TABLE 4.3-1 (SHEET 1 OF 3)

REACTOR CORE DESCRIPTION

Active core		
Equivalent diameter (in.)		132.7
Active fuel height (in.)		143.7
Height-to-diameter ratio		1.08
Total cross section area (ft^2)		96.06
H_2O/U molecular ratio, lattice, cold	LOPAR:	2.41
	VANTAGE 5:	2.73
Reflector thickness and composition		
Top - water plus steel (in.)		10
Bottom - water plus steel (in.)		10
Side - water plus steel (in.)		15
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 (lb)	LOPAR:	222,762
	VANTAGE 5:	204,231[a]
Zircaloy weight (lb) (active core)	LOPAR:	45,296
Zircaloy/ZIRLO™ weight (lb) (active core)	VANTAGE 5:	45,914
Number of grids per assembly	LOPAR:	8 R type
	VANTAGE 5:	2 nonmixing vane type, 6 mixing vane type, 3 IFM 1 protective grid
Composition of grids	LOPAR:	Inconel-718
	VANTAGE 5:	2 Inconel-718 end grids 6 Zircaloy-4/ZIRLO™ spacer grids 3 Zircaloy-4/ZIRLO™ IFM grids 1 Inconel-718 protective grid
Weight of grids in active core (lb)	LOPAR:	Inconel-718 - 2324
	VANTAGE 5:	Inconel-718 – 332 Zircaloy-4/ZIRLO™ 3547
Number of guide thimbles per assembly		24
Composition of guide thimbles		Zircaloy-4/ZIRLO™

TABLE 4.3-1 (SHEET 2 OF 3)

Diameter of guide thimbles, upper part (in.)	LOPAR:	0.450 ID x 0.482 OD
	VANTAGE 5:	0.442 ID x 0.474 OD
Diameter of guide thimbles, lower part (in.)	LOPAR:	0.397 ID x 0.430 OD
	VANTAGE 5:	0.397 ID x 0.430 OD
Diameter of instrument guide thimbles (in.)	LOPAR:	0.450 ID x 0.482 OD
	VANTAGE 5:	0.442 ID x 0.474 OD
Fuel rods		
Number		50,952
Outside diameter (in.)	LOPAR:	0.374
	VANTAGE 5:	0.360
Diameter gap (in.)	LOPAR:	0.0065
	VANTAGE 5:	0.0062
Clad thickness (in.)		0.0225
Clad material		Zircaloy-4/ZIRLO™
Fuel pellets (first cycle)		
Material		UO_2 sintered
Density (% of theoretical)		95
First Cycle fuel enrichments (weight percent)		
Region 1		2.10
Region 2		2.60
Region 3		3.10
Diameter (in.)		0.3225
Length (in.)	Unit 1:	0.530
	Unit 2:	0.387
Mass of UO_2 per ft of fuel rod (lb/ft)	Unit 1:	0.366
	Unit 2:	0.364
Fuel pellets (typical reload)		
Material		UO_2 sintered
Density (% of theoretical)		95
Diameter (in.)	LOPAR:	0.3225
	VANTAGE 5:	0.3088 (non-IFBA)
Length (in.)	LOPAR:	0.387
	VANTAGE 5:	0.370
	Axial Blanket Pellet:	0.462/0.500

TABLE 4.3-1 (SHEET 3 OF 3)

Mass of UO_2 per ft of fuel rod (lb/ft)	LOPAR: VANTAGE 5:	0.364 0.334[(a)]
RCCAs		
Neutron absorber		Hafnium or Ag-In-Cd
Diameter (in.)		0.341
Density (lb/in.^{3})		Hafnium 0.454, Ag-In-Cd 0.367
Cladding material		Type 304, cold-worked SS
Clad thickness (in.)		0.0185
Number of clusters, full-length		53
Number of absorber rods per cluster		24
BA rods (first cycle)		
Number		1518
Material		Borosilicate glass
OD (in.)		0.381
Inner tube, OD (in.)		0.1805
Clad material		SS
Inner tube material		SS
Boron loading (without B_2O_3 in glass rod)		12.5
Weight of boron-10 per foot of rod (lb/ft)		0.00419
Initial reactivity worth (%$\Delta\rho$)		~7.6 (hot) ~5.5 (cold)
Burnable Absorbers (reload cycles)		
Wet Annular Burnable Absorber Rods:		
Material		$A1_2O_3$-B_4C
OD (in.)		0.381
Inner tube, OD (in.)		0.267
Clad material		Zircaloy
Inner tube material		Zircaloy
B_{10} content (mg/cm)		6.03
Integral Fuel Burnable Absorbers:		
Material		ZrB_2
Typical B_{10} content (mg/in.)		1.50 to 2.25 (1.0X to 1.5X)
Excess reactivity (first cycle)		
Maximum fuel assembly K∞ (cold, clean, unborated water)		1.39
Maximum core reactivity (cold, zero power, beginning of cycle, zero soluble boron)		1.222

a. The decrease in fuel weight due to annular axial blanket pellets is not considered.

TABLE 4.3-2 (SHEET 1 OF 2)

NUCLEAR DESIGN PARAMETERS

(First Cycle)

Core average linear power, including densification effects (kW/ft)	5.45	
Total heat flux hot channel factor, F_Q	2.32	
Nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$	1.55	
Reactivity coefficients[a]	Design Limits	Best Estimate
Doppler-only power coefficients, see figure 15.1-5, (pcm/% power)[b]		
Upper curve	-19.4 to -12.6	-15 to -11
Lower curve	-10.2 to -6.7	-13 to -9
Doppler temperature coefficient (pcm/°F)[b]	-2.9 to -1.4	-2.4 to -1.7
Moderator temperature coefficient (pcm/°F)[b]	0 to -40	-1 to -36
Boron coefficient (pcm/ppm)[b]	-12.8 to -7.5	-16 to -7
Rodded moderator density (pcm/g/cm)^3[b]	≤0.43 x 10^5	≤0.35 x 10^5
Delayed neutron fraction and lifetime		
β_{eff} BOL, (EOL)	0.0075 (0.0044)[c]	
ℓ*, BOL, (EOL) μs		19.4 (18.1)
Control rods		
Rod requirements	See table 4.3-3.	
Maximum bank worth (pcm)	<2000	
Maximum ejected rod worth	See chapter 15.	
Bank worth HZP no overlap (pcm)[b]	BOL, Xe free	EOL Eq. Xe
Bank D	650	750
Bank C	1250	1450
Bank B	1200	1400
Bank A	500	450
Radial factor (BOL to EOL)		
Unrodded	1.37 to 1.28	
D bank	1.50 to 1.45	
D + C banks	1.60 to 1.45	
D + C + B banks	1.80 to 1.55	

TABLE 4.3-2 (SHEET 2 OF 2)

Boron concentrations (ppm)	
Zero power, k_{eff} = 0.99, cold[(d)] RCCAs out	1435
Zero power, k_{eff} = 0.99, hot[(e)] RCCAs out	1408
Design basis refueling boron concentration	2000
Zero power, $k_{eff} \leq 0.95$, cold[(d)] RCCAs in	1327
Zero power, k_{eff} = 1.00, hot[(e)] RCCAs out	1307
Full power, no xenon, k_{eff} = 1.0, hot RCCAs out	1178
Full power, equilibrium xenon, k_{eff} = 1.0, hot RCCAs out	882
Reduction with fuel burnup	
First cycle (ppm/GWd/tonne uranium)[(f)]	See figure 4.3-3.
Reload cycle (ppm/GWd/tonne uranium)	~100

a. Uncertainties are given in paragraph 4.3.3.3.

b. 1 pcm = 10^{-5} $\Delta\rho$ where $\Delta\rho$ is calculated from two statepoint values of k_{eff} by 1n (k_1/k_2).

c. Bounding lower value used for safety analysis.

d. Cold means 68°F, 1 atm.

e. Hot means 557°F, 2250 psia.

f. 1 GWd = 1000 MWd. During the first cycle, fixed BP rods are present which significantly reduce the boron depletion rate compared to reload cycles.

TABLE 4.3-3

REACTIVITY REQUIREMENTS FOR ROD CLUSTER CONTROL ASSEMBLIES

Reactivity Effects (Percent)	BOL (First Cycle)	EOL (First Cycle)	EOL Representative Equilibrium Cycle)
1. Control requirements			
Fuel temperature, Doppler (%Δρ)	1.37	1.21	1.10
Moderator temperature (%Δρ)[a]	0.15	1.15	1.15
Redistribution (%Δρ)	0.50	0.85	0.98
Rod insertion allowance (%Δρ)	0.50	0.50	0.50
2. Total control (%Δρ)	2.52	3.71	3.73
3. Estimated RCCA worth (53 rods)			
a. All full-length assemblies inserted (%Δρ)	7.54	7.42	6.76
b. All assemblies but one (highest worth) inserted (%Δρ)	6.46	6.39	5.78
4. Estimated RCCA credit with 10-percent adjustment to accommodate uncertainties, item 3b minus 10 percent (%Δρ)	5.82	5.75	5.20
5. Shutdown margin available, item 4 minus item 2 (%Δρ)	3.30	2.04	1.47[b]

a. Includes void effects.

b. The design basis minimum shutdown is 1.3 percent.

TABLE 4.3-4

DELETED

TABLE 4.3-5

AXIAL STABILITY INDEX PRESSURIZED WATER REACTOR CORE WITH A 12-FT HEIGHT

Burnup (MWd/tonne uranium)	F_Z	C_B (ppm)	Stability Index (h^{-1}) Exp	Calc
1550	1.34	1065	-0.041	-0.032
7700	1.27	700	-0.014	-0.006
5090[a]			-0.0325	-0.0255
		Radial Stability Index		
2250[b]			-0.068	-0.07

a. Four-loop plant, 12-ft core in cycle 1, axial stability test.

b. Four-loop plant, 12-ft core in cycle 1, radial (X-Y) stability test.

TABLE 4.3-6

TYPICAL NEUTRON FLUX LEVELS ($n/cm^2/s$) AT FULL POWER

	E>1.0 MeV	0.111 MeV < E <1.0 MeV	0.3 eV ≤ E <0.111 MeV	<E 0.3 eV
Core center	9.98×10^{13}	1.11×10^{14}	2.17×10^{14}	5.36×10^{13}
Core outer radius at midheight	4.24×10^{13}	4.85×10^{13}	9.52×10^{13}	2.21×10^{13}
Core top, on axis	2.62×10^{13}	2.13×10^{13}	1.31×10^{14}	4.35×10^{13}
Core bottom, on axis	2.70×10^{13}	2.25×10^{13}	1.33×10^{14}	4.74×10^{13}
Pressure vessel ID azimuthal peak,	2.08×10^{10}	2.83×10^{10}	6.18×10^{10}	1.20×10^{11}

TABLE 4.3-7

COMPARISON OF MEASURED AND CALCULATED DOPPLER DEFECTS

Plant	Fuel Type	Core Burnup (MWd/tonne uranium)	Measured (pcm)[a]	Calculated (pcm)
1	Air filled	1800	1700	1710
2	Air filled	7700	1300	1440
3	Air and helium filled	8460	1200	1210

a. pcm = 10^5 x ln (k /k)

TABLE 4.3-8

SAXTON CORE II ISOTOPICS ROD MY, AXIAL ZONE 6

Atom Ratio	Measured[a]	2 σ Precision (%)	LEOPARD Calculation
U-234/U	4.65×10^{-5}	±29	4.60×10^{-5}
U-235/U	5.74×10^{-3}	±0.9	5.73×10^{-3}
U-236/U	3.55×10^{-4}	±5.6	3.74×10^{-4}
U-238/U	0.99386	±0.01	0.99385
Pu-238/Pu	1.32×10^{-3}	±2.3	1.222×10^{-3}
Pu-239/Pu	0.73791	±0.03	0.74497
Pu-240/Pu	0.19302	±0.2	0.19102
Pu-241/Pu	6.014×10^{-2}	±0.3	5.74×10^{-2}
Pu-242/Pu	5.81×10^{-3}	±0.9	5.38×10^{-3}
Pu/U[b]	5.938×10^{-2}	±0.7	5.970×10^{-2}
Np-237/U-238	1.14×10^{-4}	±15	0.86×10^{-4}
Am-241/Pu-239	1.23×10^{-2}	±15	1.08×10^{-2}
Cm-242/Pu-239	1.05×10^{-4}	±10	1.11×10^{-4}
Cm-244/Pu-239	1.09×10^{-4}	±20	0.98×10^{-4}

a. Reported in reference 34.

b. Weight ratio.

TABLE 4.3-9

CRITICAL BORON CONCENTRATIONS (ppm) HZP, BOL

Plant Type	Measured	Calculated
2-loop, 121 assemblies, 10-ft core	1583	1589
2-loop, 121 assemblies, 12-ft core	1625	1624
2-loop, 121 assemblies, 12-ft core	1517	1517
3-loop, 157 assemblies, 12-ft core	1169	1161
3-loop, 157 assemblies, 12-ft core	1344	1319
4-loop, 193 assemblies, 12-ft core	1370	1355
4-loop, 193 assemblies, 12-ft core	1321	1306

TABLE 4.3-10

COMPARISON OF MEASURED AND CALCULATED AG-IN-CD ROD WORTH

2-Loop Plant, 121 Assemblies, 10-ft Core	Measured (pcm)	Calculated (pcm)
Group B	1885	1893
Group A	1530	1649
Shutdown group	3050	2917
ESADA critical, 0.69-in. pitch[a] 2 w/o PuO_2, 8% Pu-240, 9 control rods		
6.21-in. rod separation	2250	2250
2.07-in. rod separation	4220	4160
1.38-in. rod separation	4100	4019

BENCHMARK CRITICAL EXPERIMENT HAFNIUM CONTROL ROD WORTH

Control Rod Configuration	No. of Fuel Rods	Measured[b] Worth (Δppm B-10)	Calculated[b] Worth (Δppm B-10)
9 hafnium rods	1192	138.3	141.0

a. Reported in reference 35.

b. Calculated and measured worths are given in terms of an equivalent charge in B-10 concentration.

TABLE 4.3-11

COMPARISON OF MEASURED AND CALCULATED MODERATOR COEFFICIENTS AT HZP, BOL

Plant Type/ Control Bank Configuration	Measured α_{iso} [a] (pcm/°F)	Calculated α_{iso} (pcm/°F)
3-loop, 157-assembly, 12-ft core		
D at 160 steps	-0.50	-0.50
D in, C at 190 steps	-3.01	-2.75
D in, C at 28 steps	-7.67	-7.02
B, C, and D in	-5.16	-4.45
2-loop, 121-assembly, 12-ft core		
D at 180 steps	+0.85	+1.02
D in, C at 180 steps	-2.40	-1.90
C and D in, B at 165 steps	-4.40	-5.58
B, C, and D in, A at 174 steps	-8.70	-8.12
4-loop, 193-assembly, 12-ft core		
All Rods Out	-0.52	-1.2
D in	-4.35	-5.7
D and C in	-8.59	-10.0
D, C, and B in	-10.14	-10.55
D, C, B, and A in	-14.63	-14.45

a. Isothermal coefficients, which include the Doppler effect in the fuel.

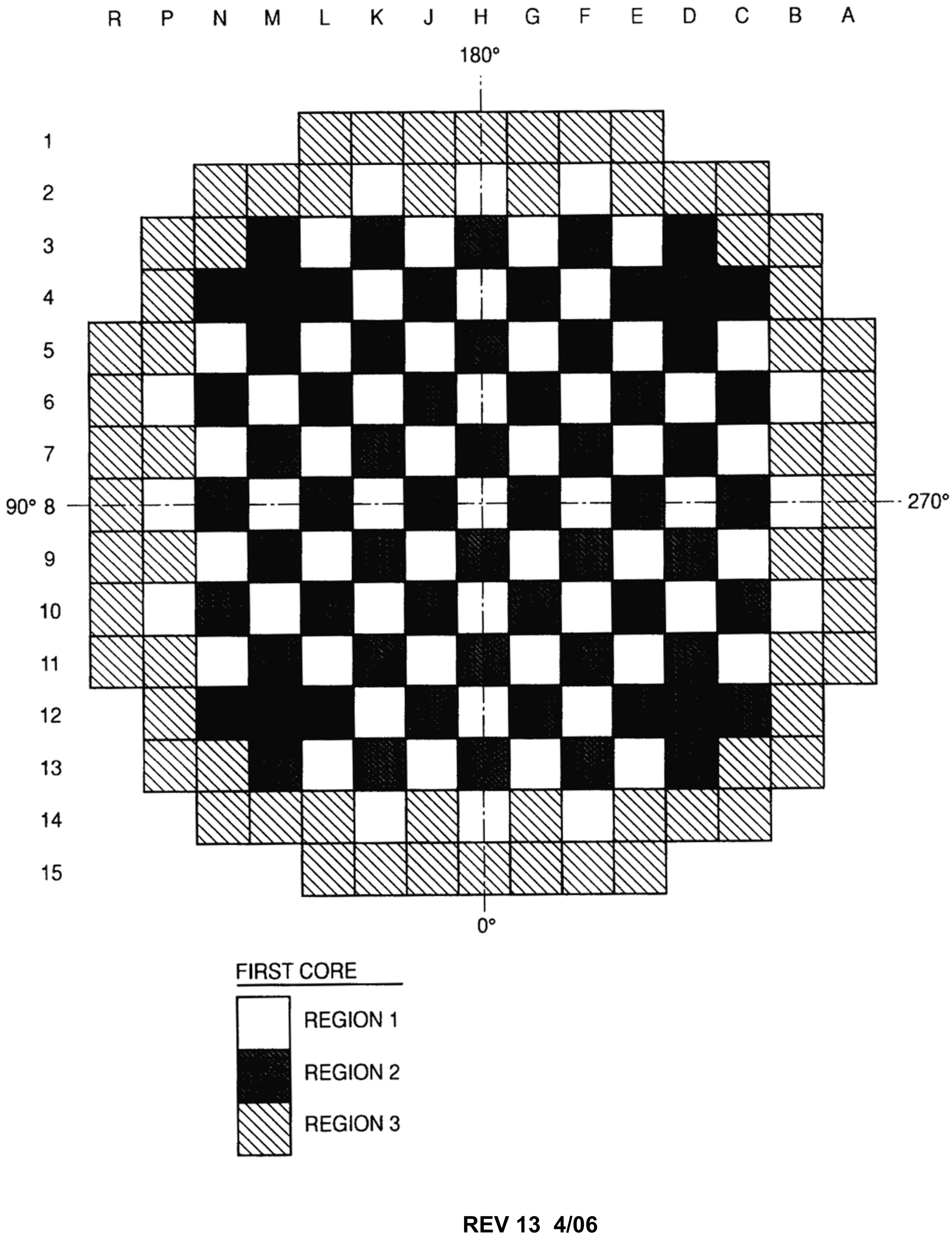
$$\alpha_{iso} = \frac{10^5 \ln \frac{k_2}{k_1}}{\Delta T^\circ F}$$

TABLE 4.3-12

BENCHMARK CRITICAL EXPERIMENTS

Description of Experiments[a]	Number of Experiments	LEOPARD k_{eff} Using Experimental Bucklings
UO_2		
Al clad	14	1.0012
SS clad	19	0.9963
Borated H_2O	7	0.9989
Subtotal	40	0.9985
U-Metal		
Al clad	41	0.9995
Unclad	20	0.9990
Subtotal	61	0.9993
Total	101	0.9990

a. Reported in reference 33.



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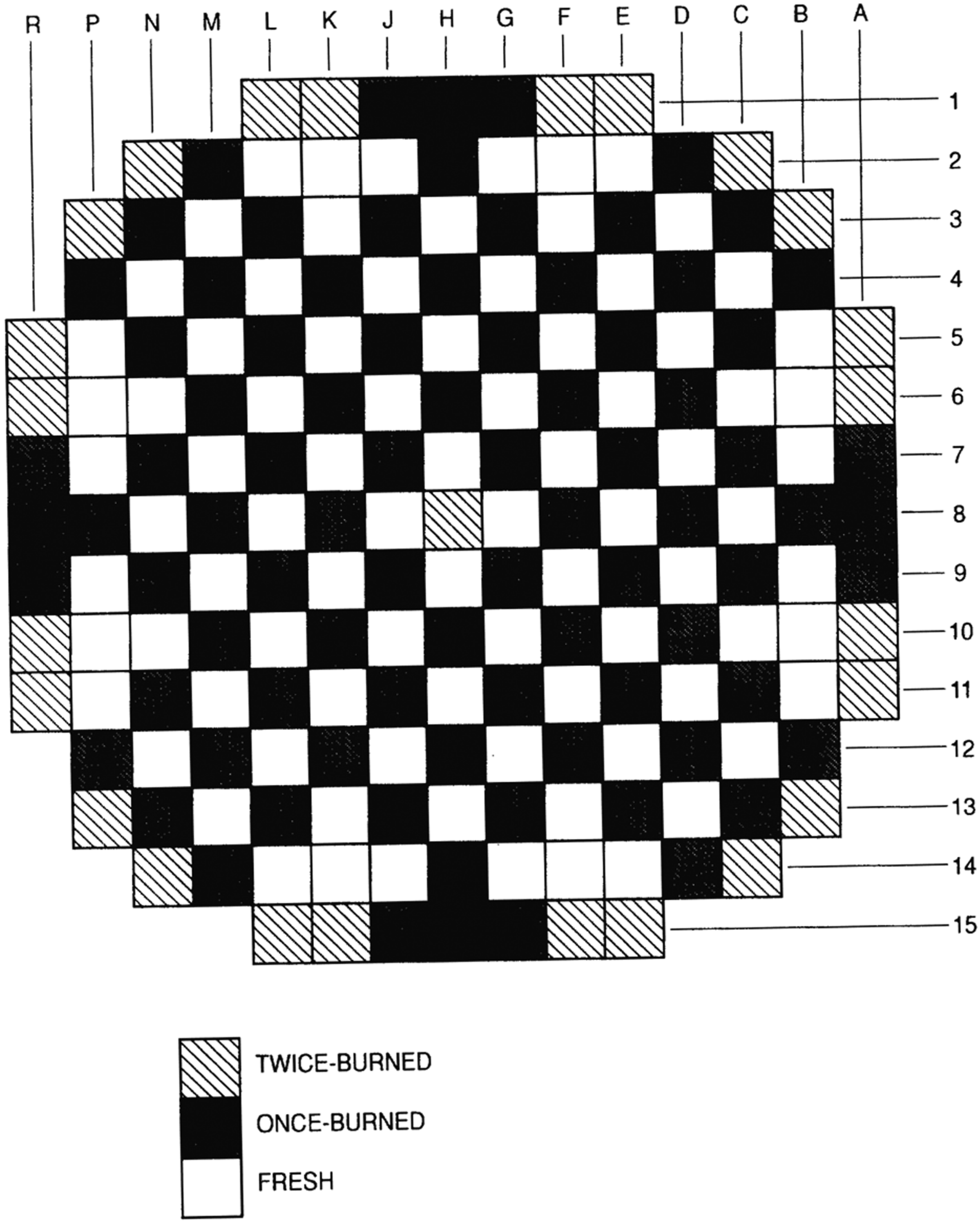
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	FUEL LOADING ARRANGEMENT (INITIAL CYCLE)
		FIGURE 4.3–1 (SHEET 1 OF 2)



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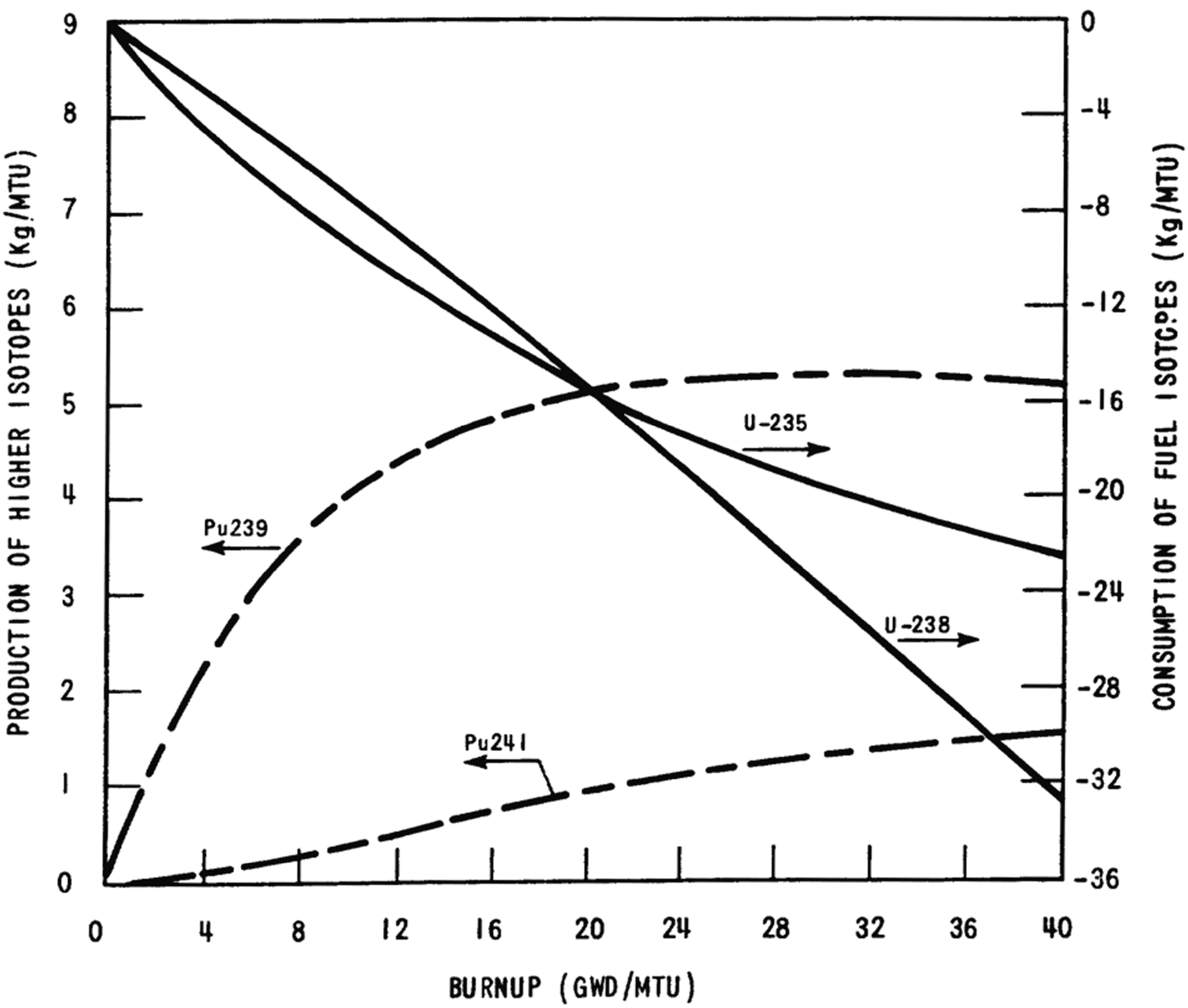
FUEL LOADING ARRANGEMENT
(TYPICAL RELOAD CYCLE)

FIGURE 4.3–1 (SHEET 2 OF 2)



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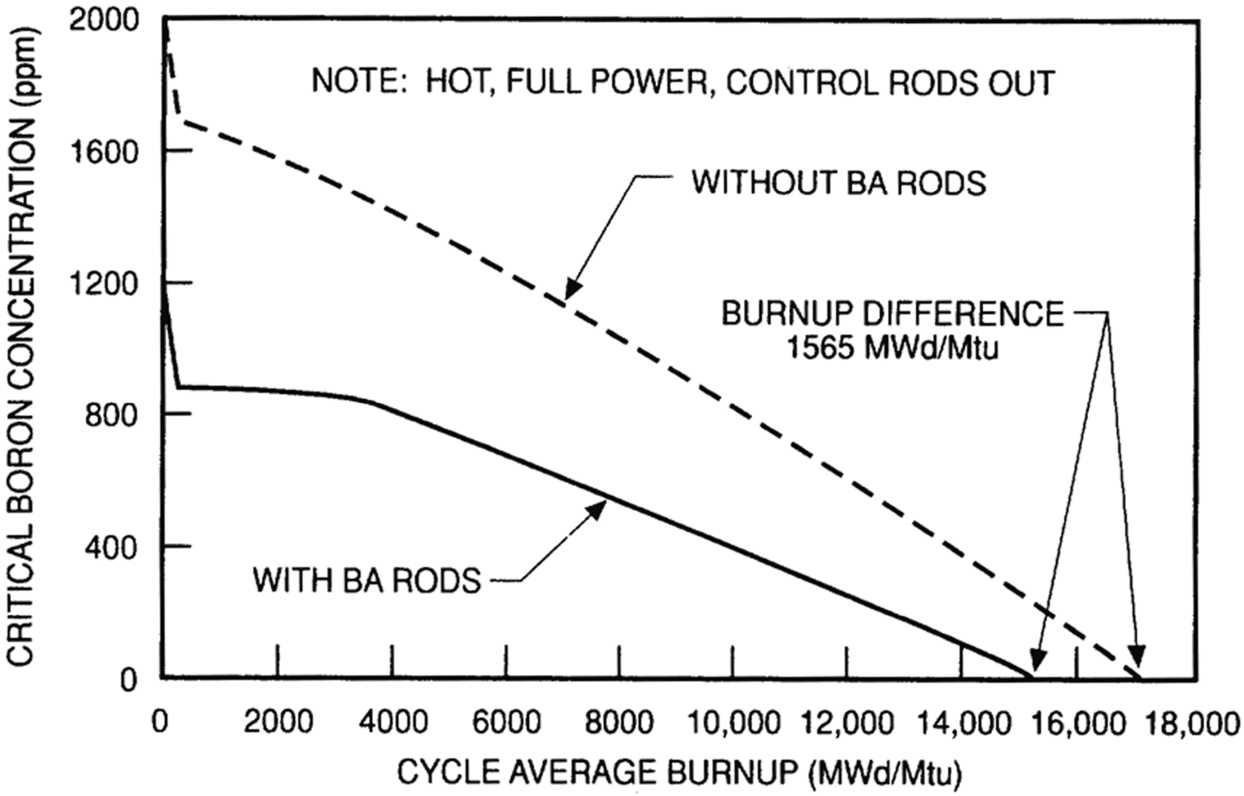
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PRODUCTION AND CONSUMPTION
OF HIGHER ISOTOPES

FIGURE 4.3–2



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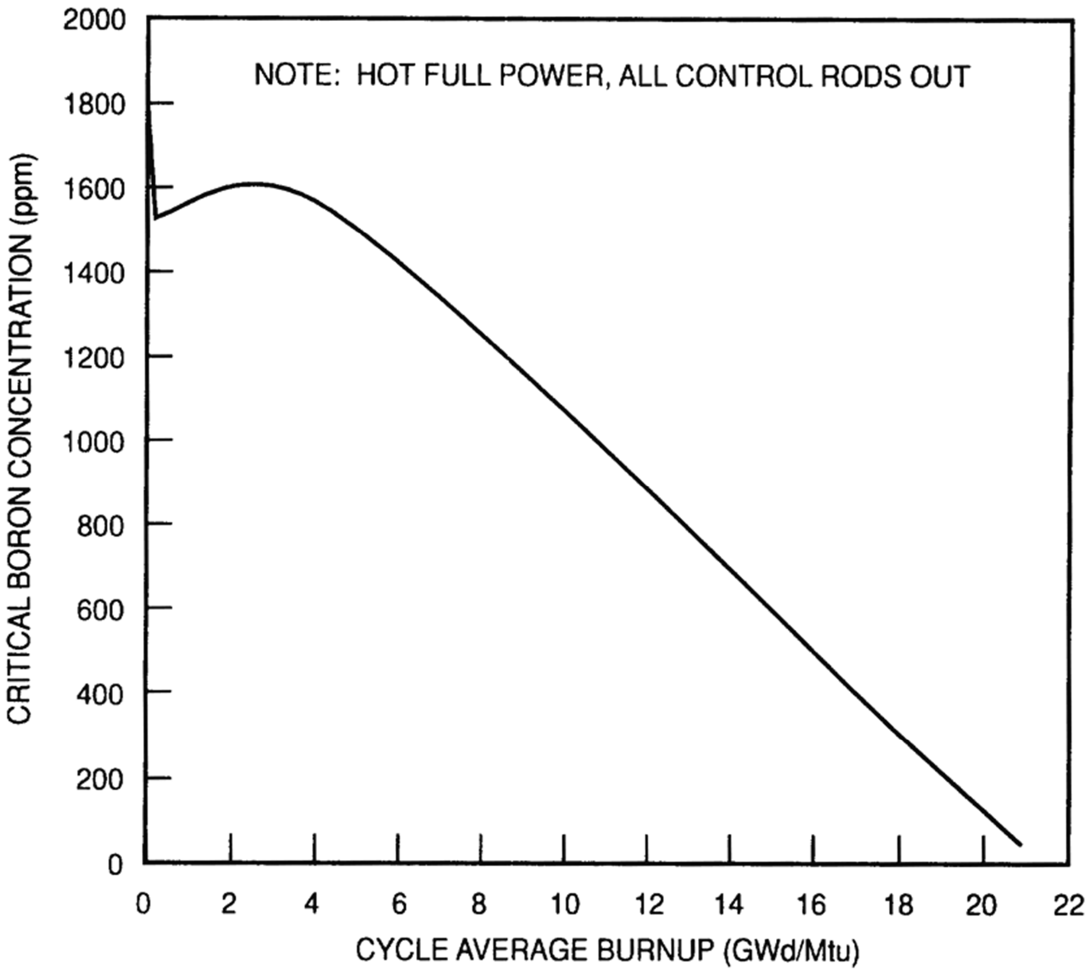
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	BORON CONCENTRATION VERSUS FIRST CYCLE BURNUP WITH AND WITHOUT BA RODS
	FIGURE 4.3–3 (SHEET 1 OF 2)



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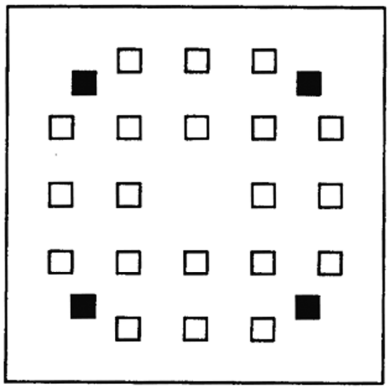
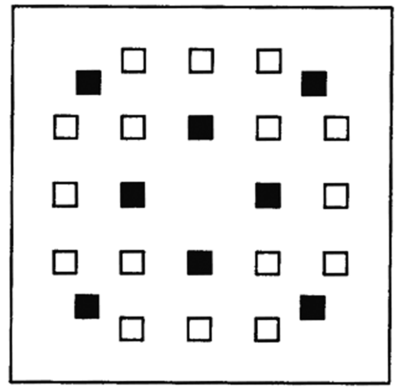
UNIT 1 AND UNIT 2

BORON CONCENTRATION VERSUS RELOAD CYCLE BURNUP WITH INTEGRAL FUEL BURNABLE ABSORBERS

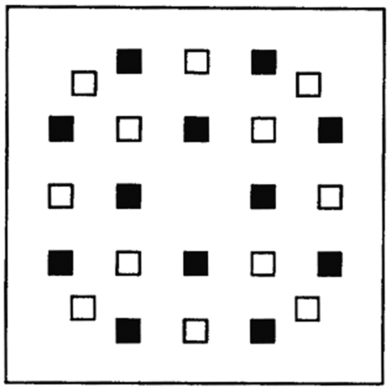
FIGURE 4.3–3 (SHEET 2 OF 2)



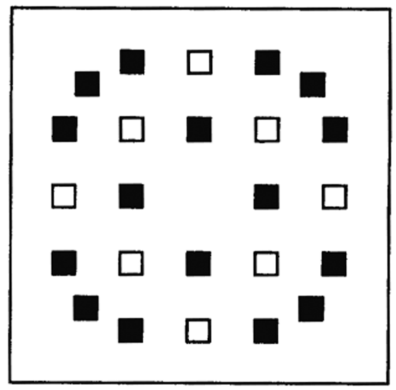
4 FRESH BA



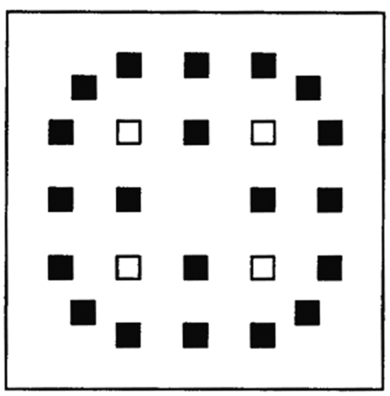
8 FRESH BA



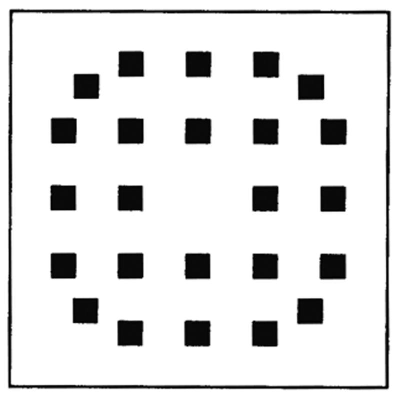
12 FRESH BA



16 FRESH BA



20 FRESH BA



24 FRESH BA

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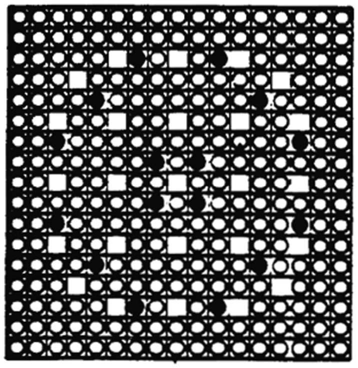
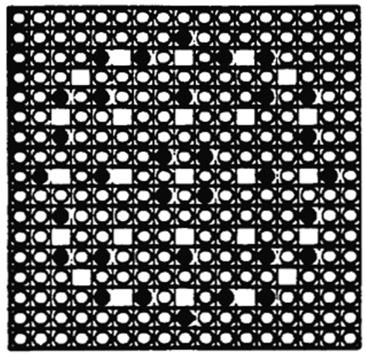
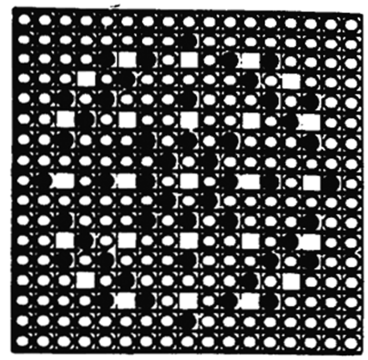
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	ROD ARRANGEMENT WITHIN AN ASSEMBLY (WABA) FIGURE 4.3–4 (SHEET 1 OF 2)



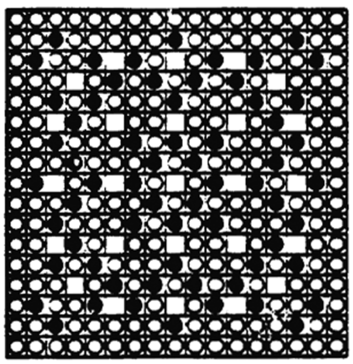
16 IFBA ROD ASSEMBLY



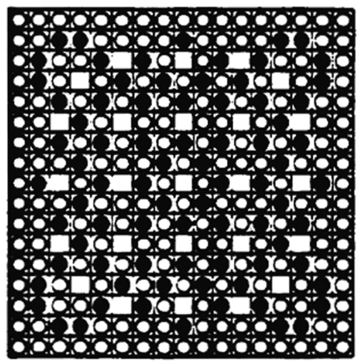
32 IFBA ROD ASSEMBLY



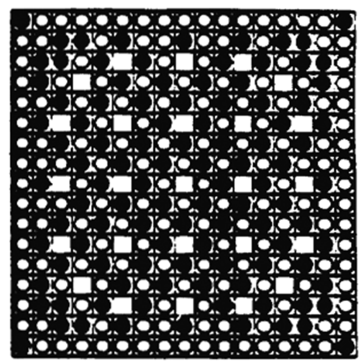
48 IFBA ROD ASSEMBLY



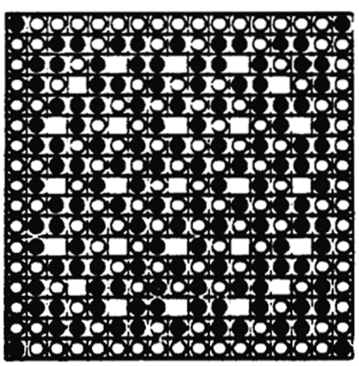
64 IFBA ROD ASSEMBLY



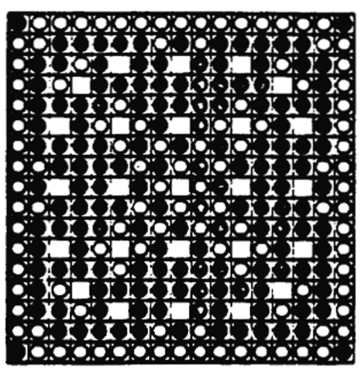
80 IFBA ROD ASSEMBLY



104 IFBA ROD ASSEMBLY



128 IFBA ROD ASSEMBLY



156 IFBA ROD ASSEMBLY

○ FUEL ROD

● FUEL ROD WITH IFBA

□ GUIDE TUBE/INST. TUBE

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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL ROD ARRANGEMENT WITHIN AN ASSEMBLY (IFBA)

FIGURE 4.3–4 (SHEET 2 OF 2)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2					4						4				
3				12		16		12 4S		16		12			
4			12		16		12		12		16		12		
5		4		16		16		12		16		16		4	
6			16		16		12		12		16		16		
7				12		12		8		12		12			
8 90°			12		12		8		8		12		12		
9				12		12		8		12		12			
10			16		16		12		12		16		16		
11		4		16		16		12		16		16		4	
12			12		16		12		12		16		12		
13				12		16		12 4S		16		12			
14					4						4				
15															

0°

NUMBER INDICATES NUMBER OF BURNABLE ABSORBER RODS

S INDICATES SOURCE ROD

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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOADING PATTERN (TYPICAL DISCRETE BA)

FIGURE 4.3–5 (SHEET 1 OF 2)

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2					48		80	128	80		48				
3				128		128		128 4S		128		128			
4			128		128		128		128		128		128		
5		48		128		128		128		128		128		48	
6			128		128		128		128		128		128		
7		80		128		128		128		128		128		80	
8		128	128		128		128		128		128		128	128	
9		80		128		128		128		128		128		80	
10			128		128		128		128		128		128		
11		48		128		128		128		128		128		48	
12			128		128		128		128		128		128		
13				128		128		128 4S		128		128			
14					48		80	128	80		48				
15															

270°

0°

NUMBER INDICATES NUMBER OF RODS WITH IFBA

S INDICATES SOURCE ROD

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UNIT 1 AND UNIT 2

LOADING PATTERN (TYPICAL IFBA)

FIGURE 4.3–5 (SHEET 2 OF 2)

0.989	1.194	1.015	1.230	1.083	1.359	1.217	0.532
1.194	1.025	1.209	1.045	1.288	1.117	1.227	0.582
1.015	1.209	1.030	1.263	1.137	1.336	1.278	0.531
1.230	1.047	1.263	1.080	1.305	1.085	1.040	0.314
1.083	1.288	1.137	1.305	1.161	1.091	0.520	
1.359	1.118	1.335	1.085	1.091	0.843	0.258	
1.217	1.222	1.273	1.039	0.520	0.258		
0.532	0.576	0.524	0.314				

1 AP	AVERAGE POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, UNRODDED, HOT FULL POWER, NO XENON

FIGURE 4.3–6

1.024	1.218	1.035	1.242	1.087	1.345	1.201	0.535
1.218	1.047	1.227	1.058	1.288	1.111	1.209	0.582
1.035	1.227	1.046	1.270	1.138	1.332	1.254	0.531
1.242	1.059	1.289	1.086	1.299	1.078	1.027	0.317
1.087	1.288	1.138	1.299	1.155	1.086	0.525	
1.345	1.111	1.321	1.079	1.086	0.844	0.265	
1.201	1.205	1.249	1.027	0.525	0.255		
0.535	0.577	0.525	0.318				

AP	ASSEMBLY POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, UNRODDED, HOT FULL POWER, EQUILIBRIUM XENON

FIGURE 4.3–7

0.831	1.190	1.040	1.262	1.110	1.379	1.234	0.550
1.190	1.037	1.236	1.072	1.310	1.133	1.239	0.598
1.040	1.236	1.054	1.277	1.141	1.333	1.273	0.541
1.262	1.072	1.277	1.071	1.264	1.060	1.028	0.320
1.110	1.310	1.141	1.253	1.045	1.037	0.512	
1.379	1.134	1.332	1.060	1.037	0.807	0.255	
1.234	1.235	1.269	1.028	0.513	0.255		
0.550	0.593	0.535	0.320				

AP	ASSEMBLY POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, GROUP D AT ROD INSERTION LIMITS, HOT FULL POWER, EQUILIBRIUM XENON

FIGURE 4.3–8

1.028	1.309	1.028	1.304	1.035	1.351	1.259	0.554
1.309	1.043	1.305	1.031	1.318	1.048	1.236	0.587
1.028	1.305	1.027	1.316	1.075	1.329	1.190	0.517
1.304	1.031	1.316	1.039	1.321	1.009	0.935	0.315
1.035	1.319	1.076	1.321	1.082	1.128	0.532	
1.351	1.048	1.329	1.009	1.128	0.895	0.289	
1.259	1.234	1.188	0.986	0.533	0.289		
0.554	0.534	0.513	0.316				

AP	ASSEMBLY POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR MIDDLE OF LIFE, UNRODDED, HOT FULL POWER, EQUILIBRIUM XENON

FIGURE 4.3–9

1.001	1.252	1.001	1.247	1.004	1.275	1.223	0.606
1.252	1.013	1.249	1.003	1.256	1.019	1.213	0.642
1.001	1.249	1.001	1.257	1.044	1.280	1.186	0.583
1.247	1.003	1.257	1.015	1.280	1.023	1.045	0.376
1.004	1.256	1.044	1.280	1.095	1.190	0.821	
1.275	1.020	1.280	1.024	1.190	1.003	0.365	
1.223	1.212	1.185	1.046	0.621	0.365		
0.506	0.640	0.580	0.377				

AP	ASSEMBLY POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR END OF LIFE, UNRODDED, HOT FULL POWER, EQUILIBRIUM XENON

FIGURE 4.3–10

0.912	1.227	1.009	1.271	1.025	1.307	1.257	0.625
1.227	1.008	1.262	1.018	1.280	1.040	1.243	0.660
1.009	1.262	1.011	1.257	1.047	1.290	1.203	0.595
1.271	1.019	1.257	1.002	1.241	1.004	1.045	0.379
1.025	1.280	1.047	1.242	0.054	1.136	0.506	
1.307	1.041	1.290	1.005	1.135	0.952	0.353	
1.257	1.241	1.202	1.046	0.507	0.353		
0.625	0.558	0.582	0.381				

AP	ASSEMBLY POWER

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UNIT 1 AND UNIT 2

NORMALIZED POWER DENSITY DISTRIBUTION NEAR END OF LIFE, GROUP D AT ROD INSERTION LIMITS, HOT FULL POWER, EQUILIBRIUM XENON

FIGURE 4.3–11

1.340	1.208	1.141	1.208	1.163	1.241	1.172	1.234	1.185	1.236	1.177	1.248	1.172	1.220	1.164	1.234	1.258
1.210	1.134	1.133	1.152	1.177	1.282	1.183	1.196	1.231	1.166	1.186	1.288	1.188	1.162	1.136	1.136	1.227
1.148	1.134	1.188	1.280	1.220		1.308	1.302		1.308	1.308		1.229	1.300	1.168	1.137	1.188
1.211	1.184	1.291		1.345	1.338	1.209	1.204	1.311	1.205	1.213	1.345	1.384		1.303	1.167	1.227
1.167	1.180	1.322	1.346	1.252	1.335	1.213	1.209	1.318	1.211	1.217	1.343	1.360	1.386	1.334	1.182	1.181
1.246	1.286		1.340	1.336		1.323	1.322		1.334	1.327		1.344	1.350		1.298	1.260
1.177	1.187	1.206	1.211	1.215	1.324	1.215	1.215	1.325	1.216	1.218	1.330	1.221	1.220	1.319	1.188	1.190
1.240	1.191	1.307	1.207	1.212	1.323	1.216	1.217	1.327	1.218	1.219	1.328	1.218	1.214	1.317	1.202	1.253
1.191	1.287		1.315	1.321		1.326	1.328		1.328	1.329		1.327	1.323		1.298	1.202
1.244	1.194	1.310	1.210	1.215	1.327	1.219	1.220	1.330	1.221	1.221	1.331	1.220	1.217	1.319	1.204	1.255
1.184	1.194	1.315	1.218	1.221	1.331	1.221	1.221	1.331	1.222	1.224	1.335	1.227	1.225	1.324	1.203	1.198
1.287	1.297		1.391	1.347		1.333	1.331		1.333	1.336		1.383	1.368		1.306	1.267
1.181	1.193	1.336	1.360	1.265	1.349	1.226	1.221	1.330	1.222	1.228	1.352	1.270	1.367	1.344	1.201	1.190
1.229	1.170	1.308		1.363	1.384	1.224	1.218	1.326	1.219	1.227	1.380	1.368		1.315	1.178	1.238
1.163	1.143	1.177	1.310	1.341		1.324	1.321		1.322	1.326		1.348	1.316	1.183	1.150	1.171
1.234	1.146	1.145	1.174	1.198	1.308	1.204	1.206	1.302	1.207	1.206	1.308	1.302	1.178	1.150	1.152	1.342
1.267	1.236	1.166	1.232	1.187	1.263	1.186	1.236	1.206	1.237	1.186	1.269	1.181	1.238	1.171	1.341	1.273

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UNIT 1 AND UNIT 2

RODWISE POWER DISTRIBUTION IN A TYPICAL ASSEMBLY (G–10) NEAR BEGINNING OF LIFE, HOT FULL POWER, EQUILIBRIUM XENON, UNRODDED CORE

FIGURE 4.3–12

1.218	1.199	1.218	1.208	1.237	1.226	1.244	1.223	1.230	1.223	1.244	1.227	1.238	1.206	1.219	1.200	1.220
1.199	1.199	1.202	1.220	1.237	1.246	1.243	1.243	1.247	1.243	1.243	1.246	1.238	1.221	1.203	1.200	1.200
1.218	1.202	1.222	1.247	1.264		1.258	1.257		1.257	1.258		1.265	1.248	1.223	1.204	1.220
1.206	1.220	1.247		1.277	1.278	1.258	1.255	1.262	1.255	1.256	1.276	1.278		1.248	1.221	1.206
1.237	1.237	1.264	1.277	1.281	1.278	1.261	1.259	1.267	1.259	1.262	1.276	1.282	1.278	1.265	1.239	1.238
1.226	1.246		1.278	1.278		1.270	1.270		1.270	1.271		1.276	1.276		1.247	1.228
1.244	1.243	1.258	1.258	1.261	1.270	1.263	1.263	1.272	1.264	1.263	1.271	1.262	1.259	1.259	1.244	1.245
1.223	1.243	1.257	1.255	1.259	1.270	1.263	1.264	1.273	1.265	1.264	1.270	1.260	1.256	1.258	1.245	1.225
1.231	1.247		1.262	1.267		1.272	1.273		1.273	1.272		1.268	1.263		1.248	1.232
1.223	1.243	1.257	1.255	1.259	1.270	1.264	1.265	1.273	1.265	1.264	1.271	1.260	1.256	1.258	1.245	1.225
1.244	1.244	1.258	1.258	1.262	1.271	1.264	1.264	1.272	1.264	1.264	1.272	1.263	1.258	1.260	1.245	1.246
1.227	1.247		1.276	1.276		1.271	1.271		1.271	1.272		1.277	1.277		1.248	1.229
1.238	1.238	1.265	1.278	1.282	1.276	1.262	1.260	1.268	1.260	1.263	1.277	1.283	1.279	1.266	1.240	1.240
1.206	1.221	1.248		1.278	1.276	1.258	1.256	1.263	1.256	1.260	1.277	1.279		1.249	1.223	1.208
1.220	1.204	1.223	1.249	1.265		1.259	1.258		1.258	1.260		1.266	1.249	1.225	1.208	1.222
1.200	1.200	1.204	1.221	1.239	1.247	1.244	1.245	1.248	1.245	1.245	1.248	1.240	1.222	1.205	1.202	1.202
1.219	1.200	1.219	1.206	1.238	1.227	1.245	1.224	1.231	1.224	1.245	1.228	1.238	1.207	1.221	1.202	1.221

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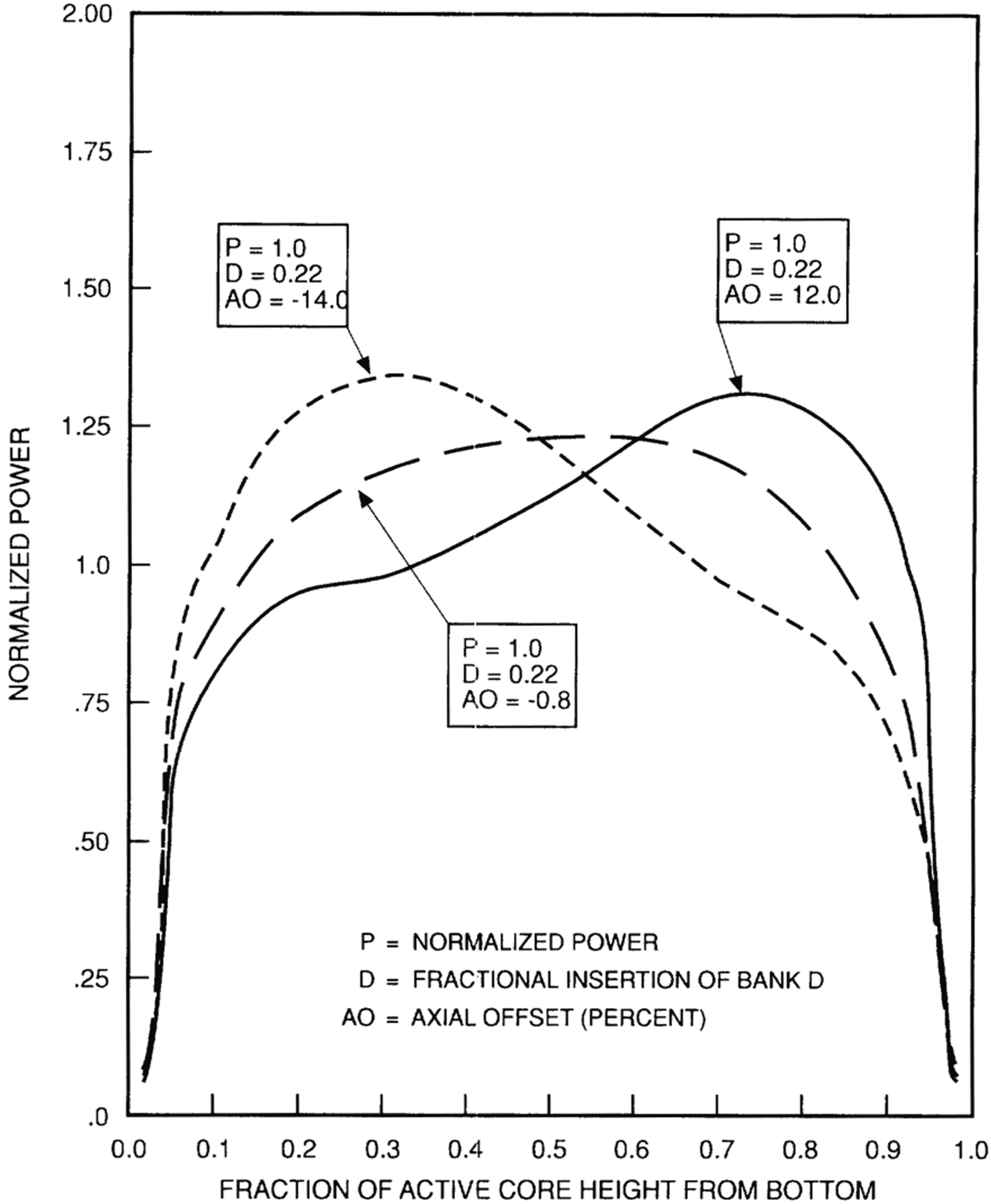
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

RODWISE POWER DISTRIBUTION IN A TYPICAL ASSEMBLY (G–10) NEAR THE END OF LIFE, HOT FULL POWER, EQUILIBRIUM XENON, UNRODDED CORE

FIGURE 4.3–13



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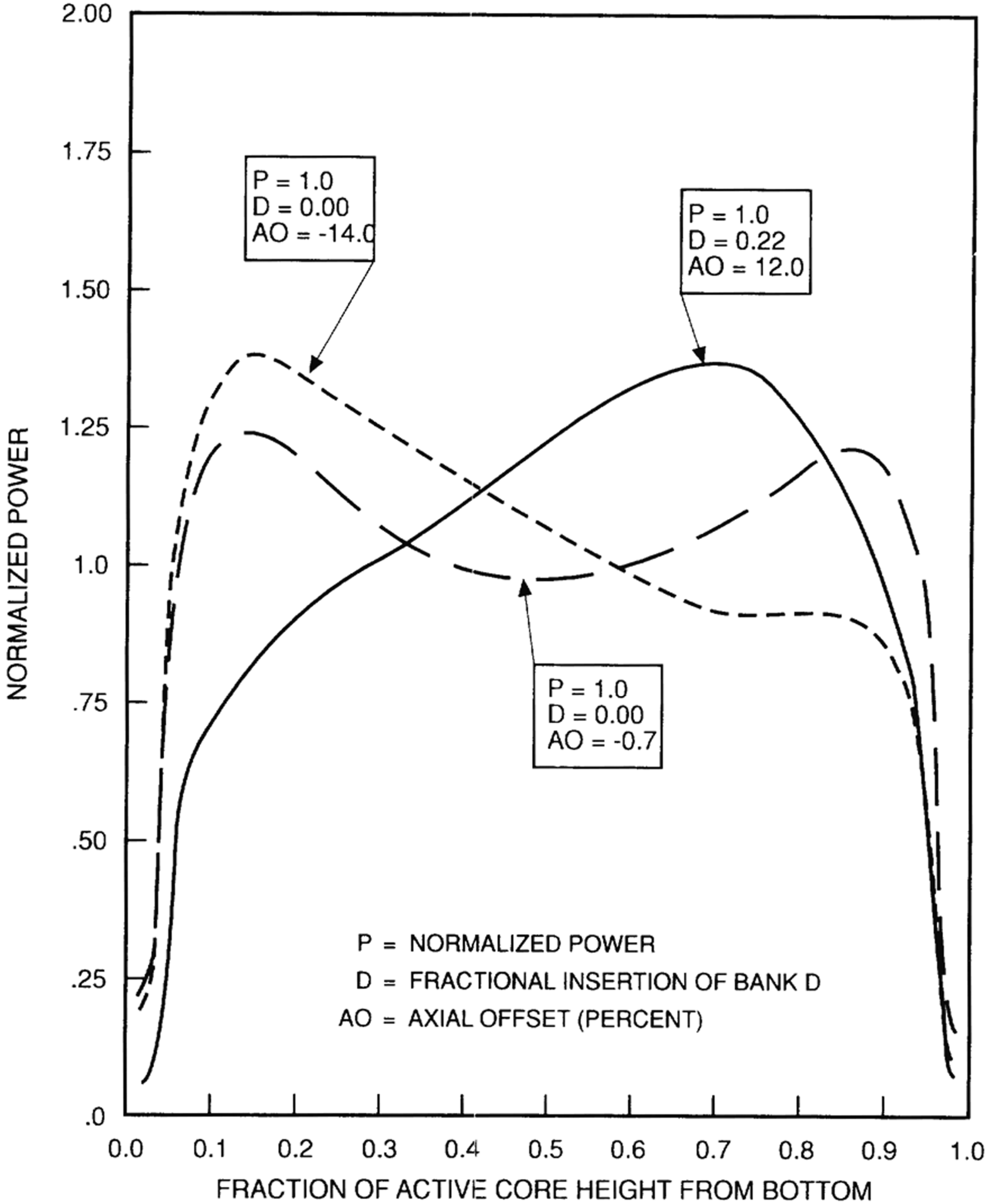
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TYPICAL AXIAL POWER SHAPES OCCURRING AT BEGINNING OF LIFE
		FIGURE 4.3–14



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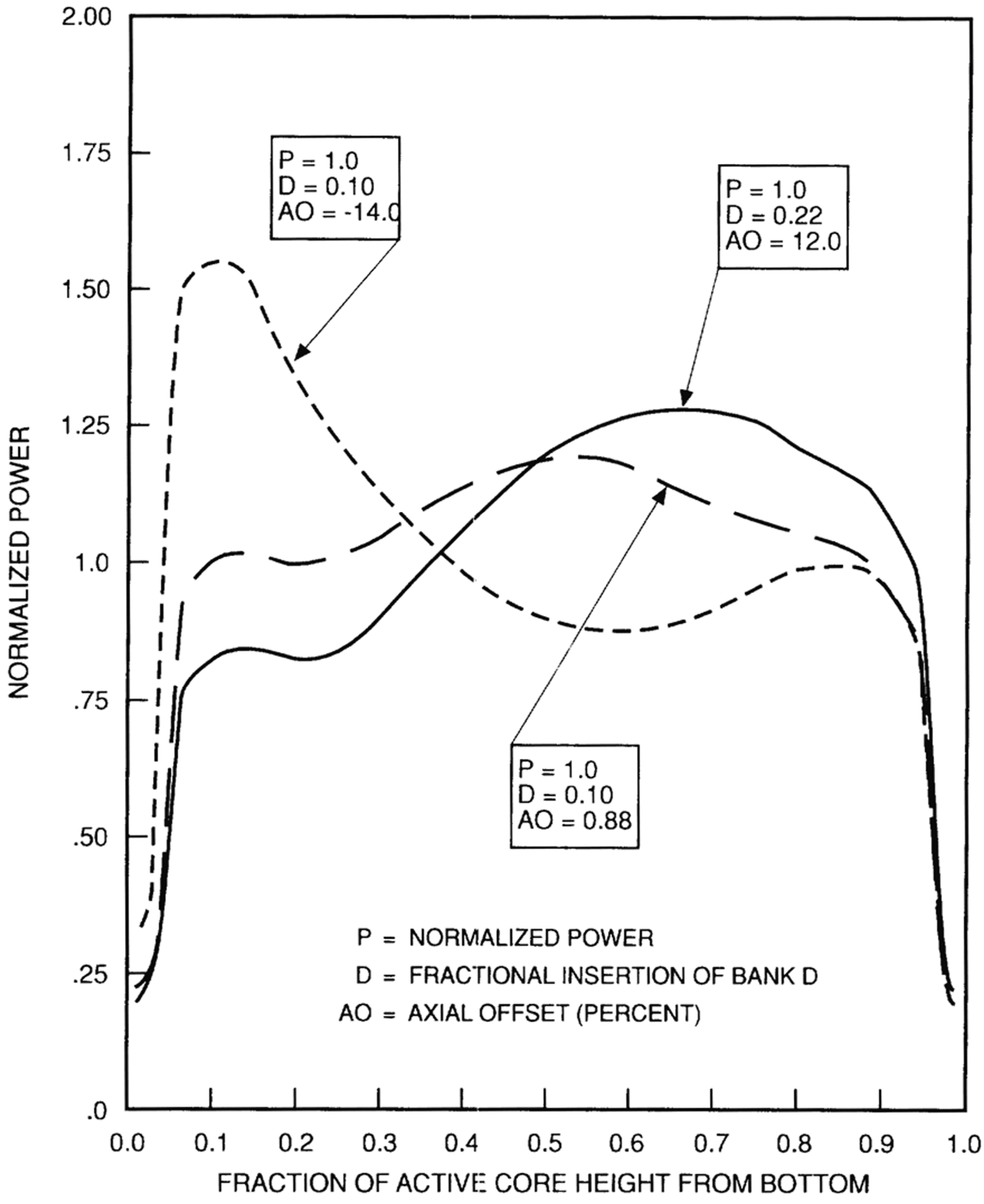
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TYPICAL AXIAL POWER SHAPES OCCURRING AT MIDDLE OF LIFE
	FIGURE 4.3–15



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UNIT 1 AND UNIT 2

TYPICAL AXIAL POWER SHAPES
OCCURRING AT END OF LIFE

FIGURE 4.3–16

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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

COMPARISON OF A TYPICAL ASSEMBLY AXIAL POWER DISTRIBUTION WITH CORE AVERAGE AXIAL DISTRIBUTION, BANK D SLIGHTLY INSERTED

FIGURE 4.3–17

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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	FLOW CHART FOR DETERMINING SPIKE MODEL FIGURE 4.3–18

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SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PREDICTED POWER SPIKE DUE TO SINGLE NONFLATTENED GAP IN THE ADJACENT FUEL FIGURE 4.3–19

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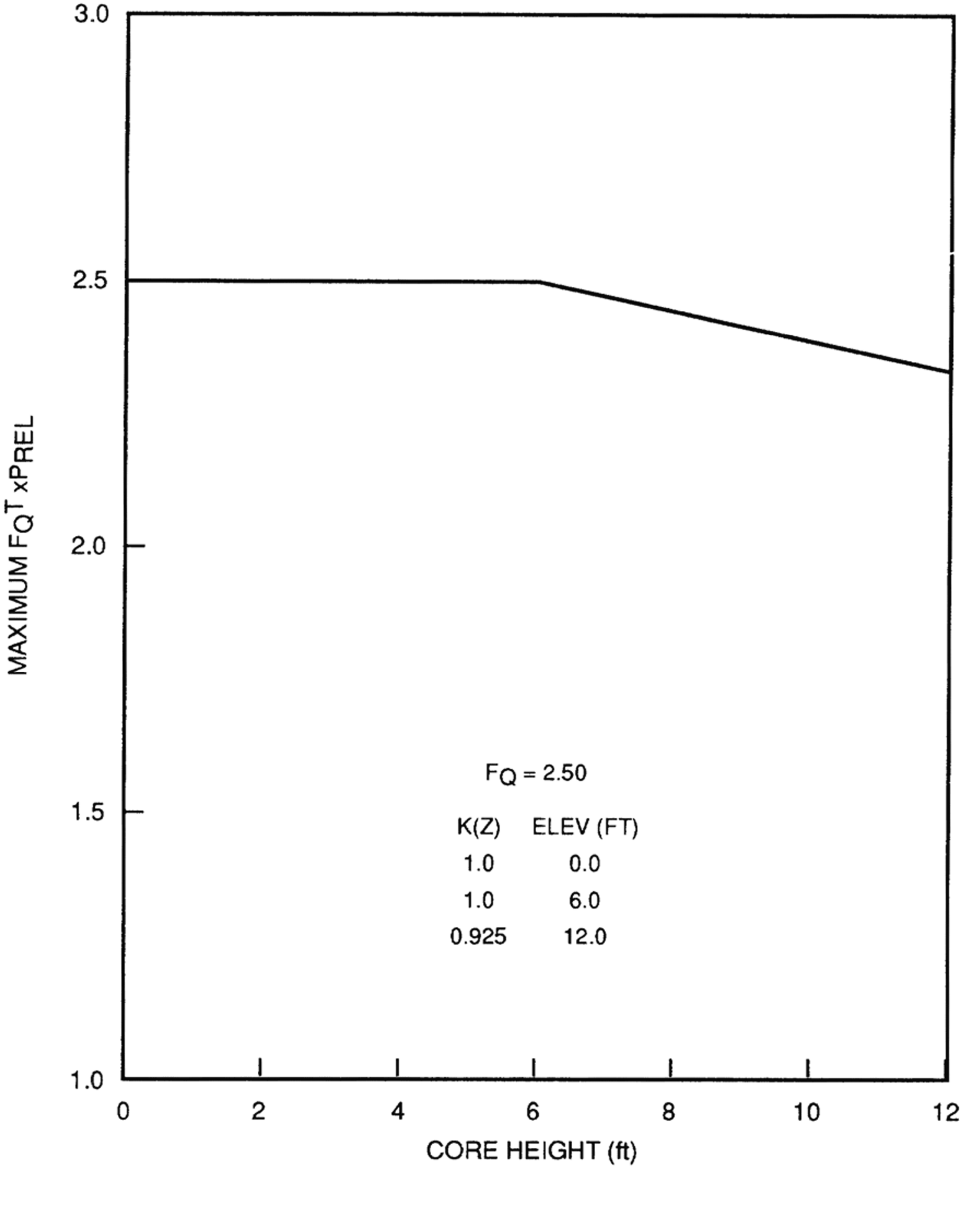
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

POWER SPIKE FACTOR AS A FUNCTION
OF AXIAL POSITION

FIGURE 4.3–20



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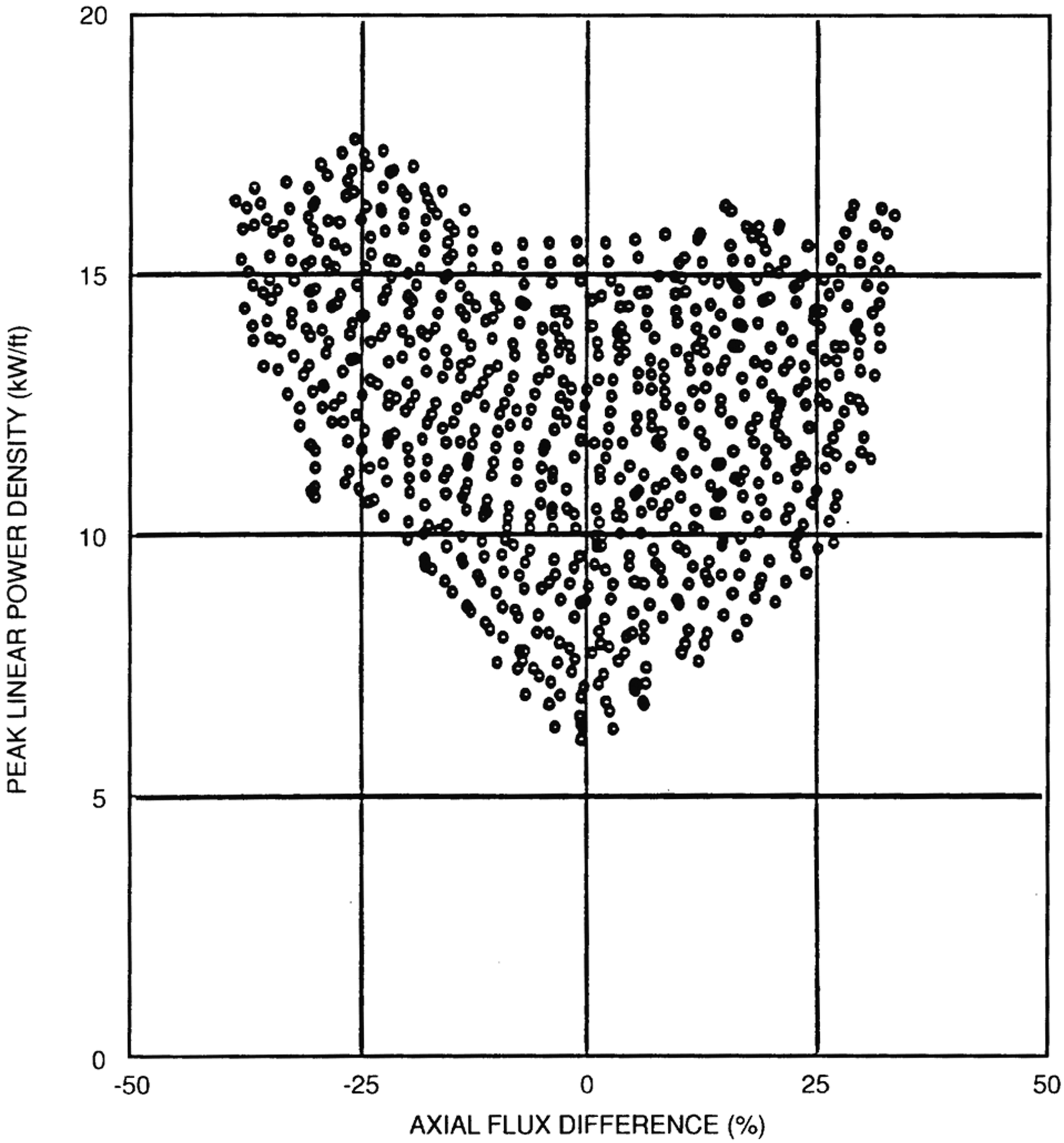
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	MAXIMUM FQ = POWER VERSUS AXIAL HEIGHT DURING NORMAL OPERATION
		FIGURE 4.3–21



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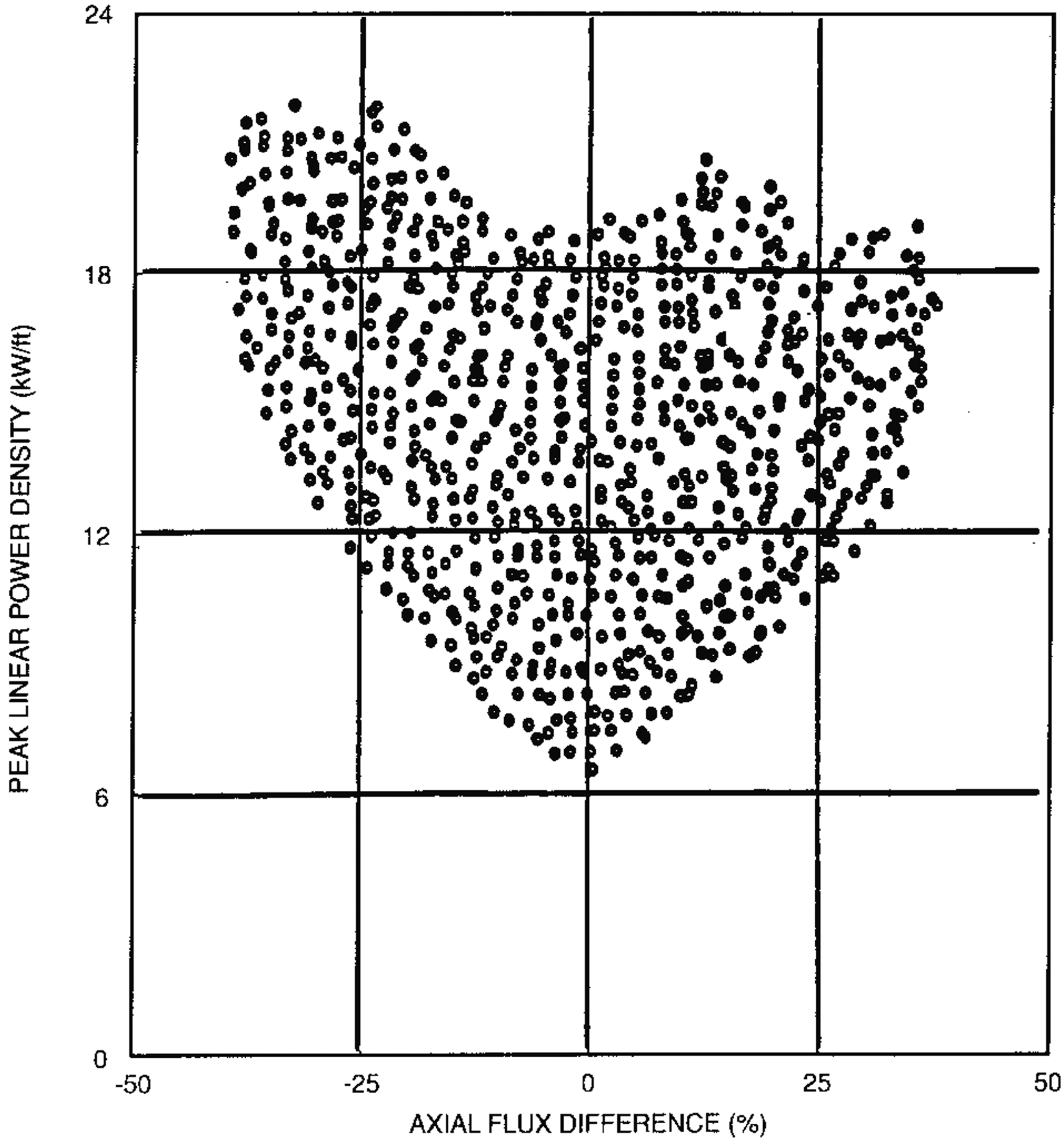
PEAK LINEAR POWER DURING CONTROL ROD MALFUNCTION OVERPOWER TRANSIENTS

FIGURE 4.3–22



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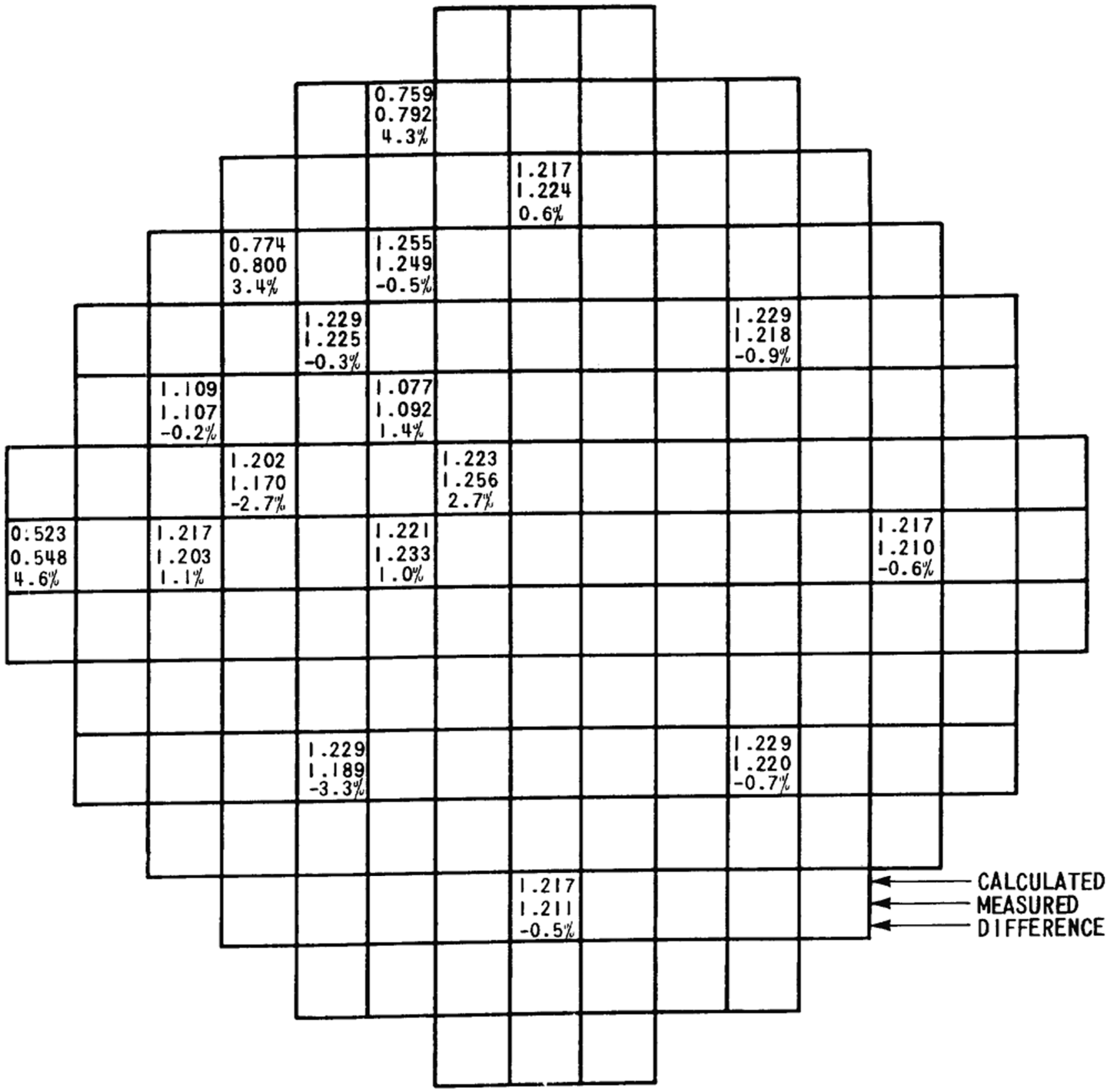
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PEAK LINEAR POWER DURING BORATION DILUTION TRANSIENTS

FIGURE 4.3–23



PEAKING FACTORS

$\bar{F}_Z = 1.5$

$F^N_{\Delta H} = 1.357$

$F^N_Q = 2.07$

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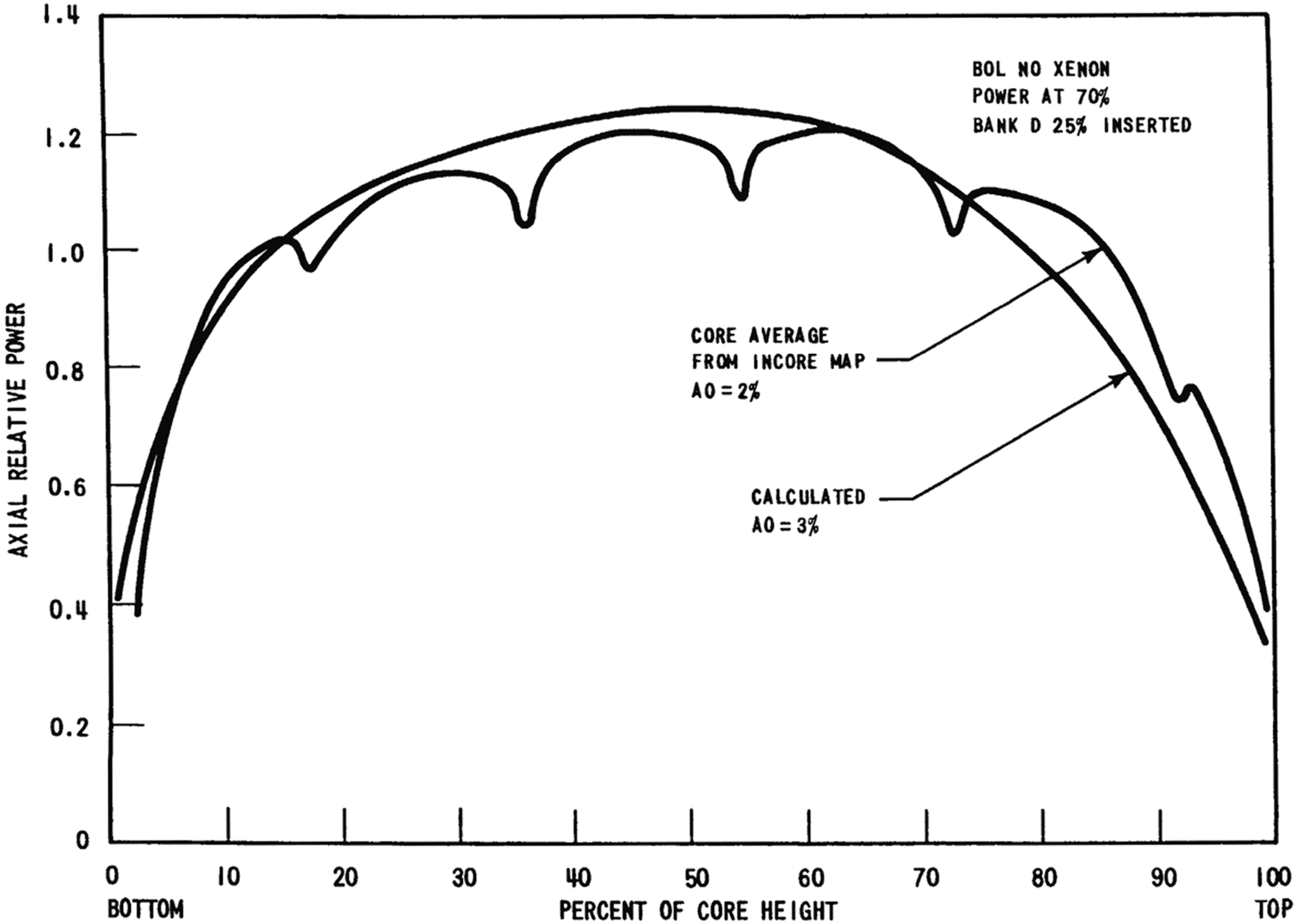
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	COMPARISON BETWEEN CALCULATED AND MEASURED RELATIVE FUEL ASSEMBLY POWER DISTRIBUTION FIGURE 4.3–24



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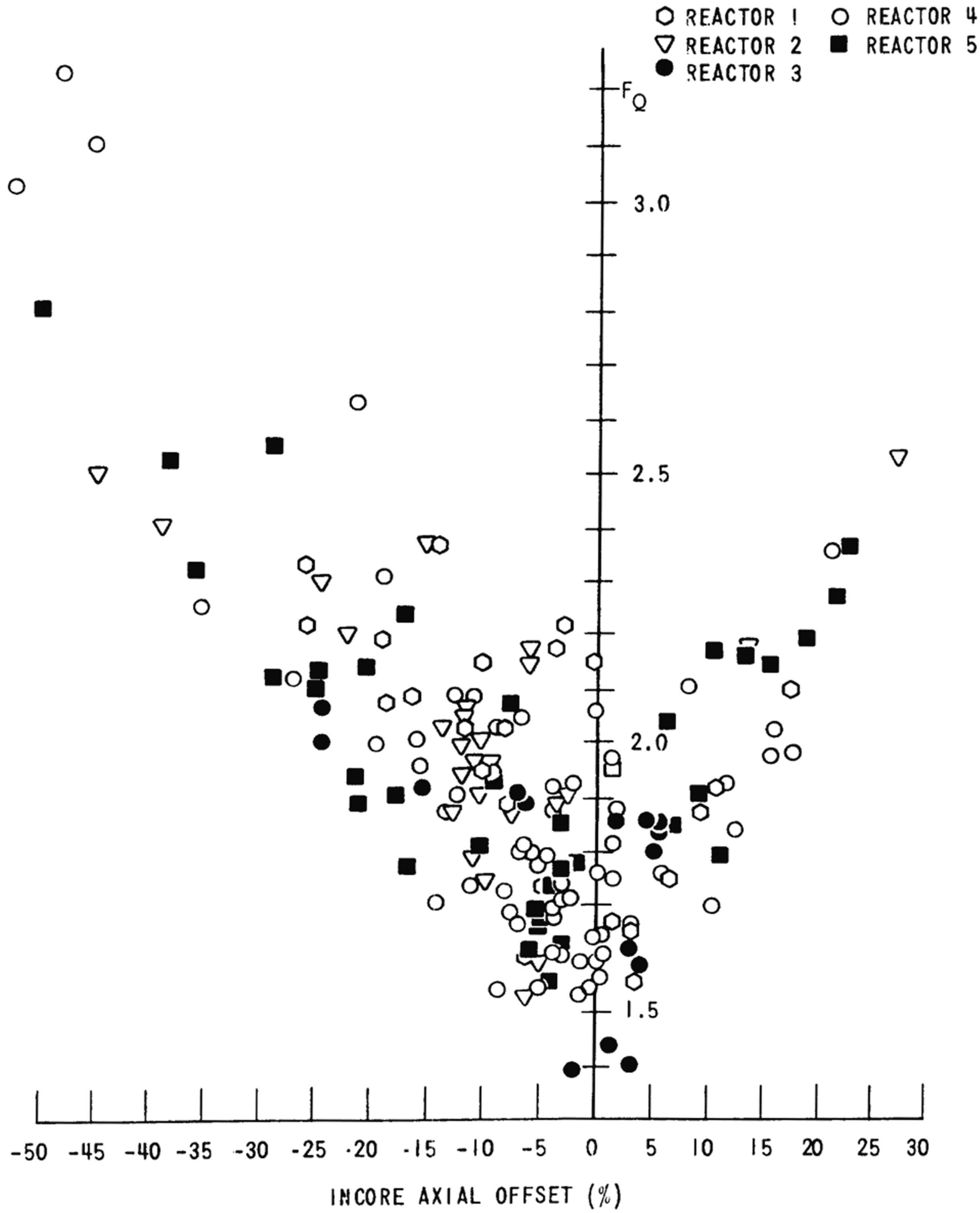
UNIT 1 AND UNIT 2

COMPARISON OF TYPICAL CALCULATED
AND MEASURED AXIAL SHAPES

FIGURE 4.3–25



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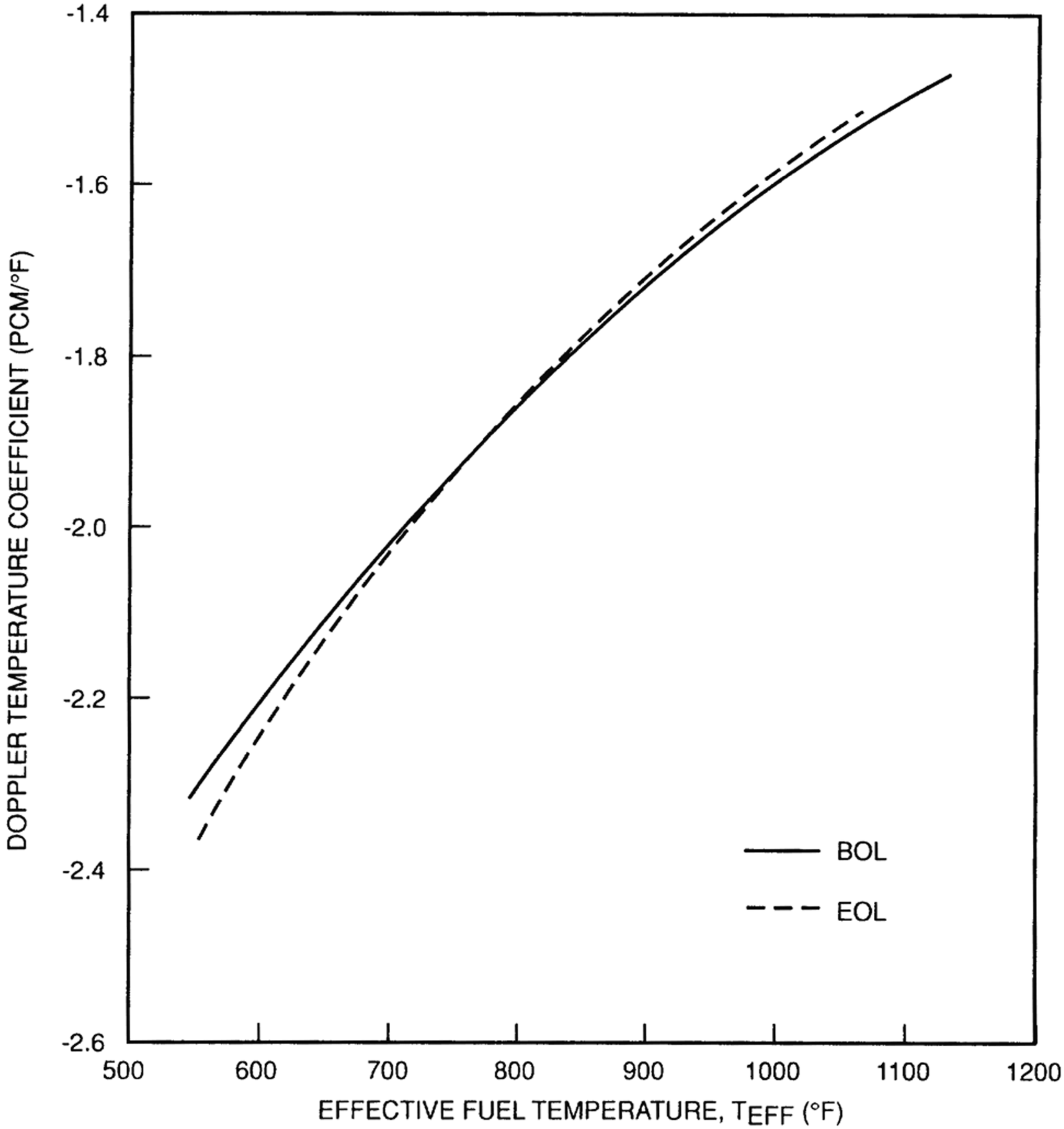
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MEASURED VALUES OF FQ FOR
FULL–POWER ROD CONFIGURATIONS

FIGURE 4.3–26



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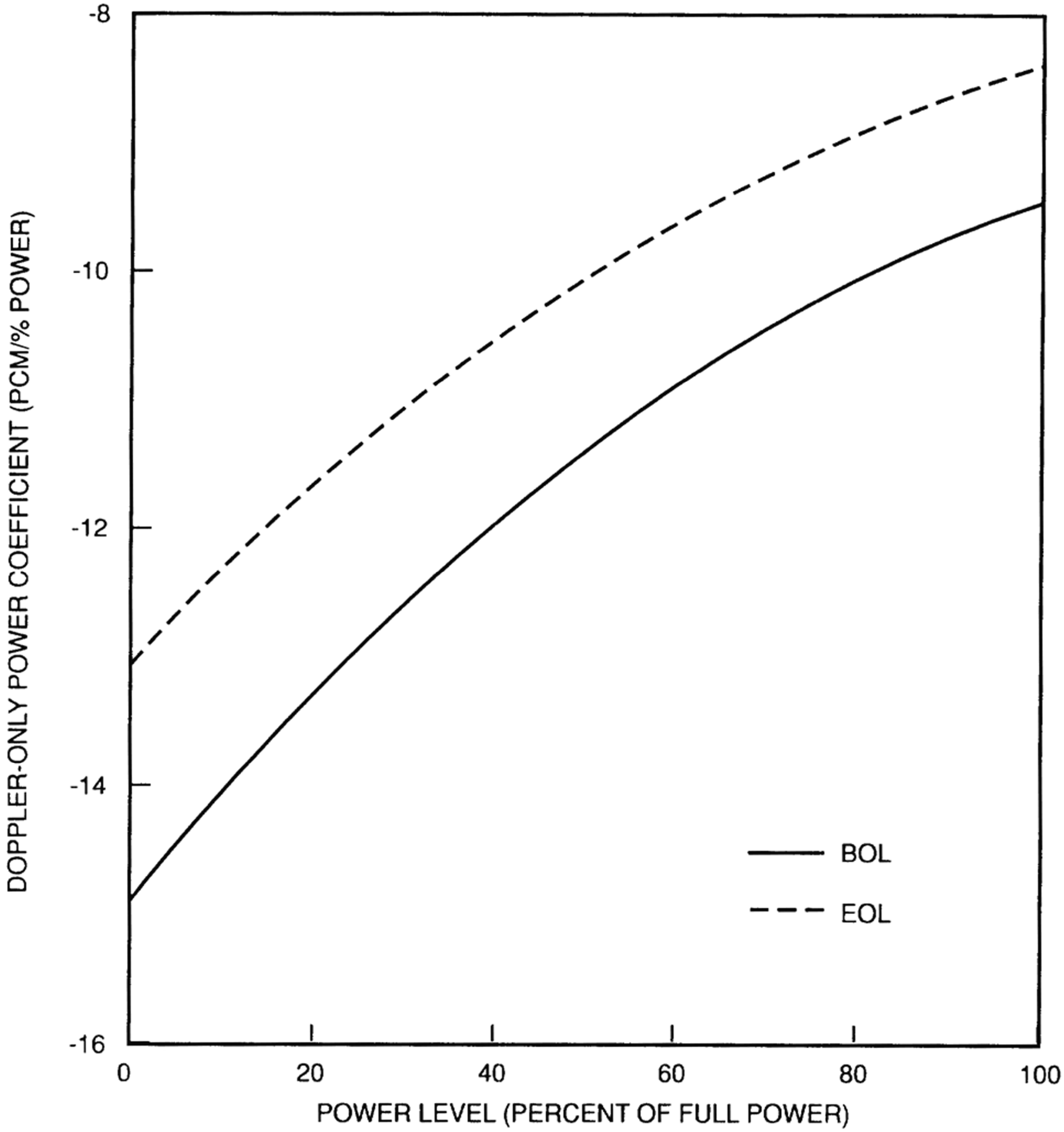
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL DOPPLER TEMPERATURE COEFFICIENT AT BEGINNING OF LIFE AND END OF LIFE

FIGURE 4.3–27



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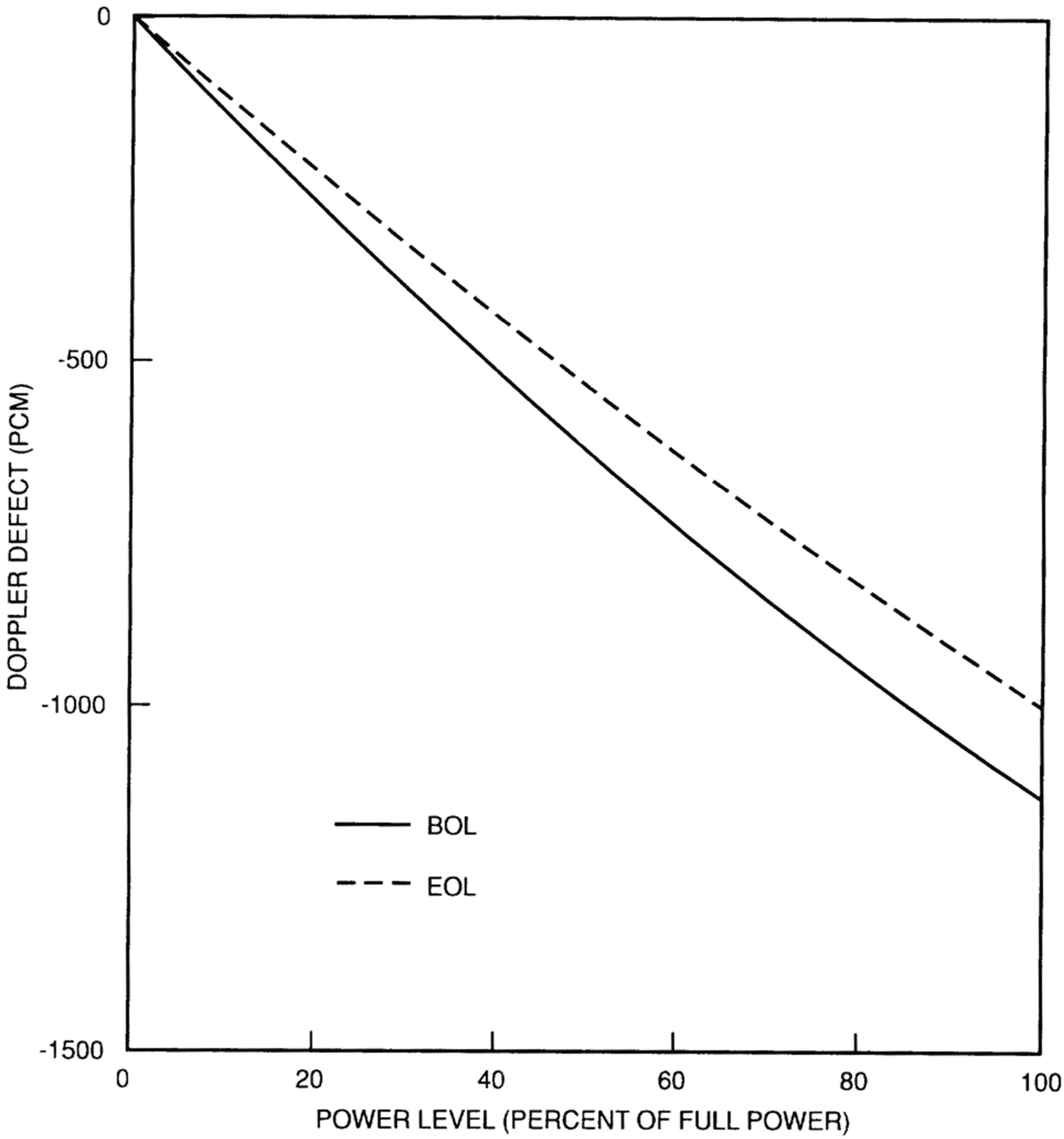
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL DOPPLER–ONLY POWER COEFFICIENT AT BEGINNING OF LIFE AND END OF LIFE

FIGURE 4.3–28



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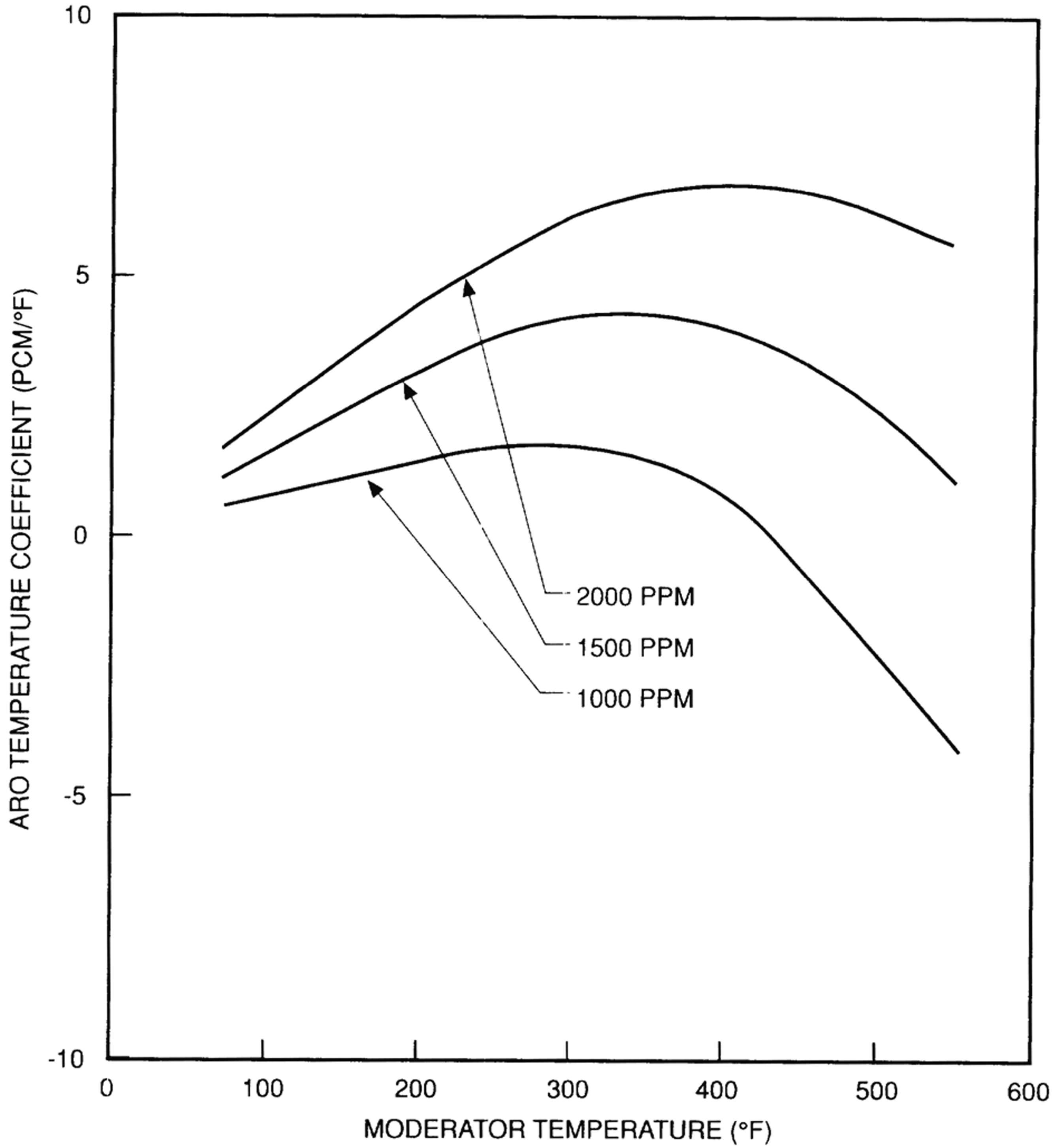
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL DOPPLER–ONLY POWER DEFECT
AT BEGINNING OF LIFE AND END OF LIFE

FIGURE 4.3–29



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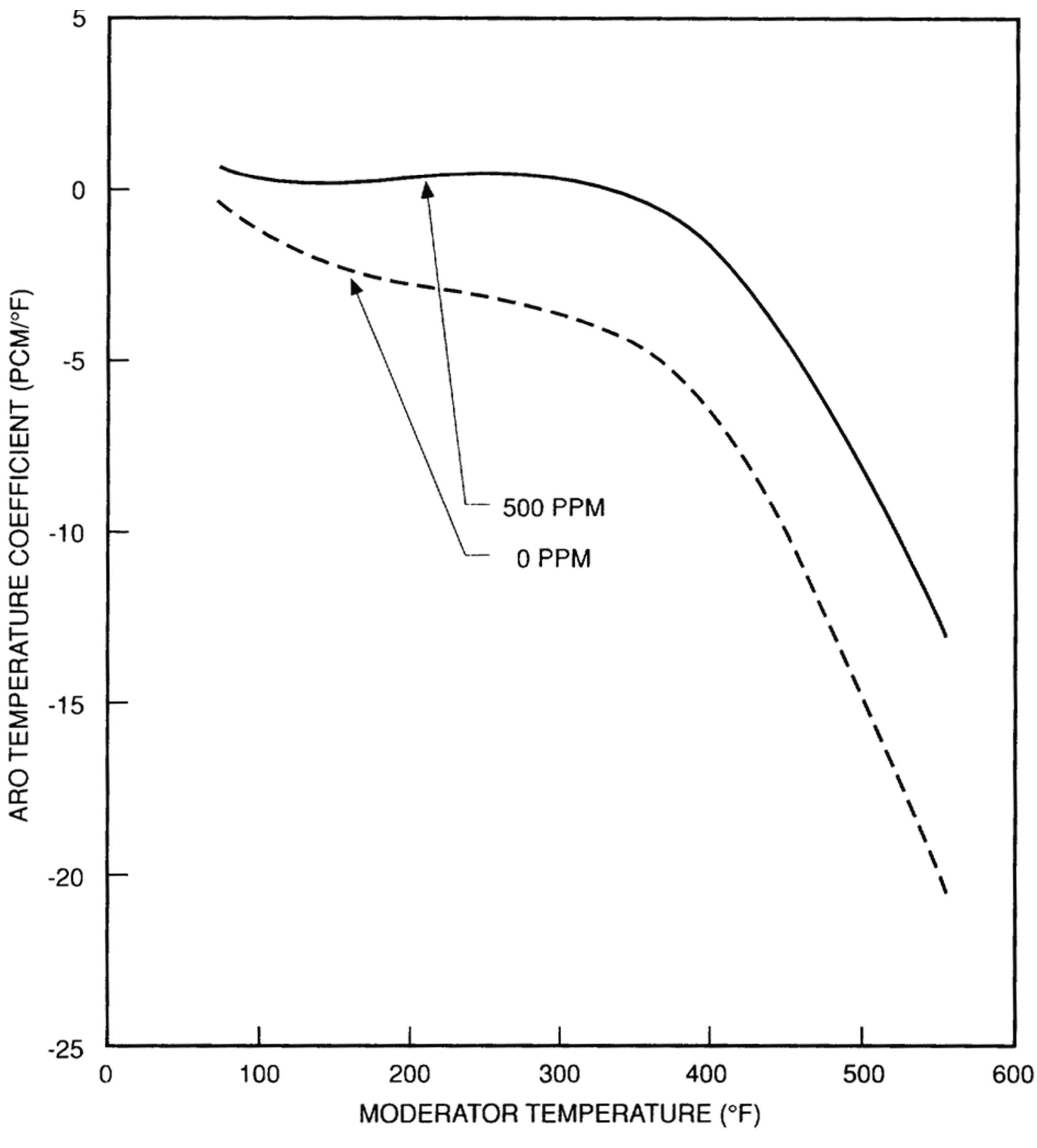
TYPICAL MODERATOR TEMPERATURE COEFFICIENT AT BEGINNING OF LIFE, UNRODDED

FIGURE 4.3–30



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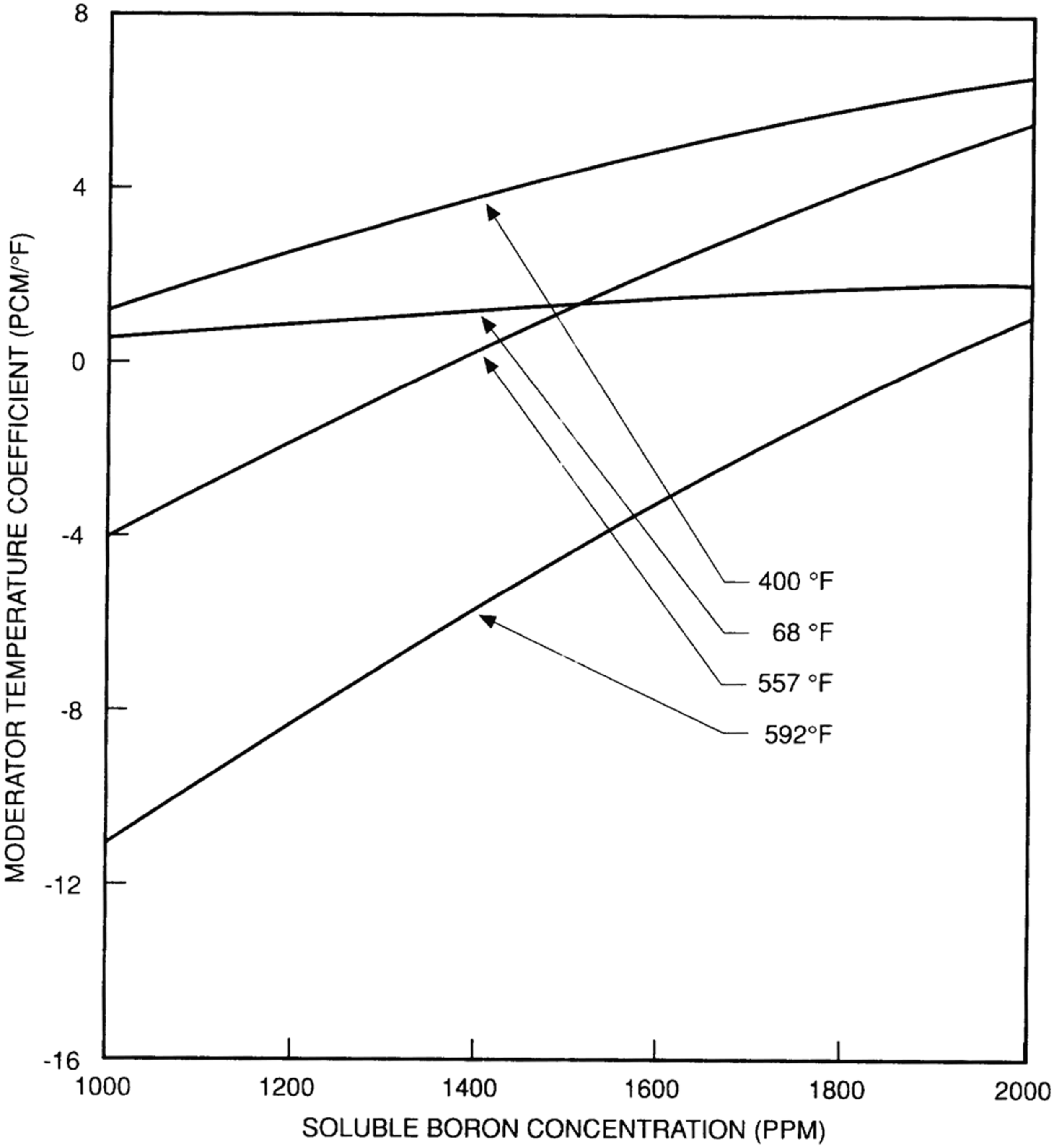
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL MODERATOR TEMPERATURE COEFFICIENT AT END OF LIFE

FIGURE 4.3–31



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TYPICAL MODERATOR TEMPERATURE COEFFICIENT AS A FUNCTION OF BORON CONCENTRATION AT BEGINNING OF LIFE, UNRODDED

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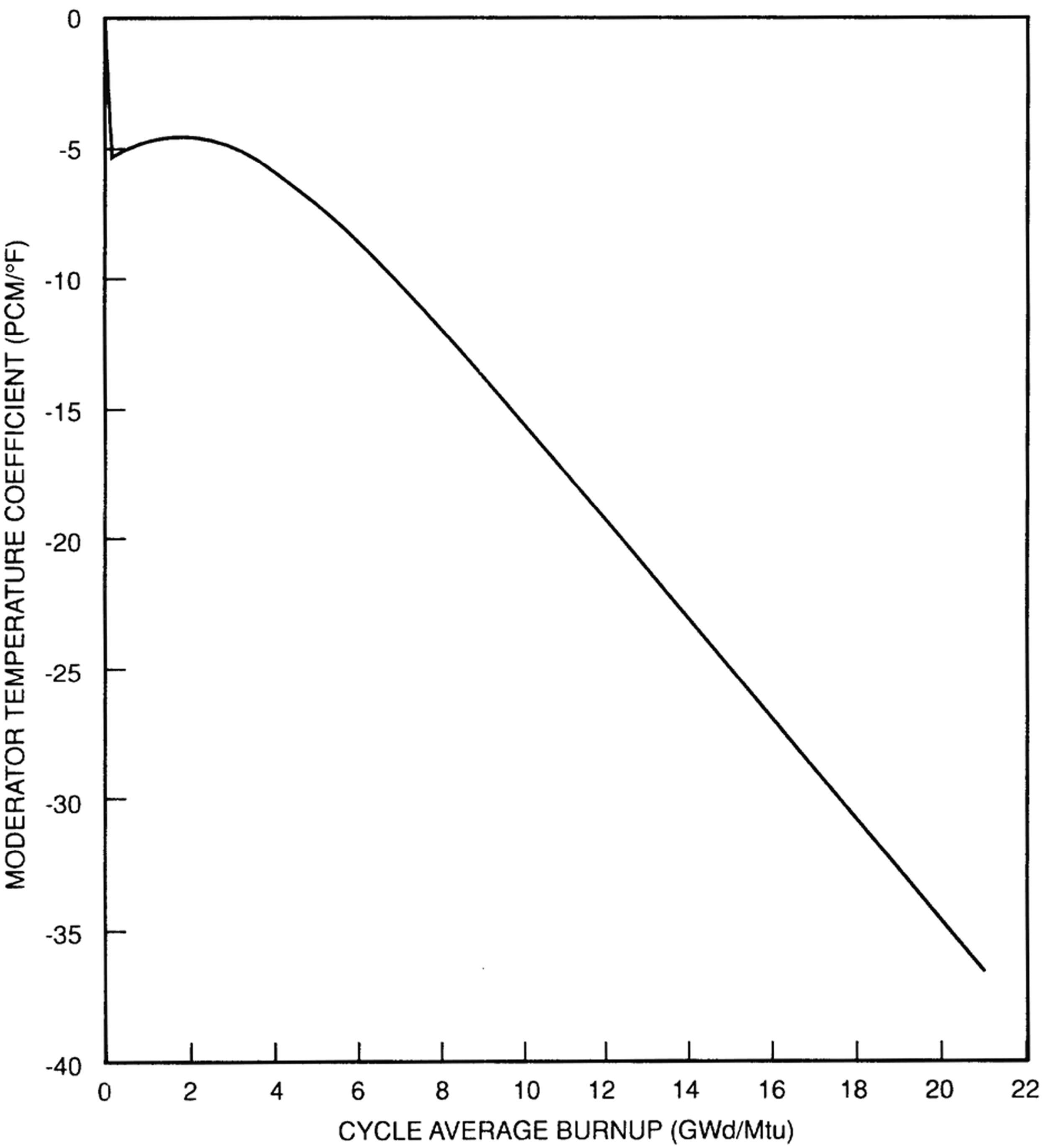
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FIGURE 4.3–32



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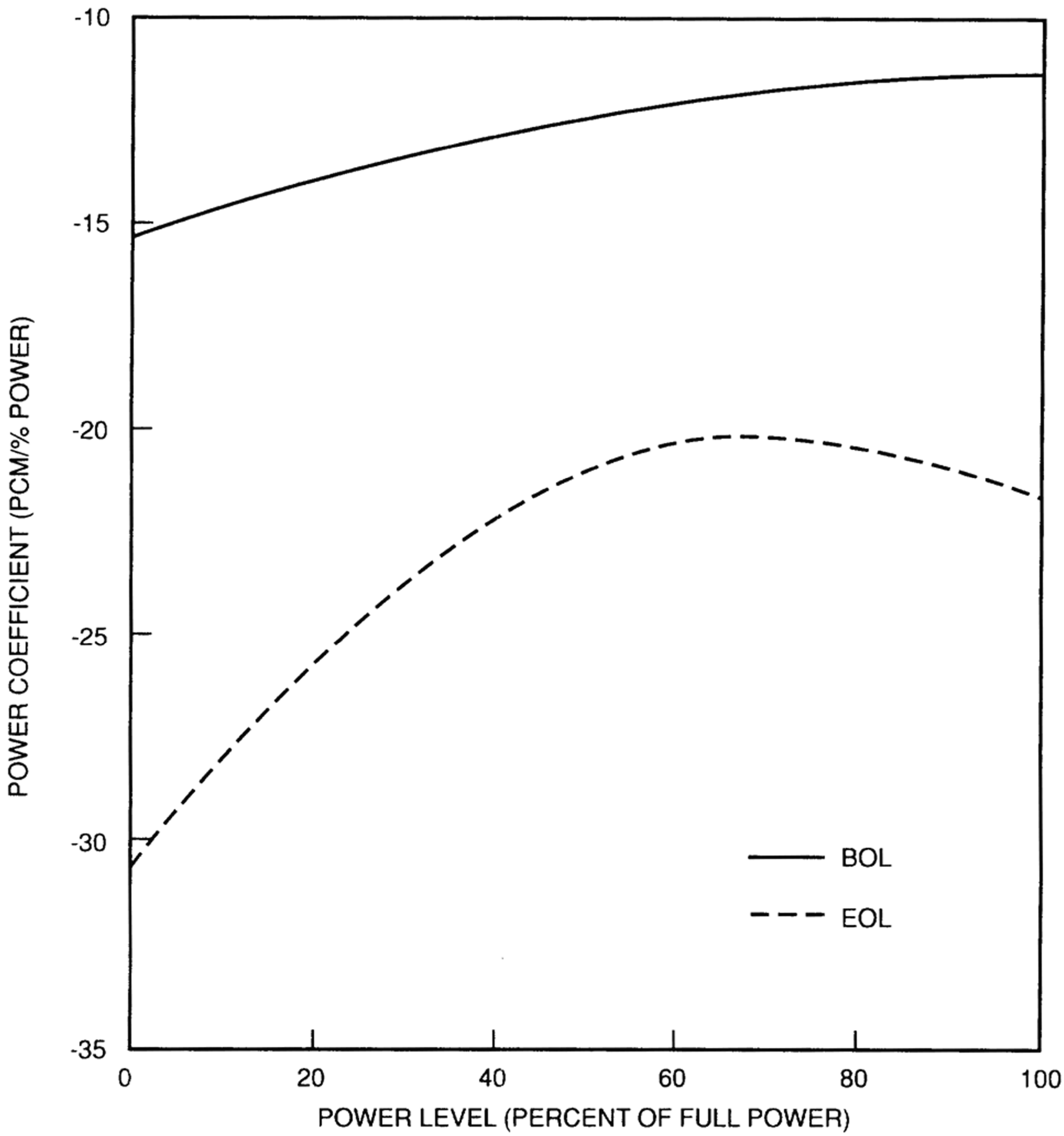
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL HOT FULL–POWER TEMPERATURE COEFFICIENT VERSUS CYCLE BURNUP

FIGURE 4.3–33



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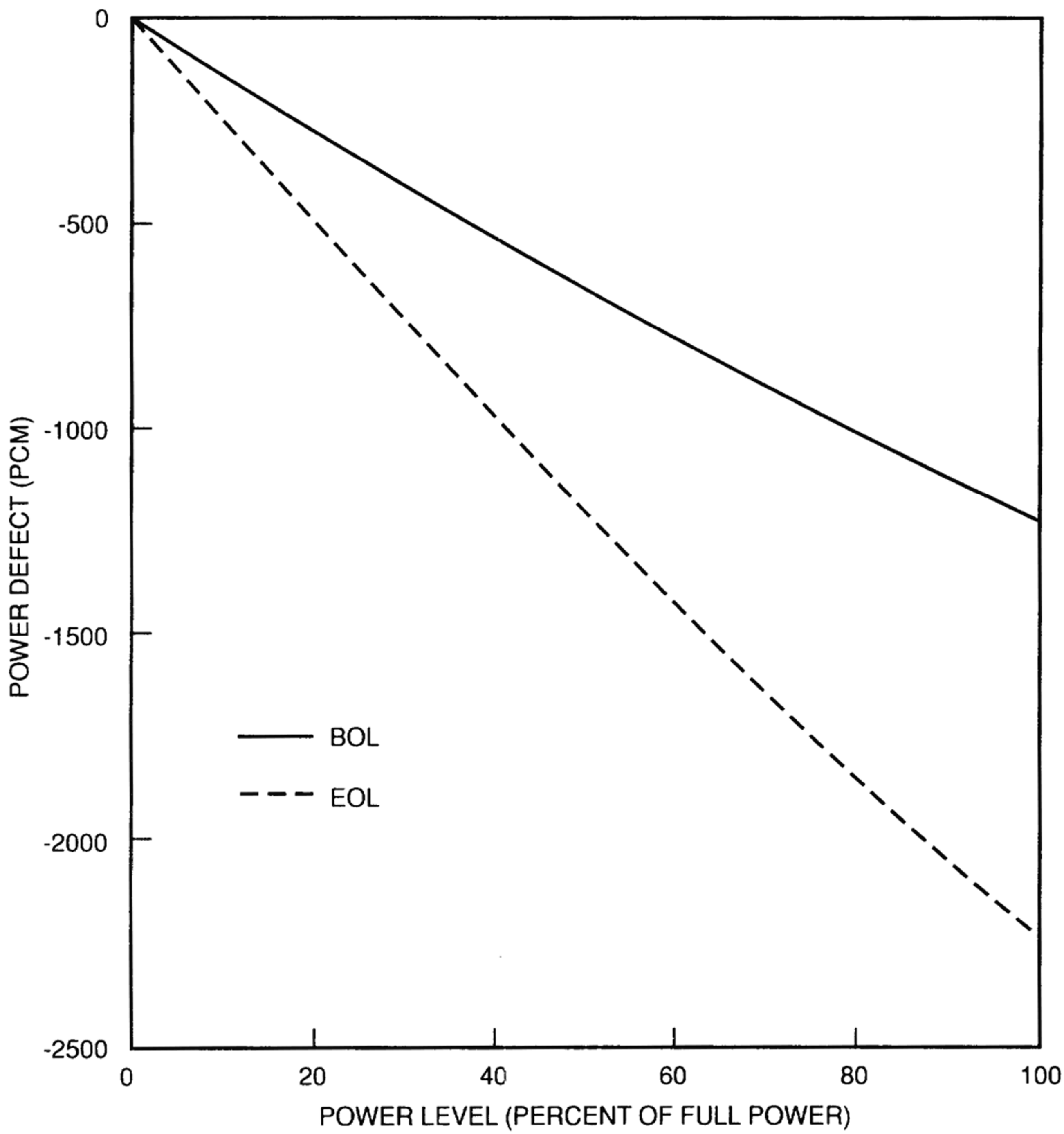
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	TYPICAL TOTAL POWER COEFFICIENT AT BEGINNING OF LIFE AND END OF LIFE
		FIGURE 4.3–34



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SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TYPICAL TOTAL POWER DEFECT AT BEGINNING OF LIFE AND END OF LIFE
	FIGURE 4.3–35

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				S_A		B		C		B		S_A			
3					S_D		S_B		S_B		S_C				
4		S_A		D				S_E				D		S_A	
5			S_C										S_D		
6		B				C		A		C				B	
7			S_B										S_B		
8		C		S_E		A		D		A		S_E		C	
9			S_B										S_B		
10		B				C		A		C				B	
11			S_D										S_C		
12		S_A		D				S_E				D		S_A	
13					S_C		S_B		S_B		S_D				
14				S_A		B		C		B		S_A			
15															

90° (left) — 270° (right)

0°

CONTROL BANK	NUMBER OF RODS
A	4
B	8
C	8
D	5
TOTAL	25

SHUTDOWN BANK	NUMBER OF RODS
S_A	8
S_B	8
S_C	4
S_D	4
S_E	4
TOTAL	28

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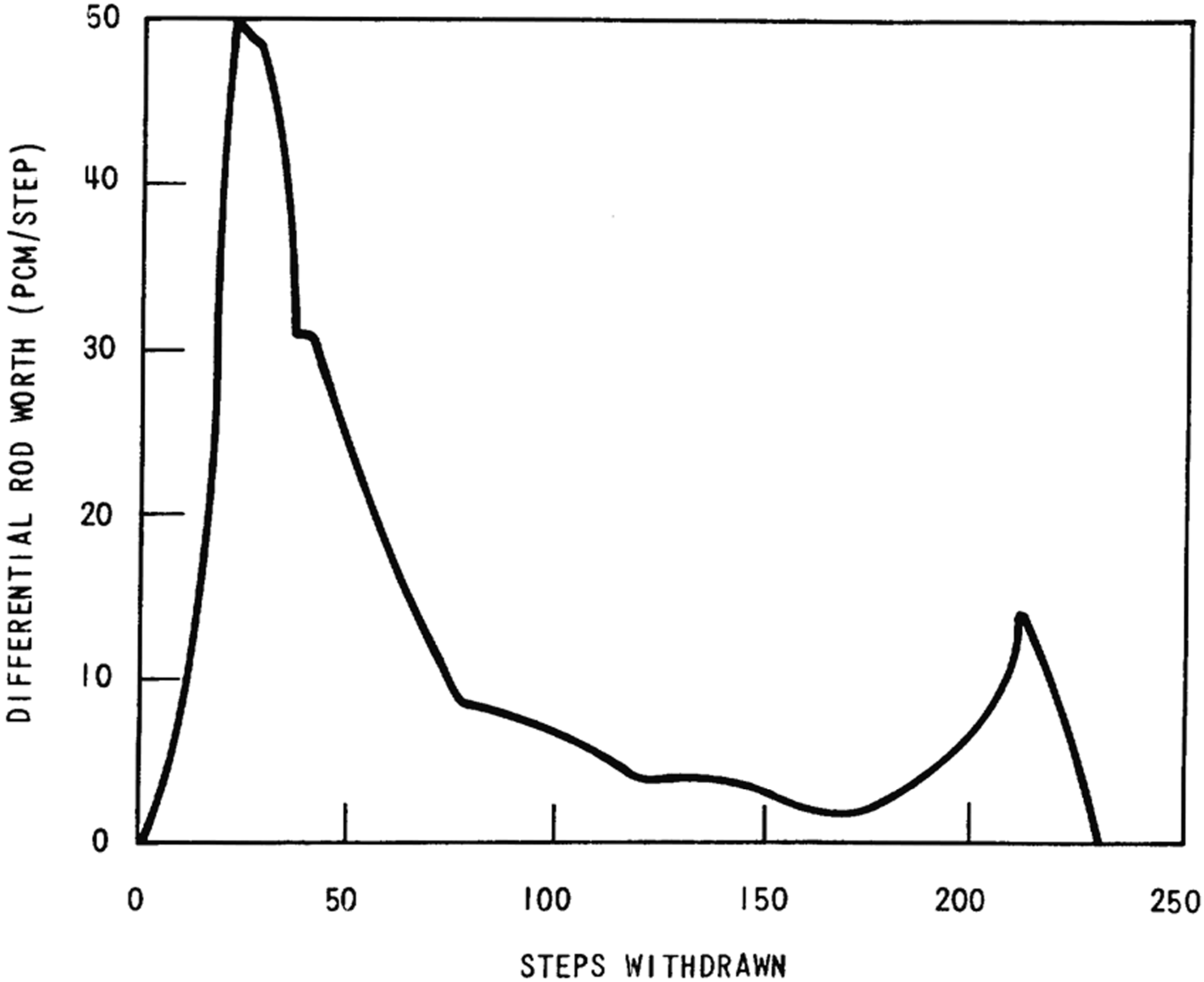
UNIT 1 AND UNIT 2

ROD CLUSTER CONTROL ASSEMBLY PATTERN

FIGURE 4.3–36



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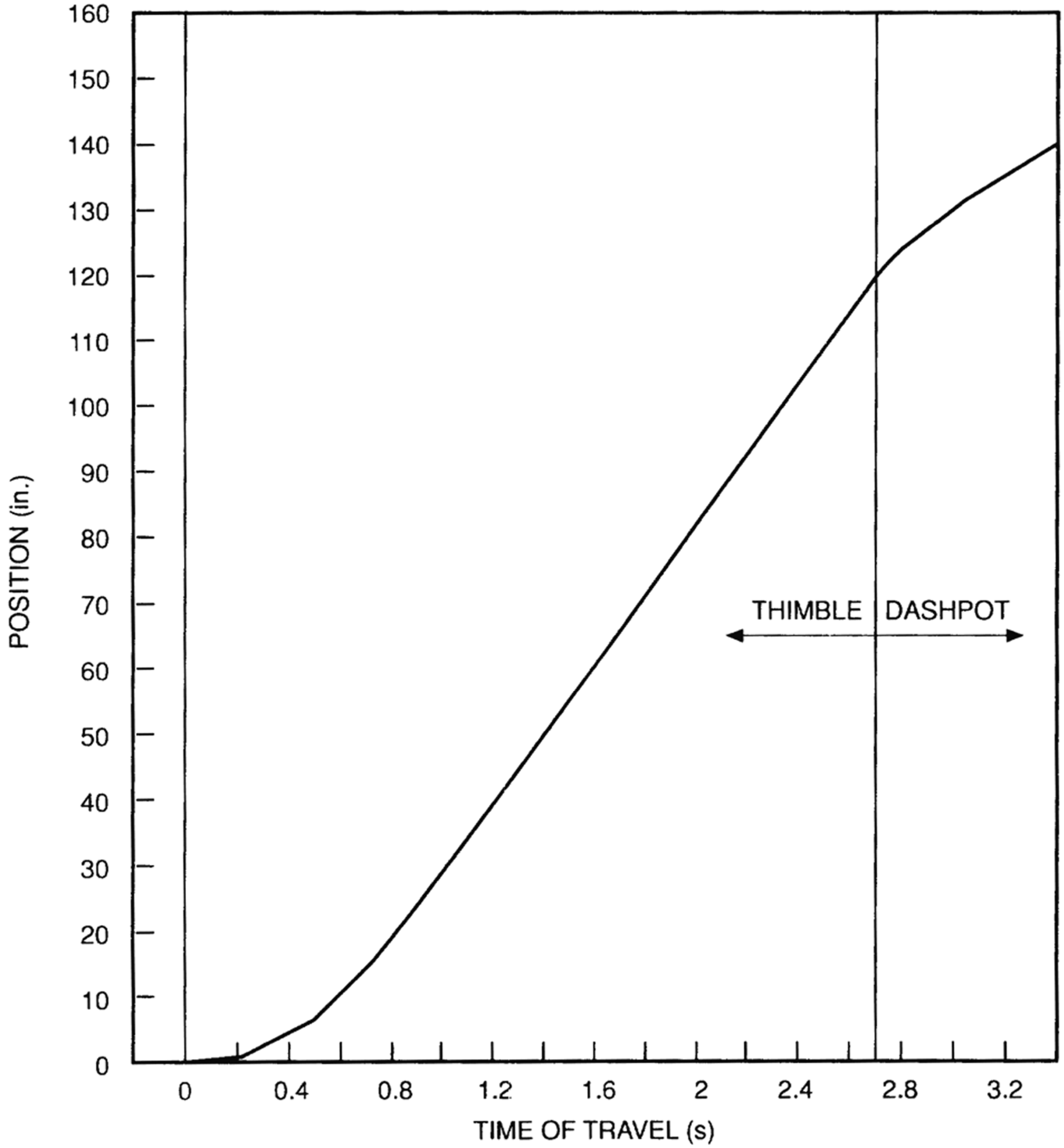
UNIT 1 AND UNIT 2

TYPICAL ACCIDENTAL SIMULTANEOUS WITHDRAWAL OF TWO CONTROL BANKS AT END OF LIFE, HOT ZERO POWER, BANK D AND B MOVING IN THE SAME PLANE

FIGURE 4.3–37



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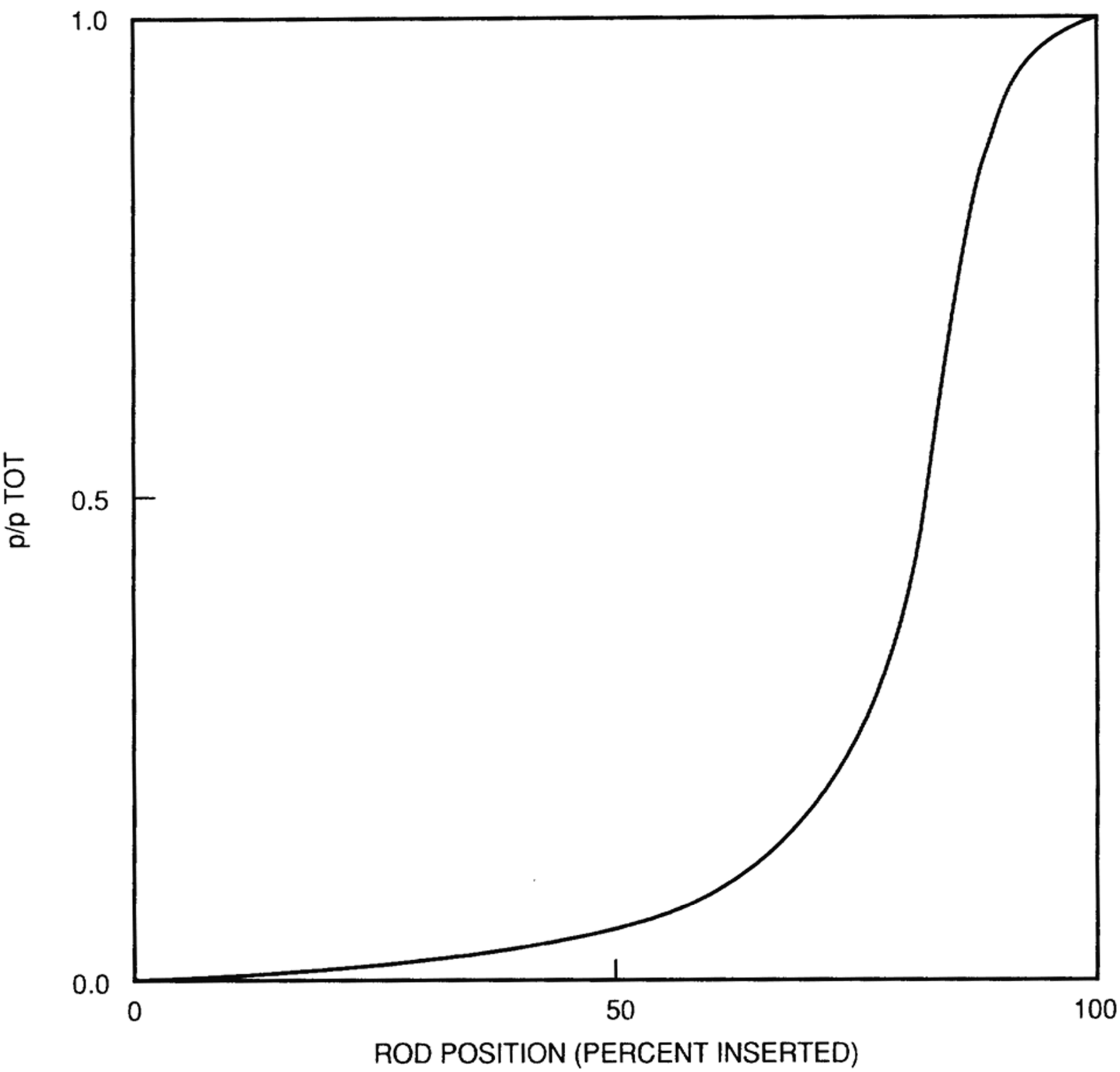
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	DESIGN TRIP CURVE FIGURE 4.3–38



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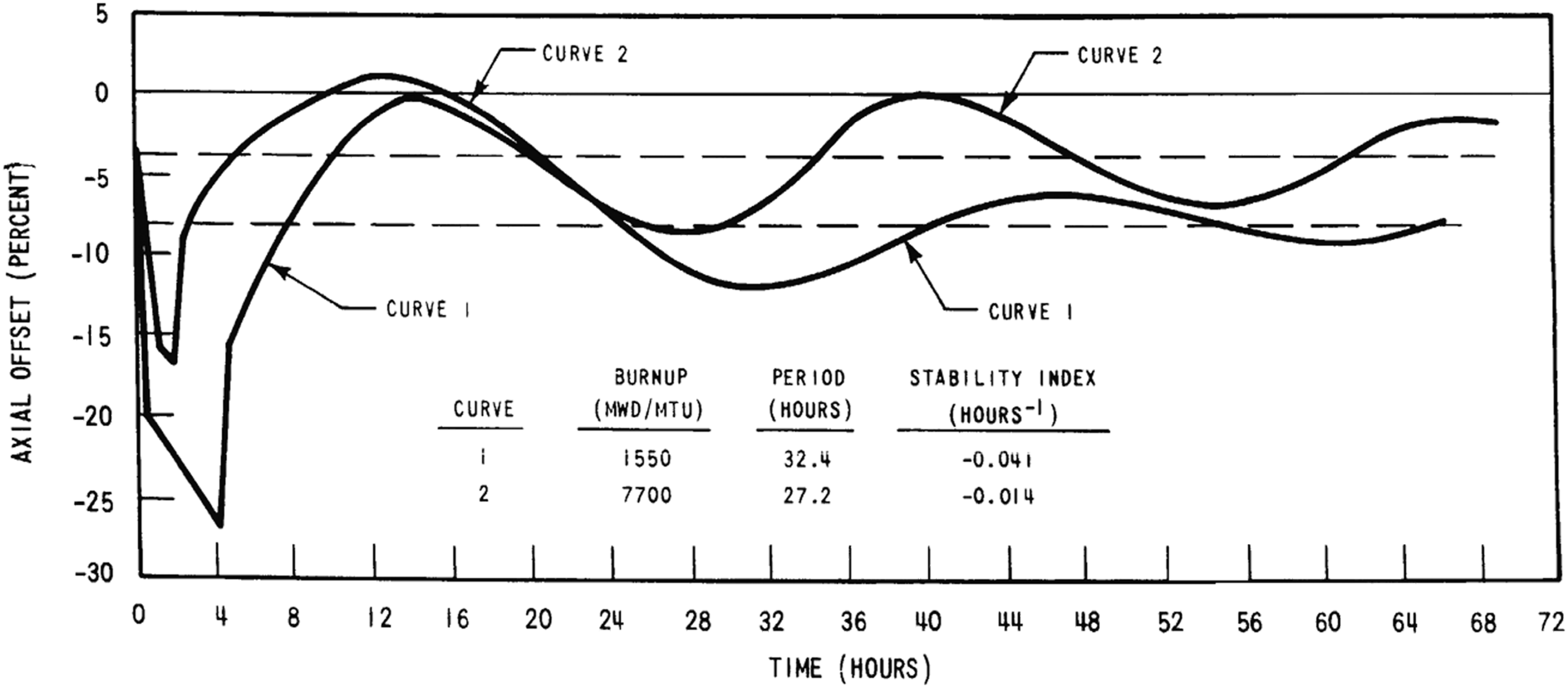
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TYPICAL NORMALIZED ROD WORTH VERSUS PERCENT INSERTION, ALL RODS OUT BUT ONE

FIGURE 4.3–39



CURVE	BURNUP (MWD/MTU)	PERIOD (HOURS)	STABILITY INDEX ($HOURS^{-1}$)
1	1550	32.4	-0.041
2	7700	27.2	-0.014

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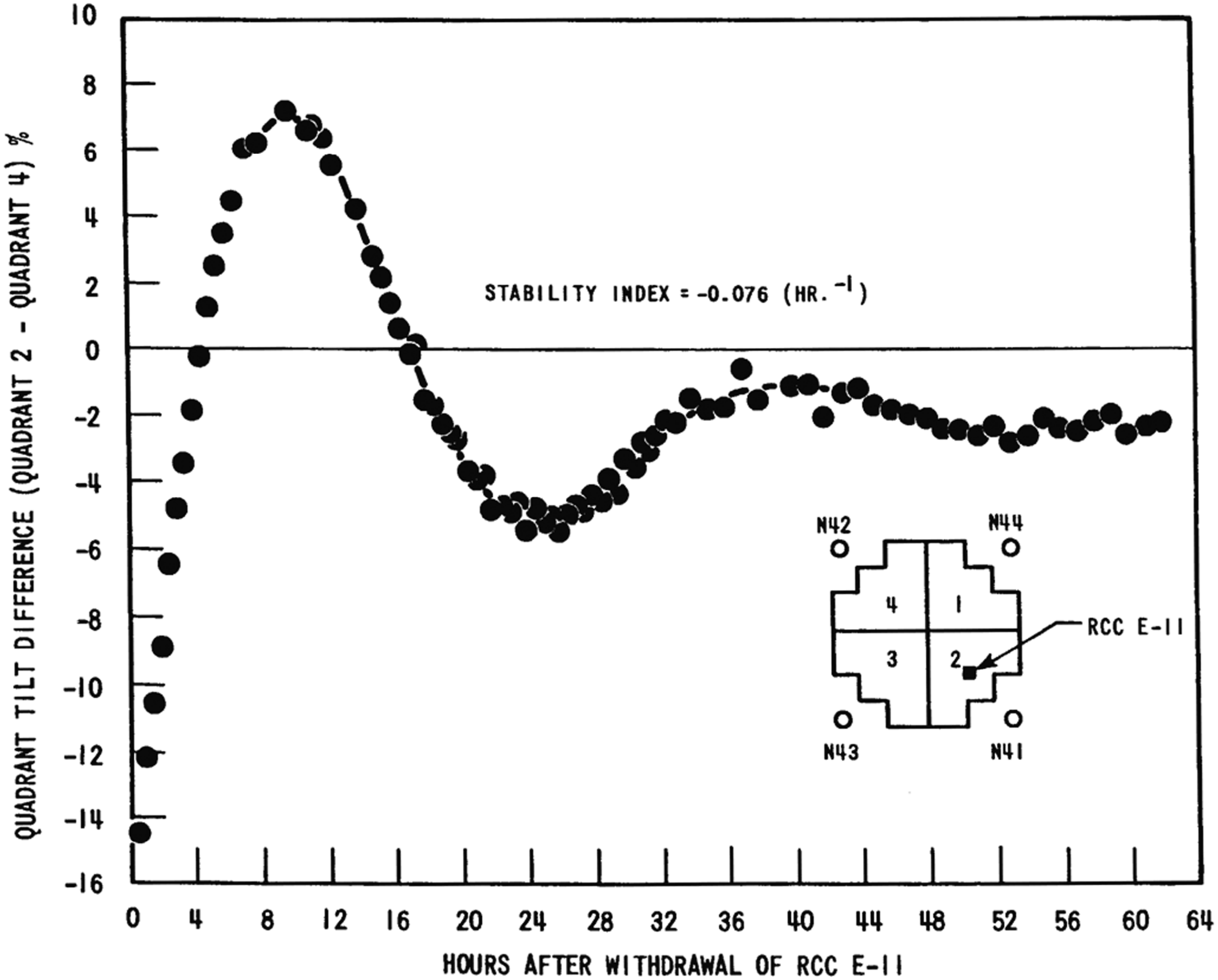
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AXIAL OFFSET VERSUS TIME, PWR CORE WITH A 12–ft HEIGHT AND 121 ASSEMBLIES

FIGURE 4.3–40



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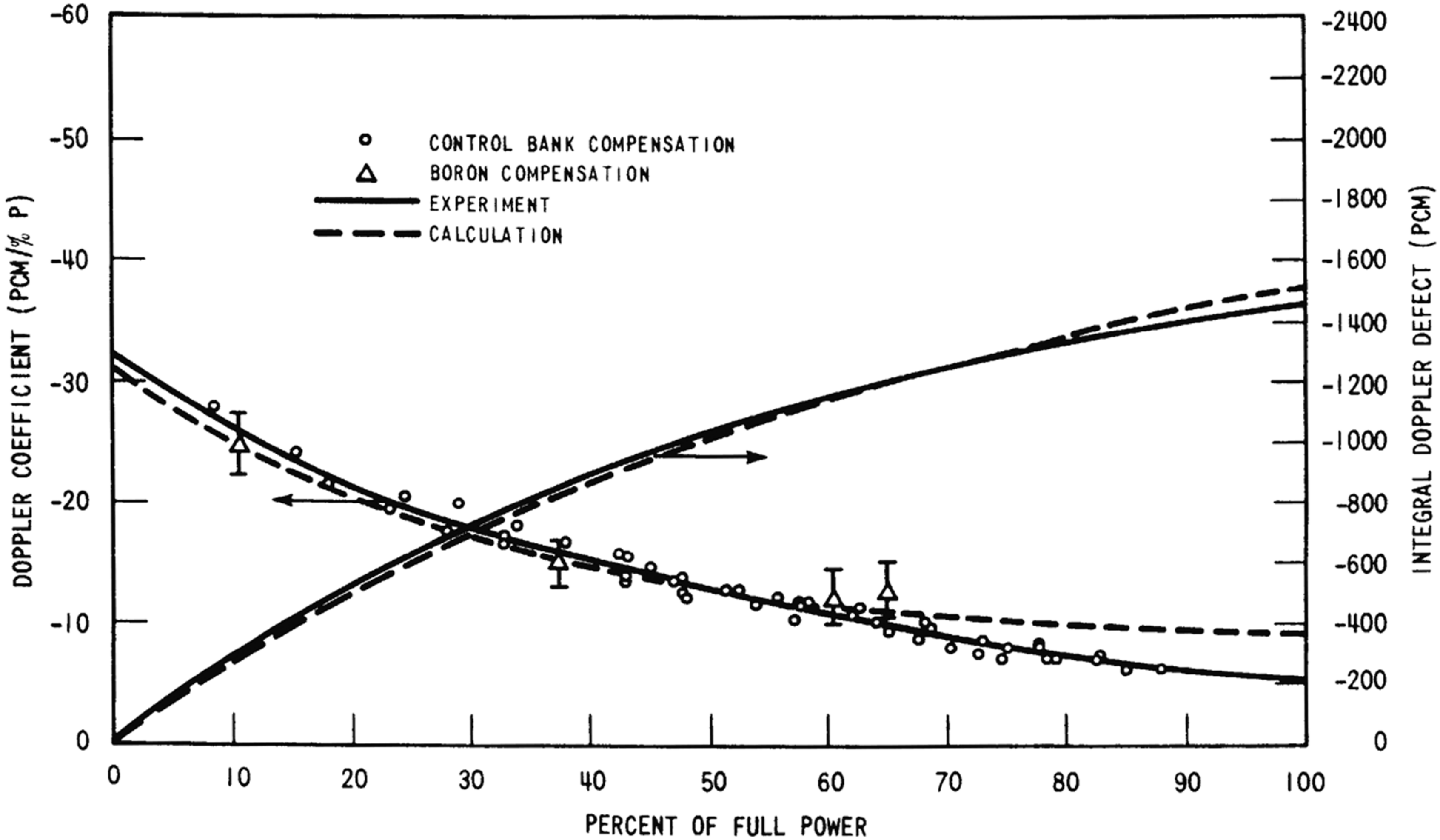
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

X–Y XENON TEST THERMOCOUPLE
RESPONSE QUADRANT TILT
DIFFERENCE VERSUS TIME

FIGURE 4.3–41



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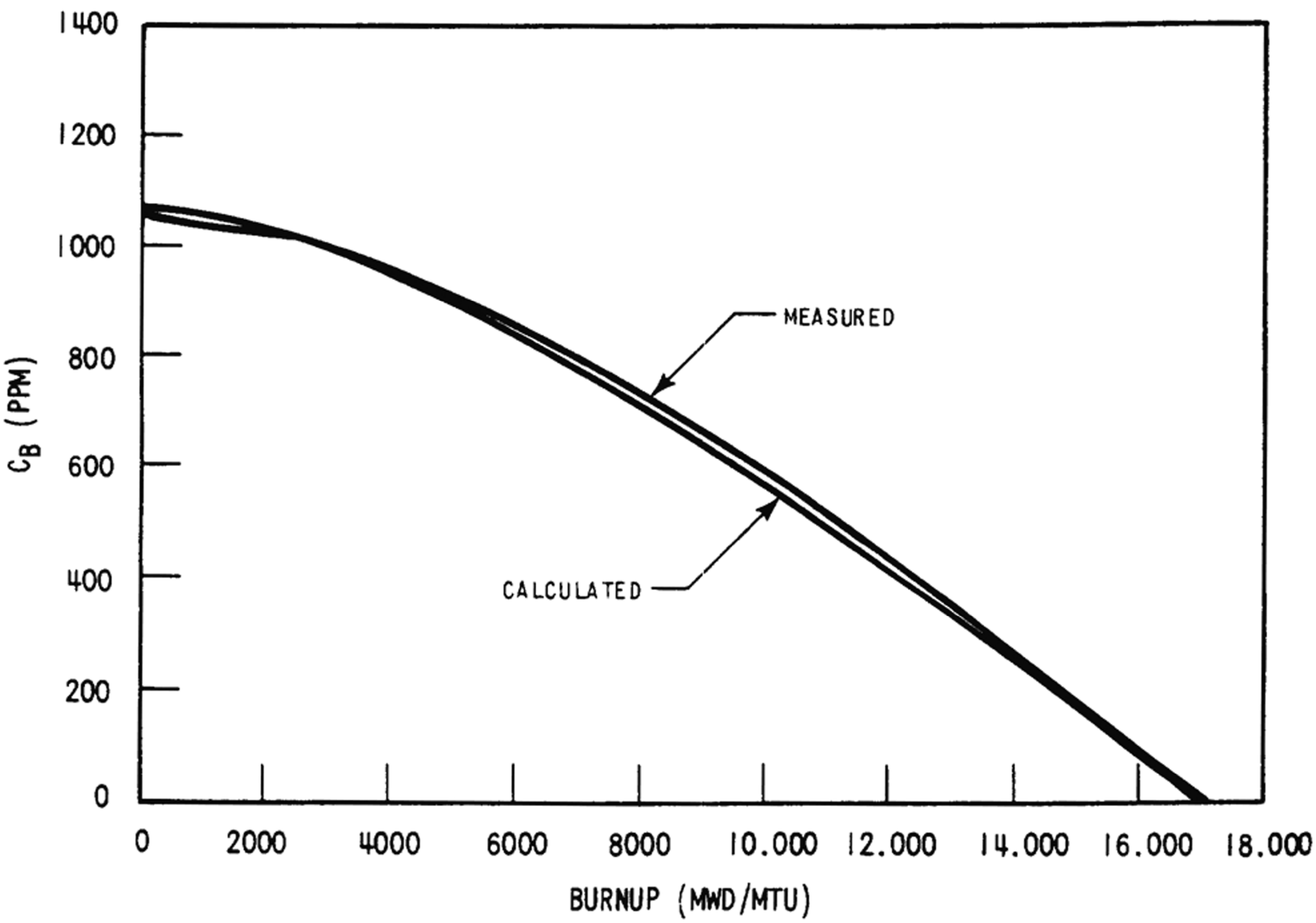
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CALCULATED AND MEASURED DOPPLER DEFECT AND COEFFICIENTS AT BEGINNING OF LIFE, 2–LOOP PLANT, 121 ASSSEMBLIES, 12–ft CORE

FIGURE 4.3–42



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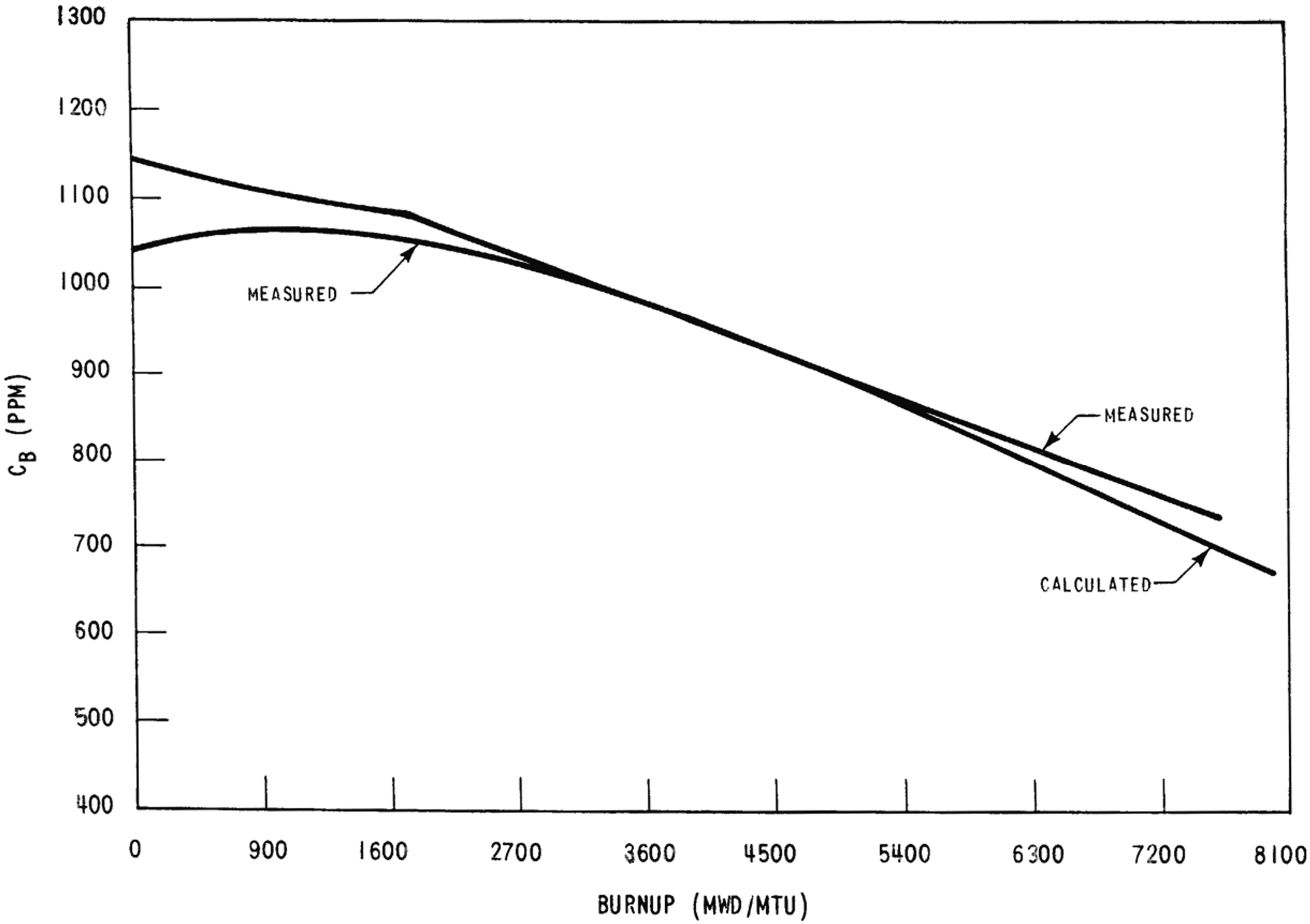
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	COMPARISION OF CALCULATED AND MEASURED BORON CONCENTRATION, 2–LOOP PLANT, 121 ASSEMBLIES, 12–ft CORE FIGURE 4.3–43



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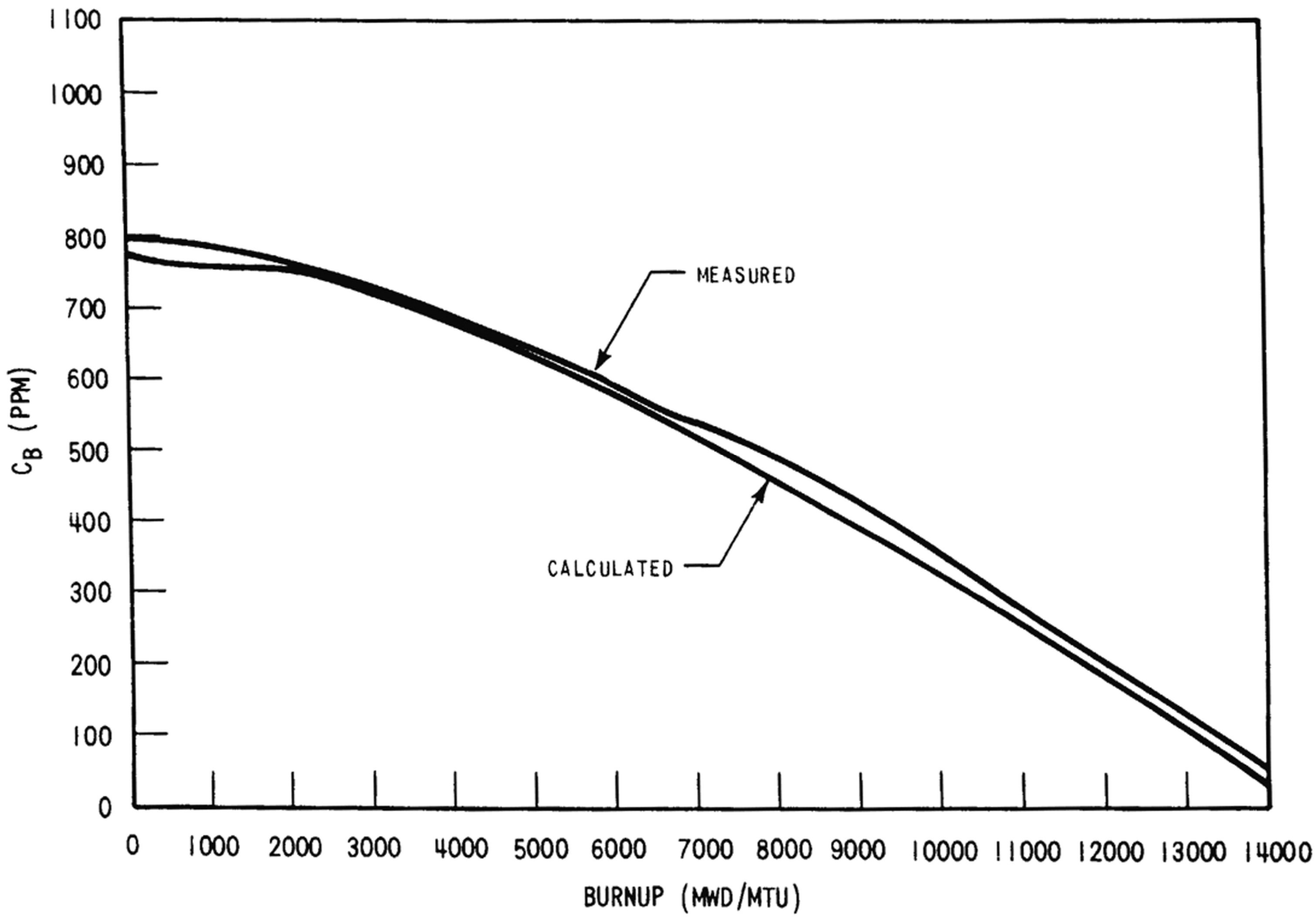
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

COMPARISON OF CALCULATED AND MEASURED C_B,
2–LOOP PLANT, 121
ASSEMBLIES, 12–ft CORE

FIGURE 4.3–44



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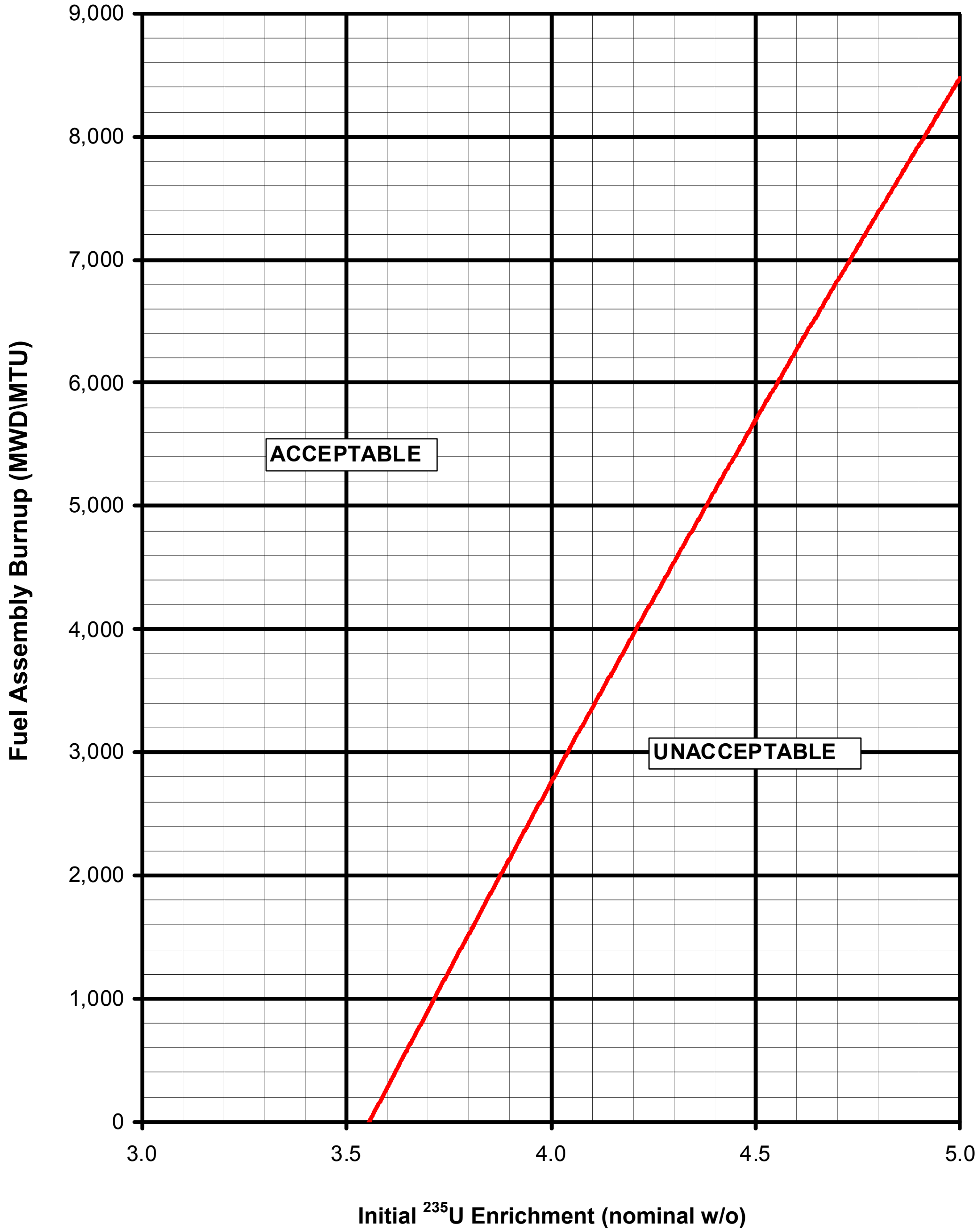
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

COMPARISON OF CALCULATED AND MEASURED C_B,
3–LOOP PLANT, 157 ASSEMBLIES, 12–ft CORE

FIGURE 4.3–45



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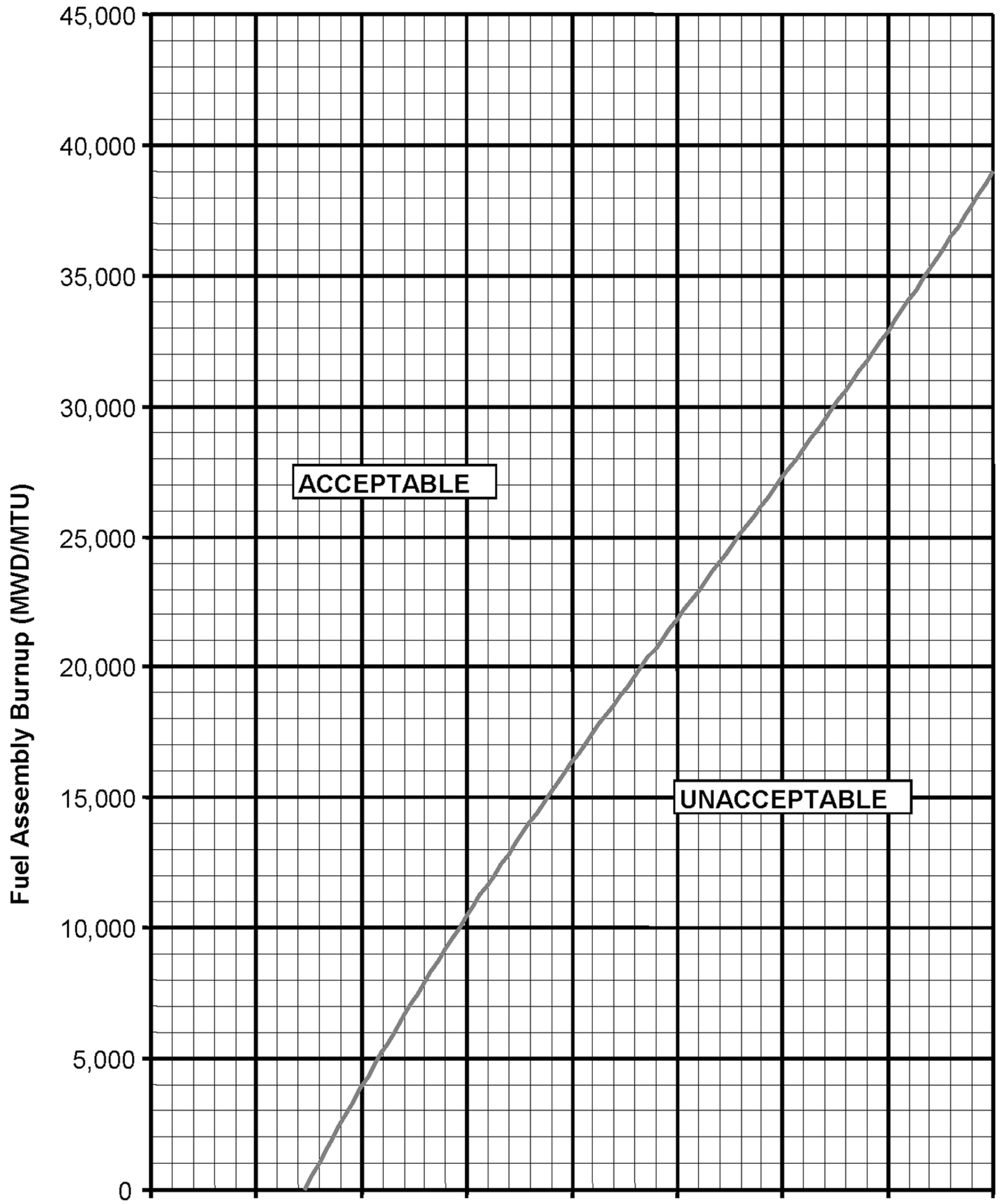
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

VOGTLE UNIT 1 BURNUP CREDIT
REQUIREMENTS FOR ALL CELL STORAGE

FIGURE 4.3–46



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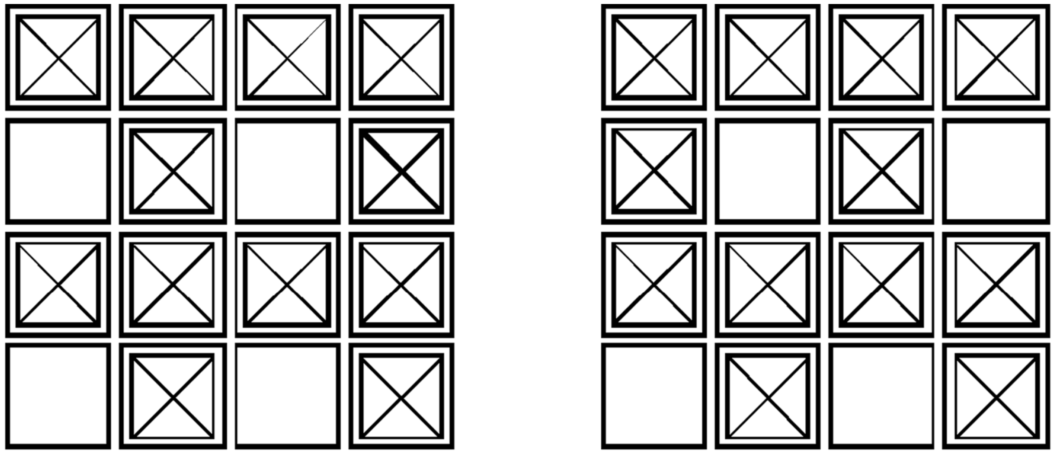
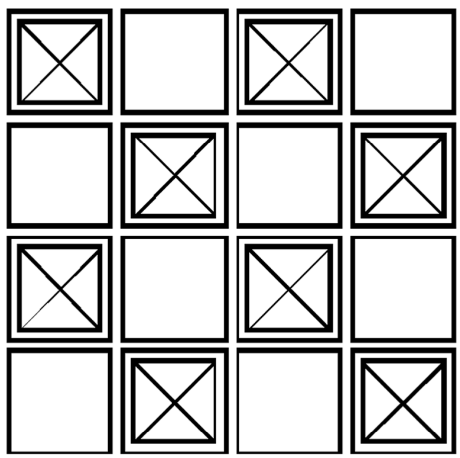
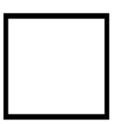
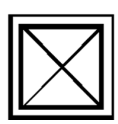
SOUTHERN COMPANY Energy to Serve Your World® VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	VOGTLE UNIT 2 BURNUP CREDIT REQUIREMENTS FOR ALL CELL STORAGE
	FIGURE 4.3–47



3-out-of-4 Checkerboard Storage (Units 1 and 2)



2-out-of-4 Checkerboard Storage (Unit 2)

 **Empty Storage Cell**  **Fuel Assembly in Storage Cell**

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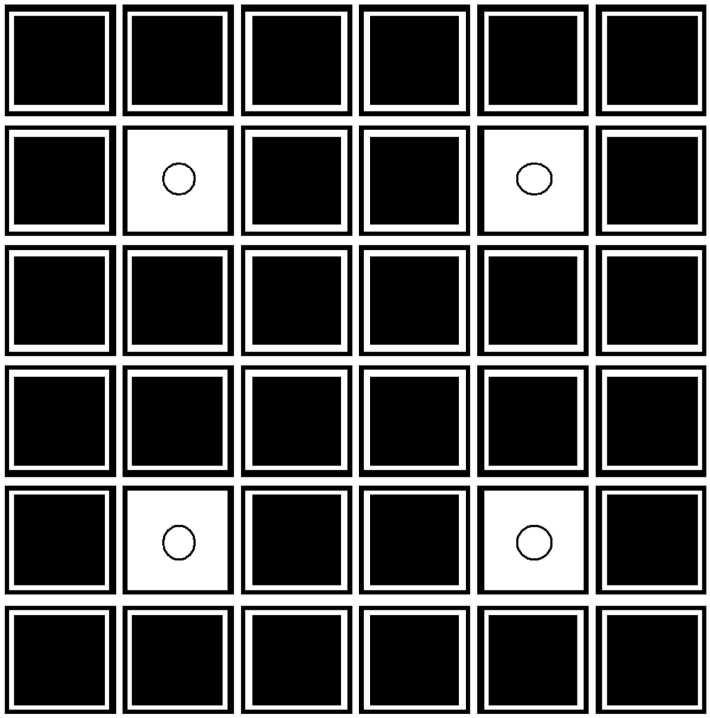
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	VOGTLE UNITS 1 AND 2 EMPTY CELL CHECKERBOARD STORAGE CONFIGURATIONS
		FIGURE 4.3–48



3x3 Checkerboard Storage

Low Enrichment Fuel Assembly in Storage Cell

High Enrichment Fuel Assembly in Storage Cell

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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

VOGTLE UNIT 2 3X3 CHECKERBOARD
STORAGE CONFIGURATION

FIGURE 4.3–49

Interface

A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
Empty	B	Empty	A	A	A
B	B	B	A	A	A
Empty	B	Empty	A	A	A

Note:
A = All Cell Enrichment
B = 3-Out-of-4 Enrichment
Empty = Empty Cell

Boundary Between All Cell Storage and 3-Out-of-4 Storage (Units 1 and 2)

Interface

A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
Empty	B	Empty	A	A	A
C	Empty	B	A	A	A
Empty	C	Empty	A	A	A

Note:
A = All Cell Enrichment
B = 3-Out-of-4 Enrichment
C= 2-Out-of-4 Enrichment
Empty = Empty Cell

Boundary Between All Cell Storage and 2-Out-of-Storage (Unit 2)

Note:

1. **A row of empty cells can be used at the interface to separate the configurations.**
2. **It is acceptable to replace an assembly with an empty cell.**

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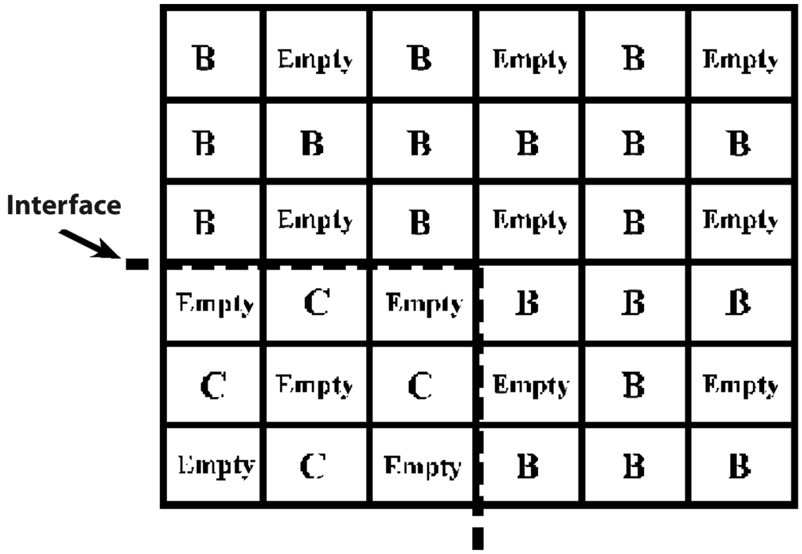
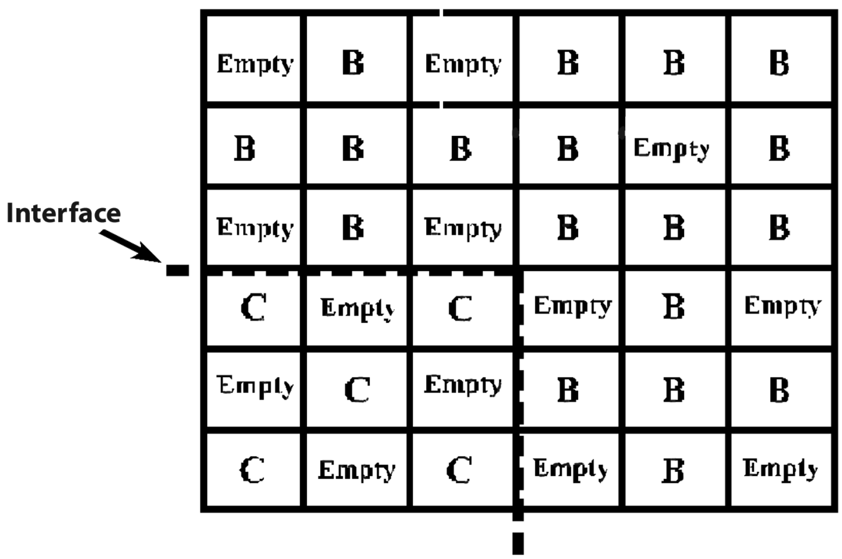


Note:
B = 3-Out-of-4 Enrichment
C = 2-Out-of-4 Enrichment
Empty = Empty Cell

Boundary Between 2-Out-of-4 Storage and 3-Out-of-4 Storage

Note:
B = 3-Out-of-4 Enrichment
C = 2-Out-of-4 Enrichment
Empty = Empty Cell



Boundary Between 2-Out-of-4 Storage and 3-Out-of-4 Storage

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assemblly with an empty cell.

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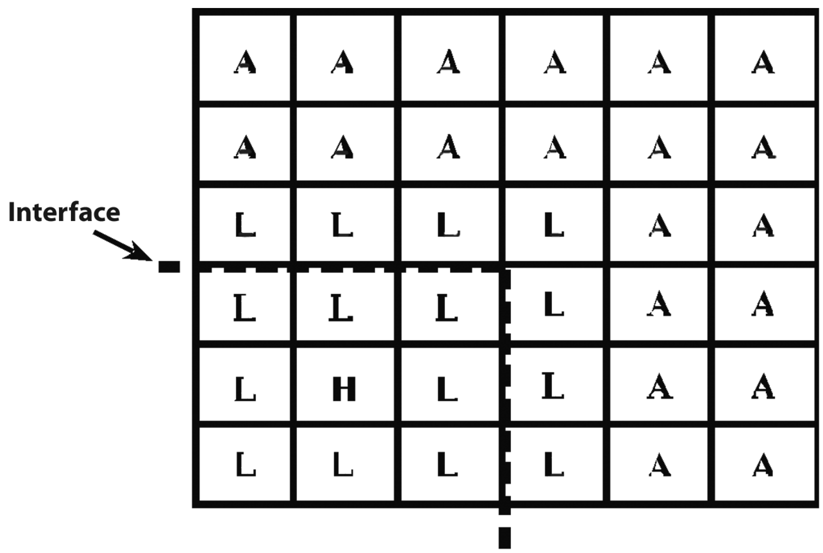
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

VOGTLE UNIT 2 INTERFACE REQUIREMENTS
(CHECKERBOARD STORAGE INTERFACE)

FIGURE 4.3–51



Note:
A = All Cell Enrichment
L = Low Enrichment of 3 x 3 Checkerboard
H = High Enrichment of 3 x 3 Checkerboard

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	VOGTLE UNIT 2 INTERFACE REQUIREMENTS (3X3 CHECKERBOARD TO ALL CELL STORAGE)
	FIGURE 4.3–52

Interface

B	B	B	B	B	B*
Empty	B	Empty	B	Empty	B
L	L	L	L	B	B
L	L	L	L	Empty	B
L	H	L	L	B	B
L	L	L	L	Empty	B

Note:
B = 3-Out-of-4 Enrichment
L= Low Enrichment of 3x3 Storage
H = High Enrichment of 3x3 Storage
Empty = Empty Cell

Boundary Between 3x3 Storage and 3-Out-of-4 Storage

Interface

C	Empty	C	Empty	C	Empty
Empty	B	Empty	B	Empty	C
L	L**	L	L**	B	Empty
L	L	L	L	Empty	C
L	H	L	L**	B	Empty
L	L	L	L	Empty	C

Note:
B = 3-Out-of-4 Enrichment
L = Low Enrichment of 3x3 Storage
H= High Enrichment of 3x3 Storage
C= 2-Out-of-4 Enrichment
Empty = Empty Cell

Boundary Between 3x3 Storage and 2-Out-of-4 Storage

Note:

1. A row of empty cells can be used at the interface to separte the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. For the 3-Out-of-4 configuration, the row beyond the Low enrichment can swap empty and B assemblies, however the next outer row must change the indicated assembly (*) to an empty cell.
4. For the 2-Out-of-4 configuration, the row beyond the Low enrichment can swap empty and B assemblies, however the next outer row of empty and C assemblies must also swap locations.
5. If empty cells are in indicated locations (**), then the face adjacent B assemblies can be C assemblies.

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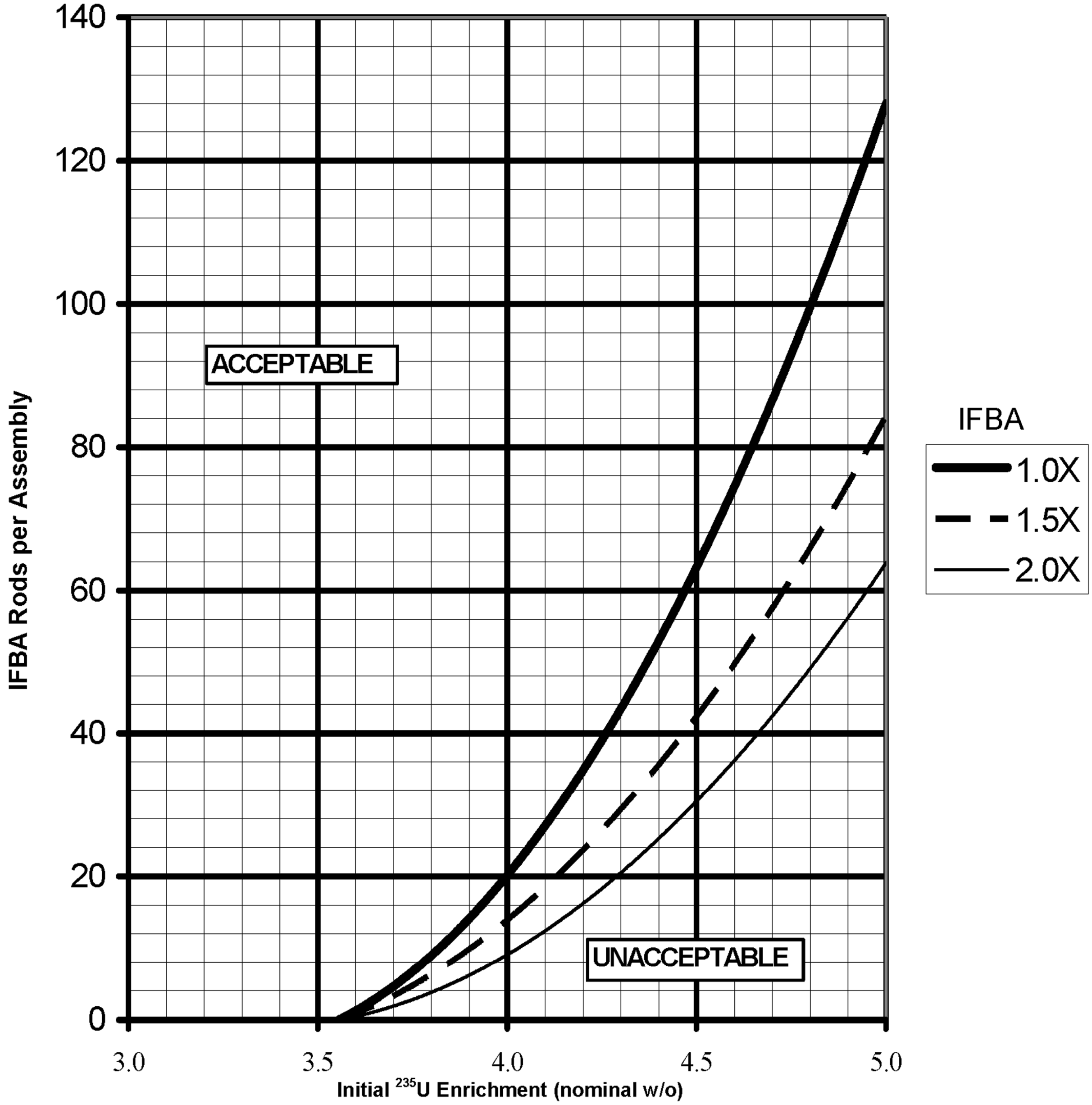
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

VOGTLE UNIT 2 INTERFACE REQUIREMENTS (3X3 TO EMPTY CELL CHECKERBOARD STORAGE)

FIGURE 4.3–53



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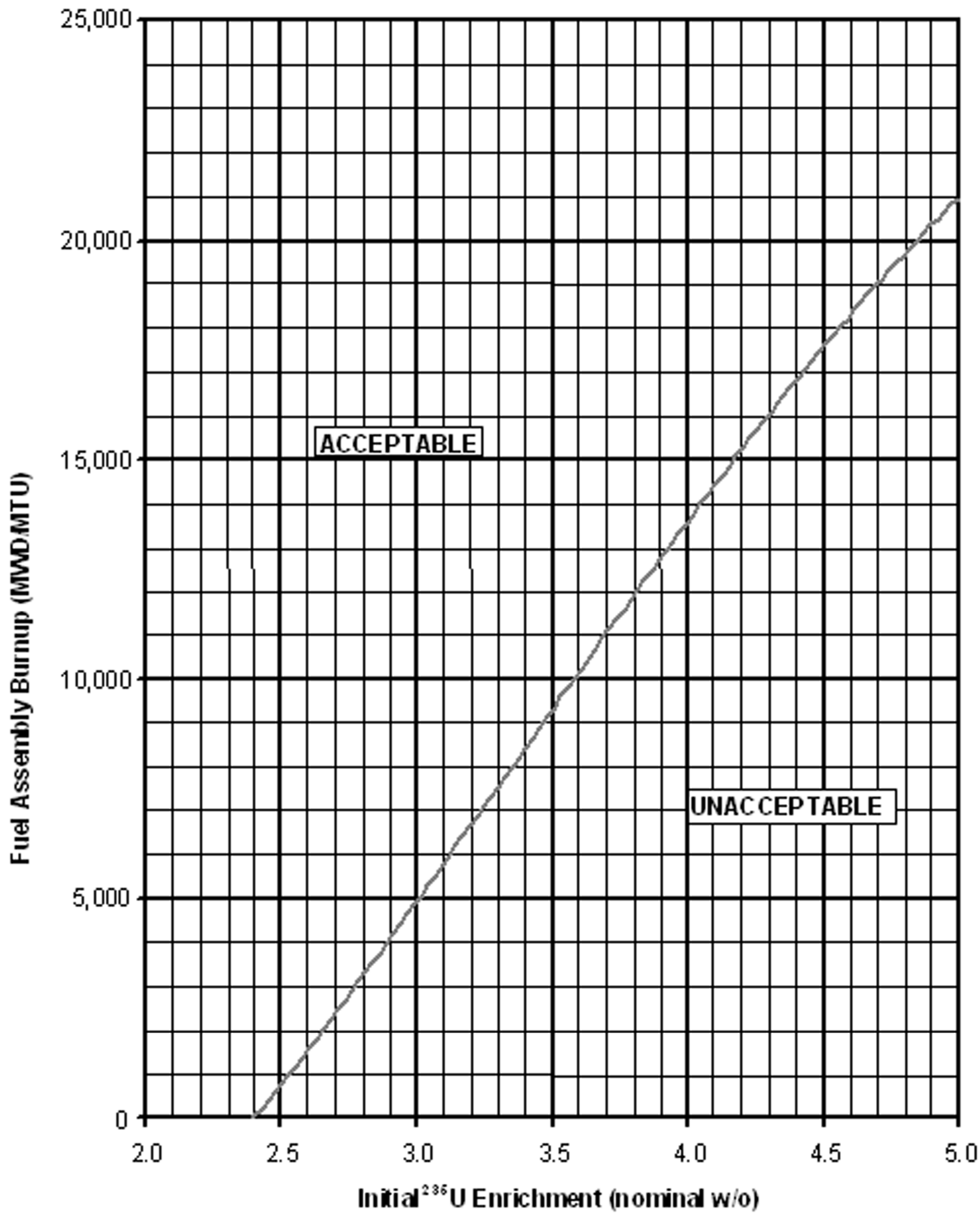
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	VOGTLE UNIT 1 IFBA CREDIT REQUIREMENTS FOR ALL CELL STORAGE
		FIGURE 4.3–54



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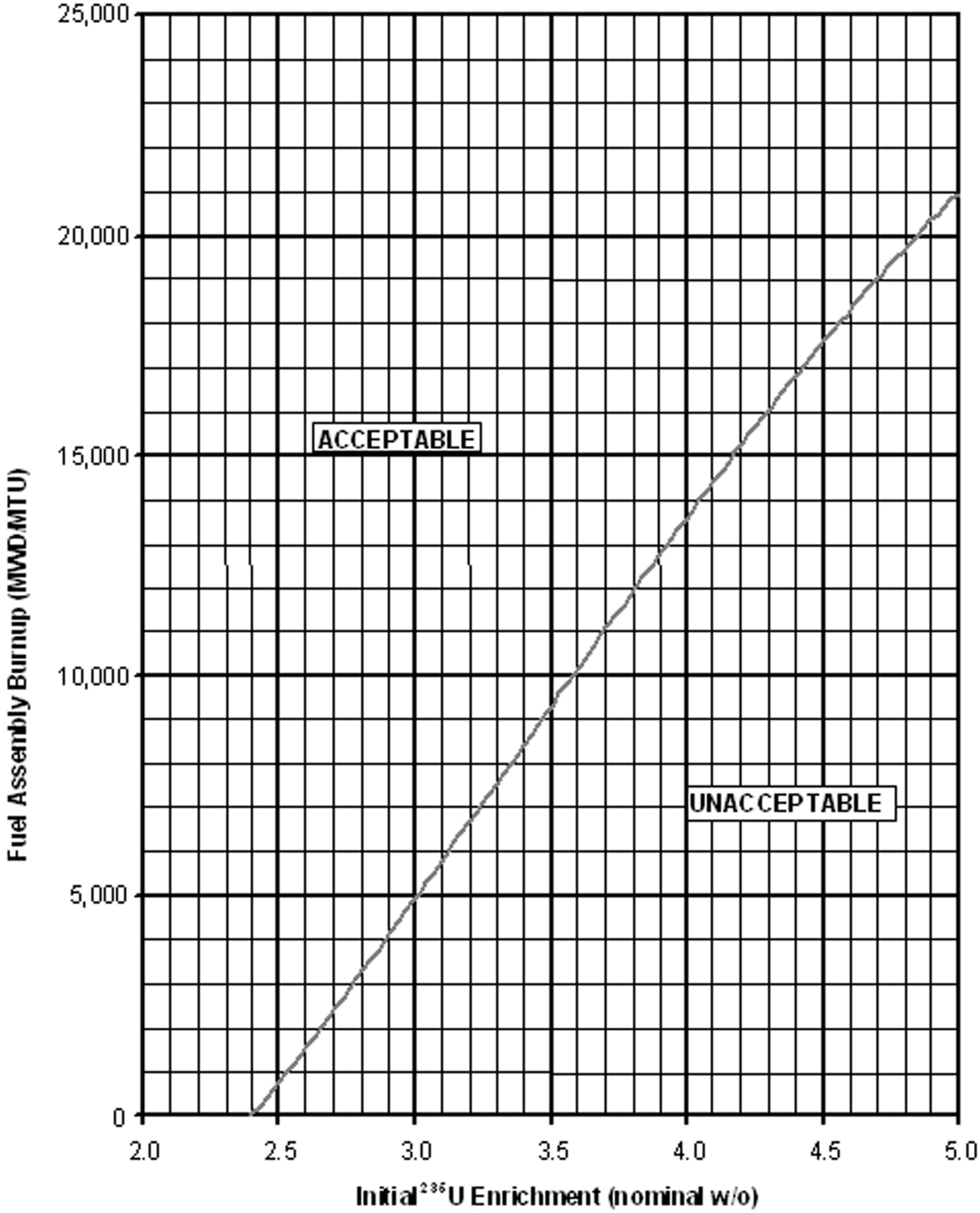
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	VOGTLE UNIT 2 BURNUP CREDIT REQUIREMENTS FOR 3-OUT-OF-4 STORAGE
		FIGURE 4.3–55



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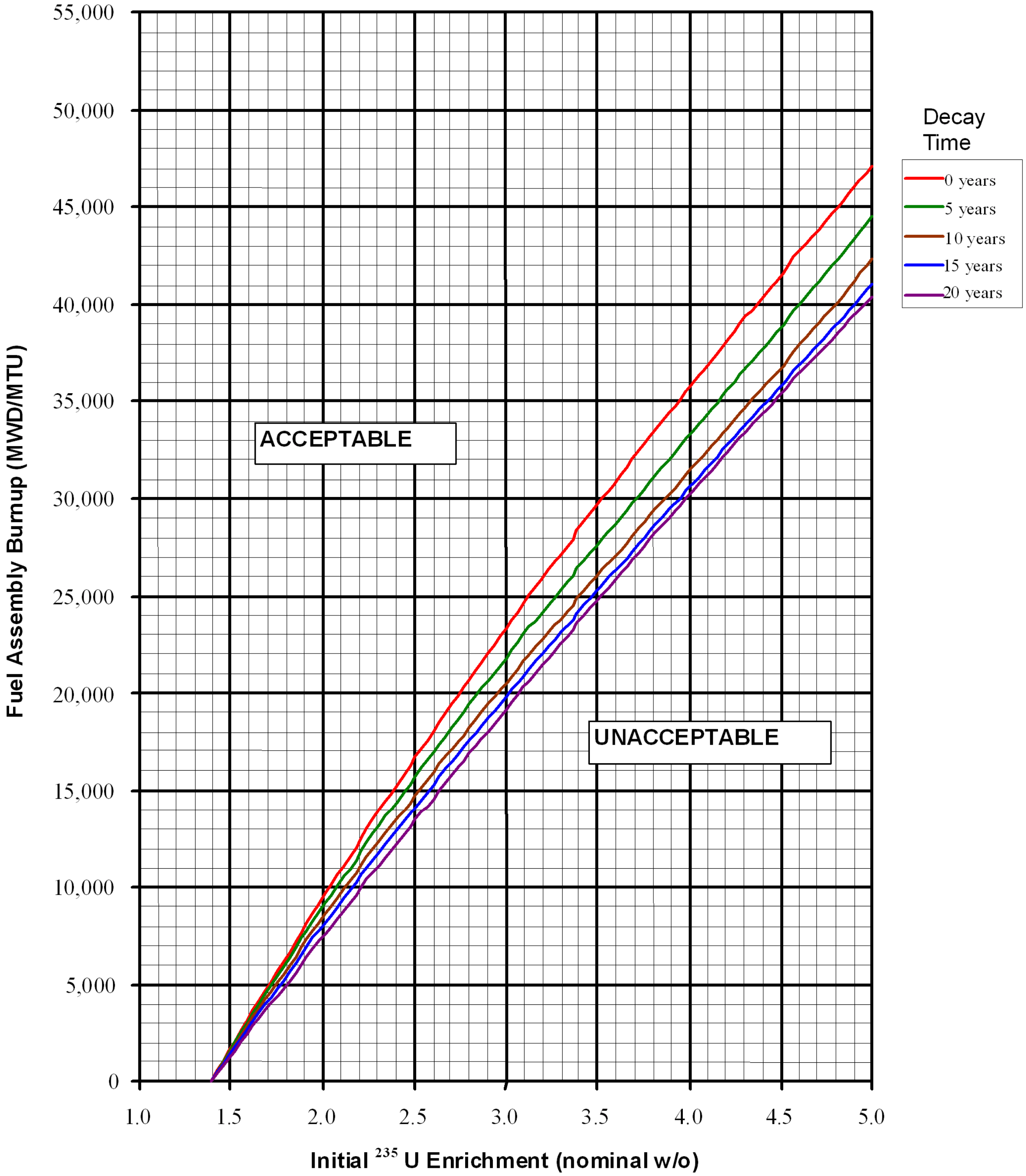
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

VOGTLE UNIT 2 IFBA CREDIT REQUIREMENTS FOR CENTER ASSEMBLY FOR 3X3 STORAGE

FIGURE 4.3–56



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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	VOGTLE UNIT 2 BURNUP CREDIT REQUIREMENTS FOR PERIPHERAL ASSEMBLIES FOR 3X3 STORAGE
	FIGURE 4.3–57

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 DESIGN BASES

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer compatible with the heat generation distribution in the core so that heat removal by the reactor coolant system (RCS) or the emergency core cooling system (ECCS), when applicable, ensures that the following performances and safety criteria requirements are met:

A. Fuel damage (defined as penetration of the fission product barrier; i.e., the fuel rod clad) is not expected during normal operation and operational transients (Condition 1) or any transient conditions arising from faults of moderate frequency (Condition 2). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

B. The reactor can be brought to a safe state following a Condition 3 event with only a small fraction of fuel rods damaged (See above definition.), although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

C. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition 4 events.

To satisfy the above requirements, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling (DNB) Design Basis

4.4.1.1.1 Basis

There will be at least a 95-percent probability that DNB will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition 1 and 2 events) at a 95-percent confidence level.

4.4.1.1.2 Discussion

The design method employed to meet the DNB design basis for the VANTAGE + / VANTAGE 5 fuel assemblies is the Revised Thermal Design Procedure (RTDP), reference 81. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty

factors, RTDP design limit DNBR values are determined such that there is at least 95-percent probability at a 95-percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2).

Uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) have been evaluated for the VEGP Units 1 and 2 with RTD bypass loops, reference 82, and for RTD bypass loops eliminated, reference 83. In the DNBR analyses with RTDP, a set of plant operating parameter uncertainties was used to bound operation with either RTD bypass loops or RTD bypass loops eliminated. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Any adverse instrumentation bias is treated as either an input to the DNBR calculation or as a direct DNBR penalty. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The RTDP design limit DNBR values are 1.24 and 1.23 for the typical and thimble cells, respectively, for VANTAGE + / VANTAGE 5 fuel.

The design limit DNBR values are used as a basis for the Technical Specifications and are considered in appropriate 10 CFR 50.59 evaluations.

To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow (paragraph 4.4.2.2.5) and transition core (paragraph 4.4.2.2.6), the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

The DNBR analyses for operation at 3626 MWt were based on the continued use of thimble plugging devices. Operation with thimble plugs in place reduces the core bypass flow through the fuel assembly thimble tubes. Bypass flow is assumed to be ineffective for core heat removal. The reduction in core bypass flow for operation with the thimble plugs in place is a DNBR benefit. All of the flow and DNBR values presented in table 4.4-1 are based the use of thimble plugs.

The option of thimble plug removal was included in all of the non-DNBR analyses performed in support of the uprating to 3626 MWt. The allocation of available DNBR margin would be required to support thimble plug removal.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method, the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analyses input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

By preventing DNB, adequate heat transfer is ensured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis, since it will be within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients

associated with Condition 2 events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

4.4.1.2 Fuel Temperature Design Basis

4.4.1.2.1 Basis

During modes of operation associated with Condition 1 and Condition 2 events, there is at least a 95-percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature at the 95-percent confidence level. The melting temperature of UO_2 is taken as 5080°F, [(1)] unirradiated and decreasing 58°F per 10,000 MWd/tonne of uranium. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations as described in paragraph 4.4.2.9.1.

4.4.1.2.2 Discussion

Fuel rod thermal evaluations are performed at rated power, at maximum overpower, and during transients at various burnups. These analyses ensure that this design basis and the fuel integrity design bases given in section 4.2 are met. They also provide input for the evaluation of Condition 3 and 4 events given in chapter 15.

4.4.1.3 Core Flow Design Basis

4.4.1.3.1 Basis

A minimum of 93.6 percent of the thermal flowrate will pass through the fuel rod region of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes and the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

4.4.1.3.2 Discussion

Core cooling evaluations are based on the thermal flowrate (minimum flow) entering the reactor vessel. A maximum of 6.4 percent of this value is allotted as bypass flow. This includes rod cluster control (RCC) guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

4.4.1.4 Hydrodynamic Stability Design Basis

Modes of operation associated with Condition 1 and 2 events shall not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The above design bases together with the fuel clad and fuel assembly design bases given in subsection 4.2.1 are sufficiently comprehensive that additional limits are not required.

Fuel rod diametral gap characteristics, moderator-coolant flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above-mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time (paragraph 4.2.3.3), and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution (paragraph 4.4.2.2) and moderator void distribution (paragraph 4.4.2.4) are included in the core thermal evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in paragraph 4.2.3.3, the fuel rod conditions change with time. A single clad temperature limit for Condition 1 or Condition 2 events is not appropriate, since it would of necessity be overly conservative. A clad temperature limit is applied to the loss-of-coolant accident (LOCA) (subsection 15.6.5), control rod ejection accident (subsection 15.4.8), and locked rotor accident (subsection 15.3.3).

4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Summary Comparison

Table 4.4-1 provides the design parameters at 3626 MWt for the 17 x 17 VANTAGE + / VANTAGE 5 fuel. The LOPAR fuel is not analyzed for use at 3626 MWt. The LOPAR design parameters at 3565 MWt are retained in table 4.4-1 for historical purposes.

4.4.2.2 Critical Heat Flux Ratio or DNBR and Mixing Technology

The minimum DNBRs for the rated power, design overpower, and anticipated transient conditions are given in table 4.4-1. The minimum DNBR in the limiting flow channel is typically downstream of the peak heat flux location (hotspot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in paragraphs 4.4.2.2.1 and 4.4.2.2.2. The VIPRE-01 computer code (discussed in paragraph 4.4.4.5) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in paragraphs 4.4.4.3.1 (nuclear hot channel factors) and 4.4.2.2.4 (engineering hot channel factors).

4.4.2.2.1 DNB Technology

The primary DNB correlation that was used for the analysis of the 17 x 17 LOPAR fuel was the WRB-1 correlation (reference 84).

The WRB-1 correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of parameters for the WRB-1 correlation is:

Pressure	$1440 \leq P \leq 2490$ psia
Local mass velocity	$0.9 \leq G_{loc}/10^6 \leq$ 3.7 lb/ft^2-hr
Local quality	$-0.2 \leq X_{loc} \leq 0.3$
Heated length, inlet to CHF location	$L_h \leq 14$ feet
Grid spacing	$13 \leq g_{sp} \leq 32$ inches
Equivalent hydraulic diameter	$0.37 \leq d_e \leq 0.60$ inches
Equivalent heated hydraulic diameter	$0.46 \leq d_h \leq 0.59$ inches

Figure 4.4-1 shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

A correlation limit DNBR of 1.17 for the WRB-1 correlation has been approved by the NRC for 17 x 17 LOPAR fuel.

The primary DNB correlation used for the analysis of the VANTAGE + / VANTAGE 5 fuel is the WRB-2 correlation (reference 85). The WRB-2 DNB correlation was developed to take credit for the VANTAGE 5 intermediate flow mixer (IFM) grid design. A limit of 1.17 is applicable for the WRB-2 correlation. Figure 4.4-1 shows measured critical heat flux plotted against predicted critical heat flux using the WRB-2 correlation.

Use of this correlation has been conservatively modified to utilize a penalty above a certain high quality threshold within the approved ranges (reference 102).

The applicable range of parameters for the WRB-2 correlation is:

Pressure	$1440 \leq P \leq 2490$ psia
Local mass velocity	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-h
Local quality	$-0.1 \leq X_{loc} \leq 0.3$
Heated length, inlet to CHF location	$L_h \leq 14$ ft
Grid spacing	$10 \leq g_{sp} \leq 26$ in.
Equivalent hydraulic diameter	$0.33 \leq d_e \leq 0.5101$ in.
Equivalent heated hydraulic diameter	$0.45 \leq d_h \leq 0.66$ in.

The W-3 DNB correlation, references 86 and 4, is used for both fuel types where the primary DNBR correlations are not applicable. The WRB-1 and WRB-2 correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlations limit is 1.45, reference 87. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30. A cold wall factor, reference 88, is applied to the W-3 DNB correlation to account for the presence of the unheated thimble surfaces.

4.4.2.2.2 Definition of DNBR

The DNB heat flux ratio (DNBR) as applied to typical cells (flow cells with all walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$\mathrm{DNBR} = \frac{q''_{\mathrm{DNB,N}}}{q''_{loc}} \tag{1}$$

where:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F} \tag{2}$$

$q''_{DNB,EU}$ = the uniform DNB heat flux as predicted by the WRB -1 DNB correlation, the WRB – 2 DNB correlation, or the W – 3 DNB correlation (typical cell only).

F = the flux shape factor to account for nonuniform axial heat flux distributions[8] with the term "C" modified as in reference 4.

q''_{loc} = the actual local heat flux.

The DNBR as applied to the W-3 DNB correlation when a cold wall (CW) is present is:

$$\mathrm{DNBR} = \frac{q''_{\mathrm{DNB,N,CW}}}{q''_{\mathrm{loc}}} \tag{3}$$

where:

$$q''_{\mathrm{DNB,N,CW}} = \frac{q''_{\mathrm{DNB,EU,Dh}} \times \mathrm{CWF}}{\mathrm{F}} \tag{4}$$

where:

$q''_{\mathrm{DNB,EU,Dh}}$ = the uniform DNB heat flux as predicted by the W - 3 cold wall DNB correlation(4) when not all flow cell walls are heated (thimble CW cell).

$$\text{CW factor}^{(4)} = 1.0 - \text{Ru}\ 13.76 - 1.372e^{1.78x} - 4.732 \left(\frac{G}{10^6}\right)^{-0.0535} - 0.0619\left(\frac{P}{1000}\right)^{0.14} - 8.509\,\text{Dh}^{0.107}$$

Ru = 1.0-De/Dh

4.4.2.2.3 Mixing Technology

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels, the local fluid density, and the flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient (TDC) which is defined as:

$$TDC = \frac{w'}{\rho V a} \tag{5}$$

where:

w' = flow exchange rate per unit length (lbm/ft-s).

ρ = fluid density (lbm/ft^3).

V = fluid velocity (ft/s).

a = lateral flow area between channels per unit length (ft^2/ft).

The application of the TDC in the THINC analysis for determining the overall mixing effect or heat exchange rate is presented in reference 10. The application of the TDC in the VIPRE-01 analysis is presented in reference 101.

Various mixing tests have been performed at Columbia University.[9] These series of tests, using the "R" mixing vane grid design on 13-, 26-, and 32-in. grid spacing, were conducted in pressurized water loops at Reynolds numbers similar to that of a pressurized water reactor (PWR) core under the following single- and two-phase (subcooled boiling) flow conditions:

- Pressure — 1500 to 2400 psia
- Inlet temperature — 332 to 642°F
- Mass velocity — 1.0 to 3.5 x 10^6 $lbm/h\text{-}ft^2$
- Reynolds number — 1.34 to 7.45 x 10^5
- Bulk outlet quality — -52.1 to -13.5 percent

TDC is determined by comparing the THINC code predictions with the measured subchannel exit temperatures. Data for 26-in. axial grid spacing are presented in figure 4.4-2, where the TDC coefficient is plotted versus the Reynolds number. TDC is found to be independent of the Reynolds number, mass velocity, pressure, and quality over the ranges tested. The two-phase data (local, subcooled boiling) fell within the scatter of the single-phase data. The effect of two-

phase flow on the value of TDC has been demonstrated by Cadek,[9] Rowe and Angle,[11][12] and Gonzalez-Santalo and Griffith.[13] In the subcooled boiling region, the values of TDC were indistinguishable from the single-phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in PWR core geometry, the value of TDC increased with quality to a point and then decreased, but never below the single-phase value. Gonzalez-Santalo and Griffith show that the mixing coefficient increased as the void fraction increased.

The data from these tests on the R-grid showed that a design TDC value of 0.038 (for 26-in. grid spacing) can be used in determining the effect of coolant mixing in the THINC or VIPRE-01 analysis. A mixing test program similar to the one described above was conducted at Columbia University for the current 17 x 17 geometry and mixing vane grids on 26-in. spacing.[14] The mean value of TDC obtained from these tests was 0.059, and all data were well above the current design value of 0.038.

Since the actual grid spacing for 17 x 17 LOPAR fuel is approximately 20 in., additional margin is available for this design, as the value of TDC increases as grid spacing decreases.[9]

The inclusion of three intermediate flow mixer grids in the upper span of the VANTAGE 5 fuel assembly results in a grid spacing of approximately 10 inches. Per Reference 85, a TDC value of 0.038 was chosen as a conservatively low value for use in VANTAGE 5 to determine the effect of coolant mixing in the core thermal performance analysis.

4.4.2.2.4 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux hot channel factor considers the local maximum linear heat generation rate at a point (the hotspot), and the enthalpy rise hot channel factor involves the maximum integrated value along a channel (the hot channel).

Each of the total hot channel factors is composed of a nuclear hot channel factor (paragraph 4.4.4.3) describing the neutron power distribution and an engineering hot channel factor, which allows for variations in flow conditions and fabrication tolerances. The engineering hot channel factors are made up of subfactors which account for the influence of the variations of fuel pellet diameter, density, enrichment, and eccentricity; inlet flow distribution; flow redistribution; and flow mixing.

A. Heat Flux Engineering Hot Channel Factor, F_Q^E

   The heat flux engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density, and enrichment and has a value of 1.03 at the 95-percent probability level with 95-percent confidence. As shown in reference 15, no DNB penalty need be taken for the short, relatively low-intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.

B. Enthalpy Rise Engineering Hot Channel Factor, $F_{\Delta H}^E$

   The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise is directly considered in the VIPRE-01 subchannel analysis

(paragraph 4.4.4.5) under any reactor operating condition. The items considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

1. Pellet Diameter, Density, and Enrichment

   Variations in pellet diameter, density, and enrichment are considered statistically in establishing the limit DNBRs (paragraph 4.4.1.1.2) for the Revised Thermal Design Procedure (reference 81) employed in this application. Uncertainties in these variables are determined from sampling of manufacturing data.

2. Inlet Flow Maldistribution

   The consideration of inlet flow maldistribution in core thermal performances is discussed in paragraph 4.4.4.2.2. A design basis of 5-percent reduction in coolant flow to the hot assembly is used in the VIPRE-01 analysis.

3. Flow Redistribution

   The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the VIPRE-01 analysis for every operating condition evaluated.

4. Flow Mixing

   The subchannel mixing model incorporated in the VIPRE-01 code and used in reactor design is based on experimental data[16] discussed in paragraphs 4.4.2.2.3 and 4.4.4.5. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

4.4.2.2.5 Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in reference 79, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained from measured plant operating parameters (such as $F^N_{\Delta H}$ or core flow), which are less limiting than those required by the plant safety analysis, can be used to offset the effect of rod bow.

For the safety analysis of the VEGP units, sufficient DNBR margin was maintained (paragraph 4.4.1.1.2) to accommodate full and low flow rod bow DNBR penalties which are based on the methodology in reference 80. The rod bow DNBR penalties that are applicable to LOPAR fuel assembly analyses using the WRB-1 DNB correlation and to VANTAGE 5 fuel assembly analyses using the WRB-2 DNB correlation were determined using the methodology in reference 80.

The maximum rod bow penalties (< 2 percent DNBR) accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWd/Mtu. At burnups greater than 24,000 MWd/Mtu, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in

fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required (reference 95).

In the upper spans of the VANTAGE 5 fuel assembly, additional restraint is provided with the intermediate flow mixer grids such that the grid-to-grid spacing in those spans with IFM grids is approximately 10 inches compared to approximately 20 inches in the other spans. Using the NRC approved scaling factor results in predicted channel closure in the limiting 10-inch spans of less than 50-percent closure. Therefore, no rod bow DNBR penalty is required in the 10-inch spans in the VANTAGE 5 safety analyses.

4.4.2.2.6 Transition Core DNB Methodology

The LOPAR and VANTAGE 5 designs have been shown to be hydraulically compatible in reference 85.

The Westinghouse transition core DNB methodology is given in references 89, 90, and 91. Using this methodology, transition cores are analyzed as if the entire core consisted of one assembly type (full LOPAR or full VANTAGE 5). The resultant DNBRs are then reduced by the appropriate transition core penalty.

The VANTAGE 5 fuel assembly has a higher mixing vane grid loss coefficient relative to the LOPAR mixing vane grid loss coefficient. In addition, the VANTAGE 5 fuel assembly has IFM grids located in spans between mixing vane grids, where no grid exists in the LOPAR assembly. The higher loss coefficients and the additional grids introduce localized flow redistribution from the VANTAGE 5 fuel assembly into the LOPAR assembly at the axial zones near the mixing vane grid and the IFM grid position in a transition core. Between the grids, the tendency for velocity equalization in parallel open channels causes flow to return to the VANTAGE 5 fuel assembly. The localized flow redistribution described above actually benefits the LOPAR assembly. This benefit more than offsets the slight mass flow bias due to velocity equalization at nongridded locations. Thus, the analysis for a full core of LOPAR is appropriate for that fuel type in a transition core. There is no transition core DNBR penalty for the LOPAR fuel.

The transition core DNBR penalty for VANTAGE 5 fuel is discussed in references 92 and 93. The transition core penalty is a function of the number of VANTAGE 5 fuel assemblies in the core, reference 94. Sufficient DNBR margin is maintained in the VANTAGE 5 safety analysis to completely offset this transition core penalty.

4.4.2.3 <u>Linear Heat Generation Rate</u>

The core average and maximum linear heat generation rates are given in table 4.4-1. The method of determining the maximum linear heat generation rate is given in paragraph 4.3.2.2.

4.4.2.4 <u>Void Fraction Distribution</u>

The VIPRE-01 calculated core average and the hot subchannel maximum and average void fractions are presented in table 4.4-2 for operation at full power. The void models used in the VIPRE-01 code are described in paragraph 4.4.2.7.3.

4.4.2.5 Core Coolant Flow Distribution

Assembly average coolant mass velocity and enthalpy at various radial and axial core locations are given in figures 4.4-3 through 4.4-5. Typical coolant enthalpy rise and flow distributions for the 4-ft elevation (one-third of core height) are shown in figure 4.4-3, for the 8-ft elevation (two-thirds of core height) in figure 4.4-4, and at the core exit in figure 4.4-5. These distributions are representative of a Westinghouse four-loop plant. The THINC code analysis for this case utilized a uniform code inlet enthalpy and inlet flow distribution. No orificing is employed in the reactor design.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

4.4.2.6.1 Core Pressure Drops

The analytical model and experimental data used to calculate the pressure drops shown in table 4.4-1 are described in paragraph 4.4.2.7. The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. The full-power operation pressure drop values shown in table 4.4-1 are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. These pressure drops are based on the best-estimate flow for actual plant operating conditions as described in subsection 5.1.4. This subsection also defines and describes the thermal design flow (minimum flow), which is the basis for reactor core thermal performance and the mechanical design flow (maximum flow), which is used in the mechanical design of the reactor vessel internals and fuel assemblies. Since the best-estimate flow is that flow which is most likely to exist in an operating plant, the calculated core pressure drops in table 4.4-1 are based on this best-estimate flow rather than the thermal design flow.

Uncertainties associated with the core pressure drop values are discussed in paragraph 4.4.2.9.2.

4.4.2.6.2 Hydraulic Loads

The fuel assembly holddown springs (figure 4.2-2) are designed to keep the fuel assemblies in contact with the lower core plate under all Condition 1 and 2 events except the turbine overspeed transient associated with a loss of external load. The holddown springs are designed to tolerate the possibility of an overdeflection associated with fuel assembly liftoff for this case and provide contact between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a LOCA. These conditions are presented in subsection 15.6.5.

Hydraulic loads at normal operating conditions are calculated considering the mechanical design flow, which is described in section 5.1, and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the cold mechanical design flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flowrate 20 percent greater than the mechanical design flow, are evaluated to be approximately twice the fuel assembly weight.

The hydraulic verification tests for the LOPAR fuel assembly and the VANTAGE 5 fuel assembly are discussed in references 18 and 85, respectively.

4.4.2.7 Correlation and Physical Data

4.4.2.7.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the familiar Dittus-Boelter correlation,[19] with the properties evaluated at bulk fluid conditions:

$$\frac{hD_e}{K} = 0.023 \frac{D_e G^{0.8}}{\mu} \frac{C_\rho \mu^{0.4}}{K} \quad (6)$$

where :

h	=	heat transfer coefficient (Btu/h-ft^2-°F).
D_e	=	equivalent diameter (ft).
K	=	thermal conductivity (Btu/h-ft-°F).
G	=	mass velocity (lbm/h-ft^2).
μ	=	dynamic viscosity (lbm/ft-h).
$C\rho$	=	heat capacity (Btu/lb-°F).

This correlation has been shown to be conservative[20] for rod bundle geometries with pitch-to-diameter ratios in the range used by PWRs.

The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's correlation.[21] After this occurrence the outer clad wall temperature is determined by:

$$\Delta T_{sat} = [0.072 \exp(-P/1260)] (q'')^{0.5} \quad (7)$$

where:

ΔT_{sat}	=	wall superheat, $T_w - T_{sat}$ (°F).
q"	=	wall heat flux (Btu/h-ft^2).
P	=	pressure (psia).
T_w	=	outer clad wall temperature (°F).
T_{sat}	=	saturation temperature of coolant at P (°F).

4.4.2.7.2 Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single-phase water. These assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flowrate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible (table 4.4-2). Two-phase flow considerations in the core thermal subchannel analyses are considered and the models are discussed in paragraph 4.4.4.2.3. Core and vessel pressure losses are calculated by equations of the form:

$$\Delta\rho_L = \left(K + F\frac{L}{D_e}\right)\frac{\rho V^2}{2g_c(144)} \qquad (8)$$

where:

$\Delta\rho_L$	=	unrecoverable pressure drop (lb/in.^{2}).
ρ	=	fluid density (lbm/ft^3).
L	=	length (ft).
De	=	equivalent diameter (ft).
V	=	fluid velocity (ft/s).
g_c	=	32.174 (lbm-ft/lb$_f$-s^2).
K	=	form loss coefficient (dimensionless).
F	=	friction loss coefficient (dimensionless).

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in table 4.4-1 for unrecoverable pressure loss across the reactor vessel, including the inlet and outlet nozzles, and across the core. The results of full-scale tests of core components and fuel assemblies were utilized in developing the core pressure loss characteristic. The pressure drop for the vessel was obtained by combining the core loss with correlation of one-seventh scale model hydraulic test data on a number of vessels(22)(23) and form loss relationships.(24) Moody(25) curves were used to obtain the single-phase friction factors.

Tests of the primary coolant loop flowrates will be made (paragraph 4.4.5.1) prior to initial criticality to verify that the flowrates used in the design, which were determined in part from the pressure losses calculated by the method described here, are conservative.

4.4.2.7.3 Void Fraction Correlation

VIPRE-01 considers two-phase flow in two steps. First, a quality model is used to compute the flowing vapor mass fraction (true quality) including the effects of subcooled boiling. Then, given the true quality, a bulk void model is applied to compute the vapor volume fraction (void fraction).

VIPRE-01 uses a profile fit model (100) for determining subcooled quality. It calculates the local vapor volumetric fraction in forced convection boiling by: 1) predicting the point of bubble departure from the heated surface, and 2) postulating a relationship between the true local vapor fraction and the corresponding thermal equilibrium value.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

4.4.2.8 Thermal Effects of Operational Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power, and axial power imbalance. Steady-state operation within these safety limits ensures that the DNBR design basis is met. Figure 15.0.6-1 shows the DNBR limit lines and the resulting overtemperature ΔT trip lines (which become part of the Technical Specifications), plotted as ΔT versus T_{avg} for various pressures. This system provides adequate protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients (e.g., uncontrolled rod bank withdrawal at power incident (subsection 15.4.2)), specific protection functions are provided as described in section 7.2. The use of these protection functions is described in chapter 15.

4.4.2.9 Uncertainties in Estimates

4.4.2.9.1 Uncertainties in Fuel and Clad Temperatures

As discussed in paragraph 4.4.2.11, the fuel temperature is a function of crud, oxide, clad, pellet-clad gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties, such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties, such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the in-pile thermocouple measurements,[29-35] by out-of-pile measurements of the fuel and clad properties,[36-47] and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is also included in the calculation of the total uncertainty.

In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in paragraph 4.3.2.2.1.

Reactor trip setpoints, as specified in the Technical Specifications, include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproducibility, temperature measurement uncertainties, noise, and heat capacity variations.

Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

4.4.2.9.2 Uncertainties in Pressure Drops

Core and vessel pressure drops based on the best-estimate flow, as described in section 5.1, are quoted in table 4.4-1. The uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drops is to determine the primary system coolant flowrates, as discussed in section 5.1. In addition, as discussed in paragraph 4.4.5.1, tests on the primary system prior to initial criticality will be made to verify that a conservative primary system coolant flowrate has been used in the design and analyses of the plant.

4.4.2.9.3 Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses are discussed in paragraph 4.4.4.2.2.

4.4.2.9.4 Uncertainty in DNB Correlation

The uncertainty in the DNB correlation (paragraph 4.4.2.2) can be written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is discussed in paragraph 4.4.2.2.2.

4.4.2.9.5 Uncertainties in DNBR Calculations

The uncertainties in the DNBRs calculated by VIPRE-01 analysis (paragraph 4.4.4.5) due to uncertainties in the nuclear peaking factors are accounted for by applying conservatively high values of the nuclear peaking factors and including measurement error allowances in the statistical evaluation of the limit DNBR (paragraph 4.4.1.1) using the Revised Thermal Design Procedure (reference 81). In addition, conservative values for the engineering hot channel factors are used as discussed in paragraph 4.4.2.2.4. The results of a sensitivity study[17] with THINC-IV show that the minimum DNBR in the hot channel is relatively insensitive to variations in the core-wide radial power distribution (for the same value of $F_{\Delta H}^{N}$). VIPRE-01 was demonstrated to be equivalent to THINC-IV in reference 101.

The ability of the VIPRE computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in paragraph 4.4.4.5 and in reference101. Studies [100, 101] have been performed to determine the sensitivity of the minimum DNBR in the hot channel to void fraction

correlation (paragraph 4.4.2.7.3) and the inlet flow distributions. The results of these studies show that the minimum DNBR is relatively insensitive to variation in these parameters. Furthermore, the VIPRE flow field model for predicting conditions in the hot channels is consistent with that used in the derivation of the DNB correlation limits, including void/quality modeling, turbulent mixing and crossflow, and two-phase friction [101].

4.4.2.9.6 Uncertainties in Flowrates

The uncertainties associated with loop flowrates are discussed in section 5.1. A thermal design flow is defined for use in core thermal performance evaluations which accounts for both prediction and measurement uncertainties. In addition, another 6.4 percent of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses the core through the various available vessel flow paths described in paragraph 4.4.4.2.1.

4.4.2.9.7 Uncertainties in Hydraulic Loads

As discussed in section 4.4.2.6.2, hydraulic loads on the fuel assembly are evaluated for a pump overspeed transient which creates flowrates 20-percent greater than the mechanical design flow. As stated in section 5.1, the mechanical design flow is greater than the best estimate or most likely flowrate value for the actual plant operating condition.

4.4.2.9.8 Uncertainty in Mixing Coefficient

The value of the mixing coefficient, TDC, used in VIPRE-01 analyses for this application is 0.038 for LOPAR fuel and VANTAGE 5 fuel.

The results of the mixing tests done on 17 x 17 LOPAR geometry, as discussed in paragraph 4.4.2.2.3, had a mean value of TDC of 0.059 and standard deviation of s equal to 0.007. Hence, the current design value of TDC is almost three standard deviations below the mean for 26-in. grid spacing.

4.4.2.10 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned rod cluster control assembly (RCCA) could cause changes in hot channel factors; however, these events are analyzed separately in chapter 15.

Other possible causes for quadrant power tilts include X-Y xenon transients, inlet temperature mismatches, enrichment variations within tolerances, and so forth.

In addition to unanticipated quadrant power tilts as described above, other readily explainable asymmetries may be observed during calibration of the excore detector quadrant power tilt alarm. During operation, incore maps are taken at least one per month and additional maps are obtained periodically for calibration purposes. Each of these maps is reviewed for deviations from the expected power distributions. Asymmetry in the core, from quadrant to quadrant, is

frequently a consequence of the design when assembly and/or component shuffling and rotation requirements do not allow exact symmetry preservation. In each case, the acceptability of an observed asymmetry, planned or otherwise, depends solely on meeting the required accident analyses assumptions. In practice, once acceptability has been established by review of the incore maps, the quadrant power tilt alarms and related instrumentation are adjusted to indicate zero quadrant power tilt ratio as the final step in the calibration process. This action ensures that the instrumentation is correctly calibrated to alarm in the event an unexplained or unanticipated change occurs in the quadrant-to-quadrant relationships between calibration intervals. Proper functioning of the quadrant power tilt alarm is significant; no allowances are made in the design for increased hot channel factors due to unexpected developing flux tilts, since all likely causes are prevented by design or procedures or are specifically analyzed. Finally, in the event that unexplained flux tilts do occur, the Technical Specifications provide appropriate corrective actions to ensure continued safe operation of the reactor.

4.4.2.11 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in subsection 4.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation. A discussion of fuel clad integrity is presented in paragraph 4.2.3.1.

The thermal-hydraulic design ensures that the maximum fuel temperature is below the melting point of UO_2 (paragraph 4.4.1.2). To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations as described in paragraph 4.4.2.9.1. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad gap, and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, etc., have been combined into a semi empirical thermal model (paragraph 4.2.3.3) which includes a model for time-dependent fuel densification as given in references 96 and 99. This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to in-pile fuel temperature measurements[29-35, 97] and melt radius data[49][50] with good results.

Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are performed several times in the fuel rod lifetime (with consideration of time-dependent densification) to determine the maximum fuel temperatures.

The principal factors employed in the determination of the fuel temperature are discussed below.

4.4.2.11.1 UO_2 Thermal Conductivity

The thermal conductivity of uranium dioxide was evaluated from data reported by Howard, et al.;[36] Lucks, et al.;[37] Daniel, et al.;[38] Feith;[39] Vogt, et al.;[40] Nishijima, et al.;[41] Wheeler, et al.;[42] Godfrey, et al.;[43] Stora, et al.;[44] Bush;[45] Asamoto, et al.;[46] Kruger;[47] and Gyllander.[51]

At the higher temperatures, thermal conductivity is best obtained by utilizing the integral conductivity to melt which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate is:

$$\int_{0}^{2800} Kdt = 93\,W/cm$$

This conclusion is based on the integral values reported by Gyllander;[51] Lyons, et al.;[52] Copolin, et al.;[53] Duncan;[49] Bain;[54] and Stora.[55]

The design curve for the thermal conductivity is shown in figure 4.4-6. The section of the curve at temperatures between 0°C and 1300°C is in excellent agreement with the recommendation of the International Atomic Energy Agency (IAEA) panel.[56] The section of the curve above 1300°C is derived for an integral value of 93 W/cm.[49][51][55]

Thermal conductivity for UO_2 at 95-percent theoretical density can be represented best by the following equation:

$$K = \frac{1}{11.8 + 0.0238\,T} + 8.775 \times 10^{-13}\ T^3 \qquad (9)$$

where:

K = W/cm-°C.

T = °C.

4.4.2.11.2 Radial Power Distribution in UO_2 Fuel Rods

An accurate description of the radial power distribution as a function of burnup is needed for determining the power level for incipient fuel melting and other important performance parameters, such as pellet thermal expansion, fuel swelling, and fission gas release rates. Radial power distribution in UO_2 fuel rods is determined with the neutron transport theory code, LASER. The LASER code has been validated by comparing the code predictions on radial burnup and isotopic distributions with measured radial microdrill data.[57][58] A radial power depression factor, f, is determined using radial power distributions predicted by LASER. The factor f enters into the determination of the pellet centerline temperature, T, relative to the pellet surface temperature, T, through the expression:

$$\int_{T_s}^{T_c} K(T)\, dT = \frac{q' f}{4\pi}$$

where:

K(T)	=	the thermal conductivity for UO_2 with a uniform density distribution.
q'	=	the linear power generation rate.

4.4.2.11.3 Gap Conductance

The temperature drop across the pellet-clad gap is a function of the gap size and the thermal conductivity of the gas in the gap. The gap conductance model is selected such that when combined with the UO_2 thermal conductivity model, the calculated fuel centerline temperatures

reflect the in-pile temperature measurements. A more detailed discussion of the gap conductance model is presented in references 96 and 99.

4.4.2.11.4 Surface Heat Transfer Coefficients

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in paragraph 4.4.2.7.1.

4.4.2.11.5 Fuel Clad Temperatures

The outer surface of the fuel rod at the hotspot operates at a temperature of approximately 660°F for steady-state operation at rated power throughout core life due to the onset of nucleate boiling. Initially (beginning of life (BOL)), this temperature is that of the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of clad temperature.

4.4.2.11.6 Treatment of Peaking Factors

The total heat flux hot channel factor, FQ is defined by the ratio of the maximum to core average heat flux. The design value of FQ as discussed in paragraph 4.3.2.2.6 is 2.50 for normal operation. This results in a peak linear power of 14.47 kW/ft at full-power conditions.

As described in paragraph 4.3.2.2.6, the peak linear power resulting from overpower transients/operator errors [assuming a maximum overpower of 120 percent] does not exceed 22.4 kW/ft. The centerline fuel temperature must be below the UO_2 melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is discussed in paragraph 4.4.1.2 and results in a maximum allowable calculated centerline temperature of 4700°F. The peak linear power for prevention of centerline melt is 22.4 kW/ft for VANTAGE 5 fuel and 22.5 kW/ft for LOPAR fuel. The centerline temperature at the peak linear power resulting from overpower transients/operator errors [assuming a maximum overpower of 120 percent] is below that required to produce melting.

4.4.3 DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE RCS

4.4.3.1 <u>Plant Configuration Data</u>

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided as appropriate in chapters 5, 6, and 9. Implementation of the ECCS is discussed in chapter 15. Some specific areas of interest are the following:

A. Total coolant flowrates for the RCS and each loop are provided in table 5.1.2-1. Flowrates employed in the evaluation of the core are presented throughout section 4.4.

B. Total RCS volume including pressurizer and surge line and RCS liquid volume including pressurizer water at steady-state power conditions are given in table 5.1.2-1.

C. The flow path length through each volume may be calculated from physical data provided in the above-referenced table.

D. The height of fluid in each component of the RCS may be determined from the physical data presented in section 5.4. The components of the RCS are water filled during power operation with the pressurizer being approximately 60-percent water filled.

E. Components of the ECCS are to be located so as to meet the criteria for net positive suction head (NPSH) described in section 6.3.

F. Line lengths and sizes for the safety injection system (SIS) are determined so as to guarantee a total system resistance which will provide, as a minimum, the fluid delivery rates assumed in the safety analyses described in chapter 15.

G. The parameters for components of the RCS are presented in section 5.4.

H. The steady-state pressure drops and temperature distributions through the RCS are presented in table 5.1.2-1.

4.4.3.2 Operating Restrictions on Pumps

The minimum NPSH and minimum seal injection flowrate must be established before operating the reactor coolant pumps. With the minimum 6-gal/min labyrinth seal injection flowrate established, the operator will have to verify that the system pressure satisfies NPSH requirements.

4.4.3.3 Power-Flow Operating Map (Boiling Water Reactor (BWR))

This paragraph is not applicable to VEGP.

4.4.3.4 Temperature-Power Operating Map (PWR)

The relationship between RCS temperature and power is shown in figure 4.4-8.

The effects of reduced core flow due to inoperative pumps is discussed in subsections 5.4.1 and 15.2.6 and section 15.3. Natural circulation capability of the system is discussed in paragraph 5.4.2.3.2.

4.4.3.5 Load Following Characteristics

Load follow using control rod motion and dilution or boration by the boron system is discussed in paragraph 4.3.2.4.16. The RCS is designed on the basis of steady-state operation at full-power heat load. The reactor coolant pumps utilize constant speed drives as described in section 5.4, and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in section 7.7.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal and hydraulic characteristics are given in tables 4.1-1, 4.4-1, and 4.4-2.

4.4.4 EVALUATION

4.4.4.1 Critical Heat Flux

The critical heat flux correlation utilized in the core thermal analysis is explained in detail in subsection 4.4.2.

4.4.4.2 Core Hydraulics

4.4.4.2.1 Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths for core bypass flow are considered:

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

B. Flow entering into the RCC guide thimbles.

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

D. Flow introduced between the baffle and the barrel for the purpose of cooling these components and not considered available for core cooling.

E. Flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall.

The above contributions are evaluated to confirm that the design value of the core bypass flow is met. The design value of core bypass flow for the VEGP is equal to 6.4 percent of the total vessel flow.

Of the total allowance, 2.1 percent is associated with the core and the remainder is associated with the internals (items A, C, D, and E above). Calculations have been performed using

drawing tolerances in the worst direction and accounting for uncertainties in pressure losses. Based on these calculations, the core bypass is no greater than the design value quoted above.

Flow model test results for the flow path through the reactor are discussed in paragraph 4.4.2.7.2.

4.4.4.2.2 Inlet Flow Distributions

Data has been considered from several one-seventh scale hydraulic reactor model tests[22][23][61] in arriving at the core inlet flow maldistribution criteria to be used in the VIPRE-01 analyses (paragraph 4.4.4.5). 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.[62] The same design basis of 5-percent reduction to the hot assembly inlet is used in VIPRE-01 analyses.

The experimental error estimated in the inlet velocity distribution has been considered as outlined in reference 17, where the sensitivity of changes in inlet velocity distributions to hot channel thermal performance is shown to be small. Studies[17] made with THINC-IV show that it is adequate to use the 5-percent reduction in inlet flow to the hot assembly for a loop out of service, based on the experimental data in references 22 and 23. VIPRE-01 was demonstrated to be equivalent to THINC-IV in reference 101.

The effect of the total flowrate on the inlet velocity distribution was studied in the experiments of reference 22. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flowrate.

4.4.4.2.3 Empirical Friction Factor Correlations

Empirical friction factor correlations are used in the VIPRE-01 code (described in paragraph 4.4.4.5).

The friction factor in the axial direction, parallel to the fuel rod axis, is evaluated using a correlation for the smooth tube [101]. The effect of two-phase flow on the friction loss is expressed in terms of the single-phase friction pressure drop and a two-phase friction multiplier. The multiplier is calculated directly using the homogeneous equilibrium flow model.

The flow in the lateral directions, normal to the fuel rod axis, views the reactor core as a large tube bank. Thus, the lateral friction factor proposed by Idel'chik[24] is applicable. This correlation is of the form:

$$F_L = A\,Re_L^{-0.2} \qquad (11)$$

where:

A = a function of the rod pitch and diameter as given in reference 24.

Re_L = the lateral Reynolds number based on the rod diameter.

Extensive comparisons of VIPRE-01 predictions using these correlations to THINC-IV predictions are given in reference 101; they verify the applicability of these correlations in PWR design.

4.4.4.3 Influence of Power Distribution

The core power distribution, which is largely established at BOL by fuel enrichment, loading pattern, and core power level, is also a function of variables such as control rod worth and position, and fuel depletion through lifetime. Radial power distributions in various planes of the core are often illustrated for general interest; however, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater importance for DNB analyses. These radial power distributions, characterized by $F_{\Delta H}^{N}$ (defined in paragraph 4.3.2.2.1), as well as axial heat flux profiles are discussed in the following two paragraphs.

4.4.4.3.1 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{N}$

Given the local power density q' (kW/ft) at a point x, y, z in a core with N fuel rods and height H,

$$F_{\Delta H}^{N} = \frac{\text{hot rod power}}{\text{average rod power}} = \frac{\text{MAX} \int_{O}^{H} q'(x_o, y_o, z_o)\, dz}{\frac{1}{N} \sum_{\substack{\text{all} \\ \text{rods}}} \int_{O}^{H} q'\,(x, y, z)\, dz} \tag{12}$$

The way in which $F_{\Delta H}^{N}$ is used in the DNB calculation is important. The location of minimum DNBR depends on the axial profile, and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod integral is used to identify the most likely rod for minimum DNBR. An axial power profile is obtained which, when normalized to the design value of $F_{\Delta H}^{N}$, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical distributions found in hot assemblies. In this manner, worst-case axial profiles can be combined with worst-case radial distributions for reference DNB calculations.

It should be noted again that $F_{\Delta H}^{N}$ is an integral and is used as such in DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal power shapes throughout the core. The sensitivity of the DNB calculations to radial power shapes is discussed in reference 17. The VIPRE-01 analyses were based on the design radial power distributions discussed in reference 17.

For operation at a fraction of full power, the design $F_{\Delta H}^{N}$ used is given by:

$$F_{\Delta H}^{N} = F_{\Delta H}^{RTD} \left[1 + PF_{\Delta H}\left(1 - P\right)\right] \tag{13}$$

$F_{\Delta H}^{RTP}$ is the limit at rated thermal power (RTP) specified in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the power factor multiplier for $F_{\Delta H}^{N}$ specified in the COLR.

P is the fraction of rated thermal power.

The permitted relaxation of $F_{\Delta H}^{N}$ is included in the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits,[65] thus allowing greater flexibility in the nuclear design.

4.4.4.3.2 Axial Heat Flux Distributions

As discussed in paragraph 4.3.2.2, the axial heat flux distribution can vary as a result of rod motion or power change or as a result of a spatial xenon transient which may occur in the axial direction. Consequently, it is necessary to measure the axial power imbalance by means of the excore nuclear detectors (as discussed in paragraph 4.3.2.2.7) and to protect the core from excessive axial power imbalance. The reference axial shape used in establishing core DNB limits (that is, overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.55. The reactor trip system provides automatic reduction of the trip setpoints on excessive axial power imbalance. To determine the magnitude of the setpoint reduction, the reference shape is supplemented by other axial shapes skewed to the bottom and top of the core.

The course of those accidents in which DNB is a concern is analyzed in chapter 15 assuming that the protection setpoints have been set on the basis of these shapes. In many cases, the axial power distribution in the hot channel changes throughout the course of the accident due to rod motion, coolant temperature, and power level changes.

The initial conditions for the accidents for which DNB protection is required are assumed to be those permissible within the specified axial offset control limits described in paragraph 4.3.2.2. In the case of the loss-of-flow accident, the hot channel heat flux profile is very similar to the power density profile in normal operation preceding the accident. It is therefore possible to illustrate the calculated minimum DNBR for conditions representative of the loss-of-flow accident as a function of the flux difference initially in the core. A typical plot of this type is provided in figure 4.4-9. As noted on this figure, all power shapes were evaluated with a full-power radial peaking factor $F_{\Delta H}^{N}$ of 1.55. The radial contribution to the hot rod power shape is conservative both for the initial condition and for the condition at the time of minimum DNBR during the loss of flow transient. Also shown is the minimum DNBR calculated for the design power shape for nonoverpower/overtemperature DNB events. It can be seen that this design shape results in calculated DNBR that bounds all the normal operation shapes.

4.4.4.4 Core Thermal Response

A general summary of the steady-state thermal-hydraulic design parameters including thermal output, flowrates, etc., is provided in table 4.4-1.

As stated in subsection 4.4.1, the design bases of the application are to prevent DNB and to prevent fuel melting for Condition 1 and 2 events. The protective systems described in chapter 7 are designed to meet these bases. The response of the core to Condition 2 transients is given in chapter 15.

4.4.4.5 Analytical Methods

4.4.4.5.1 Core Analysis

The objective of reactor core thermal design is to determine the maximum heat-removal capability in all flow subchannels and to show that the core safety limits are not exceeded using the most conservative power distribution. The thermal design takes into account local variations in dimensions, power generation, flow redistribution, and mixing.

Prior to the power uprate to 3626 MWt, the THINC-IV code [(17) (48) (67) (98)] was used for the core thermal design. Commencing with the power uprate to 3626 MWt, the VIPRE-01 code is used for the core thermal design. VIPRE-01 is a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels [(100)]. VIPRE-01 modeling of a PWR core is based on one-pass modeling approach. [(101)] In the one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow, and pressure drop.

4.4.4.5.2 Steady-State Analysis

The VIPRE-01 core model as approved by the NRC, reference 101, is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions. The VIPRE-01 code is described in detail in reference 100, including discussions on code validation with experimental data. The VIPRE-01 modeling method is described in reference101, including empirical models and correlations used. The effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the VIPRE-01 evaluations. However, conservative treatment by the VIPRE-01 modeling method has been demonstrated to bound this effect in DNBR calculations [(101)].

The VIPRE-01 model has been demonstrated in reference 101 to be equivalent to the THINC-IV code. The DNBR limits for the DNB correlations in paragraph 4.4.2.2.1 remain unchanged with the VIPRE-01 model [(101)].

Estimates of uncertainties are discussed in paragraph 4.4.2.9.

4.4.4.5.3 Experimental Verification

Extensive additional experimental verification of VIPRE-01 is presented in reference 100.

The VIPRE-01 analysis is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel

elements. The use of the VIPRE-01 analysis provides a realistic evaluation of the core performance and is used in the thermal analyses as described above.

4.4.4.5.4 Transient Analysis

VIPRE-01 is capable of transient DNB analysis. The conservation equations in the VIPRE-01 code contain the necessary accumulation terms for transient calculations. The input description can include one or more of the following time dependent arrays:

1. Inlet flow variation,

2. Core heat flux variation,

3. Core pressure variation,

4. Inlet temperature or enthalpy variation.

At the beginning of the transient, the calculation procedure is carried out as in the steady-state analysis. The time is incremented by an amount determined either by the user or by the time step control options in the code itself. At each new time step the calculations are carried out with the addition of the accumulation terms which are evaluated using the information from the previous time step. This procedure is continued until a preset maximum time is reached.

At time intervals selected by the user, a complete description of the coolant parameter distributions as well as DNBR is printed out. In this manner the variation of any parameter with time can be readily determined.

The methods for evaluating fuel rod thermal response are described in subsection 15.0.11.

4.4.4.6 Hydrodynamic and Flow Power Coupled Instability

Boiling flow may be susceptible to thermo hydrodynamic instabilities.[68] These instabilities are undesirable in reactors, since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design criterion was developed which states that modes of operation under Condition 1 and 2 events shall not lead to thermo hydrodynamic instabilities.

Two specific types of flow instabilities are considered for VEGP operation. These are the Ledinegg (or flow excursion) type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flowrate from one steady state to another. This instability occurs[68] when the slope of the RCS pressure drop-flowrate curve:

$$\left(\left. \frac{\partial \Delta P}{\partial G} \right|_{\text{internal}} \right)$$

becomes algebraically smaller than the loop supply (pump head) pressure drop-flowrate curve:

$$\left(\frac{\partial \Delta P}{\partial G} \bigg|_{external} \right)$$

The criterion for stability is thus:

$$\frac{\partial \Delta P}{\partial G} \bigg|_{internal} \geq \frac{\partial \Delta P}{\partial G} \bigg|_{external}$$

The W pump head curve has a negative slope ($\partial \Delta P / \partial G$ external less than 0), whereas the RCS pressure drop-flow curve has a positive slope ($\partial \Delta P / \partial G$ internal greater than 0) over the Condition 1 and Condition 2 operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody.(69) Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region and causes quality or void perturbations in the two-phase regions which travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii(70) for parallel closed-channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor designs(71)(72)(73) under Conditions 1 and 2 operation. The results indicate that a large margin to density wave instability exists; e.g., increases on the order of 150% of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii(70) to Westinghouse reactor designs is conservative due to the parallel open-channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high-power density. There is also energy transfer from channels of high-power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open-channel configuration is more stable than the above closed-channel analysis under the same boundary conditions. Flow stability tests(74) have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross-connected at several locations. The cross-connections were such that the resistance to channel cross-flow and enthalpy perturbations would be greater than that which would exist in a PWR core which has a relatively low resistance to cross-flow.

Flow instabilities which have been observed have occurred almost exclusively in closed-channel systems operating at low pressures relative to the Westinghouse PWR operating pressures. Kao, Morgan, and Parker(75) analyzed parallel closed-channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many Westinghouse rod bundles have been tested

over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermo hydrodynamic instabilities will not occur under Condition 1 and 2 modes of operation for Westinghouse PWR reactor designs. A large power margin exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant-to-plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power-to-flow ratios, fuel assembly length, etc., will not result in gross deterioration of the above power margins.

4.4.4.7 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior are more pronounced than external blockages of the same magnitude. In both cases the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV or VIPRE-01 codes. Inspection of the DNB correlations (paragraph 4.4.2.2 and references 4, 84, 85, 86) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The VIPRE-01 code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. In reference 100, it is shown that for a fuel assembly similar to the Westinghouse design, VIPRE-01 accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 in. downstream of the blockage. With the reactor operating at the nominal full-power conditions specified in table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching the DNBR limit.

From a review of the open literature, it is concluded that flow blockage in open-lattice cores similar to the Westinghouse cores causes flow perturbations which are local to the blockage. For instance, Ohtsubo, et al.,[76] show that the mean bundle velocity is approached asymptomatically about 4 in. downstream from a flow blockage in a single flow cell. Similar results were also found for two and three cells completely blocked. Basmer, et al.,[78] tested an open-lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 in. for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence, the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. The reduction in local mass velocity is then the main parameter which affects the DNBR. If the plants were operating at full power and nominal steady-state conditions as specified in table 4.4-1, a reduction in local mass velocity greater than 60 percent in the VANTAGE + / VANTAGE 5 fuel would be required to reduce the DNBR to the DNBR limit. The above mass velocity effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality a local flow blockage is expected to promote turbulence and thus would not likely affect DNBR at all.

Coolant flow blockages induce local cross-flows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high cross-flow component. Fuel rod vibration could occur, caused by this cross-flow component, through vortex shedding or turbulent mechanisms. If the cross- flow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The cross-flow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the cross-flow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Cross-flow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow-induced vibration is considered in the fuel rod fretting evaluation (section 4.2).

4.4.5 TESTING AND VERIFICATION

4.4.5.1 Tests Prior to Initial Criticality

A reactor coolant flow test is performed following fuel loading but prior to initial criticality. Coolant loop elbow tap pressure data is obtained in this test. This data allows determination of the coolant flowrates at reactor operating conditions. This test verifies that proper coolant flowrates have been used in the core thermal and hydraulic analysis.

4.4.5.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels (chapter 14). These tests are used to ensure that conservative peaking factors are used in the core thermal and hydraulic analysis.

Additional demonstration of the overall conservatism of the THINC analysis was obtained by comparing THINC predictions to incore thermocouple measurements.[77] These measurements were performed on the Zion reactor. No further inreactor testing is planned. In reference 101, the VIPRE-01 code was confirmed to be as conservative as the THINC code.

4.4.5.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are described in subsection 4.2.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors in the design analyses (paragraph 4.4.2.2.4) are met.

4.4.6 INSTRUMENTATION REQUIREMENTS

4.4.6.1 Incore Instrumentation

The incore instrumentation system consists of chromel-alumel thermocouples at fixed core outlet positions and movable miniature neutron detectors (fission chambers) at selected fuel assemblies. The thermocouples are monitored by the plant safety monitoring system. The movable detectors can perform flux mapping at various core quadrants to obtain a flux map for any region of the core.

The thermocouples measure coolant outlet temperatures at preselected positions, and the fission chamber detectors positioned in guide thimbles which run the length of selected fuel assemblies measure the neutron flux distribution. Figure 4.4-10 shows the location of instrumented assemblies in the core. Provisions are made for 50 core exit thermocouples, divided into 2 trains of 25 each. Adequate provision has been made for inoperable or failed thermocouples because the number required for post accident monitoring functions described in subsections 7.5.3 and 7.5.4 are much less than 25 per channel. Information from the core exit thermocouples may be used to supplement information from the movable incore detectors, but it is not required for conducting flux maps and may not substitute for flux maps.

The incore instrumentation system is described in more detail in paragraph 7.7.1.9.

The movable incore detectors obtain data for the determination of incore fission power density distribution, coolant enthalpy distribution, and fuel burnup distribution. The core exit thermocouples provide input into the plant safety monitoring system (as described in appendix 4A) for indication of inadequate core cooling and core subcooling margin monitoring. The alternate shutdown panel also utilizes a core exit temperature signal from the core quadrants associated with loops 2 and 3.

The monitoring of core exit temperature following an accident is described in subsections 7.5.2 and 7.5.3. Core exit temperature is a category 1 parameter for monitoring types A, B, and C variables as indicated on table 7.5.4-1. This function is met by the two per quadrant per train requirement and reflected in the Technical Specifications. In accordance with NUREG-0737, the two thermocouples per train must be located such that they indicate the radial temperature gradient.

The use of core exit thermocouples as a backup to hot leg temperature measurement can be achieved by 16 thermocouples per train equally distributed across the core. Since no credit is taken for this function, table 7.5.2-1 indicates a need for eight thermocouples per train and two per quadrant.

4.4.6.2 Overtemperature and Overpower ΔT Instrumentation

The overtemperature ΔT trip protects the core against low DNBR. The overpower ΔT trip protects against excessive power (fuel rod rating protection).

As discussed in paragraph 7.2.1.1.2, factors included in establishing the overtemperature ΔT and overpower ΔT trip setpoints include the reactor coolant temperature in each loop and the axial distribution of core power through use of the two section excore neutron detectors.

4.4.6.3 Instrumentation to Limit Maximum Power Output

The output of the three ranges (source, intermediate, and power) of detectors, with the electronics of the nuclear instruments, is used to limit the maximum power output of the reactor within their respective ranges.

There are six radial locations containing a total of eight neutron flux detectors installed around the reactor in the primary shield. Fission chambers used for both source and intermediate range are installed on opposite "flat" portions of the core containing the startup sources at an elevation one-half of the core height. All four fission chambers are used for the source range. Two are used for intermediate range. Four dual-section uncompensated ionization chamber assemblies for the power range are installed vertically at the four corners of the core and located equidistant from the reactor vessel at all points and, to minimize neutron flux pattern distortions, within 1 ft of the reactor vessel. Each power range detector provides two signals corresponding to the neutron flux in the upper and in the lower sections of a core quadrant. The three ranges of detectors are used as inputs to monitor neutron flux from a completely shutdown condition to 120% of full power.

The output of the power range channels is used for:

A. The rod speed control function.

B. Alerting the operator to an excessive power unbalance between the quadrants.

C. Protecting the core against the consequences of rod ejection accidents.

D. Protecting the core against the consequences of adverse power distributions resulting from dropped rods.

Details of the neutron detectors and nuclear instrumentation design and the control and trip logic are given in chapter 7. The limits on neutron flux operation and trip setpoints are given in the Technical Specifications.

4.4.6.4 Digital Metal Impact Monitoring System *(DMIMS-DX™)*

General System Description

The metal impact monitoring system *(DMIMS-DX™)* is designed to detect loose parts in the reactor coolant system. The system consists of sensors, preamplifiers, signal conditioners, signal processors, and a display. It contains 12 active instrument channels, each comprised of a piezoelectric accelerometer (sensor), signal conditioning and diagnostic equipment. Conformance with Regulatory Guide 1.133, Revision 1, is discussed in paragraph 1.9.133.2.

Two redundant sensors are fastened mechanically to the RCS at each of the following potential loose parts collection regions:

- Reactor pressure vessel: upper head region
- Reactor pressure vessel: lower head region
- Each steam generator: reactor coolant inlet region

The output signal from each accelerometer is passed through a preamplifier and an amplifier. The amplified signal is processed through a discriminator to eliminate noises and signals that are not indicative of loose parts. The processed signal is compared to a preset alarm setpoint. Loose parts detection is accomplished at a frequency of 1 kHz to 20 kHz, where background signals from the RCS are acceptable. Spurious alarming from control rod stepping is prevented by a module that detects CRDM motion commands and automatically inhibits alarms during control rod stepping.

If measured impact signals exceed the preset alarm level, audible and visible alarms in the control room are activated. Digital signal processors record the times that the first and subsequent impact signals reach various sensors. This timing information provides a basis for locating the loose part. The *DMIMS-DX™* also has a provision for audio monitoring of any channel. The audio signal can be compared to a previously recorded audio signal, if desired.

The online sensitivity of the *DMIMS-DX™* is such that the system will detect a loose part that weighs from 0.25 to 30 lb and impacts with a kinetic energy of 0.5 ft-lb on the inside surface of the RCS pressure boundary within 3 ft of a sensor.

The *DMIMS-DX™* audio and visual alarm capability will remain functional after an Operating Basis Earthquake (OBE). All of the *DMIMS-DX™* components are qualified for structural integrity during a Safe Shutdown Earthquake (SSE) and will not mechanically impact any safety-related equipment. In addition, the equipment inside containment is designed to remain functional through normal radiation exposures anticipated during a 40-year operating lifetime[a]. Physical separation of the two instrument channels, associated with the redundant sensors at each reactor coolant system location, exists from each sensor to the output of the incontainment signal conditioning devices. The incontainment signal conditioning devices are accessible during power operation. The *DMIMS-DX™* components outside containment are located in a mild environment. Capabilities exist for subsequent periodic online channel checks and channel functional tests and for offline channel calibrations at refueling outages. Figure 4.4-11 shows a block diagram of the loose parts monitoring system.

<u>Key Features, Components and Architecture</u>

Key features of system components and architecture are discussed in the following sections.

Sensors

The sensors are piezoelectric accelerometers that convert acceleration to electric charge. The acoustic waves created by an impacting metallic object can be detected by the piezoelectric accelerometers. While the excitation of the impact produces a very wideband frequency response, the frequency range of interest for most loose parts is 1 kHz to 20 kHz.

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

Piezoelectric accelerometers are high output impedance devices that convert acceleration to electric charge. The flat frequency response range for the accelerometers used in *DMIMS-DX™* is from 5 Hz to 10 kHz, and they have a useful frequency upper limit of over 20 kHz. The resonant frequency of the accelerometers is greater than 30 kHz. The accelerometers are designed to operate at high temperature (nominally 625 °F) and have high radiation capability.

The piezoelectric elements in the accelerometers are electrically isolated from the component to which they are attached in order to prevent unwanted noise due to ground loops. The accelerometers typically have an integral 4 foot mineral-insulated ("hardline") cable and a large triax connector. This hardline cable is also built to withstand high temperatures, while the connector allows for interfacing to lower temperature softline cables.

Softline Cable

Because the charge output of an accelerometer is a very low level signal, and normal cables can emit charge upon being vibrated, a special low-noise, radiation-resistant softline cable is used between the accelerometer and preamplifier.

Preamplifier

The remote preamplifier is mounted in a sealed metal enclosure inside containment. The charge signal from the accelerometer is converted to a voltage signal. The preamplifier operates in a "charge" amplifier mode such that the capacitance of the cable between the high-output-impedance accelerometer and the preamplifier has very little effect on the signal or its calibration. The charge preamplifier output voltage is then a normal, low-impedance millivolt instrument signal requiring only normal cabling and shielding considerations.

Signal Conditioner

The signal conditioner module provides power to the remote preamplifier, provides final amplification of the signal to a calibrated full scale range, and provides lowpass and highpass filtering.

Audio Subsystem

The audio patch panel, audio amplifier, and speakers make up the audio subsystem. Listening by a trained ear can be a very effective tool for evaluation and validation of signal characteristics. The system is designed such that any channel may be selected at any time for audio monitoring.

Digital Signal Processing (DSP) Processor

In the Digital Signal Processing (DSP) processor, the signals are converted from analog to digital at a high rate, and the impact detection algorithm is applied by a special microprocessor optimized for digital signal processing. The board contains a buffer memory that can store the complete impact signal time history for its monitored channels. Upon the detection of an impact, the data are normally transferred to the main Central Processing Unit (CPU) process for further evaluation, waveform storage, and alarm generation. However, if for some reason the CPU processor fails, the DSP processor has the capability for generating alarms on its own.

Central Processing Unit (CPU) Processor

The CPU processor is a personal computer architecture device. It takes the data from the DSP processors, controls the mass storage devices, provides displays of monitoring system information, drives the printer, and generates alarms. The CPU uses a PCI bus for high speed communication with the other processor modules and drives the tape and disk peripherals by means of a parallel Small Computers System Interface (SCSI) interface. Addition of the peripherals provides for mass data storage onto high speed digital tape and writeable CDs.

Display

The display is a qualified, high-resolution, color panel that is overlaid with a high-resolution touchscreen surface. The display shows the system and alarm statuses at a glance, presents the waveforms used in impact analysis, and shows the analysis conclusions. By means of the touchscreen, which has all of the capabilities of a standard mouse, many system functions can be run without opening the keyboard drawer.

Alarm Panel

The alarm panel provides continuous indication of alarm or trouble status, allowing the color display to be turned off when not being viewed. The panel contains red LEDs for alarm indication, orange LEDs for trouble indication, yellow LEDs that flash each time an impact event is detected by their respective channels, and green LEDs for indication of proper DSP operation.

Printer

A high-resolution laser printer is provided for printout of system status, waveform graphs, and other data for the generation of reports.

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TABLE 4.4-1 (SHEET 1 OF 3)

THERMAL AND HYDRAULIC COMPARISON TABLE

Design Parameters	LOPAR[m]	VANTAGE + / VANTAGE 5
Reactor core heat output (MWt)	3565[m]	3626[l]
Reactor core heat output (10^6 Btu/h)	12,164	12,372
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2250	2250
System pressure, minimum steady-state (psia)	2200	2200
Minimum DNBR at nominal conditions		
Typical flow channel	3.20	2.45
Thimble (cold wall) flow channel	3.01	2.35
Minimum DNBR for design transients		
Typical flow channel	1.23	1.24
Thimble (cold wall) flow channel	1.22	1.23
DNB correlation[a]	WRB-1	WRB-2
Coolant conditions[b]		
Vessel minimum measured flowrate(MMF)[c] 10^6 lbm/h	142.9	143.0
gpm	384,000	384,000
Vessel thermal design flowrate (TDF) 10^6 lbm/h	139.4	139.5
gpm	374,400	374,400
Effective flowrate for heat transfer (based on TDF) 10^6 lbm/h	130.5	130.6
gpm	350,440	350,440
Effective flow area for heat transfer (ft^2)[d]	51.08	54.13
Average velocity along fuel rods (ft/s)[d]	16.3	15.3
Average mass velocity (10^6 lbm/h-ft^2) (based on TDF)[d]	2.55	2.41

TABLE 4.4-1 (SHEET 2 OF 3)

Design Parameters	LOPAR[m]	VANTAGE + / VANTAGE 5
Coolant temperature		
Nominal inlet (°F)	556.8	556.3
Average rise in vessel (°F)	63.2	64.2
Average rise in core (°F)	66.9	68.0
Average in core (°F)	592.2	592.3
Average in vessel (°F)	588.4	588.4
Heat transfer		
Active heat transfer surface area (ft^2)[d]	59,742	57,505
Average heat flux (Btu/h-ft^2)[d]	198,370	209,612
Maximum heat flux for normal[d,e] operation (Btu/h-ft^2)	495,925	524,030
Average linear power (kW/ft)[f]	5.69	5.788
Peak linear power for normal operation (kW/ft)[e,f]	14.2	14.47
Peak linear power resulting from overpower transients/operator errors, assuming a maximum overpower of 120%(kW/ft)	<22.4	<22.4
Peak linear power for prevention of centerline melt (kW/ft)[h]	22.5	22.4
Power density (kW/l of core)[i]	109.2	111.1
Specific power (kW/kg uranium)[d,i]	39.0	43.2
Fuel central temperature		
Peak at peak linear power for prevention of centerline melt (°F)	4,700	4,700
Pressure drop		
Across core (psi)[d]	23.3 ± 2.3	28.6 ± 2.9[j,k]
Across vessel, including nozzle (psi)[d]	46.2 ± 4.6	48.5 ± 4.9[k]

TABLE 4.4-1 (SHEET 3 OF 3)

---

a. See paragraph 4.4.2.2.1 for the use of the W-3 correlation.

b. Flowrates are based on 10 percent steam generator tube plugging.

c. Inlet temperature = 557.0°F

d. Assumes all LOPAR or VANTAGE + / VANTAGE 5 Core.

e. Based on 2.50 F_Q peaking factor.

f. Based on densified active fuel length.

g. See paragraph 4.3.2.2.6

h. See paragraph 4.4.2.11.6.

i. Based on cold dimensions and 95 percent of theoretical density fuel.

j. Based on a best-estimate reactor flowrate of 102,100 gpm/loop and an inlet temperature of 558.8°F.

k. With thimble plugs.

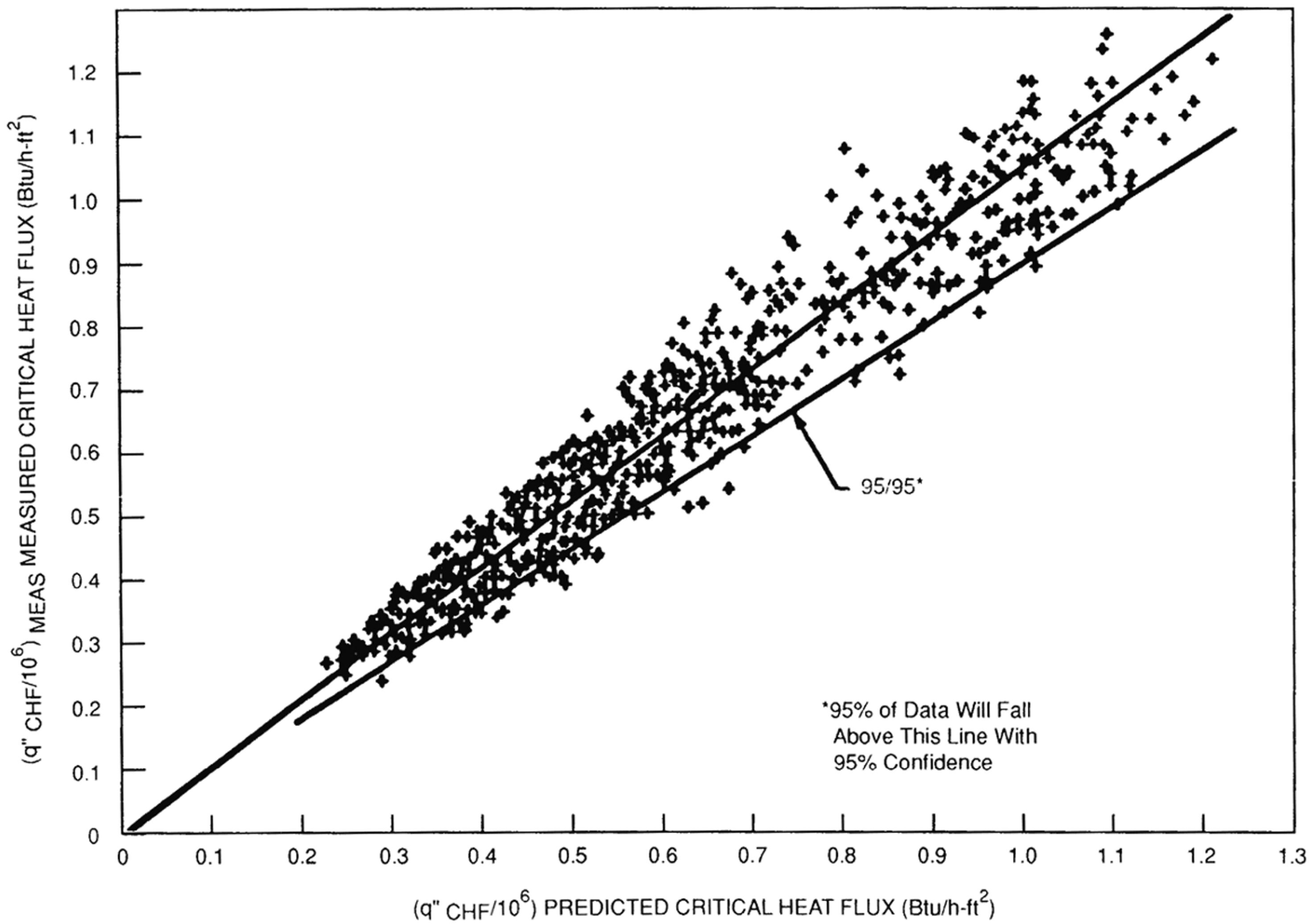
l. The VEGP MUR 1.7% power uprate increases the licensed reactor core power level from 3565 MWt to 3625.6 MWt.

m. LOPAR fuel is not analyzed for the power uprate to 3626 MWt. The LOPAR values at 3565 MWt are retained for historical purposes.

TABLE 4.4-2

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS

		Average	Maximum
Core	(VANTAGE + / VANTAGE 5)	< 0.01%	- -
Hot subchannel	(VANTAGE + / VANTAGE 5)	1.8%	7.0%



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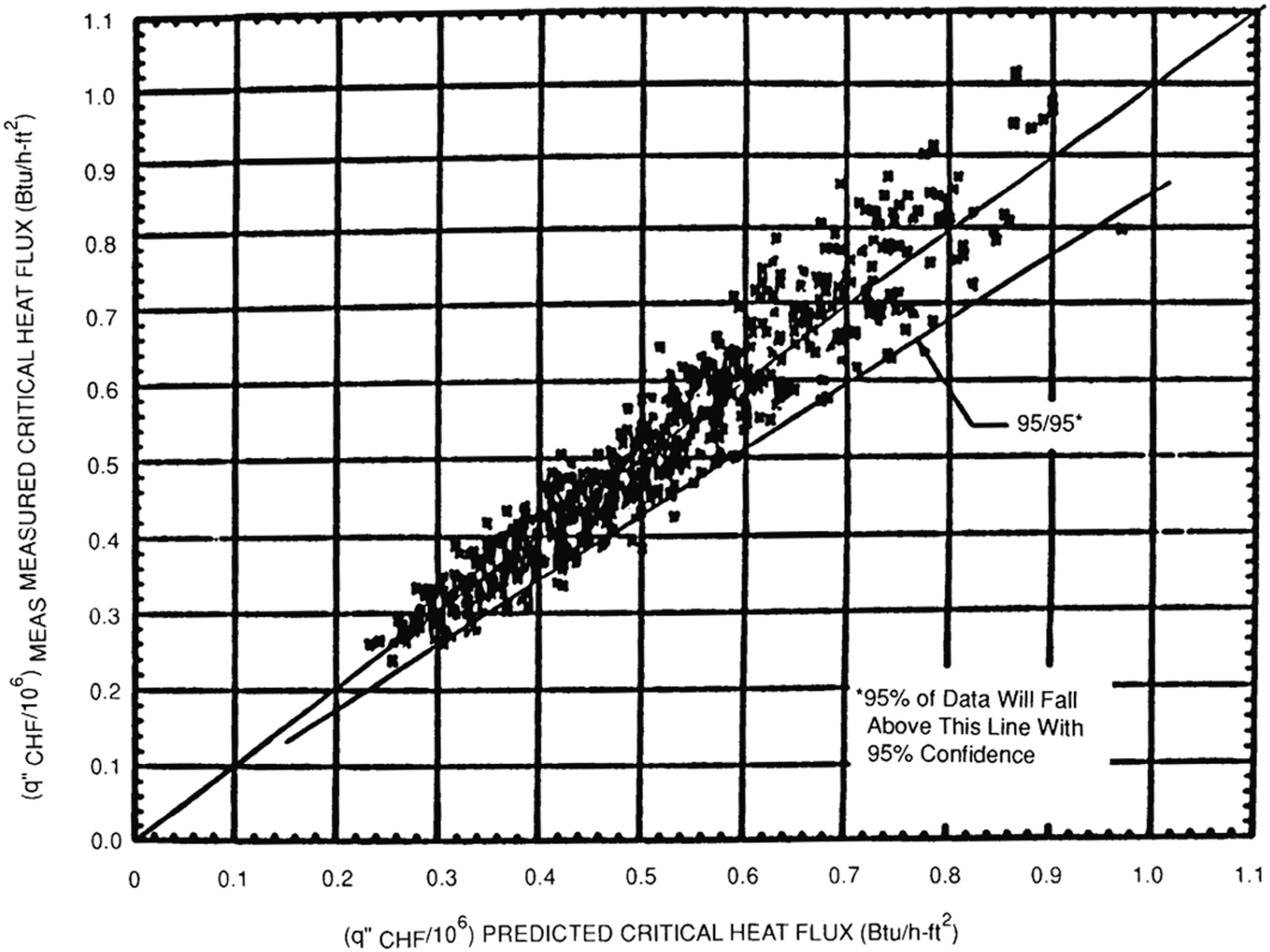
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	MEASURED VERSUS PREDICTED CRITICAL HEAT FLUX (WRB–1 CORRELATION) FIGURE 4.4–1 (SHEET 1 OF 2)



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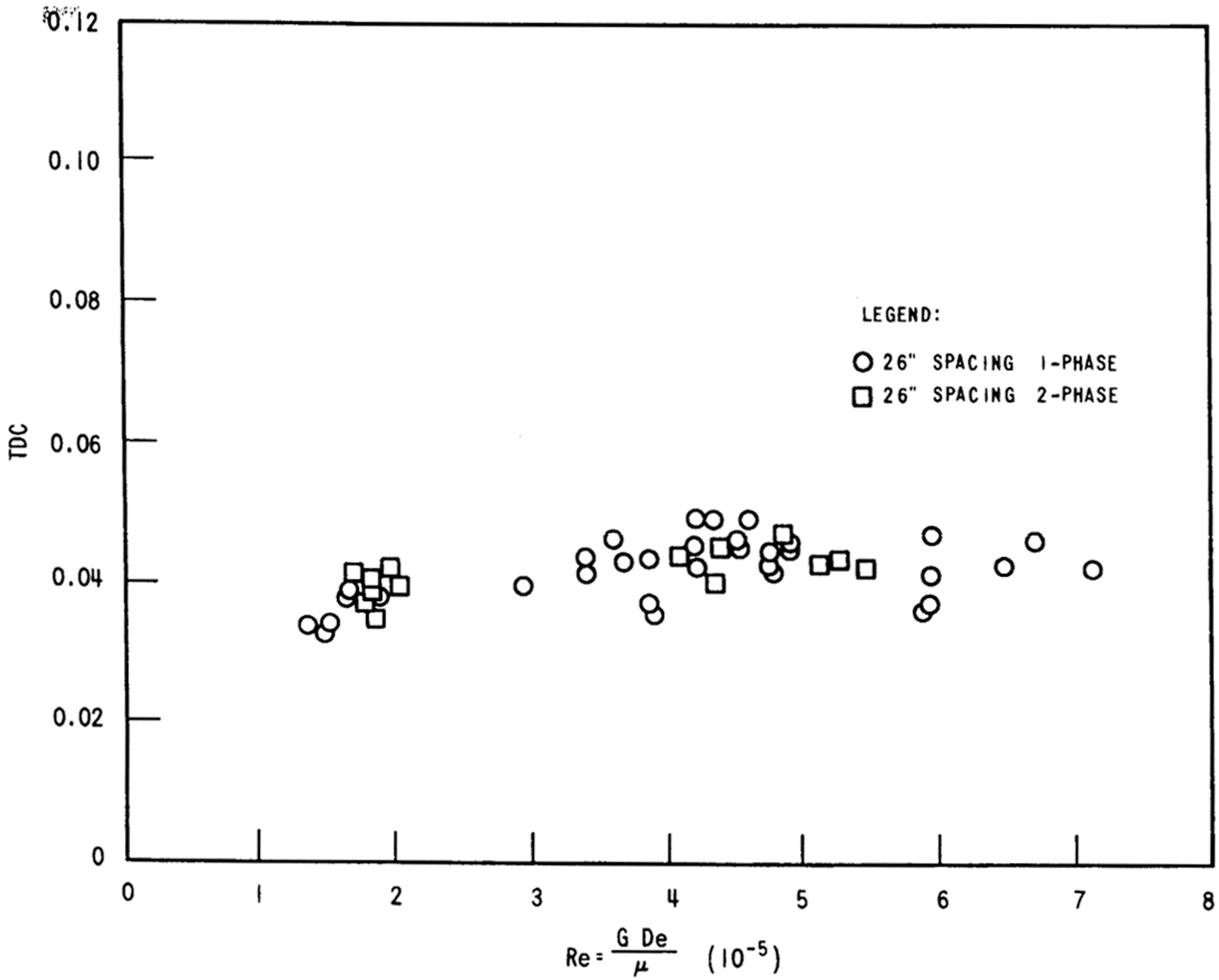
UNIT 1 AND UNIT 2

MEASURED VERSUS PREDICTED CRITICAL HEAT FLUX (WRB–2 CORRELATION)

FIGURE 4.4–1 (SHEET 2 OF 2)



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TDC VERSUS REYNOLDS NUMBER FOR
26–in. GRID SPACING

FIGURE 4.4–2

℄							
1.096 1.001							KEY: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$
1.029 1.001	1.120 1.001						
1.153 1.000	1.074 1.001	1.185 1.000					
1.166 1.000	1.209 1.000	1.162 1.001	1.185 1.000				
1.223 1.000	1.170 1.001	1.188 1.000	1.065 1.002	1.238 1.000			
1.126 1.001	1.161 1.001	1.093 1.002	1.086 1.002	0.916 1.001	0.967 1.001		
1.025 1.002	1.025 1.002	0.990 1.002	0.975 1.002	0.823 1.000	0.466 0.997		
0.717 0.999	0.780 0.999	0.664 0.998	0.563 0.997				

FOR RADIAL POWER DISTRIBUTION NEAR BEGINNING OF LIFE, HOT FULL POWER, EQUILIBRIUM XENON CALCULATED $F^N_{\Delta H} = 1.34$

REV 14 10/07

1.096 0.996							KEY: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$
1.026 1.002	1.120 0.994						
1.155 0.991	1.072 0.998	1.188 0.988					
1.169 0.990	1.212 0.986	1.165 0.990	1.189 0.998				
1.228 0.985	1.173 0.989	1.191 0.988	1.065 0.998	1.242 0.981			
1.129 0.992	1.165 0.990	1.095 0.995	1.086 0.996	0.916 1.009	0.963 1.006		
1.024 1.000	1.025 1.000	0.989 1.003	0.973 1.004	0.819 1.018	0.468 1.018		
0.715 1.022	0.777 1.019	0.663 1.021	0.562 1.019				

FOR RADIAL POWER DISTRIBUTION NEAR BEGINNING OF LIFE. HOT FULL POWER, EQUILIBRIUM XENON CALCULATED $F^N_{\Delta H}$ = 1.34

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UNIT 1 AND UNIT 2

NORMALIZED RADIAL FLOW AND ENTHALPY DISTRIBUTION AT 8–ft ELEVATION

FIGURE 4.4–4

1.097 0.995							KEY: $\Delta h/\overline{\Delta h}$ $G/\overline{G}$
1.026 0.999	1.121 0.993						
1.157 0.991	1.073 0.996	1.189 0.989					
1.170 0.990	1.215 0.980	1.166 0.991	1.190 0.990				
1.231 0.987	1.175 0.990	1.193 0.989	1.066 0.997	1.243 0.987			
1.130 0.993	1.165 0.991	1.095 0.995	1.087 0.996	0.914 1.005	0.961 1.003		
1.023 1.000	1.024 1.000	0.987 1.002	0.971 1.003	0.817 1.011	0.469 1.030		
0.711 1.016	0.774 1.013	0.660 1.019	0.560 1.025				

FOR RADIAL POWER DISTRIBUTION NEAR BEGINNING OF LIFE. HOT FULL POWER, EQUILIBRIUM XENON
CALCULATED $F^N_{\Delta H} = 1.34$

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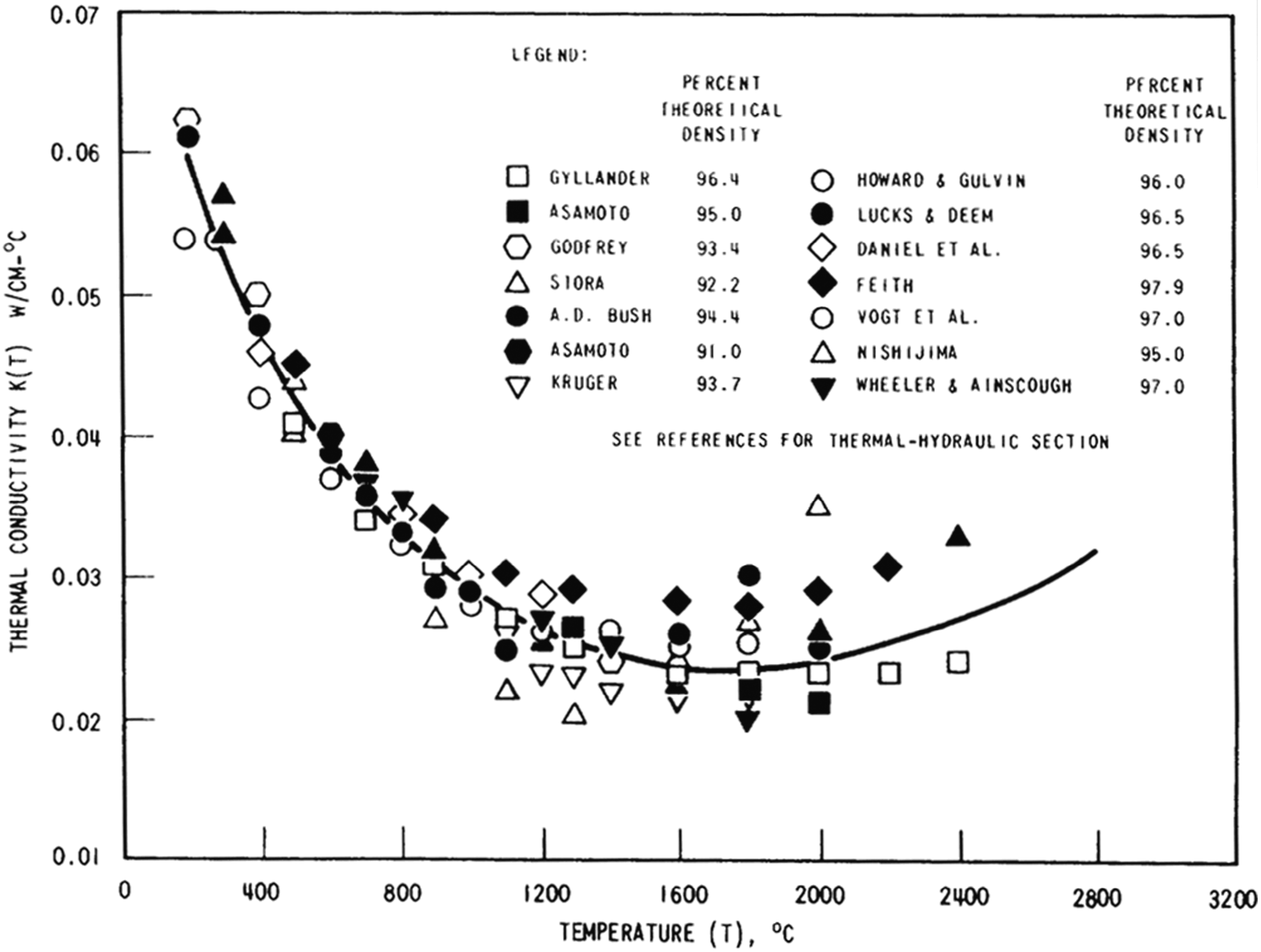
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NORMALIZED RADIAL FLOW AND ENTHALPY DISTRIBUTION AT 12–ft ELEVATION, CORE EXIT

FIGURE 4.4–5



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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	THERMAL CONDUCTIVITY OF UO_2 (DATA CORRECTED TO 95% THEORETICAL DENSITY)
	FIGURE 4.4–6

DELETED

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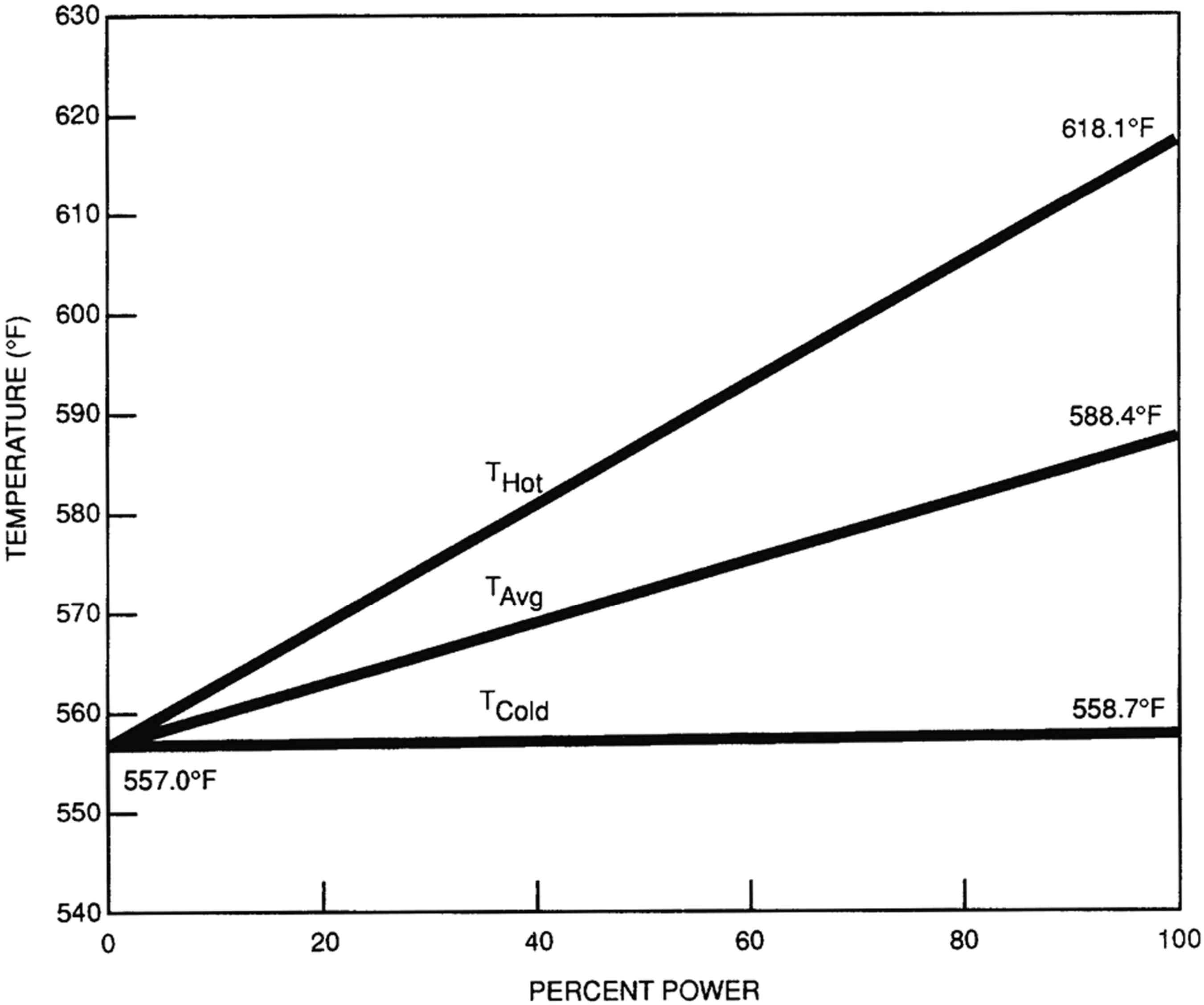
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

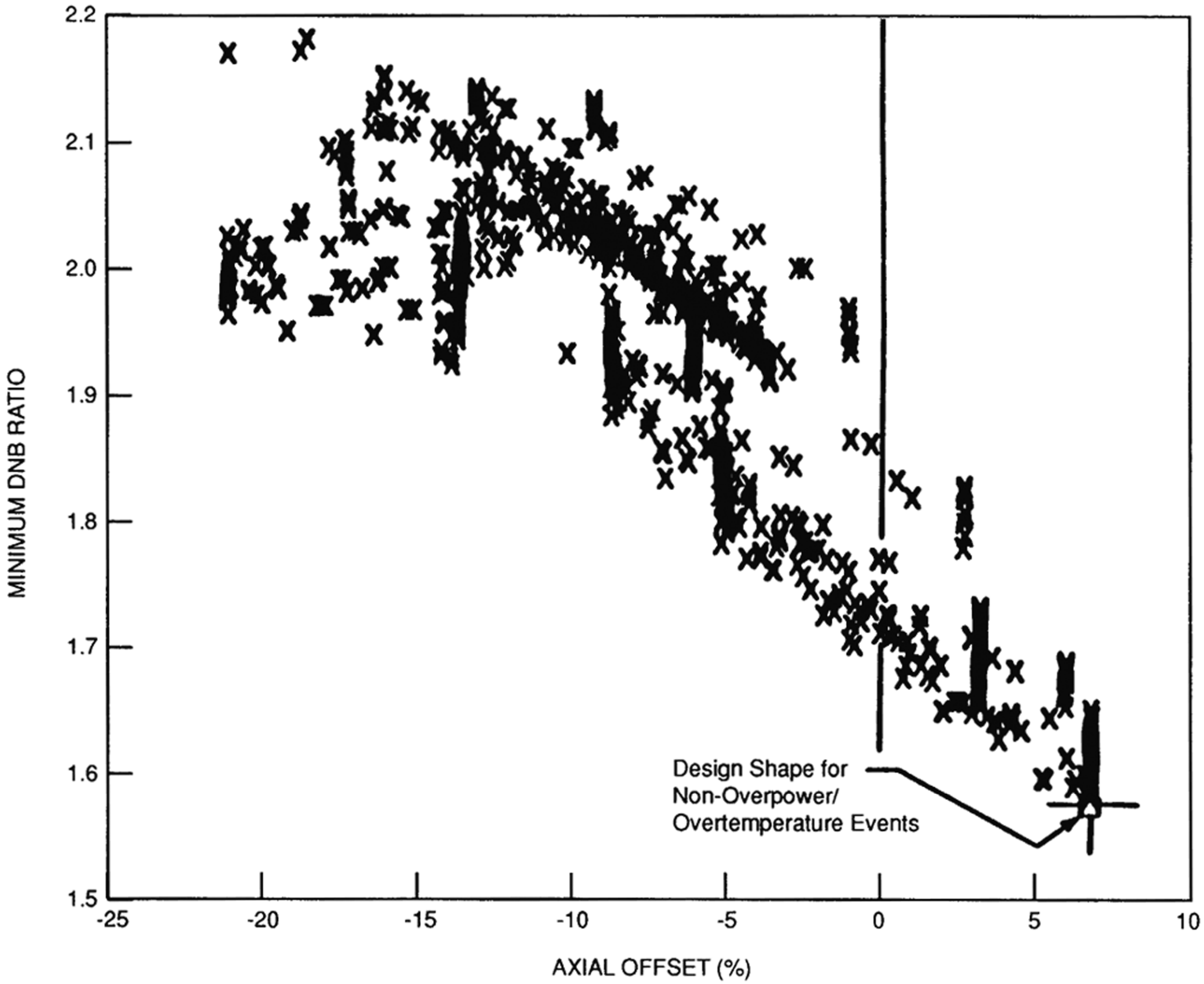
DELETED

FIGURE 4.4–7



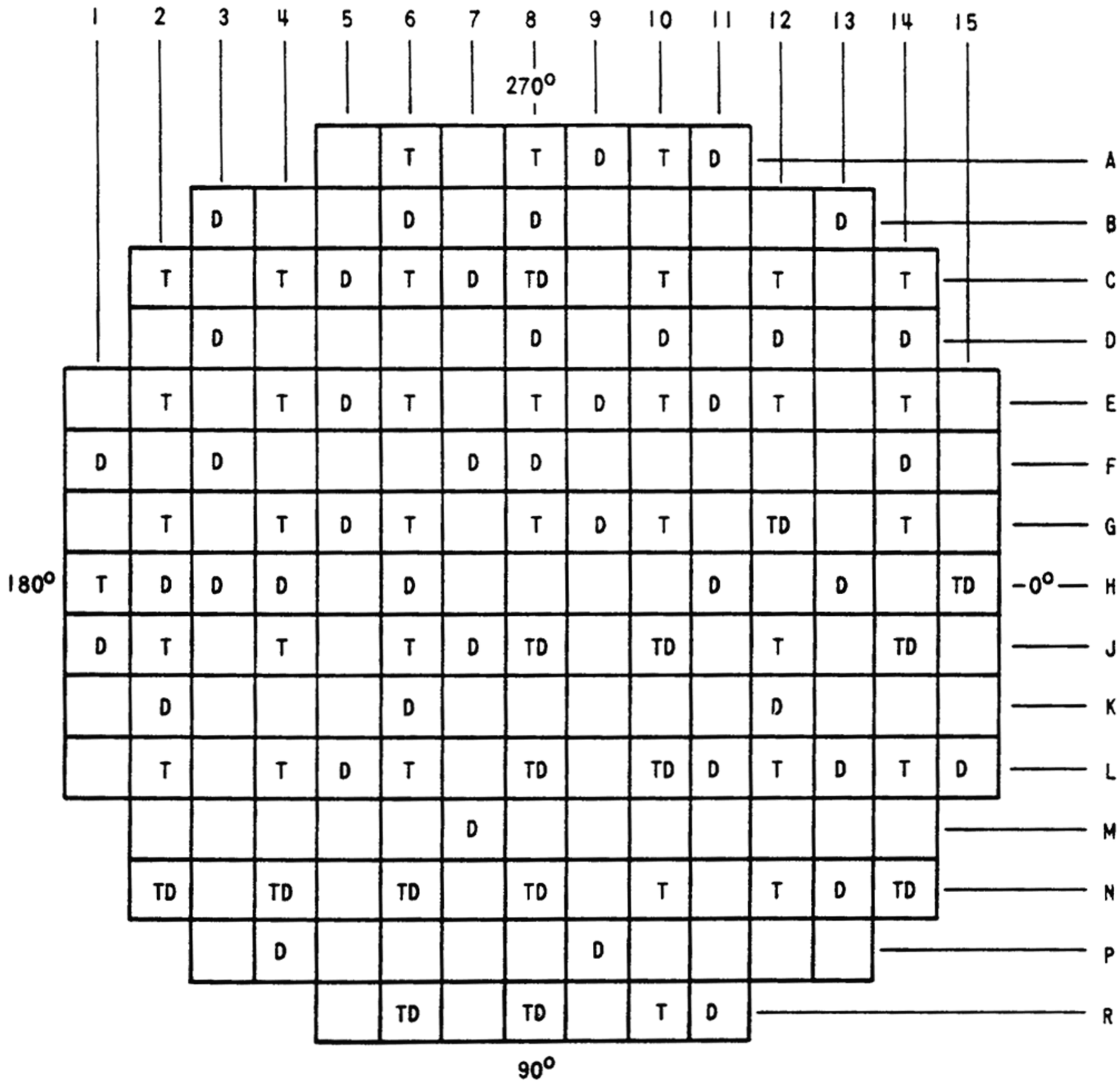
REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	RCS TEMPERATURE–PERCENT POWER MAP
		FIGURE 4.4–8



REV 15 4/09

SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	100 PERCENT POWER SHAPES EVALUATED AT CONDITIONS REPRESENTATIVE OF LOSS OF FLOW, ALL SHAPES EVALUATED WITH $F^{N}_{\Delta H} = 1.55$
	FIGURE 4.4–9

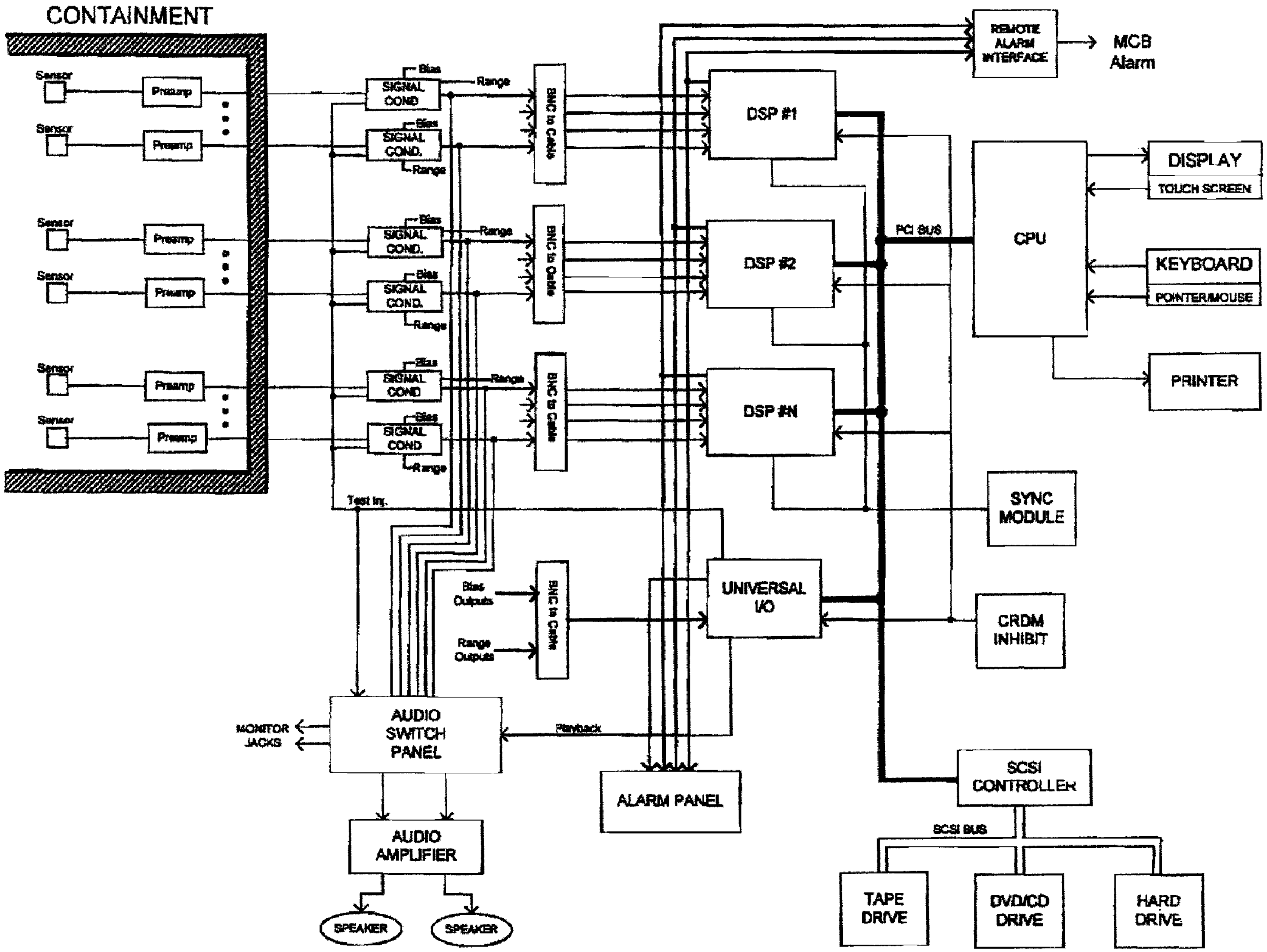


T = THERMOCOUPLE LOCATIONS

D = MOVABLE INCORE DETECTOR LOCATIONS

REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	DISTRIBUTION OF INCORE INSTRUMENTATION FIGURE 4.4–10



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOOSE PARTS MONITORING SYSTEM BLOCK DIAGRAM

FIGURE 4.4-11

4.5 REACTOR MATERIALS

4.5.1 CONTROL ROD DRIVE SYSTEM STRUCTURAL MATERIALS

4.5.1.1 Materials Specifications

All parts exposed to reactor coolant are made of metals which resist the corrosive action of the water. Three types of metals are used exclusively: stainless steels, nickel-chromium-iron, and cobalt-based alloys. In the case of stainless steels, only austenitic and martensitic stainless steels are used. For pressure boundary parts, martensitic stainless steels are not used in the heat-treated conditions which cause susceptibility to stress-corrosion cracking or accelerated corrosion in Westinghouse pressurized water reactor chemistry. Pressure boundary parts/components are made of type 304 or equivalent material.

Internal latch assembly parts are fabricated of heat-treated martensitic stainless steel. Heat treatment is such that susceptibility to stress-corrosion cracking is not initiated.

A. Pressure Vessel

All pressure-containing materials comply with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and are fabricated from austenitic (type 304) stainless steel.

B. Coil Stack Assembly

The coil housings require a magnetic material. Both low carbon cast steel and ductile iron have been successfully tested for this application. The choice, made on the basis of cost, indicates that ductile iron will be specified on the control rod drive mechanism (CRDM). The finished housings are zinc plated or flame sprayed to provide corrosion resistance.

Coils are wound on bobbins of molded Dow Corning type 302 material, with double glass insulated copper wire. Coils are then vacuum impregnated with silicon varnish. A wrapping of mica sheet is secured to the coil outside diameter. The result is a well-insulated coil capable of sustained operation at 200°C.

C. Latch Assembly

Magnetic pole pieces are fabricated from type 410 stainless steel. All nonmagnetic parts, except pins and springs, are fabricated from type 304 stainless steel. Haynes-25 is used to fabricate link pins. Springs are made from nickel-chromium-iron alloy (Inconel-X). Latch arm tips are clad with Stellite-6 to provide improved wearability. Hard chrome plate and Stellite-6 are used selectively for bearing and wear surfaces.

D. Drive Rod Assembly

The drive rod assembly utilizes a type 410 stainless steel drive rod. The coupling is machined from type 403 stainless steel. Other parts are type 304 stainless steel with the exception of the springs, which are nickel-chromium-iron alloy; the locking button, which is Haynes-25; and the belleville washers, which are Inconel-718. Several small parts (screws and pins) are Inconel-600.

4.5.1.2 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in subsection 5.2.3, concerning the processes, inspections, and tests on austenitic stainless steel components to ensure freedom from increased susceptibility to intergranular corrosion caused by sensitization, and the discussions provided in subsection 5.2.3, concerning the control of welding of austenitic stainless steels especially control of delta ferrite, are applicable to the austenitic stainless steel pressure-housing components of the CRDM.

4.5.1.3 Contamination Protection and Cleaning of Austenitic Stainless Steel

The CRDMs are cleaned prior to delivery in accordance with the guidance of American National Standards Institute (ANSI) 45.2.1. Process specifications in packaging and shipment are discussed in subsection 5.2.3. Westinghouse personnel conduct surveillance of these operations to ensure that manufacturers and installers adhere to appropriate requirements as discussed in subsection 5.2.3.

4.5.1.4 Other Materials

Haynes-25 is used in small quantities to fabricate link pins. The material is ordered in the solution-treated, cold-worked condition. Stress-corrosion cracking has not been observed in this application over the last 15 years.

The CRDM springs are made from nickel-chromium-iron alloy (Inconel-750) ordered to MIL-S-23192 or MIL-N-24114 Class A No. 1 temper-drawn wire. Operating experience has shown that springs made of this material are not subject to stress-corrosion cracking.

4.5.2 REACTOR INTERNALS MATERIALS

4.5.2.1 Materials Specifications

All the major material for the reactor internals is type 304 stainless steel. Parts not fabricated from type 304 stainless steel include bolts and dowel pins, which were fabricated from type 316 stainless steel, and radial support key bolts, which were fabricated of Inconel-750. Radial support clevis inserts are Inconel-600, and the holddown spring is type 403 stainless steel. These materials are listed in table 5.2.3-2. There are no other materials used in the reactor internals or core support structures which are not otherwise included in the ASME Code, Section III, Appendix I.

4.5.2.2 Controls on Welding

The discussions provided in subsection 5.2.3 are applicable to the welding of reactor internals and core support components.

4.5.2.3 Nondestructive Examination of Tubular Products and Fittings

The nondestructive examination of wrought seamless tubular products and fittings is in accordance with Section III of the ASME Code.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in subsection 5.2.3 and section 1.9 verify conformance of reactor internals and core support structures with Regulatory Guide 1.44.

The discussions provided in subsection 5.2.3 and section 1.9 verify conformance of reactor internals and core support structures with Regulatory Guide 1.31.

The discussion provided in section 1.9 verifies conformance of reactor internals with Regulatory Guide 1.34.

The discussion provided in section 1.9 verifies conformance of reactor internals and core support structures with Regulatory Guide 1.71.

4.5.2.5 Contamination Protection and Cleaning of Austenitic Stainless Steel

The discussions provided in subsection 5.2.3 and section 1.9 are applicable to the reactor internals and core support structures and verify conformance with ANSI 45 specifications and Regulatory Guide 1.37.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 INFORMATION FOR CONTROL ROD DRIVE SYSTEM

The control rod drive system (CRDS) is described in paragraph 3.9.4.1. Figures 3.9.4-1 and 3.9.4-2 provide the details of the control rod drive mechanisms, and figure 4.2-8 provides the layout of the CRDS. No hydraulic system is associated with its functioning. The instrumentation and controls for the reactor trip system are described in section 7.2, and the reactor control system is described in section 7.7.

4.6.2 EVALUATIONS OF THE CRDS

The CRDS has been analyzed in detail in the failure mode and effects analysis.[1] This study and the analyses presented in chapter 15 demonstrate that the CRDS performs its intended safety function, a reactor trip, by putting the reactor in a subcritical condition when a safety system setting is reached, with any assumed credible failure of a single active component. The essential elements of the CRDS (those required to ensure reactor trip) are isolated from nonessential portions of the CRDS (the rod control system) as described in section 7.2. The essential portion of the CRDS is protected from the effects of postulated moderate- and high-energy line breaks.

Despite the extremely low probability of a common mode failure impairing the ability of the reactor trip system to perform its safety function, analyses have been performed in accordance with the requirements of WASH-1270. These analyses, documented in references 2 and 3, have demonstrated that acceptable safety criteria would not be exceeded even if the CRDS were rendered incapable of functioning during a reactor transient for which its function would normally be expected.

The design of the control rod drive mechanism (CRDM) is such that failure of the CRDM cooling system will, in the worst case, result in an individual control rod trip or a full reactor trip (section 7.2).

4.6.3 TESTING AND VERIFICATION OF THE CRDS

The CRDS is extensively tested prior to its operation. These tests may be subdivided into five categories:

- Prototype tests of components.
- Prototype CRDS tests.
- Production tests of components following manufacture and prior to installation.
- Onsite preoperational and initial startup tests.
- Periodic inservice tests.

These tests, which are described in paragraph 3.9.4.4, sections 4.2 and 14.2, and chapter 16, are conducted to verify the operability of the CRDS when called upon to function.

4.6.4 INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY SYSTEMS

As is indicated in chapter 15, the only postulated events which assume credit for reactivity control systems, other than a reactor trip to render the plant subcritical, are the steam line break, feedwater line break, and loss-of-coolant accident (LOCA). The reactivity control systems for which credit is taken in these accidents are the reactor trip system and the safety injection system (SIS). Additional information on the CRDS is presented in subsection 3.9.4 and on the SIS in section 6.3. Note that no credit is taken for the boration capabilities of the chemical and volume control system (CVCS) as a system in the analysis of transients presented in chapter 15. Information on the capabilities of the CVCS is provided in subsection 9.3.4. The adverse boron dilution possibilities due to the operation of the CVCS are investigated in subsection 15.4.6. Prior proper operation of the CVCS has been presumed as an initial condition to evaluate transients, and appropriate technical specifications and requirements in the Technical Requirements Manual have been prepared to ensure the correct operation or remedial action.

4.6.5 EVALUATION OF COMBINED PERFORMANCE

The evaluation of the steam line break, the feedwater line break, and the LOCA, which presumes the combined actuation of the reactor trip system to the CRDS and the SIS, is presented in subsections 15.1.5, 15.2.8, and 15.6.5. Reactor trip signals and safety injection signals for these events are generated from functionally diverse sensors and actuate diverse means of reactivity control, i.e., control rod insertion and injection of soluble poison.

Nondiverse but redundant types of equipment are utilized only in the processing of the incoming sensor signals into appropriate logic which initiates the protective action. This equipment is described in detail in sections 7.2 and 7.3. In particular, note that protection from equipment failures is provided by redundant equipment and periodic testing. Effects of failures of this equipment have been extensively investigated.[4] The failure mode and effects analysis described in reference 4 verifies that any single failure will not have a deleterious effect upon the engineered safety features actuation system. Adequacy of the emergency core cooling system and SIS performance under faulted conditions is verified in section 6.3.

4.6.6 REFERENCES

1. Shopsky, W. E., "Failure Mode and Effects Analysis of the Solid State Full-Length Rod Control System," WCAP-8976, September 1977.

2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

3. Gangloff, W. C., and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Nonproprietary), February 1971.

4. Eggleston, F. T., Rawlins, D. H., and Petrow, J. R., "Failure Mode and Effects Analysis of the Engineering Safeguard Features Actuation System," WCAP-8584 (Proprietary) and WCAP-8760 (Nonproprietary), April 1976.

APPENDIX 4A

RESPONSE TO NUREG-0737, II.F.2, INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

4A.1 THE INADEQUATE CORE COOLING (ICC) MONITORING SYSTEM INSTALLED AT VEGP WILL INCLUDE THE FOLLOWING:

- Core exit thermocouple (T/C) monitoring.
- Core subcooling margin monitoring.
- Reactor vessel level monitoring.

A detailed electrical and layout description of each of the above ICC monitoring subsystems is given below.

A. Core Exit Thermocouple System

The core exit thermocouple monitoring system consists of two redundant independent trains that monitor all operable core exit thermocouples (each train monitoring up to 25 channels). A layout sketch of the system is shown in figure 4A.1. The core exit thermocouples are mounted at the top of the core support plate. They are then routed to four upper head core exit thermocouple nozzle assemblies (CETNA) penetrations. After exiting the CETNA penetrations, the thermocouple wires proceed through a swagelok and then to qualified connectors to facilitate disconnection during removal of the upper head. Upon exiting the reactor vessel cavity, the cables are routed in a manner consistent with the requirements of Regulatory Guide 1.75 to the in-containment qualified reference junction boxes. Each reference junction box includes three redundant platinum resistance temperature detectors (RTDs) imbedded in a block of copper to reflect the temperature at the junction of the chromel alumel to copper wire. The uncompensated core exit thermocouple signals (up to 25 per train) and the reference junction box temperatures (3) are routed to remote processing units (RPUs) A3 and B3. The signals from both RPUs are routed to both display processing units (DPUs) for calculation of the compensated core exit thermocouple value. The value chosen for the reference junction box temperature is a function of the data quality of each of the RTD signals. Following the calculation of all operable compensated thermocouple values, the information from both DPUs are transmitted to both seismically qualified flat panel plant safety monitoring system (PSMS) displays. The displays are located on section D of the Plant Vogtle control board. (See figure 18.1-1.) DPU-A and display A are powered by train A and DPU-B, and display B are powered by train B. The cabling between the RPUs, DPUs, and displays meets the requirements of Regulatory Guide 1.75.

B. Core Subcooling Margin Monitor

The inputs to the core subcooling margin monitor include the following:

- Wide range reactor coolant system (RCS) pressure (4 channels)
- Core exit compensated thermocouple values (50 channels, excluding invalid channels)

- Reference junction box RTD values (6 channels)

The electrical layout of the subcooling margin monitor is shown in figure 4A-2. One channel of wide range RCS pressure is input into each RPU channel (A2, A3, B2, and B3). Also the uncompensated thermocouple channels and the corresponding three reference junction box RTD signals are input into RPUs A3 and B3. The outputs of each of the RPUs are routed to each DPU. The RCS subcooling margin is then calculated based upon the wide range RCS pressure and compensated core exit thermocouple readings. The value of RCS pressure utilized in the calculation is a function of the quality of the pressure readings. The value of core exit thermocouple temperature is based upon the auctimeered high quadrant thermocouple average reading. The subcooling margin calculated values are routed to both displays (A and B). The cable routing from sensor input to display meet the requirements of Regulatory Guide 1.75. The PSMS displays are the same display panels utilized in displaying the core exit thermocouple information.

C. Reactor Vessel Level Instrumentation System

The reactor vessel level instrumentation system (RVLIS) consists of two redundant independent trains that monitor the water level in the reactor vessel.

The wide range RVLIS reading provides an indication of reactor vessel water level from the bottom of the vessel to the top of the vessel during natural circulation conditions. The narrow range RVLIS reading provides an indication of reactor vessel water level from the middle of the hot leg pipe to the top of the reactor vessel head during natural circulation conditions. The dynamic head RVLIS reading provides an indication of reactor core, internals, and outlet nozzle pressure drop for any combination of operating reactor coolant pumps. Comparison of the measured pressure drop with the normal, single phase pressure drop provides an approximate indication of the relative void content of the circulating fluid. The inputs to the RVLIS system include the following:

1. RCS hot leg wide range RTD's (2 channels)
2. Wide range RCS pressure (4 channels)
3. Differential pressure (6 channels)
4. Reference leg temperature values (14 channels)
5. Reactor coolant pump status (4 channels)

A fluid diagram of one train of the VEGP RVLIS system is shown in figure 4A-3 for the inputs associated solely with the RVLIS system. The electrical block diagram associated with the RVLIS system is shown in figure 4A-4.

As discussed, the RCS hot leg wide range RTD signals are input to RPUs A2 and B1. Also, one wide range RCS pressure channel is input into each RPU (A2, A3, B2, and B3).

In addition, one of two sets of three differential pressure signals (wide range, narrow range, and dynamic head) are input into RPU A3 and B3, respectively. Also seven reference leg compensating temperature inputs from each train of RVLIS are input into RPUs A3 and B3. Finally, to determine the appropriate RVLIS indication, the running status of each reactor coolant pump is input into the non-1E RPU N1.

4A.2 Several analyses have been performed to verify the design of the RVLIS system described in item 4a.1c. The results of these are discussed in the following documents:

A. Summary Report, Westinghouse Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling, December 1980, submitted to the NRC via T. M. Anderson to Darrell G. Eisenhut, NS-TMA-2358 dated December 23, 1980.

B. Responses to NRC Request for Additional Information on the Westinghouse RVLIS, Summary Report.

C. Supplemental Information on the Westinghouse RVLIS, submitted to the NRC via E. P. Rahe to L. E. Phillips, NS-EPR-2579 dated March 19, 1982.

In addition to the analyses conducted in the three references above, the hydraulic components of the RVLIS system were installed at the Semiscale Test Facility in Idaho so that transient response characteristics could be obtained during small-break loss-of-coolant accident (LOCA) and other accident conditions. A description of the tests conducted and a discussion of the test results are presented in the following documents:

D. Westinghouse Evaluation of RVLIS Performance at the Semiscale Test Facility, December 1981, submitted to the NRC via E. P. Rahe to L. E. Phillips, NS-EPR-2526 dated December 8, 1981.

E. Westinghouse Evaluation of RVLIS Performance at the Semiscale Test Facility for Test S-UT-8, January 1982, submitted to the NRC via E. P. Rahe to L. E. Phillips, NS-EPR-2542 dated January 13, 1982.

F. Westinghouse Evaluation of RVLIS performance at the Semiscale Test Facility for Test S-IB-7 submitted to the NRC via E. P. Rahe to L. E. Phillips, SED-SA-00081 dated June 28, 1982.

4A.3 A description of the tests conducted on the Westinghouse RVLIS system and the results of the tests are presented in references D, E, and F listed above.

Hardware (from sensor to computer inputs) similar to that installed on VEGP is currently functioning at several operating plants for monitoring inadequate core cooling. The algorithms for computing the core exit temperature, core subcooling margin, and reactor vessel level utilized in hardware at the operating plants is similar to that implemented at VEGP.

4A.4 Response to II.F.2, Attachment I,[a] Design and Qualification Criteria for Pressurized Water Reactor Incore Thermocouples

A. Attachment I provides design of the display package on the PSMS. The display package hierarchy, as summarized from Attachment I, includes the following:

   1. Top level plant status summary

[a] Westinghouse copyrighted 1985, not included as part of the FSAR.

2. Four lower level graphic displays

    a. Core temperature map

    b. Pressure-temperature operating limits

    c. Reactor vessel water level

    d. Nuclear power

3. Four pages of menu display

    a. Primary Data Trend Menu

    b. Secondary Data Trend Menu

    c. Containment Data Trend Menu

    d. Detailed Data Menu

4. Four multi-page sets of data

    a. Six-page set of primary data trends

    b. Five-page set of secondary data trends

    c. Two-page set of containment data trends

    d. Eight-page set of detailed data

B. The following provides a top down display of the core exit thermocouple information:

1.
    a. Maximum core exit thermocouple temperature.

    b. Quadrant core exit thermocouple maximum, average and maximum temperature. Also provides a comparison between the RCS hot leg RTDs and the quadrant T/C data.

    c. Spatially oriented core exit thermocouple map showing each thermocouple temperature.

    d. Alphanumeric listing of core exit thermocouple location, tag designation, and temperature reading per quadrant.

    e. A 2-h trend history of the four core exit thermocouple quadrant maximum temperatures.

2. The core exit thermocouple display pages are designed such that any numeric thermocouple readout greater than 1200°F will be flashed at a frequency of 1 hertz.

C. The following provides a summary of the top down display of the core subcooling margin (based upon core exit thermocouples):

1. a. Core subcooling margin based upon core exit thermocouples.

   b. RCS pressure-temperature plot exhibiting plant approach to saturation.

   c. Alphanumeric listing of both trains of core subcooling margin.

   d. A 2-h trend history of the core subcooling margin.

2. The core subcooling margin will indicate "SUBCOOL" when the maximum core exit thermocouple temperature is at or below the RCS coolant saturation point. "SUPERHEAT" and the appropriate numeric value in degrees F will be displayed in reverse video when the maximum core exit thermocouple temperature exceeds the coolant saturation temperature.

D. The following provides a summary of the top down display of the RVLIS system.

1. Displays appropriate RVLIS narrow and wide range and dynamic head readings depending upon RCP status.

2. Mimic of analog meters indicating RVLIS narrow, wide, and dynamic readings with respect to reactor vessel. Only displays appropriate ranges based upon RCP status.

3. Alphanumeric listing of appropriate ranges for both trains of RVLIS system.

4. A 2-h trend history of all three RVLIS ranges. Also presents a trend of RCP status.

E. Since the VEGP PSMS display system features two redundant independent displays, one display console is considered the primary display and the other display console is considered the backup display. As such, the backup display console for ICC monitoring is also a qualified display.

F. The ranges of the ICC instrumentation are given in table 7.5.2-1.

4A.5 Response to II.F.2, Appendix B, Design and Qualification Criteria for Accident Monitoring Instrumentation

A. Equipment Qualification

1. Core Exit Thermocouple Monitoring

Listed below are the appropriate documents indicating the qualification tests conducted on the PSMS subsystems.

Subsystem	Document
Al-Ch Connectors	ESE-43B,C
Reference junction box	ESE-44A
Microprocessors	ESE-53

Plasma display	ESE-63B

2. Core Subcooling Margin Monitoring

Subsystem	Document
Wide range RCS pressure	ESE-2
Core exit thermocouples	See item above
Microprocessors	ESE-53
Plasma display	ESE-63B

3. RVLIS Monitoring System

Subsystem	Document
Wide range RCS pressure	ESE-1A
Differential pressure	ESE-4
Wide range RTDs	ESE-6
High volume pressure sensor	ESE-48A
Hydraulic isolator	ESE-49A
Reference leg RTDs	ESE-42A
Microprocessors	ESE-53
Plasma display	ESE-63B

B. Single Failure Criteria

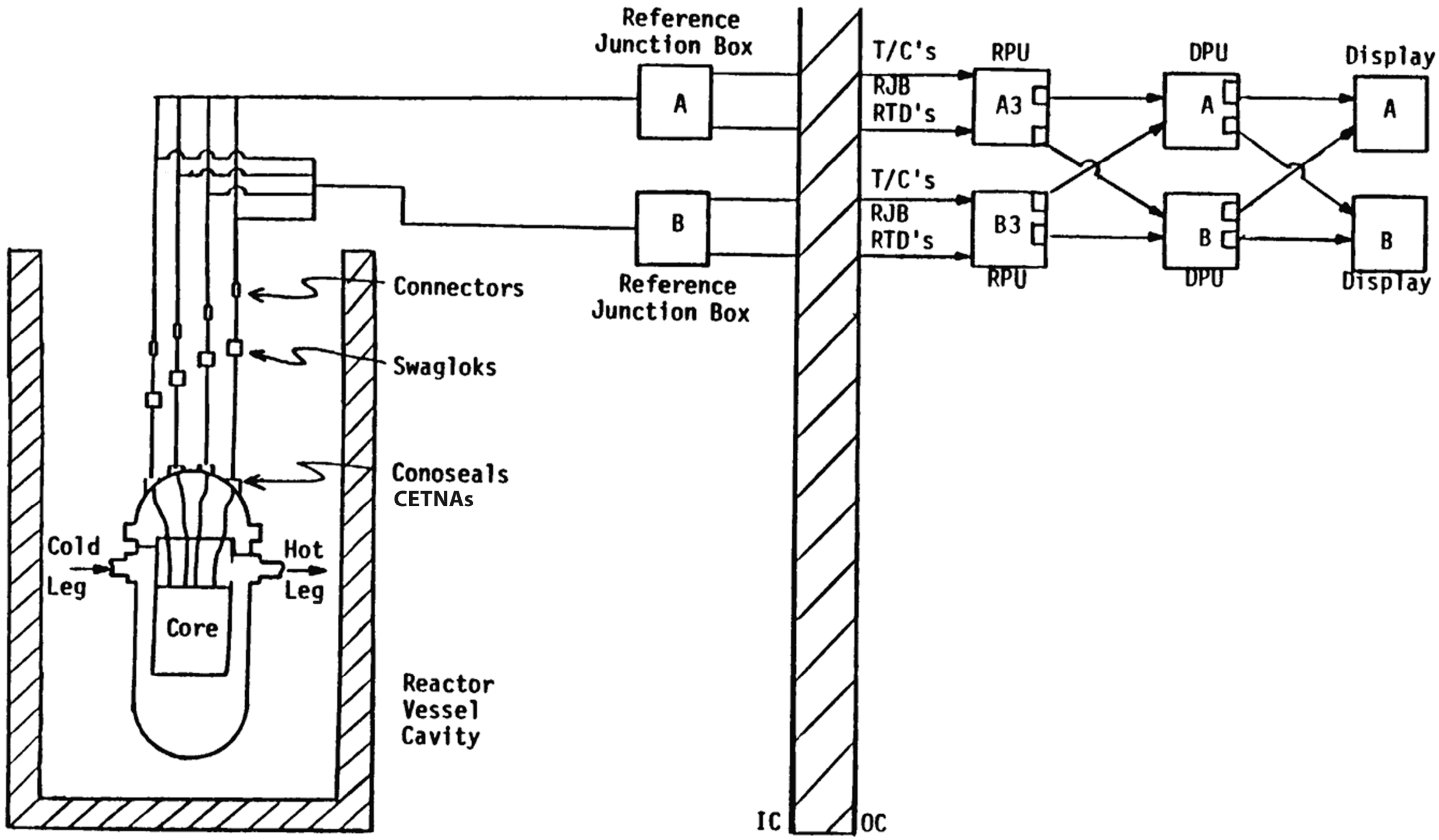
A detailed discussion of the Regulatory Guide 1.97, Post Accident Monitoring Design Basis, is presented in section 7.5 of the VEGP FSAR. Included in the discussion is a justification for the number of channels selected and the diverse variable identified where necessary. Discussed in FSAR section 7.5 is a detailed description of the characteristics associated with each ICC monitoring system.

C. RPUs A1 and A2, DPU-A, and Display A are provided by inverter power bus I. RPUs B1 and B2, DPU-B, and display B are powered by inverter power bus II. RPU A3 is powered by inverter power bus III, and RPU B3 is powered by inverter power bus IV.

A sketch of signal flows between the protection channels, RPUs, DPUs, and displays is shown in figure 4A-5.

4A.6 VEGP is adopting the format and content of the Westinghouse Owners Group (WOG) Emergency Response Guidelines for writing the plant specific procedures. Attachment II illustrates the generic WOG. These procedures provide for a critical safety function status tree for monitoring the status of plant core cooling. All variables necessary to implement the core cooling status tree are provided by the VEGP ICC instrumentation system. The functional restoration guideline, to which the operator is directed based upon the logic dictated by the tree, also utilizes the information provided by the ICC instrumentation.

4A.7 Evaluation of the acceptability of the location of the plant safety monitoring system (PSMS) displays will be included as part of the detailed control room design review.



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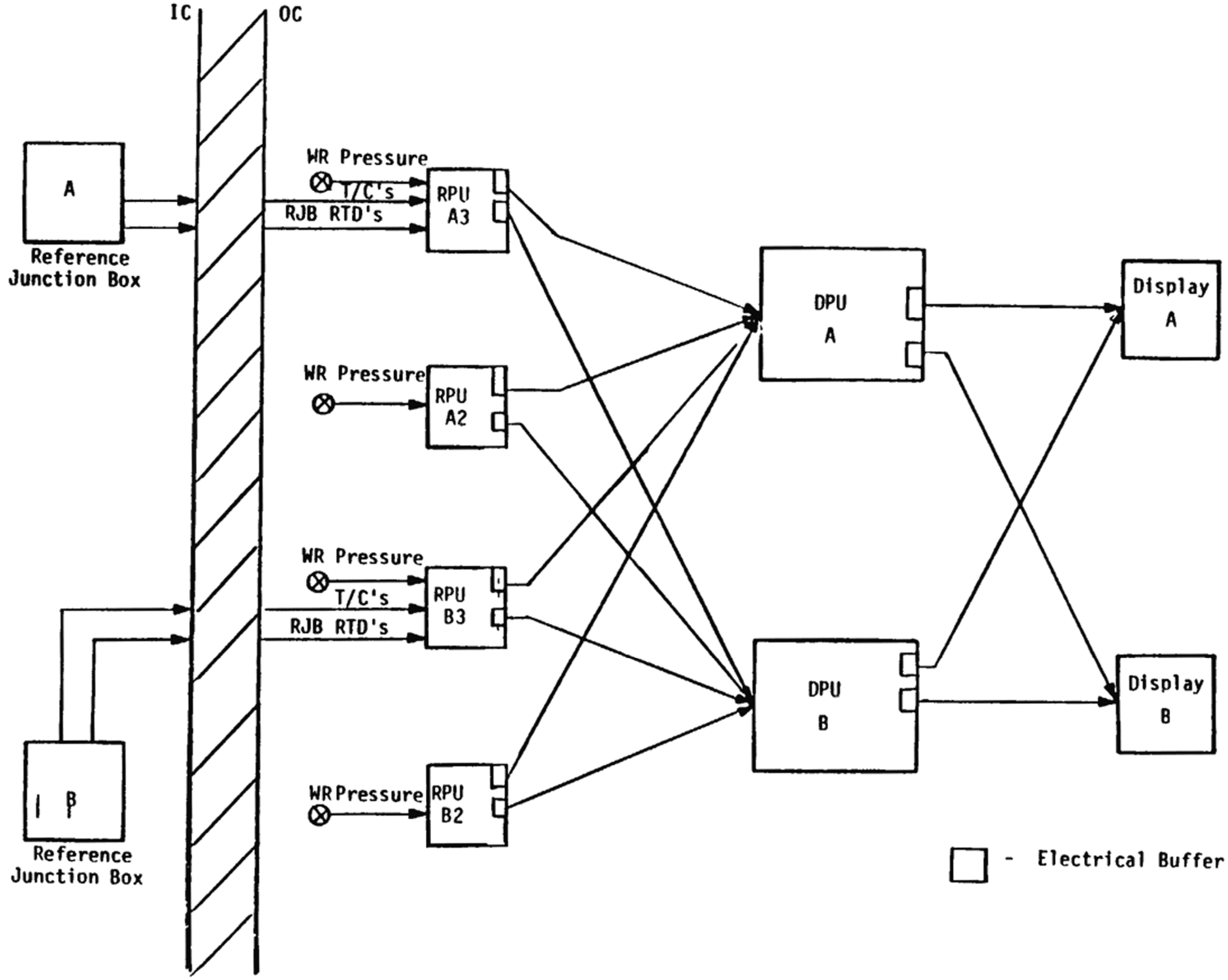
CORE EXIT THERMOCOUPLE
MONITORING SYSTEM (LAYOUT SKETCH)

FIGURE 4A–1



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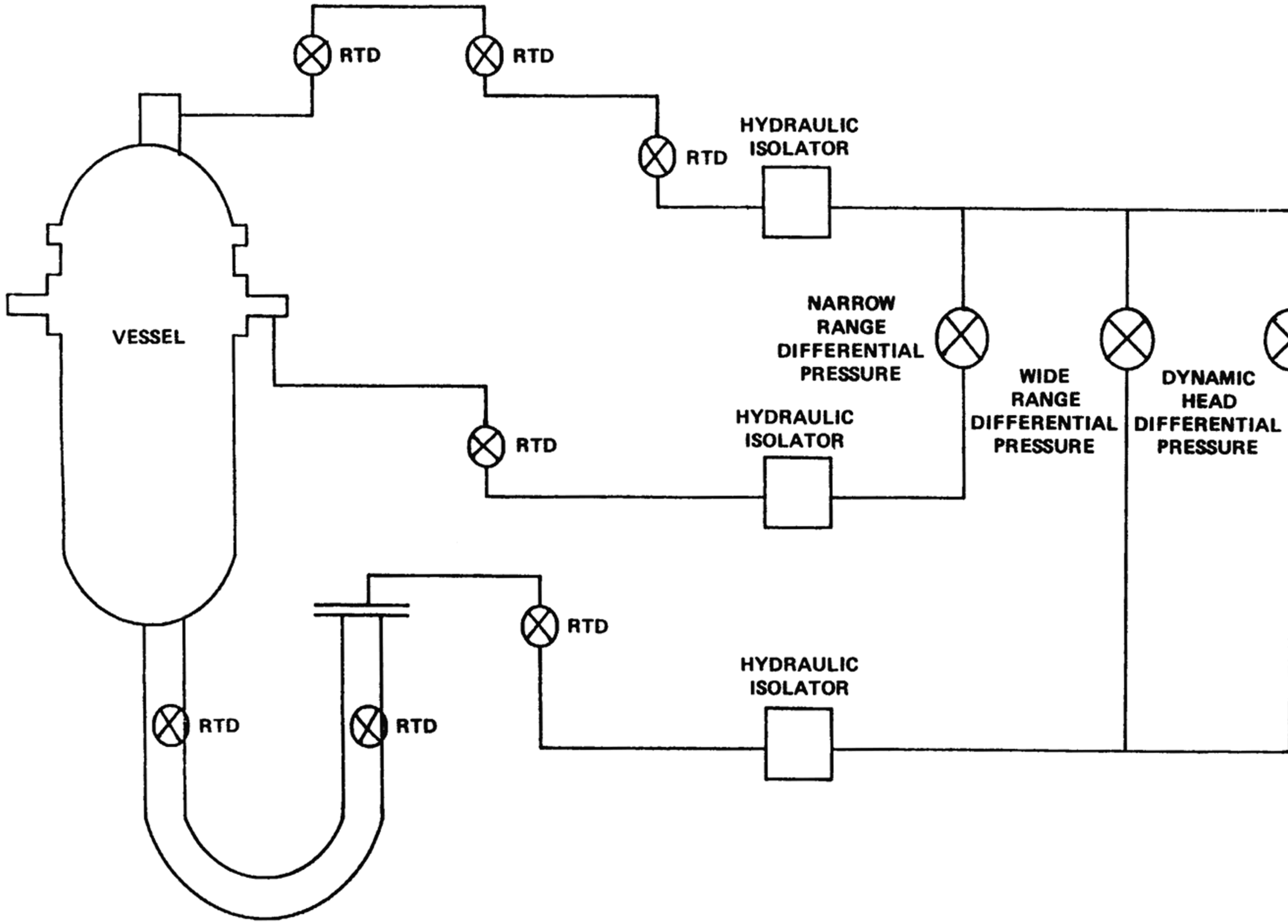
CORE SUBCOOLING MARGIN SYSTEM

FIGURE 4A–2



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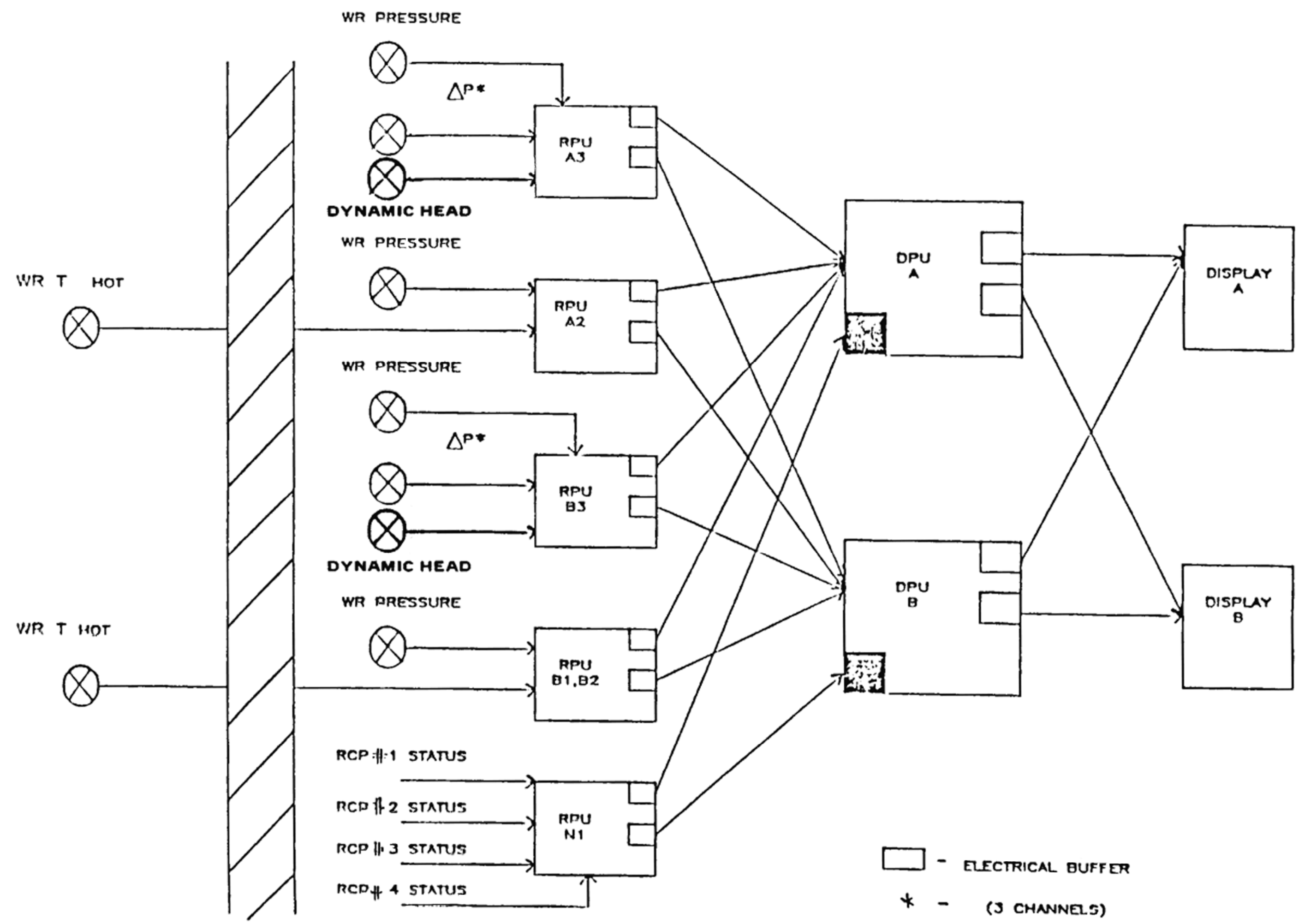
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR VESSEL LEVEL INSTRUMENTATION FLUID LAYOUT DRAWING FIGURE 4A–3



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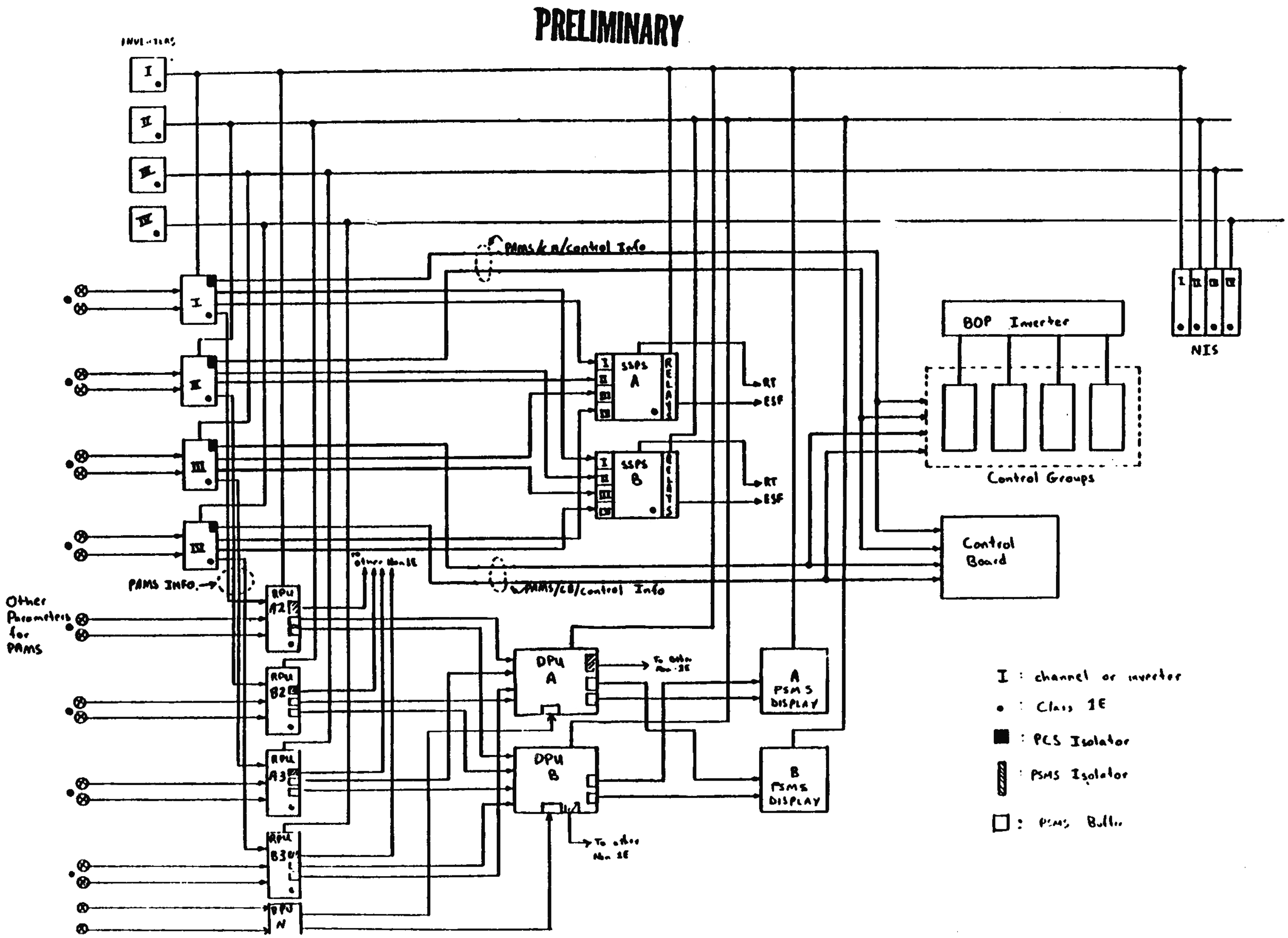
REACTOR VESSEL LEVEL
INSTRUMENTATION SYSTEM

FIGURE 4A–4



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VOGTLE PSMS ELECTRICAL INTERCONNECTION

FIGURE 4A–5

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

This section describes the reactor coolant system (RCS) and includes a schematic flow diagram (figure 5.1.2-1), a piping and instrumentation diagram (drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, 1X4DB113, and 2X4DB113), and an elevation drawing (drawings 1X4DL4A17 and 2X4DL4A17).

5.1.1 DESIGN BASES

The performance and safety design bases of the RCS and its major components are interrelated. These design bases are listed below:

A. The RCS has the capability to transfer to the steam and power conversion system the heat produced during power operation and when the reactor is subcritical, including the initial phase of plant cooldown.

B. The RCS has the capability to transfer to the residual heat removal system the heat produced during the subsequent phase of plant cooldown and cold shutdown.

C. The RCS heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, ensures no fuel damage within the operating bounds permitted by the reactor control and protection systems.

D. The RCS provides the water used as the core neutron moderator and reflector and as a solvent for the neutron absorber used as chemical shim control.

E. The RCS maintains the homogeneity of the soluble neutron poison concentration and the rate of change of the coolant temperature so that uncontrolled reactivity changes do not occur.

F. The RCS pressure boundary is capable of accommodating the temperatures and pressures associated with operational transients.

G. The reactor vessel supports the reactor core and control rod drive mechanisms.

H. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant.

I. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.

J. The steam generators provide high-quality steam to the turbine. The tube and tube sheet boundary are designed to prevent the transfer of radioactivity generated within the core to the secondary system.

K. The RCS piping contains the coolant under operating temperature and pressure conditions and limits leakage (and activity release) to the containment atmosphere. The RCS piping contains demineralized borated water that is circulated at the flowrate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

L. The RCS is monitored for loose parts, as described in subsection 4.4.6.

M. Applicable industry standards and equipment classifications of RCS components are identified in table 3.2.2-1.

N. The reactor vessel is provided with a head vent that meets the requirements of TMI Action Item II.B.1. (See subsections 5.4.7 and 5.4.15.)

O. Unisolable sections of safety injection, normal and alternate charging, and auxiliary spray lines interconnected with the reactor coolant system, two 12-in. residual heat removal suction lines attached to the reactor coolant loop, and the pressurizer surge line are instrumented with resistance temperature detectors (RTDs) strapped on the pipe to detect thermal stratification. (See paragraph 5.4.3.3.4.)

5.1.2 DESIGN DESCRIPTION

The reactor coolant system (RCS), shown in drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113, consists of four similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, the system includes a pressurizer, pressurizer relief and safety valves, interconnecting piping, and instrumentation necessary for operational control. All the above components are located in the containment building.

During operation, the RCS transfers the heat generated in the core to the steam generators, where steam is produced to drive the turbine-generator. Borated demineralized water is circulated in the RCS at a flowrate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector and as a solvent for the neutron absorber used in chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

The RCS pressure is controlled by the use of the pressurizer where water and steam are maintained at saturation conditions by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power-operated relief valves connected to the pressurizer provide for steam discharge from the RCS. Discharged steam is piped to the pressurizer relief tank (pressurizer relief discharge system), where the steam is condensed and cooled by mixing with water.

The extent of the RCS is defined as:

- The reactor vessel, including control rod drive mechanism housings.
- The portion of the steam generators containing reactor coolant.
- The reactor coolant pumps.

- The pressurizer.
- The safety and relief valves.
- The head vent.
- The interconnecting piping, valves, and fittings between the principal components listed above.
- The piping, fittings, and valves leading to connecting auxiliary or support systems.

The RCS schematic flow diagram is shown in figure 5.1.2-1. Included with this figure is a tabulation of principal pressures, temperatures, and flowrates of the system under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate flow at the pump discharge. The RCS volume under these conditions is presented in table 5.1.2-1.

A piping and instrumentation diagram of the RCS is shown in drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113. This diagram shows the extent of the systems located within the containment and the points of separation between the RCS and the secondary (heat utilization) system. Drawings 1X4DE312, 1X4DE313, 1X4DE314, 1X4DE317, 1X4DE320, 1X4DE322, 1X2D48E007 and 1X2D48E008 provide plan and elevation views of the containment, and drawings 1X4DL4A17 and 2X4DL4A17 show plan and section of the reactor coolant loops. These figures show principal dimensions of RCS components in relationship to supporting and surrounding steel and concrete structures and demonstrate the protection provided to the RCS by its physical layout.

5.1.3 SYSTEM COMPONENTS

The major components of the reactor coolant system are as follows:

A. Reactor Vessel

The reactor vessel is cylindrical and has a welded, hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

B. Steam Generators

The steam generators are vertical shell and U-tube evaporators with integral moisture-separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

C. Reactor Coolant Pumps

The reactor coolant pumps (RCPs) are single-speed centrifugal units driven by water/air-cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pump. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The flow inlet is at the bottom of the pump, and the discharge is on the side.

D. Piping

The reactor coolant piping is seamless, stainless steel piping. The hot leg is defined as the piping between the reactor vessel outlet nozzle and the steam generator. The mechanical stress improvement process (MSIP) has been applied to the Unit 1 and Unit 2 hot legs near the reactor vessel outlet nozzle. The cold leg is defined as the piping between the RCP outlet and the reactor vessel. The crossover leg is defined as the piping between the steam generator outlet and the RCP inlet.

E. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. The pressurizer is connected to the hot leg of one of the coolant loops by a surge line. Electrical heaters are installed through the bottom head of the vessel. The spray nozzle and relief and safety valve connections are located in the top head of the vessel.

F. Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop type. The valves are spring loaded and self-activated with backpressure compensation. The power-operated relief valves are solenoid-operated valves. They are operated automatically or by remote manual control. Remotely operated gate valves are provided to isolate the inlet to the power-operated relief valves if excessive leakage occurs. Position-indicating lights are provided in the control room for these valves.

5.1.4 SYSTEM PERFORMANCE CHARACTERISTICS

Design and performance characteristics of the reactor coolant system are provided in table 5.1.2-1.

A. Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established by a detailed design procedure supported by operating plant performance data and component hydraulics experimental data. The procedure establishes a best-estimate flow as well as conservatively high and low flows for the applicable mechanical and thermal design considerations. In establishing the range of design flows, the procedure accounts for the uncertainties in the component flow resistances and the pump head-flow capability, established by analysis of the available experimental data. The procedure also accounts for the uncertainties in the technique used to measure flow in the operating plant.

Definitions of the three reactor coolant flows applied in various plant design considerations are presented in the following paragraphs.

B. Best-Estimate Flow

The best-estimate flow is considered to be the most likely value for the plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator, and piping flow resistances and on the best estimate of the reactor coolant pump (RCP) head-flow capability, with no known uncertainties assigned to either the system flow resistance or the pump head. The best-estimate flow provides the basis for the establishment of the other design flows required for the system and component design. System pressure losses based on best-estimate flow are presented in table 5.1.2-1.

The best-estimate flow analysis has been based on extensive experimental data, including accurate flow and pressure drop data from one operating plant, flow resistance measurements from several fuel assembly hydraulics tests, and hydraulic performance measurements from several pump impeller model tests. Since operating plant flow measurements have been shown to be in close agreement with the calculated best-estimate flows, the flows established with this design procedure can be applied to the plant design with a high level of confidence.

Although the best-estimate flow is the most likely value to be expected in operation, more conservative flowrates are applied in the thermal and mechanical designs.

C. Thermal Design Flow

Thermal design flow is the flowrate used as a basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. The thermal design flow accounts for the uncertainties in flow resistances (reactor vessel, steam generator, and piping), RCP head, and the methods used to measure flowrate. The thermal design flow is approximately 5.8% less than the best-estimate flow with 10% equivalent steam generator plugging. The thermal design flow is confirmed when the plant is placed in operation. Tabulations of important design parameters based on the thermal design flow are provided in table 5.1.2-1.

D. Mechanical Design Flow

Mechanical design flow is a conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. The mechanical design flow is based on a reduced system resistance and on increased pump head capability. The mechanical design flow is approximately 1.7% greater than the best-estimate flow with 0% equivalent steam generator tube plugging.

Pump overspeed due to a turbine generator overspeed of 20% results in a peak reactor coolant flow of 120% of the mechanical design flow. The overspeed condition is applicable only to operating conditions when the reactor and turbine generator are at power.

TABLE 5.1.2-1 (SHEET 1 OF 2)

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life (years)	40[g]	
Nominal operating pressure (psig)	2235	
Total system volume, including pressurizer and surge line (ft^3)	12,347[(a)]	
System liquid volume, including pressurizer water at maximum guaranteed power (ft^3)	11,594[(a)]	
Pressurizer spray rate, maximum (gal/min)	900	
Pressurizer heater capacity (kW)	1800 / 1661 (Unit 2)	
Minimum pressurizer heater capacity (kW) required to survive 3/8-in. line break w/o Rx trip or ECCS actuation (does not reflect 100 kW heat loss reduction)	1624	
<u>System Thermal and Hydraulic Data</u>	<u>4 Pumps Running</u>	
	(b)	(e)
Nuclear steam supply system power (MW)	3643	
Reactor power (MW)	3626	
Thermal design flows (gal/min)		
Active loop	93,600	
Reactor	374,400	
Total reactor flow (10^6 lb/h)	139.45	143.1
Temperatures (°F)		
Reactor vessel outlet	620.6	603.8
Reactor vessel inlet	556.2	537.6
Steam generator outlet	555.9	537.3
Steam generator steam	537.3	520.6
Feedwater	448.7	448.7
Steam pressure (psia)	941	817

TABLE 5.1.2-1 (SHEET 2 OF 2)

Total steamflow (10^6 lb/h)	16.31	16.22
Best-estimate flows (gal/min)		
Active loop	99,400	102,500
Reactor	397,600	410,000
Mechanical design flows (gal/min)		
Active loop	104,200	
Reactor	416,800	
System pressure drops[f]		
Reactor vessel ΔP (psi)	46.5	47.8
Steam generator ΔP (psi)	45.5	41.3
Hot leg piping ΔP (psi)	1.2	1.3
Crossover leg piping ΔP (psi)	3.1	3.4
Cold leg piping ΔP (psi)	3.3[d]	3.6[d]
Pump head (ft)	309	295

a. Nominal volumes reflecting 0% equivalent steam generator tube plugging.

b. Parameters based on full power operation with 10% equivalent steam generator tube plugging and reactor vessel average temperature of 588.4°F.

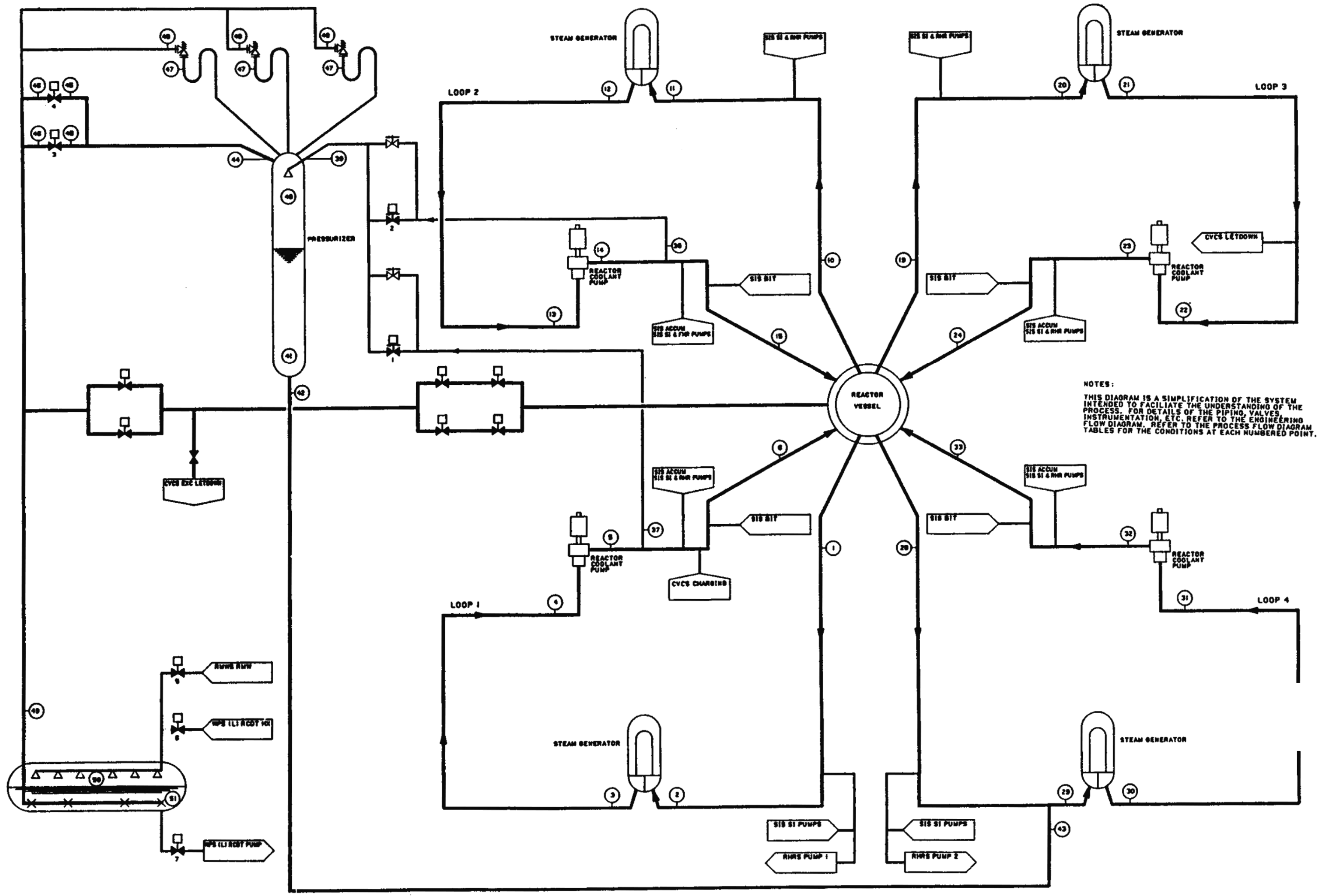
c. Includes core, internals, and nozzles.

d. Includes pump weir ΔP of 2.0 psi.

e. Parameter based on full power operation with 0% equivalent steam generator tube plugging and reactor vessel average temperature of 570.7.

f. System pressure drops are based on best estimate flow.

g. The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).



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RCS PROCESS FLOW DIAGRAM

FIGURE 5.1.2–1 (SHEET 1 OF 3)

Steady-State, Full-Power Operation[a]

Location	Fluid	Pressure (psig)	Nominal Temperature	Flow[b] (gal/min)	Flow (lb/hx10)	Volume (ft^3)
1	Reactor coolant	2246.6	618.2	111,570	37.0430	-
2	Reactor coolant	2245.2	618.2	111,680	37.0783	-
3	Reactor coolant	2206.0	558.6	99,930	37.0783	-
4	Reactor coolant	2202.9	558.6	100,100	37.1397	-
5	Reactor coolant	2295.6	558.8	99,980	37.1372	-
6	Reactor coolant	2292.3	558.8	99,910	37.1086	-
10-16	Reactor coolant	See loop No. 1 specifications.				
19-24	Reactor coolant	See loop No. 1 specifications.				
28-33	Reactor coolant	See loop No. 1 specifications.				
37	Reactor coolant	2296.5	558.8	1.0	0.00037	-
38	Reactor coolant	2295.5	558.8	1.0	0.00037	-
39	Reactor coolant	2235.0	652.7	1.0	0.00074	-
40	Steam	2235.0	652.7	-	-	720

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RCS PROCESS FLOW DIAGRAM

FIGURE 5.1.2–1 (SHEET 2 OF 3)

Location	Fluid	Pressure (psig)	Nominal Temperature	(b) Flow (gal/min)	(lb/hx10)	Volume (ft^3)
41	Reactor coolant	2244.0	652.7	-	-	1080
42	Reactor coolant	2244.0	652.7	2.50	0.00074	-
44	Steam	2235.0	652.7	0	0	Minimize
45	Reactor coolant	2235.0	652.7	0	0	Minimize
46	N_2	3.0	120	0	0	-
47	Reactor coolant	2235.0	652.7	0	0	Minimize
48	N_2	3.0	120	0	0	-
49	N_2	3.0	120	0	0	-
50	N_2	3.0	120	0	0	450
51	Pressurizer relief tank water	3.0	120	-	-	1350

a. Simplifying assumption is made that letdown and charging flows are distributed over all loops. Charging, letdown, and continuous spray flows are ignored in overall balance.

b. At the conditions specified.

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RCS PROCESS FLOW DIAGRAM

FIGURE 5.1.2–1 (SHEET 3 OF 3)

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime[a]. Section 50.2 of 10 CFR 50 defines the RCPB as extending to the outermost containment isolation valve in system piping which penetrates the containment and is connected to the reactor coolant system (RCS). This section is limited to a description of the components of the RCS, as defined in section 5.1, unless otherwise noted. Components which are part of the RCPB (as defined in 10 CFR 50) but are not described in this section are described in the following sections:

A. Section 6.3 - RCPB components which are part of the emergency core cooling system.

B. Subsection 9.3.4 - RCPB components which are part of the chemical and volume control system.

C. Subsection 3.9.N.1 - Design loading, stress limits, and analyses applied to the RCS and American Society of Mechanical Engineers (ASME) Code Class 1 components.

D. Subsection 3.9.N.3 - Design loadings, stress limits, and analyses applied to ASME Code Class 2 and 3 components.

The abbreviation RCS, as used in this section, is as defined in section 5.1. When the term RCPB is used in this section, its definition is that of Section 50.2 of 10 CFR 50.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR 50.55a

RCS components are designed and fabricated in accordance with 10 CFR 50, Section 50.55a, Codes and Standards. The addenda of the ASME code applied in the design of each component are listed in table 5.2.1-1.

5.2.1.2 Applicable Code Cases

Regulatory Guides 1.84 and 1.85 are discussed in section 1.9. The following discussion addresses only unapproved or conditionally approved code cases (per Regulatory Guides 1.84 and 1.85) used on Class 1 components.

Code Case 1528 (SA-508, Class 2a) material has been used in the manufacture of the VEGP steam generators and pressurizers. Regulatory Guide 1.85 presently reflects a conditional Nuclear Regulatory Commission (NRC) approval of Code Case 1528. Westinghouse has conducted a test program which demonstrates the adequacy of Code Case 1528 material. The

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

results of the test program are documented in reference 1. Reference 1 and a request for approval (reference 2) of the use of Code Case 1528 have been submitted to the NRC.

5.2.1.3 References

1. Letter NS-CE-1228, dated October 4, 1976, C. Eicheldinger (Westinghouse) to J. F. Stolz (NCR).

2. Letter NS-CE-173, dated March 17, 1978, C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC).

5.2.2 OVERPRESSURE PROTECTION

Reactor coolant system (RCS) overpressure protection is provided by the pressurizer safety valves, steam generator safety valves, and the reactor protection system and associated equipment. Combinations of these systems ensure compliance with the overpressure requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

The only portions of an auxiliary system connected to the RCS that are utilized for overpressure protection of the RCS are the liquid relief valves of the residual heat removal system (RHRS). These valves protect the RCS at low temperatures when the RHRS is in operation.

5.2.2.1 Design Bases

Overpressure protection is provided for the RCS by the pressurizer safety valves. This protection is afforded for the following events which envelope those credible events that could lead to overpressure of the RCS if adequate overpressure protection is not provided:

- Loss of electrical load and/or turbine trip.
- Uncontrolled rod withdrawal at power.
- Loss of reactor coolant flow.
- Loss of normal feedwater.
- Loss of offsite power to the station auxiliaries.

The sizing of the pressurizer safety valves is based on analysis of a complete loss of steamflow to the turbine with the reactor operating at 102 percent of engineered safeguards design power. In this analysis, feedwater flow is also assumed to be lost, and no credit is taken for operation of the pressurizer power-operated relief valves (PORV), pressurizer level control system, pressurizer spray system, rod control system, steam dump system, or steam line PORV. The reactor is maintained at full power (no credit for direct reactor trip on turbine trip), and steam relief through the steam generator safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient.

This sizing procedure results in a safety valve capacity well in excess of the capacity required to prevent exceeding 110 percent of system design pressure for the events listed above.

Overpressure protection for the steam system is provided by steam generator safety valves. The steam system safety valve capacity is based on providing enough relief to remove the engineered safeguards design steamflow. This must be done while limiting the maximum steam system pressure to less than 110 percent of the steam generator shell side design pressure.

Blowdown and heat dissipation systems of the nuclear steam supply system (NSSS) connected to the discharge of these pressure relieving devices are discussed in subsection 5.4.11.

Steam generator blowdown systems are discussed in subsection 10.4.8.

5.2.2.2 Design Evaluation

A description of the pressurizer safety valves performance characteristics along with the design description of the incidents, assumptions made, method of analysis, and conclusions are discussed in chapter 15.

The relief capacities of the pressurizer and steam generator safety valves are determined from the postulated overpressure transient conditions in conjunction with the action of the reactor protection system. An evaluation of the functional design of the overpressure protection system and an analysis of the capability of the system to perform its function for a typical plant are presented in reference 1. The report describes in detail the types and number of pressure relief devices employed, relief device description, locations in the systems, reliability history, and the details of the methods used for relief device sizing based on typical worst-case transient conditions and analysis data for each transient condition. An overpressure protection report specifically for the VEGP is prepared in accordance with Article NB-7300 of Section III of the ASME Code. The description of the analytical model used in the analysis of the overpressure protection system and the basis for its validity are discussed in reference 2.

The capacities of the pressurizer safety and relief valves are discussed in subsection 5.4.13. The setpoints and reactor trip signals which occur during overpressure transients are discussed in subsection 5.4.10.

5.2.2.3 Piping and Instrumentation Diagrams

Overpressure protection for the RCS is provided by the pressurizer safety and relief valves shown in drawing 1X4DB112. These valves discharge to the pressurizer relief tank through a common manifold.

The steam system safety valves are discussed in section 10.3 and are shown in drawings 1X4DB159-1, 1X4DB159-2, and 1X4DB159-3.

5.2.2.4 Equipment and Component Description

The operation, significant design parameters, number and types of operating cycles, and environmental conditions of the pressurizer safety valves are discussed in section 3.11.N and subsections 3.9.N.1 and 5.4.13.

Section 10.3 contains a discussion of the equipment and components of the steam system overpressure protection features.

5.2.2.5 Mounting of Pressure Relief Devices

The design and installation of the pressure relief devices for the RCS are described in subsection 5.4.11. The design basis for the assumed loads for the primary and secondary side pressure relief devices are described in section 3.9.N. Subsection 10.3.2 provides a discussion of the main steam safety valves and the power-operated atmospheric steam relief valves.

5.2.2.6 Applicable Codes and Classification

The requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are met.

Piping, valves, and associated equipment used for overpressure protection are classified in accordance with American National Standards Institute (ANSI) N18.2a-1975, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. These safety class designations are delineated in table 3.2.2-1 and shown in drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113.

5.2.2.7 Material Specifications

Refer to subsection 5.2.3 for a description of material specifications.

5.2.2.8 Process Instrumentation

Each pressurizer safety valve discharge line incorporates a control board temperature indicator and alarm to notify the operator of steam discharge due to either leakage or actual valve operation. For a further discussion on process instrumentation associated with the system, refer to chapter 7.

5.2.2.9 System Reliability

The reliability of the pressure relieving devices is discussed in section 4 of reference 1.

5.2.2.10 RCS Pressure Control During Low-Temperature Operation[a]

An important aspect of RCS overpressure protection at low temperatures is the use of administrative controls which are discussed in some detail in paragraph 5.2.2.10.2. Although specific alarms do not exist to invoke specific administrative procedures, annunciation is provided to alert the operator to arm the cold overpressure mitigation system. Operating procedures maximize the use of a pressurizer steam bubble, since a steam bubble reduces the maximum pressure reached for some transients, and slows the rate of pressure increase for others, and aids the operator in controlling RCS pressure during low temperature operation.

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[a] The cold overpressure protection system setpoint calculation was evaluated as a time-limited aging analysis (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in paragraph 19.4.6.4.

When the RCS is at temperatures below approximately 350°F, it is opened to the RHRS for the purposes of removing residual heat from the core, providing a path for letdown to the purification subsystem, and controlling the RCS pressure when the plant is operating in a water solid mode. The RHRS is provided with self-actuated water relief valves to prevent overpressure in this relatively low design pressure system caused either within the system itself or from transients transmitted from the RCS. The RHRS relief valves mitigate pressure transients originated in the RCS to maximum pressure values determined by the relief valve set pressure.

The low design pressure RHRS is normally isolated from the high design pressure RCS during reactor power operation at temperatures above approximately 350°F by two isolation valves in series. Therefore, the RHRS can be inadvertently isolated from the RCS by these same isolation valves. The PORVs and associated logic provide overpressure mitigation for those transients which might occur if the RHRS isolation valves were inadvertently closed. The PORV logic is manually armed at the system setpoint.

Two pressurizer PORVs are each supplied with actuation logic. The logic for each PORV continuously monitors RCS temperature and pressure, converts an auctioneered RCS temperature to the Appendix G allowable pressure, and then compares the allowable pressure to the actual RCS pressure. As the actual RCS pressure approaches the allowable pressure, a main control board alarm is annunciated. If the RCS pressure continues to increase, an actuation signal is transmitted to a PORV and the valve opens to mitigate the transient.

As described in subsection 5.4.13, the VEGP PORVs are safety related. They were designed in accordance with the ASME Code and are qualified via the Westinghouse pump and valve operability program which is described in paragraph 3.10.N.2.2.

The hardware and logic associated with this function will operate following an operating basis earthquake. Offsite power is not required for the system to function. The actuation logic in the system is testable. However, the PORVs and RHR relief valves are not exercised with the reactor at power. They are capable of being tested as required by the ASME Code and the VEGP Technical Specifications.

5.2.2.10.1 Transient Evaluation

Potential overpressurization transients to the RCS, while at relatively low temperatures, can be caused by either of two types of events to the RCS; i.e., mass input or heat input. Both types result in more rapid pressure changes when the RCS is water solid.

Anticipated mass and heat input transients are evaluated to demonstrate conformance with Appendix G. The most limiting heat input transient is an inadvertant reactor coolant pump startup in a loop where the steam generator secondary temperature is 50°F higher than the primary temperature in any loop. The most limiting mass input transient is a charging-letdown mismatch where two emergency core cooling system (ECCS) centrifugal charging pumps and one normal centrifugal charging pump are charging water into the reactor coolant system with the letdown path isolated.

It should be noted that the following transient is also addressed. With the plant in a cooled down and depressurized condition in which the cold overpressure protection system is required to be operable and with charging and letdown established and RHRS open to RCS, a dc vital bus fails. This failure causes normal letdown to isolate and also results in the loss of one of the two PORVs. However, RHRS relief valves mitigate the transient.

5.2.2.10.2 Administrative Controls

During plant operation the following precautions are observed:

A. At least one RHR inlet line from the reactor coolant loop is not isolated unless there is a steam bubble in the pressurizer. This precaution ensures that there is a relief path from the reactor coolant loop to a RHR suction line relief valve when the RCS is at low temperature and is water solid.

B. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, letdown should include flow from the operating RHR loop through the RHRS cleanup to the letdown heat exchanger valve.

C. One RCP should normally be running anytime RCS temperature is changed by more than 10°F in 1 h. Additionally, RCPs should not be started if steam generator secondary water temperature is greater than 10°F above the RCS temperature.

D. During a typical plant cooldown, operable steam generators should be connected to the steam header to ensure a uniform cooldown of the reactor coolant loops.

E. To preclude inadvertent emergency core cooling system (ECCS) actuation during heatup and cooldown, blocking of the high pressurizer pressure, and low steam line pressure safety injection, signal actuation logic at 1970 psig is required. These manual blocking features are further discussed in paragraph 7.3.1.2.2.6.

   During further cooldown, closure and power lockout of the accumulator isolation valves is performed at 1000 psig and power lockout to the safety injection pumps is performed at approximately 220°F in the RCS.

F. Periodic ECCS pump performance testing requires the testing of the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

   Should cold shutdown testing of the pumps be required, the test is done when the vessel is open to the atmosphere, again precluding overpressurization potential.

   If cold shutdown testing with the vessel closed is necessary, the procedures require only one pump to be tested with ECCS discharge valve closure and RHRS alignment to both isolate potential ECCS pump input and to provide backup benefit of the RHRS relief valves.

   The SI signal circuitry testing, if performed during cold shutdown, also requires RHRS alignment and safety injection pumps power lockout to preclude developing cold overpressurization transients.

G. A steam bubble will be maintained in the pressurizer when the RCS temperature is greater than 220°F.

5.2.2.11 <u>Consequences of a Postulated Loss of a dc Bus Coupled with a Single Failure Disabling a PORV Allowing a Cold Pressurization Event</u>

This discussion addresses the following postulated event:

With the plant in a cooled down and depressurized condition in which the cold overpressure protection system is required to be operable and with charging and letdown established, a dc vital bus fails. This failure causes normal letdown to isolate and also results in the loss of one of the two PORVs.

In addition to the dc bus failure, an additional random failure of the second PORV is postulated to occur. This sequence of events places the plant in a condition in which letdown is isolated, the automatic cold overpressure protection system is inoperable, and charging flow is filling the pressurizer, increasing system pressure towards the Appendix G limits.

To begin this discussion the limitations placed on plant operation by the VEGP Technical Specifications are addressed.

A. With RCS temperature below 200°F (i.e., cold shutdown) one RHR pump is required to be in operation. This requirement ensures that at least one RHR suction relief valve is available for overpressure protection of the RCS. This valve will relieve the flow from two ECCS centrifugal charging pumps and one normal centrifugal charging pump at the valve lift setting pressure.

B. Whenever the cold overpressure protection system is required to be operable, only the two ECCS centrifugal charging pumps and the normal centrifugal charging pumps are allowed to be operable. The safety injection pumps will be inoperable. This ensures that the maximum charging letdown mismatch will be that stated in 5.2.2.10.1.

C. Considering these requirements, anytime RHR is in operation and the RCS is in a condition requiring the cold overpressure protection to be operable, there is no overpressure event as a result of the prescribed event. Assuming the event as described[a] did occur, the RHR relief valve would prevent RCS pressure from reaching the Appendix G limit by relieving all charging flow.

Typically at least one RHR train is in operation, or at a minimum one RHR loop suction valve is open, providing an open path from the RCS to a RHR suction relief valve, whenever RCS temperature is below 350°F. For this reason an overpressure event resulting from the prescribed event is very unlikely; however, the discussion is extended to the infrequent case where the RHRS is isolated from the RCS, and the cold overpressure protection system is required to be operable.

To gain a better understanding of the results of the event, it is necessary to address the functions of some of the chemical and volume control system control valves. As stated earlier, the letdown valves fail closed on loss of dc power isolating letdown. Between the charging pumps and the normal charging isolation valves are two normally throttled valves which receive their power from the process and control racks powered by the essential ac instrument buses. These valves are unaffected by a dc bus failure and continue to work normally during the event.

One of these valves is the charging flow control valve (1-FCV-121), which automatically regulates flow to maintain a prescribed pressurizer level. Assuming this valve continues to function normally, as pressurizer level rises, charging flow is reduced until the charging flow is limited to that required for seal injection (32 gal/min) plus a minimal amount (15 gal/min) required for regenerative heat exchanger cooling. At this flowrate, ample time is provided (as discussed below) to allow appropriate operator action. If valve control is in manual, the valve position remains unchanged. The other valve is the charging flow backpressure regulator

[a] As additional information, during RHR operation letdown is typically taken from the discharge of the RHR pumps and is not isolated by the dc bus failure.

(1-HCV-182), which is manually positioned to regulate flow to the seal. This valve remains in its initial position. The effect of these two valves is to limit charging flow to its value at the beginning of the event. Assuming maximum letdown at the initiation of the event, total flow (charging plus seal injection) to the RCS is limited to approximately 130 gal/min.

An additional consideration is that, with the plant in the hot shutdown condition and RHR isolated from the RCS, normal operation is to have a steam bubble in the pressurizer of approximately 1350 ft^3. At a maximum charging rate of 130 gal/min, it would take in excess of 30 min to reach the Appendix G limit at 200°F, the temperature corresponding to the coldest RCS temperature at which RHR is permitted to be isolated. As an extreme case, with a bubble of only half the normal size, the corresponding time available for appropriate action would be in excess of 15 min.

To summarize:

A. The postulated event is unlikely to occur since the dc buses have a battery as an emergency power supply, and should the dc bus fail, it must be coupled with the additional failure of the second PORV for overpressurization.

B. In the unlikely event that the prescribed event did occur, RHR would normally be online and capable of mitigating any potential overpressure resulting from two ECCS centrifugal charging pumps and one normal centrifugal charging pump.

C. In the highly unlikely event that the prescribed event should occur when RHR is isolated from the RCS, the operator would have sufficient time to mitigate the event.

D. The Appendix G curves are excessively conservative for their intended purpose of ensuring vessel integrity during cold shutdown.

No further action is necessary to address this postulated event, and the existing plant design and operational techniques result in successful event mitigation.

5.2.2.12 Testing and Inspection

Testing and inspection of the overpressure protection components are discussed in paragraph 5.4.13.4 and chapter 14. Testing capabilities of the overpressurization protection system are consistent with testing principles for systems' electronics in paragraph 7.2.2.2.3. Operational surveillance procedures will demonstrate the operability of the overpressure protection system when system operation is required. Inservice inspection is performed in accordance with Section XI of the ASME Code. Relief requests, augmented examinations, and alternate techniques will be utilized as appropriate.

5.2.2.13 References

1. Cooper, L., Miselis, V., and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972 (also letter NS-CE-622 dated April 16, 1975, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), Additional Information on WCAP-7769, Revision 1).

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, October 1972.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY (RCPB) MATERIALS

5.2.3.1 Material Specifications

Typical material specifications used for the principal pressure-retaining applications in Class 1 primary components and for Class 1 and 2 auxiliary components in systems required for reactor shutdown and for emergency core cooling are listed in table 5.2.3-1. Typical material specifications used for the reactor internals required for emergency core cooling, for any mode of normal operation or under postulated accident conditions, and for core structural load bearing members are listed in table 5.2.3-2.

Tables 5.2.3-1 and 5.2.3-2 may not be totally inclusive of the material specifications used in the listed applications; however, the listed specifications are representative. Identification of actual materials is available in VEGP quality assurance records.

The materials utilized conform to the applicable American Society of Mechanical Engineers (ASME) code rules.

The welding materials used for joining the ferritic base materials of the RCPB conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining the austenitic stainless steel base materials of the RCPB conform to ASME Material Specifications SFA 5.4 and 5.9. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. They are qualified to the requirements of the ASME Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant[a]

The reactor coolant system (RCS) chemistry specifications are given in table 5.2.3-3.

The RCS water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications.

The chemical and volume control system (CVCS) provides a means for adding chemicals to the RCS which perform the following functions:

- Control the pH of the coolant during prestartup testing and subsequent operation.
- Scavenge oxygen from the coolant during heatup.
- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during all power operations subsequent to startup.

[a] The Water Chemistry Control Program is credited as a license renewal aging management program (see subsection 19.2.28).

The normal limits for chemical additives and reactor coolant impurities for power operation are shown in table 5.2.3-3.

The pH control chemical utilized is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/ Inconel systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and transferred to the chemical additive tank. Reactor makeup water is then used to flush the solution to the suction header of the charging pumps. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizer.

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent.

The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen also reacts with oxygen and nitrogen introduced into the RCS as impurities under the impetus of core radiation. Sufficient partial pressure of hydrogen is maintained in the volume control tank so that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A self-contained pressure control valve maintains a minimum pressure in the vapor space of the volume control tank.

This can be adjusted to provide the correct equilibrium hydrogen concentration.

Boron, in the chemical form of boric acid, is added to the RCS for long-term reactivity control of the core.

A soluble zinc compound may be added to the reactor coolant as a means to reduce radiation fields within the primary system. The zinc used may be either natural zinc or zinc depleted of ^{64}Zn. When used, the target system zinc concentration is normally maintained to a concentration no greater than 40 ppb.

Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

All of the ferritic low-alloy and carbon steels which are used in principal pressure-retaining applications have corrosion-resistant cladding on all surfaces that are exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation-hardened stainless steel. These corrosion-resistant cladding materials may be subjected to the ASME code required postweld heat treatment for ferritic base materials.

Ferritic low-alloy and carbon steel nozzles have safe ends of either stainless steel wrought materials, stainless steel weld metal analysis A-7 (designated A-8 in the 1974 edition of the ASME code), or nickel-chromium-iron alloy weld metal F-Number 43. The latter buttering material requires further safe ending with austenitic stainless steel base material after completion of the postweld heat treatment when the nozzle is larger than a 4-in. nominal inside diameter and/or the wall thickness is greater than 0.531 in.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure-retaining applications are used in the solution annealed condition. These heat treatments are as required by the material specifications.

During subsequent fabrications, these materials are not heated above 800°F other than locally by welding operations. The solution-annealed surge line material is subsequently formed by hot bending followed by a resolution annealing heat treatment.

Components employing stainless steel sensitized in the manner expected during component fabrication and installation operate satisfactorily under normal plant chemistry conditions in pressurized water reactor (PWR) systems because chlorides, fluorides, and oxygen are controlled to very low levels.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, all of the materials listed in table 5.2.3-1 which are used in principal pressure-retaining applications and are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the RCPB is either reflective stainless steel type or made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Section 1.9 indicates the degree of conformance with Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel.

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., only materials that are compatible with the coolant are used. These are as shown in table 5.2.3-1. Ferritic materials exposed to coolant leakage can be readily observed as part of the inservice visual and/or nondestructive inspection program to ensure the integrity of the component for subsequent service.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the RCPB components meet the requirements of the ASME Code, Section III, Paragraphs NB, NC, and ND-2300, as appropriate.

The fracture toughness properties of the reactor vessel materials are discussed in section 5.3.

Limiting steam generator and pressurizer reference temperature for a nil ductility transition (RT_{NDT}) temperatures are guaranteed at 60°F for the base materials and the weldments.

These materials meet the 50-ft-lb absorbed energy and 35-mils lateral expansion requirements of the ASME Code, Section III, at 120°F. The actual results of these tests are provided in the ASME material data reports which are supplied for each component and submitted to the owner at the time of shipment of the component.

Calibration of temperature instruments and Charpy impact test machines are performed to meet the requirements of the ASME Code, Section III, Paragraph NB-2360.

Westinghouse has conducted a test program to determine the fracture toughness of low-alloy ferritic materials with specified minimum yield strengths greater than 50,000 psi to demonstrate compliance with Appendix G of the ASME Code, Section III. In this program, fracture toughness properties were determined and shown to be adequate for base metal plates and forgings, weld metal, and heat-affected zone metal for higher strength ferritic materials used for components of the RCPB. The results of the program are documented in reference 1, which has been submitted to the Nuclear Regulatory Commission (NRC) for review.

5.2.3.3.2 Control of Welding

All welding is conducted utilizing procedures qualified according to the rules of Sections III and IX of the ASME Code. Control of welding variables, as well as examination and testing, during procedure qualification and production welding is performed in accordance with ASME Code requirements.

Westinghouse practices for storage and handling of welding electrodes and fluxes comply with ASME Code, Section III, Paragraph NB-2400.

Section 1.9 indicates the degree of conformance of the ferritic materials components of the RCPB with Regulatory Guides 1.34, Control of Electroslag Weld Properties, 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components, 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel, and 1.71, Welder Qualification for Areas of Limited Accessibility.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Paragraphs 5.2.3.4.1 through 5.2.3.4.5 address Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, and present the methods and controls utilized by Westinghouse to avoid sensitization and prevent intergranular attack (IGA) of austenitic stainless steel components. Also, section 1.9 indicates the degree of conformance with Regulatory Guide 1.44.

5.2.3.4.1 Cleaning and Contamination Protection Procedures

It is required that all austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules covering these controls are stipulated in Westinghouse process specifications. As applicable, these process specifications supplement the equipment specifications and purchase order requirements of every individual austenitic stainless steel component or system which Westinghouse procures for the VEGP nuclear steam supply systems (NSSSs), regardless of the ASME Code classification. Westinghouse process specifications are also given to Bechtel, the architect-engineer, and to Georgia Power Company for recommended use within their scope of supply and activity.

The process specifications that define these requirements and that follow the guidance of the American National Standards Institute (ANSI) N-45 committee specifications include the following:

Number	Process Specification
82560HM	Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steels.
83336KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment.
83860LA	Requirements for Marking of Reactor Plant Components and Piping.
8435OHA	Site Receiving Inspection and Storage Requirements for Systems, Material, and Equipment.
8435INL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials.
853I0QA	Packaging and Preparing Nuclear Components for Shipment and Storage.
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS.
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures.
597760	Cleanliness Requirements During Storage Construction, Erection, and Startup Activities of Nuclear Power System.

Section 1.9 indicates the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

5.2.3.4.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in tables 5.2.3-1 and 5.2.3-2 are utilized in the final heat-treated condition required by the respective ASME Code, Section II materials specification for the particular type or grade of alloy.

5.2.3.4.3 Material Testing Program

Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products that do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. When testing is required, the tests are performed in accordance with ASTM A 262, Practice A or E, as amended by Westinghouse Process Specification 84201MW.

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized austenitic stainless steels are subject to IGA provided that three conditions are present simultaneously. These are:

- An aggressive environment; e.g., an acidic aqueous medium containing chlorides or oxygen.
- A sensitized steel.
- A high temperature.

If any one of the three conditions described above is not present, IGA will not occur. Since high temperatures cannot be avoided in all components in the NSSS, reliance is placed on the elimination of the other two conditions to prevent IGA on wrought stainless steel components.

This is accomplished by:

- Control of primary water chemistry to ensure a benign environment.
- Utilization of materials in the final heat-treated condition and the prohibition of subsequent heat treatments in the 800°F and 1500°F temperature range.
- Control of welding processes and procedures to avoid heat-affected zone sensitization.
- Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and the reactor internals do not result in the sensitization of heat-affected zones.

Further information on each of these steps is provided in the following paragraphs.

The water chemistry in the RCS is controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations are 0.1 ppm and 0.15 ppm, respectively. Table 5.2.3-3 lists the recommended reactor coolant water chemistry specifications. The precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage are stipulated in the appropriate process specifications. The use of hydrogen overpressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long-term exposure of severely sensitized stainless steels to reactor coolant environments in early Westinghouse PWRs has not resulted in any sign of IGA. Reference 2 describes the laboratory experimental findings and reactor operating experience. The additional years of operations since the issuing of reference 2 have provided further confirmation of the earlier conclusions that severely sensitized stainless steels do not undergo any IGA in Westinghouse PWR coolant environments.

Although there is no evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the NSSS components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock is used for components that are part of:

- The RCPB.
- Systems required for reactor shutdown.
- Systems required for emergency core cooling.

- Reactor vessel internals relied upon to permit adequate core cooling for normal operation or under postulated accident conditions.

The wrought austenitic stainless steel stock is utilized in one of the following conditions:

- Solution annealed and water quenched.
- Solution annealed and cooled through the sensitization temperature range within less than approximately 5 min.

It is generally accepted that these practices will prevent sensitization. Westinghouse has verified this by performing corrosion tests on wrought material as it was received.

The heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800°F to 1500°F. However, severe sensitization (i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion) can be avoided by controlling welding parameters and welding processes. The heat input and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments.

Heat input is calculated according to the formula:

$$H = \frac{(E)(I)(60)}{S}$$

where:

H = joules/in.

E = volts.

I = amperes.

S = travel speed (in./min).

Of 25 production and qualification weldments tested, representing all major welding processes and a variety of components and incorporating base metal thicknesses from 0.10 to 4.0 in., only portions of two were severely sensitized. Of these, one involved a heat input of 120,000 J, and the other involved a heavy socket weld in relatively thin-walled material. In both cases, sensitization was caused primarily by high-heat inputs relative to the section thickness. In only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment. The component has been redesigned, and a material change has been made to eliminate this condition.

The heat input in all austenitic pressure boundary weldments has been controlled by:

- Prohibiting the use of block welding.
- Limiting the maximum interpass temperature to 350°F.
- Exercising approval rights on all welding procedures.

5.2.3.4.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

As described in the previous section, it is not normal Westinghouse practice to expose unstabilized austenitic stainless steels to the sensitization range of 800°F to 1500°F during fabrication into components. If during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, 800°F to 1500°F, the material may be tested in

accordance with ASTM A 262, as amended by Westinghouse Process Specification 84201MW, to verify that it is not susceptible to IGA, except that testing is not required for:

A. Cast metal or weld metal with a ferrite content of 5 percent or more.

B. Material with a carbon content of 0.03 percent or less that is subjected to temperatures in the range of 800°F to 1500°F for less than 1 h.

C. Material exposed to special processing, provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

If it is not verified that such material is not susceptible to IGA, the material is resolution annealed and water quenched or rejected.

5.2.3.4.6 Control of Welding

The following paragraphs address Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, and present the methods used, and the verification of these methods, for austenitic stainless steel welding.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there is insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0- and 3-percent delta ferrite.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated, or stamped in accordance with the ASME Code, Section III, Class 1 and 2, and core support components. Delta ferrite control is appropriate for the above welding requirements, except where no filler metal is used or where for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedure and welder qualification in accordance with Section III and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5-percent delta ferrite (the equivalent ferrite number may be substituted for percent delta ferrite) as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of Sections III and IX.

The results of all the destructive and nondestructive tests are reported in the procedure qualification record in addition to the information required by Section III.

The starting welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7 (designated A-8 in the 1974 edition of the ASME Code), type 308 or 308L for all applications. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA 5.9, and are procured to contain not less than 5-percent delta ferrite according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA 5.4 or 5.9 and are procured in a wire-flux combination to be capable of providing not less than 5-percent delta ferrite in the deposit according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heat and lots of starting welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of starting materials, qualification records, and welding parameters. Welding systems are also subject to:

- Quality assurance audit including calibration of gauges and instruments.
- Identification of starting and completed materials.
- Welder and procedure qualifications.
- Availability and use of approved welding and heat-treating procedures.
- Documentary evidence of compliance with materials, welding parameters, and inspection requirements.

Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules.

To ensure the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in reference 3. This program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position on Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which do support the hypothesis presented in reference 3, are summarized in reference 4.

Section 1.9 indicates the degree of conformance of the austenitic stainless steel components of the RCPB with Regulatory Guides 1.34, Control of Electroslag Properties, and 1.71, Welder Qualification for Areas of Limited Accessibility.

5.2.3.5 References

1. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.
2. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.
3. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

4. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY[a]

Inservice inspection and testing of Class 1 pressure-retaining components such as vessels, piping, pumps, valves, bolting, and supports within the reactor coolant pressure boundary shall be performed in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Code including any applicable addenda in accordance with 10 CFR 50.55a(g)(4)(ii) (specific edition and any applicable addenda of the code will be delineated in each program), with certain exceptions whenever specific written relief is granted by the Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.55a(a)3 and 10 CFR 50.55a(g)(6)(i). The inservice testing of pumps and valves in accordance with the requirements of Articles IWP and IWV of the code is discussed in subsection 3.9.6. Class 2 and 3 components examinations are addressed in section 6.6.

The preservice inspection program requirements for each unit were completed prior to the commercial operation date for each of the respective units. The preservice inspection program for Unit 1 complied with the ASME Code, Section XI, 1980 Edition including addenda through Winter 1980, except that reactor pressure vessel examinations were performed using the 1980 Edition including addenda through Winter 1981. The preservice inspection program for Unit 2 complied with the ASME Code, Section XI, 1983 Edition including addenda through Summer 1983, except that reactor pressure vessel examinations were performed using the 1980 Edition including addenda through Winter 1981. Certain preservice inspection requirements of the ASME Code, Section XI were determined to be impractical and relief requests were granted by the NRC, pursuant to 10 CFR 50.55a(g)(i). The relief requests were supported by information pursuant to 10 CFR 50.55a (a) (3). The inservice inspection program and inservice test program were submitted to the NRC prior to commercial operation. These programs comply with applicable inservice inspection provisions of 10 CFR 50.55a(g) and the NRC guidelines attached as an appendix to section 121.0 of review questions entitled "Guidance for Preparing Preservice and Inservice Inspection Programs and Relief Requests Pursuant to 10 CFR 50.55a(g)." Where compliance with code requirements is not practical, relief requests have been submitted to the NRC for review and approval. The inservice programs detail the areas subject to examination and method, extent, and frequency of examinations. Additionally, component supports and snubber testing requirements are included in the inspection programs.

5.2.4.1 System Boundary Subject to Inspection

In addition to the reactor pressure vessel, all Class 1 components such as vessels, piping, pumps, valves, bolting, and supports shall be inspected to the extent practical, in accordance with Article IWB of ASME Code, Section XI. Class 1 pressure-retaining components and their specific boundaries are identified in the inspection plan documents.

[a] The Inservice Inspection Program is credited as a license renewal aging management program (see subsection 19.2.13).

5.2.4.2 Arrangement and Accessibility

The physical arrangement of components was designed to allow personnel and equipment access to the extent practical to perform the required inservice examinations. Removable insulation and shielding was provided on those piping systems requiring volumetric and surface examination. Temporary or permanent working platforms, scaffolding, and ladders are provided to facilitate access to piping welds.

An inservice inspection design review was undertaken to identify exceptions to the access requirements of the code with subsequent design modifications and/or inspection technique development to ensure code compliance to the extent practical. Additional exceptions may be identified and reported to the NRC after plant operation, as specified in 10 CFR 50.55a(g)(5)(iv).

Space has been provided to handle and store insulation, structural members, shielding, and other materials related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment were installed at appropriate locations. The reactor pressure vessel (RPV) inspections are performed primarily from the vessel internal surfaces. Other areas of the RPV such as the closure head are accessible from the outer surfaces of the vessel for inspection. Closure studs, nuts, and washers are removed to a dry location for direct inspection.

5.2.4.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination techniques, procedures, and special techniques are in accordance with the requirements of subarticle IWA-2200 and table IWB-2500-1 of the ASME Code, Section XI except where compliance with code requirements is not practical and relief has been requested from the NRC. Liquid penetrant methods and/or magnetic particle methods are used for surface examinations. Radiography and/or ultrasonic techniques, whether manual or remote, are used for volumetric examinations. A special vessel inspection tool is used to inspect the RPV welds from the vessel internal surfaces. Welds located in the reactor vessel beltline region are examined to meet the requirements of Regulatory Guide 1.150 to the extent practicable. Other RPV welds are examined to meet the requirements of Regulatory Guide 1.150, with the exception of the near surface examination, to the extent practicable. Other examination techniques may be used provided that the results are demonstrated to be equivalent or superior to the above techniques.

5.2.4.4 Inspection Intervals

Inspection intervals are as defined in subarticles IWA-2400 and IWB-2400 of ASME Code, Section XI. The interval may be extended by as much as 1 year to permit inspections to be concurrent with plant outages. It is intended that inservice examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements are in accordance with subarticle IWB-2500 and table IWB-2500-1 of ASME Code, Section XI. The preservice examinations complied with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Examination results are evaluated in accordance with IWB-3000, with flaw indications in accordance with IWB-3400 and table IWB-3410. Repair procedures are in accordance with IWB-4000 of ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests comply with IWA-5000 and IWB-5000 of ASME Code, Section XI.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary (RCPB) leakage detection systems monitor leaks from the reactor coolant and associated systems. These systems provide information which permit the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.

5.2.5.1 Design Bases

The leak detection systems are designed in accordance with the requirements of 10 CFR 50 and the general design criterion 30 to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage. The systems conform with Regulatory Guide 1.45. Main systems that monitor the environmental condition of the containment include the sump level monitoring system, the airborne particulate radioactivity monitoring systems, and the containment fan cooler condensate measuring system. In addition to the above systems, the humidity, temperature, pressure, and radiogas monitors provide indirect indication of leakage to the containment.

Associated systems and components connected to the reactor coolant system have intersystem leakage monitoring devices.

These leakage detection systems are qualified for all seismic events not requiring a shutdown. The airborne radioactivity monitoring system is qualified for a safe shutdown earthquake (SSE).

5.2.5.1.1 Leakage Classification

RCPB leakage is classified as either identified or unidentified leakage. Identified leakage includes: leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank; or leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB; or leakage into auxiliary systems and secondary systems. Unidentified leakage is all other leakage.

5.2.5.1.2 Limits for Reactor Coolant Leakage

Limits for reactor coolant leakage are identified in the Technical Specifications.

5.2.5.2 Identified Intersystem Leakage Detection

Unidentified leakage into closed primary systems is directed to the reactor coolant drain tank or pressurizer relief tank. Identified leakage, such as pump seal or valve packing leakage, is directed to the reactor coolant drain tank where it is monitored by tank pressure, temperature, level, and flow instrumentation on the reactor coolant drain tank discharge lines.

Identified leakage, such as leakage past the pressurizer safety valves or power-operated relief valves (PORVs), is directed to the pressurizer relief tank. This leakage is monitored by temperature instrumentation in the piping system and tank pressure, temperature, and level instrumentation. Leakage collected in the pressurizer relief tank is directed to the reactor coolant drain tank for subsequent treatment and discharge.

An important identified leakage path for reactor coolant into other systems is through the steam generator tubes into the secondary side of the steam generator. Identified leakage to the steam generators is detected by means of the steam generator sample liquid or condenser air ejector radiation monitors. Two additional primary-to-secondary leak detection systems are also provided: a noble gas detector and a system utilizing N16 as the detection medium. The N16 detector is installed in the turbine building main steam pipe chase, between the two main steam pipes. The N16 leak monitor is independent of the primary loop fission and corrosion product radioactivity, so when the tube leak rates increase without considerable changes in the secondary side radioactivity levels, the system can still detect small leaks. The noble gas detector is located in the condenser steam jet air ejector discharge header immediately prior to the filtration unit. For details of these radiation monitors, see subsection 11.5.2.

Auxiliary systems connected to the RCPB incorporate design and administrative provisions that serve to limit leakage. These provisions include isolation valves designed for low seat leakage, periodic testing of RCPB check valves (paragraph 6.3.4.2), and inservice inspection (subsection 5.2.4 and section 6.6). Leakage is detected by increasing auxiliary system level, temperature, and pressure indications or lifting of relief valves accompanied by increasing values of monitored parameters in the relief valve discharge path. These systems are isolated from the RCS by normally closed valves and/or check valves.

5.2.5.2.1 Description and Operation of Identified Leak Detection System

A. Residual Heat Removal System (RHRS) (Suction Side)

   The RHRS is isolated from the RCS on the suction side by motor-operated valves HV-8701A/B and HV-8702A/B. Leakage past these valves is detected by lifting of relief valves PSV-8708A or PSV-8708B, accompanied by increasing pressurizer relief tank level, pressure, and temperature indications and alarms on the main control board.

B. Safety Injection System (SIS)/Accumulators

   The accumulators are isolated from the RCS by check valves 1204-U6-083 through -086 and 1204-U6-079 through -082. Leakage past these valves and into the accumulator subsystem is detected by redundant control room accumulator pressure and level indications and alarms.

C. SIS/RHR Discharge Subsystem

   The RHR pump portion of the SIS is isolated from the RCS by check valves 1204-U6-083 through -086, 1204-U6-147 through -150, 1204-U6-125 and -126, 1204-U6-128 and -129, and normally closed motor-operated valve HV-8840.

Leakage past these valves will eventually pressurize the RHR discharge header and the pump suction header through the normally open pump miniflow isolation valves FV-610 or FV-611. A continued increase in RHR pump discharge pressure will be indicated in the control room and ultimately result in lifting relief valves PSV-8708A and PSV-8708B in the suction header.

D. SIS/Safety Injection Pump Subsystem

The safety injection pump portion of the SIS is isolated from the RCS by check valves 1204-U4-083 through -086, 1204-U4-143 through -146, 1204-U6-124 through -127, 1204-U4-120 through -123, and normally closed motor- operated valves HV-8802A/B. Leakage past these valves will pressurize the safety injection pump discharge header, resulting in control room indication of increasing pressure and, eventually, lifting of relief valve PSV-8851 or PSV-8853A/B.

E. SIS/Centrifugal Charging Pump Subsystem

The charging pump subsystem of the SIS is isolated from the RCS by check valves 1204-U4-026 through -029, 1204-U6-013, and motor-operated valves HV-8801A/B. Leakage past valves HV-8801A/B is not possible, since the valve inlets are pressurized by the operating charging pump.

F. Head Gasket Monitoring Connections

The reactor vessel flange and head are sealed by two metallic O-rings. These gaskets are of the hollow self-energizing type in which pressure of the fluid being sealed enters the interior of the gasket. The O-rings are fastened to the closure head by a mechanical connection to facilitate removal.

Seal leakage is detected by means of two leak-off connections: one between the inner and outer ring, and one outside the outer O-ring. A manual isolation valve is installed just outside the missile barrier of each leak-off line. Downstream of these valves the lines are headered before being routed to the reactor coolant drain tank in the waste processing system. An air-operated isolation valve, actuated from the control board, is installed in the common line. During normal plant operation, the leak-off piping is aligned such that leakage across the inner O-ring passes through valves 1201-U4-088 and HV-8032 into the drain tank. A surface mounted, resistance temperature detector, installed on the bottom of the common pipe, signals leakage at an alarm setpoint. A blind flanged branch line containing isolation valve 1201-U4-089 is provided to confirm and establish the magnitude of the leakage.

Once inner O-ring leakage is discovered, valve 1201-U4-087 should be opened and valve 1201-U4-088 closed so that possible leakage across the second O-ring would be monitored.

In addition, during plant refueling operations both the inner and outer reactor vessel flange leak-off valves are closed. This prevents possible gas leakage from the reactor coolant drain tank to the containment atmosphere. Refer to drawings 1X4DB111 and 2X4DB111 for the flow diagram representation.

The reactor vessel is the only flanged vessel within the RCPB that is provided with leak-off collection provisions.

Leakage past the reactor vessel head gaskets results in temperature indication and alarm in the control room.

G. Component and Auxiliary Component Cooling Water Systems

Leakage from the RCS to the component cooling water (CCW) and auxiliary component cooling water (ACCW) systems, which service all RCPB associated components that require cooling, is detected by the CCW and ACCW radioactivity monitoring system (subsection 11.5.2) and/or increasing surge tank level. Components serviced by these auxiliary cooling systems include: reactor coolant pump thermal barriers, RHR heat exchangers, letdown line heat exchangers, reactor coolant pump seal water heat exchangers.

5.2.5.3 Unidentified Leakage Detection

Normally, unidentified leakage from the RCS is very low. The RCS is an all-welded system, with the exception of the connections on the pressurizer safety valves, reactor vessel head, pressurizer and steam generator manways, and reactor vessel head vent, which are flanged.

In general, valves in the RCS that are 2 in. and under are of the packless type. All valves larger than 2 in. have dual packing with a leak-off connection to the reactor coolant drain tank between the two packings or a reduced packing configuration with the valve stem leakoff line capped.

Primary indications of unidentified coolant leakage to the containment are provided by air particulate radioactivity monitors, gaseous radioactivity monitors, fan cooler condensate flow monitors, and containment sump level monitors.

In normal operation, these primary monitors show a background level that is indicative of the normal level of unidentified leakage inside the containment. Variations in airborne radioactivity or specific humidity above the normal level signify an increase in unidentified leakage rates and signal to the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage.

RCS unidentified leakage may also be indicated by increasing charging pump flowrate compared with normal RCS inventory changes and by unscheduled increases in reactor makeup water usage.

Reactor coolant inventory monitoring provides an indication of system leakage. Net level changes in the pressurizer and volume control tank are indicative of system leakage, since the chemical and volume control system is a closed loop connected to the RCS. Monitoring net makeup to the chemical and volume control system, as well as net collected leakage, provides an important method of obtaining information for use in establishing a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can be indicative of increased system leakage.

The sensitivity and response time of the detection equipment for unidentified leakage is such that a leakage rate, or its equivalent, of 1 gal/min can be detected in approximately 1 h.

The above methods are supplemented by visual and ultrasonic inspections of the RCPB during plant shutdown periods, in accordance with the inservice inspection program (subsection 5.2.4).

5.2.5.3.1 Description and Operation of Main Unidentified Leak Detection Systems

Systems employed for detecting leakage to the containment from unidentified sources are:

- Containment airborne particulate radioactivity monitor.
- Containment gaseous radioactivity monitor.

- Containment air cooler condensate flow monitor.
- Containment sump level monitor.

Additionally, humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment.

A. Containment Airborne Particulate Radioactivity Monitoring System

An air sample is drawn outside the containment into a closed system by a sample pump and is then consecutively passed through a particulate filter with detector and a gaseous monitor chamber with detector. The filter collects 99 percent of the particulate matter greater than 1 mm in size. The sample transport system includes:

- A pump to obtain the air sample.
- A flow control valve to provide flow adjustment.
- A flow meter to indicate the flowrate.
- A flow alarm assembly to provide high- and low-flow alarm signals.

The particulate filter is continuously monitored by a scintillation crystal with a photomultiplier tube that provides an output signal proportional to the activity collected on the filter. The particulate monitor has a minimum detectable concentration of 10^{-11} $\mu Ci/cm^3$and a range of 10^{-11} to 10^{-6} $\mu Ci/cm^3$. More details concerning the particulate monitors can be found in subsection 11.5.2.

Particulate activity can be correlated with the coolant fission and corrosion product activities. Any increase of more than two standard deviations above the count rate for background would indicate a possible leak. The total particulate activity concentration above background, due to an abnormal leak and natural decay, increases almost linearly with time for the first several hours after the beginning of a leak. As shown in figure 5.2.5-1, with 0.01-percent failed fuel, containment background airborne particulate radioactivity equivalent to 10^{-3} percent/day, and a partition factor equal to 0.001, a 1-gal/min leak would be detected in approximately 1 h. Larger leaks would be detected in proportionately shorter times (exclusive of sample transport time, which remains constant). The detection capabilities and response times are shown in figure 5.2.5-1.

The activity is indicated on displays and electronically recorded. High-activity alarm indications are displayed on the radiation monitoring cabinets. Local alarms provide operational status of supporting equipment such as pumps, motors, and flow and pressure controllers.

The leakage flowrate can be determined by performing a water inventory balance of the reactor coolant system when the count rate indicates a possible leak as explained above.

B. Containment Gaseous Radioactivity Monitoring System

The containment gaseous radioactivity monitor determines gaseous radioactivity in the containment by monitoring continuous air samples from the containment atmosphere. After passing through the gas monitor, the sample is returned via the closed system to the containment atmosphere.

Each sample is continuously mixed in a fixed, shielded volume where its activity is monitored. The monitor has a range of 10^{-7} to 10^{-2} µCi/cm^3 and a minimum detectable concentration of 5 x 10^{-7} µCi/cm^3.

The containment gaseous radioactivity monitors are described in subsection 11.5.2.

Gaseous radioactivity can be correlated with the gaseous activity of the reactor coolant. Any increase more than two standard deviations above the count rate for background would indicate a possible leak. The total gaseous activity level above background increases almost linearly for the first several hours after the beginning of the leak. As specified in figure 5.2.5-1, with 0.01-percent failed fuel, containment background airborne gaseous radioactivity equivalent to 1 percent/day, and a partition factor equal to 1, a 1-gal/min leak would be detected in approximately 1 h. Larger leaks would be detected in proportionately shorter times (exclusive of the sample transport time, which remains constant). The detection capabilities and response times are shown in figure 5.2.5-1.

The detector outputs are transmitted to the radiation monitoring system cabinets in the control room, where the activity is indicated by displays and electronically recorded. High-activity alarm indications are displayed on the control board annunciator in addition to the radiation monitoring system cabinets. Local alarms annunciate the operational status of the supporting equipment.

The leakage flowrate can be determined by performing a water inventory balance of the reactor coolant system when the count rate indicates a possible leak as explained above.

The containment purge system radioactivity monitors (subsection 11.5.2) serve as backup to the containment air particulate and gaseous airborne radioactivity monitoring system while the purge is in operation.

The containment purge monitors function in the same manner as the containment air particulate and gaseous radioactivity monitors, except that the purge monitors sample from the containment purge exhaust line.

C. Containment Air Cooler Condensate Monitoring System

The condensate monitoring system permits measurements of the liquid runoff from the containment cooler units. It consists of a containment cooler drain collection header, a vertical standpipe, valving, and standpipe level instrumentation. The condensation from the containment coolers flows via the collection header to the vertical standpipe. A differential pressure transmitter provides standpipe level signals. The system provides measurements of low leakages by monitoring standpipe level increase versus time.

The condensate flowrate is a function of containment humidity, nuclear service cooling water (NSCW) temperature, and containment purge rate. The water vapor dispersed by a 1-gal/min leak is usually greater than the water vapor brought in with the outside air. Air brought in from the outside is heated to 60°F before it enters the containment.

After the air enters the containment, it mixes with the containment atmosphere and is heated to between 100°F and 120°F while the relative humidity drops. The most important factor in condensing the water vapor is the temperature of the NSCW which is supplied to the containment coolers. This water is assumed to vary in temperature between 35°F and 95°F.

Drainage flowrate from the units due to normal condensation is calculated for the ambient (background) atmospheric conditions present within the containment. With the initiation of an additional or abnormal leak, the containment atmosphere humidity and condensation runoff rate both begin to increase, the water level rises in the vertical pipe, and the high condensate flow alarm is actuated. Level changes of as little as 0.25 in. in the cooler condensate standpipes can be detected.

Figure 5.2.5-1 shows the detection capabilities of the system for various conditions. Normal background leakage increases containment humidity to the point where the condensation rate will increase, which improves the detection capabilities of this system. As shown in figure 5.2.5-1, a sensitivity of 1 gal/min in approximately 1 h can be achieved when supplying cold NSCW to the containment coolers or with the initial background leakage.

The rate of leakage can be determined when the precise NSCW, outside air, and containment air temperatures, and the outside relative humidity are known.

D. Containment Sump Level Monitoring System

Since a leak in the primary system would result in reactor coolant flowing into the containment normal or reactor cavity sumps, leakage would be indicated by a level increase in the sump. Indication of increasing sump level is transmitted from the sump to the control room level indicator by means of a sump level transmitter. The system provides measurements of low leakages by monitoring level increase versus time.

The detection capabilities of the containment normal sump and reactor cavity sump are shown in figure 5.2.5-1, assuming that the water from the leak is collected in the sump.

The actual reactor coolant leakage rate can be established from the increase above the normal rate of change of sump level. A check of other instrumentation would be required to eliminate possible leakage from nonradioactive systems as a cause of an increase in sump level. The leakage rate can also be determined from the frequency of sump pump operation.

Under normal conditions, the containment normal sump pumps operate infrequently and reactor cavity sump pump operates very infrequently. Gross leakage can be surmised from unusual frequency of pump operation. Sump level and pump running indication are provided in the control room to alert the operators.

5.2.5.3.2 Additional Unidentified Leakage Detection Methods

Other methods available for detecting leakage are:

A. Charging Pump Operation

During normal operation, one of the charging pumps is in operation. If a gross increase in reactor coolant leakage occurs, the flowrate of the charging pump would increase, indicating leakage from the RCS. This leakage must be sufficient to cause a decrease in pressurizer or volume control tank level that is within the sensitivity range of the level indicators. The flowrate of the charging

pump would automatically increase to try to maintain pressurizer level. Charging pump discharge flow indication is provided in the control room.

The leakage rate can be determined by the amount that the charging pump flowrate increases above the letdown flowrate to maintain constant pressurizer level. Any significant increase in the charging flowrate is a possible indication of a leak.

B. Containment Humidity Monitoring System

The containment humidity system, utilizing temperature-compensated humidity detectors, is provided to determine the water-vapor content of the containment atmosphere. An increase in the humidity of the containment atmosphere indicates release of water within the containment. The range of the containment humidity measuring system is 5- to 99-percent relative humidity at 80°F with a temperature range of 40 to 120°F. The accuracy of the humidity detectors is ±3 percent.

The response of the containment humidity under various outside air conditions and no leakage falls within the extremes shown in figure 5.2.5-1. The humidity monitor supplements the condensate monitor. It is most sensitive under conditions when there is no condensation.

A rapid increase of humidity over the background level by more than 10 percent can be taken as a probable indication of a leak.

The leakrate can be determined when the outside air temperature and humidity and the containment atmosphere temperature are known.

C. Liquid Inventory

The operators can surmise gross leakage from changes in the reactor coolant inventory. Noticeable decreases in the pressurizer level not associated with known changes in operation are investigated. Likewise, makeup water usage information which is available from the plant computer is checked frequently for unusual makeup rates not due to plant operations.

5.2.5.4 <u>Safety Evaluation</u>

The leak detection system has no safety design basis; however, the containment atmosphere radioparticulate and radiogas monitors are qualified for an SSE per the recommendation of Regulatory Guide 1.45.

5.2.5.5 <u>Tests and Inspections</u>

Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, frictional tests, and channel checks. A description of calibration and maintenance procedures and frequencies for the containment radioactivity monitoring system is presented in subsection 11.5.2.

The humidity detector and condensate measuring system are also periodically tested to ensure proper operation and verify sensitivity.

Inservice inspection criteria, the equipment used, procedures involved, the frequency of testing, inspection, surveillance, and examination of the structural and leaktight integrity of RCPB components are described in detail in subsection 5.2.4.

5.2.5.6 Instrumentation Applications

The following indications are provided in the control room to allow operating personnel to monitor for leakage:

A. Containment air particulate monitor - air particulate activity.

B. Containment gaseous activity monitor - gaseous activity.

C. Containment cooler condensate monitoring system - standpipe level.

D. Containment humidity measuring system - containment humidity.

E. Containment normal sump level and reactor cavity sump level.

F. Gross leakage detection methods - charging pump flowrate, letdown flowrate, pressurizer level, and reactor coolant temperatures are available for the charging pump flow method. Containment sump levels and pump operation are available for the sump pump operation method. Total makeup waterflow is available from the plant computer for liquid inventory.

TABLE 5.2.1-1

APPLICABLE CODE ADDENDA FOR RCS COMPONENTS

Reactor vessel	ASME III, 1971 Edition through Summer 1972 Addenda
Steam generator	ASME III, 1971 Edition through Summer 1972 Addenda
Pressurizer	ASME III, 1971 Edition through Summer 1972 Addenda
Control rod drive mechanism (CRDM) housing	ASME III, 1974 Edition through Summer 1975 Addenda
CRDM head adapter	ASME III, 1971 Edition through Summer 1972 Addenda
Reactor coolant pump	ASME III, 1971 Edition through 1972 Winter Addenda
Reactor coolant pipe	ASME III, 1974 Edition through Winter 1975 Addenda
Surge line	ASME III, 1974 Edition through Winter 1975 Addenda
Valves	
Pressurizer safety	ASME III, 1971 Edition through Winter 1972 Addenda
Motor operated	ASME III, 1974 Edition through Summer 1974 Addenda
Manual (3 in. and larger)	ASME III, 1974 Edition through Summer 1974 Addenda
Control	ASME III, 1971 Edition through Summer 1972 Addenda

TABLE 5.2.3-1 (SHEET 1 OF 5)

PRIMARY AND AUXILIARY COMPONENTS MATERIAL SPECIFICATIONS

Reactor Vessel Components

Shell and head plates (other than core region)	SA-533, Grade A, B, or C, Class 1 or 2 (vacuum treated)
Shell plates (core region)	SA-533, Grade A or B, Class 1 (vacuum treated)
Shell, flange, and nozzle forgings, nozzle safe ends	SA-508, Class 2 or 3; SA-182; Grade F304 or F316
Control rod drive mechanism (CRDM) and/or ECCS appurtenances, upper head	SB-166 or SB-167 and SA-182 Grade F304
Instrumentation tube appurtenances, lower head	SB-166 or SB-167 and SA-182, Grade F304, F304L, or F316
Closure studs, nuts, washers, inserts, and adaptors	SA-540, Class 3, Grade B23 or B24
Core support pads	SB-166 with carbon less than 0.10 percent
Monitor tubes and vent pipe	SA-312 or SA-376, Grade TP304 or TP316 or SB-166, SB-167, or SA-182, Grade F316
Vessel supports, seal ledge, and head lifting lugs	SA-516, Grade 70 (quenched and tempered) or SA-533, Grade A, B, or C, Class 1 or 2 (Vessel supports may be of weld metal buildup of equivalent strength of the nozzle material.)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43

TABLE 5.2.3-1 (SHEET 2 OF 5)

Steam Generator Components	
Pressure plates	SA-533, Grade A, B, or C, Class 1 or 2
Pressure forgings (including nozzles and tube sheet)	SA-508, Class 1, 2, 2a, or 3
Nozzle safe ends	Stainless Steel Weld Metal Analysis A-8
Channel heads	SA-533, Grade A, B, or C, Class 1 or 2 or SA-216, Grade WCC
Tubes	SB-163 (Ni-Cr-Fe annealed)
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure studs/nuts	SA-193, Grade B7/SA-194 Gr 7
Pressurizer Components	
Pressure plates	SA-533, Grade A, Class 2
Pressure forgings	SA-508, Class 2 or 2a
Nozzle safe ends	SA-182, Grade F316L
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure studs/nuts	SA-193, Grade B7/SA-194 Gr 7
Reactor Coolant Pump	
Pressure forgings	SA-182, Grade F304, F316, F347, or F348
Pressure casting	SA-351, Grade CF8, CF8A, or CF8M
Tube and pipe	SA-213; SA-376 or SA-312, Seamless, Grade TP304 or TP316

TABLE 5.2.3-1 (SHEET 3 OF 5)

Pressure plates	SA-240, Type 304 or 316
Bar material	SA-479, Type 304 or 316
Closure bolting	SA-193, SA-320, SA-540, or SA-453, Grade 660
Flywheel	SA-533, Grade B, Class 1
<u>Reactor Coolant Piping</u>	
Reactor coolant pipe	SA-351, Grade CF8A Centrifugal Casting
Reactor coolant fittings, branch nozzles	SA-351, Grade CF8A and SA-182, (Code Case 1423-2) Grade 316N
Surge line	SA-376, Grade TP304, TP316, or F304N
Auxiliary piping	SA-312 and SA-376 Grades TP304 and TP316 to ANSI B36.10 or B36.19
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5
<u>Full-Length CRDM</u>	
Latch housing	SA-182, Grade F304 or SA-351, Grade CF8
Rod travel housing	SA-182, Grade F304 or SA-336 Class 304
Cap	SA-479, Type 304
Welding materials	Stainless Steel Weld Metal Analysis A-8
<u>Valves</u>	
Bodies	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M

TABLE 5.2.3-1 (SHEET 4 OF 5)

Bonnets	SA-182, Grade F316 or SA-351, Grade CF8 or CF8M
Discs	SA-182, Grade F316 or SA-564, Grade 630 or SA-351, Grade CF8 or CF8M
Stems	SA-182, Grade F316 or SA-564, Grade 630
Pressure retaining bolting	SA-453, Grade 660
Pressure retaining nuts	SA-453, Grade 660 or SA-194, Grade 6
Auxiliary Heat Exchangers	
Heads	SA-240, Type 304
nozzle necks	SA-182, Grade F304, SA-240 and 312, Type 304
Tubes	SA-213, Grade TP304; SA-249 Type 304
Tube sheets	SA-182, Grade F304; SA-240, Type 304 and SA-516 GR 70 clad with Stainless Steel Weld Metal Analysis A-8
Shells	SA-240 and SA-312, Grade TP304
Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells and heads	SA-240, Type 304 or SA-351 Grade CFA8, or SA264 (consisting of SA-537, Class 1 with Stainless Steel Weld Metal Analysis A-8 Cladding)
Flanges and nozzles	SA-182, Grade F304 and SA-105 or SA-350, Grade LF2 and LF3 with Stainless Steel Weld Metal Analysis A-8 Cladding
Piping	SA-312 Type 304 and SA-240, Grade TP304 or TP316

TABLE 5.2.3-1 (SHEET 5 OF 5)

Pipe fittings	SA-403, Grade WP304 Seamless
Closure bolting and nuts	SA-193, Grade B7 and SA-194, Grade 2H or Grade 7
<u>Auxiliary Pumps</u>	
Pump casing and heads	SA-351, Grade CF8 or CF8M; SA-182, Grade F304 or F316
Flanges and nozzles	SA-182, Grade F304 or F316; SA-403 Grade WP316L Seamless
Piping	SA-312, Grade TP304 or TP316 Seamless
Stuffing or packing box cover	SA-351, Grade CF8 or CF8M; SA-240 Type 304 or 304L or 316
Pipe fittings	SA-403, Grade WP316L Seamless
Closure bolting and nuts	SA-193, Grade B6, B7, or B8M; SA-194, Grade 2H or 8M; SA-453 Grade 660, and Nuts, SA-194 Grade 2H, 6 and 8M

TABLE 5.2.3-2

REACTOR VESSEL INTERNALS MATERIAL SPECIFICATIONS

Forgings	SA-182, Grade F304
Plates	SA-240, Type 304
Pipes	SA-312, Grade TP304 Seamless or SA-376, Grade TP304
Tubes	SA-213, Grade TP304 or ASTM A-511 Grade MT304, Code Case1618
Bars	SA-479, Type 304
Castings	SA-351, Grade CF8
Bolting	SA-193, Class 2 (65-90 YS/90 MTS) Code Case 1618 Inconel-750; SA-637, Grade 688, Type 2
Nuts	SA-193, Grade B8
Locking devices	SA-479, Type 304

TABLE 5.2.3-3 (SHEET 1 OF 2)

RECOMMENDED[a] REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical conductivity	Determined by the concentration of boric acid and alkali present. Expected range is <1 to 40 μmhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) and 10.5 (low boric acid concentration) at 25°C. Values will be 5.0 or greater at normal operating temperatures.
Oxygen[b]	0.10 ppm, maximum
Chloride[c]	0.15 ppm, maximum
Fluoride[c]	0.15 ppm, maximum
Hydrogen[d]	25 to 50 cm^3 (STP)/kg H_2O
Suspended solids[e]	0.2 ppm, maximum
pH control agent (Li_7OH)	Lithium is coordinated with boron in accordance with the principles of the EPRI PWR Primary Water Chemistry Guidelines, Volume 1.
Boric acid	Variable from 0 to 4000 ppm as boron
Zinc[g]	0.04 ppm, maximum
Silica[f]	1.0 ppm, maximum
Aluminum[f]	0.05 ppm, maximum
Calcium[f] + magnesium	0.05 ppm, maximum
Magnesium[f]	0.025 ppm, maximum

TABLE 5.2.3-3 (SHEET 2 OF 2)

a. Refer to the Technical Requirements Manual for required reactor coolant chemistry limits.

b. Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant by scavenging with hydrazine prior to plant operation above 250°F. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration will not exceed 0.005 ppm.

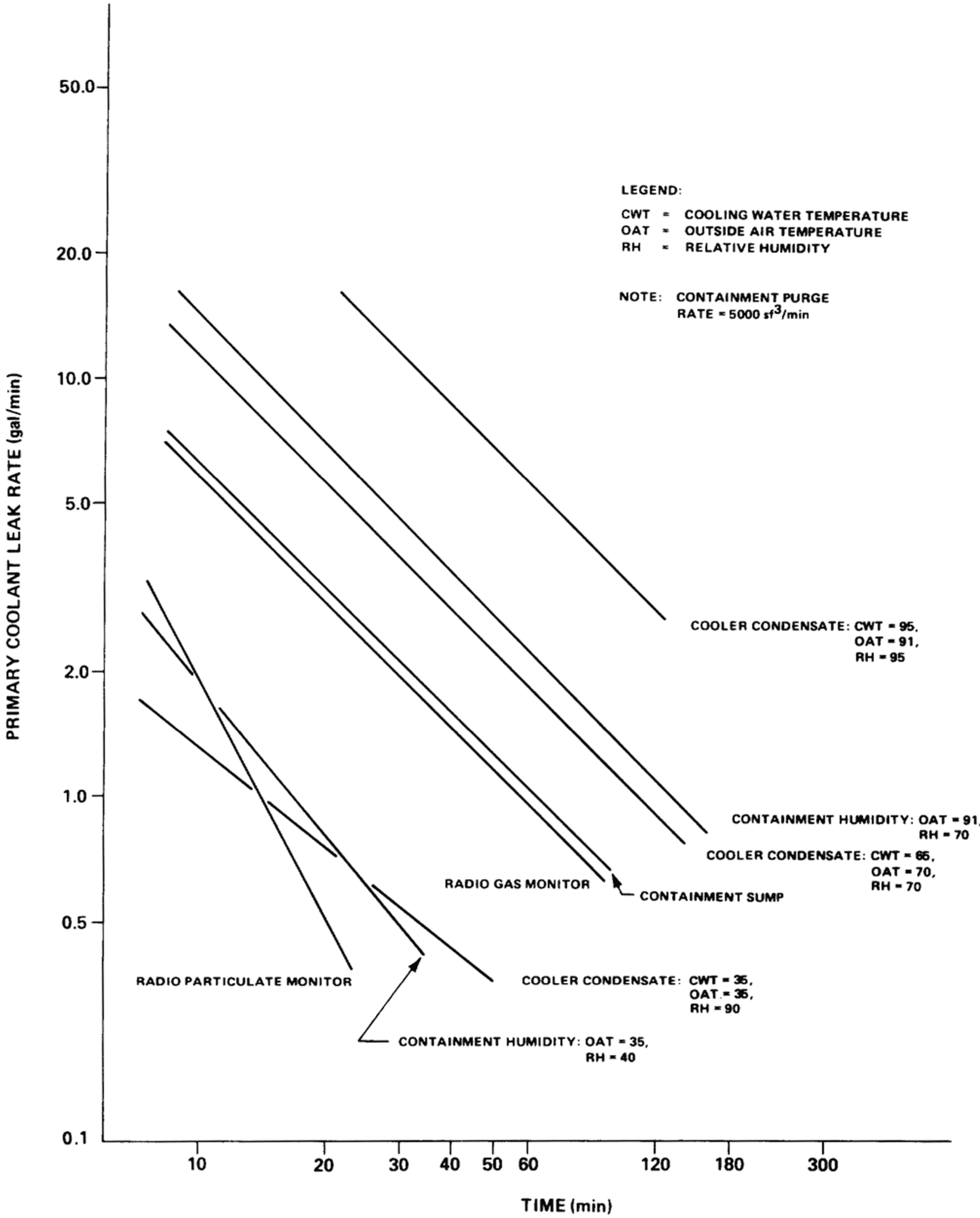
c. Halogen concentrations must be maintained below the specified values at all times when fuel is in the reactor vessel regardless of system temperature.

d. Hydrogen must be maintained in the reactor coolant for all plant operations with nuclear power above 1 MW.

e. Solids concentration determined by filtration through filter having 0.45-μm pore size.

f. These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity within the limits shown for these species is judged desirable with the current data base to minimize fuel clad crud deposition which affects the corrosion resistance and heat transfer of the clad.

g. Specification is applicable during power operation when zinc is being injected. Zinc target concentrations are maintained at the lower of 0.04 ppm or that specified in the reload safety analysis.



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PRIMARY COOLANT LEAK DETECTION
RESPONSE TIME

FIGURE 5.2.5–1

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Material Specifications

Material specifications are in accordance with the American Society of Mechanical Engineers (ASME) Code requirements and are given in subsection 5.2.3. All ferritic reactor vessel materials comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR 50.

The ferritic materials of the reactor vessel beltline are restricted to the following maximum limits of copper and phosphorus to reduce sensitivity to irradiation embrittlement in service:

Element	Base Metal (percent)	As Deposited Weld Metal (percent)
Copper	0.10 (ladle)	0.10
	0.12 (check)	
Phosphorus	0.012 (ladle)	0.020
	0.017 (check)	

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

A. The vessel is Safety Class 1. Design and fabrication of the reactor vessel is carried out in strict accordance with ASME Code, Section III, Class 1 requirements. The vessel head, flanges, and nozzles are manufactured as forgings. The cylindrical portion of the vessel is made of several shells, each consisting of formed plates joined by full penetration longitudinal and girth weld seams. The hemispherical heads are made from dished plates. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes.

B. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by either a select choice of material or by programming the method of assembly.

C. The control rod drive mechanism (CRDM) head adapter threads and surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts.

D. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.

E. The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during inservice inspection.

F. The stainless steel clad surfaces are sampled to ensure that composition requirements are met.

G. Freedom from underclad cracking is ensured by special evaluation of the procedure qualification for cladding applied on low-alloy steel (SA-508, Class 2).[a]

H. Minimum preheat requirements have been established for pressure boundary welds using low-alloy material. The preheat is maintained until either an intermediate postweld heat treatment or a full postweld heat treatment is completed or until the completion of welding.

I. A field weld is made, after the reactor vessel has been set, to install the permanent reactor vessel cavity seal ring. This stainless steel filler weld joins the seal ring to the reactor vessel seal ledge. A minimum preheat is specified for this weld in compliance with the ASME Code requirements.

5.3.1.3 Special Methods for Nondestructive Examination

The nondestructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code, Section III requirements; also, numerous examinations are performed in addition to ASME Code, Section III requirements. The NDE of the vessel is discussed in the following paragraphs, and the reactor vessel quality assurance program is given in table 5.3.1-1.

5.3.1.3.1 Ultrasonic Examination

A. In addition to the required ASME Code straight beam ultrasonic examination, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

B. In addition to the ASME Code, Section III NDE, all full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test is performed upon completion of the welding and intermediate heat treatment but prior to the final postweld heat treatment.

C. After hydrotesting, all full penetration ferritic pressure boundary welds in the reactor vessel, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are also performed in addition to the ASME Code, Section III NDE.

5.3.1.3.2 Penetrant Examinations

The partial penetration welds for the CRDM head adapters and the bottom instrumentation tubes are inspected by dye penetrant after the root pass, in addition to code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each 1/2 in. of weld metal. All clad surfaces and other vessel and head internal surfaces are inspected by dye penetrant after the hydrostatic test.

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[a] Underclad cracking of the reactor pressure vessel was evaluated as a time-limited aging analysis (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in paragraph 19.4.6.5.

5.3.1.3.3 Magnetic Particle Examination

The magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code.

All magnetic particle examinations of materials and welds are performed in accordance with the following:

- Prior to the final postweld heat treatment, only by the prod, coil, or direct contact method.

- After the final postweld heat treatment, only by the yoke method.

The following surfaces and welds are examined by magnetic particle methods. The acceptance standards are in accordance with Section III of the ASME Code.

A. Surface Examinations

1. Magnetic particle examination of all exterior vessel and head surfaces after the hydrostatic test.
2. Magnetic particle examination of all exterior closure stud surfaces and all nut surfaces after final machining or rolling. Continuous circular and longitudinal magnetization is used.
3. Magnetic particle examination of all inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection is performed after forming and machining and prior to cladding.

B. Weld Examination

Magnetic particle examination of the welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each 1/2 in. of weld metal is deposited. All pressure boundary welds are examined after back-chipping or back-grinding operations.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferrite steels and austenitic stainless steels is discussed in subsection 5.2.3. Subsection 5.2.3 includes discussions which indicate the degree of conformance with Regulatory Guide 1.44. Section 1.9 discusses the degree of conformance with Regulatory Guides 1.43, 1.50, 1.71, and 1.99.

5.3.1.5 Fracture Toughness[a]

Assurance of adequate fracture toughness of ferritic materials in the reactor vessel (ASME Code, Section III, Class 1 component) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline (including welds) are 75 ft-lb, as required by Appendix G of 10 CFR 50. The vessel

[a] Reactor vessel neutron embrittlement was evaluated as a TLAA for license renewal in accordance with 10 CFR 54.21 (see subsection 19.4.1).

fracture toughness data for Units 1 and 2 are given in tables 5.3.1-2 and 5.3.1-3, respectively. The end-of-life RT_{NDT} and upper shelf energy projections estimated using Regulatory Guide 1.99 for the end-of-life neutron fluence at the 1/4 T and ID reactor vessel locations for Units 1 and 2 are given in tables 5.3.3-2 and 5.3.3-3.

5.3.1.6 Material Surveillance[a]

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in FSAR tables 5.3.1-8 and 5.3.1-9. The results of these examinations shall be used to update figures in the Pressure Temperature Limits Report (PTLR).[16] In the surveillance program, the evaluation of radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and post irradiation testing of Charpy V-notch, tensile, and 1/2-T compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program conforms to American Society of Testing Materials (ASTM) E-185-82, Conducting Surveillance Tests for Light-Water-Cooled Nuclear Reactor Vessels, and 10 CFR 50, Appendix H.

[HISTORICAL] The reactor vessel surveillance program uses six specimen capsules. The capsules are located in guide baskets welded to the outside of the neutron shield pads and positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The six capsules contain reactor vessel steel specimens, oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel and associated weld metal and weld heat-affected zone metal. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat-affected zone material), and 72 CT specimens. Archive material sufficient for two additional capsules is retained at Westinghouse. The surveillance program withdrawal schedule, lead factor, test samples, and materials in the reactor vessel are given in tables 5.3.1-7 and 5.3.1-8. [HISTORICAL]

Removal of these specimen capsules was completed for Unit 1 at refueling outage 1R14 and for Unit 2 at refueling outage 2R14. Refer to paragraph 5.3.1.6.1.4 for a description of the external neutron monitoring system also known as external vessel neutron dosimetry system (EVNDS) used following removal of the last specimen capsules.

Dosimeters, as described below, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent is made for surveillance material and as deposited weld metal. Each of the six capsules contains the following specimens:

[a] The Reactor Vessel Surveillance Program is credited as a license renewal aging management program (see subsection 19.2.25).

Material	Number of Charpys	Number of Tensiles	Number of CTs
Limiting base material[a]	15	3	4
Limiting base material[b]	15	3	4
Weld metal[(c)]	15	3	4
Heat-affected zone	15	-	-

The following dosimeters and thermal monitors are included in each of the six capsules:

A. Dosimeters

1. Iron.
2. Copper.
3. Nickel.
4. Cobalt-aluminum (0.15-percent cobalt).
5. Cobalt-aluminum (cadmium shielded).
6. Uranium-238 (cadmium shielded).
7. Neptunium-237 (cadmium shielded).

B. Thermal Monitors

1. 97.5-percent lead, 2.5-percent silver, (579°F melting point).
2. 97.5-percent lead, 1.75-percent silver, 0.75-percent tin (590°F melting point).

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in paragraph 5.3.1.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on all capsules withdrawn. The schedule for removal of the capsules for postirradiation testing conforms with ASTM E-185-82 and Appendix H of 10 CFR 50.

5.3.1.6.1 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the

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[a] Specimens oriented in the major rolling or working direction.
[b] Specimens oriented normal to the major rolling or working direction.

[(c)] Weld metal to be selected in accordance with ASTM E-185-82.

measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level and, hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

- The measured specific activity of each sensor.
- The physical characteristics of each sensor.
- The operating history of the reactor.
- The energy response of each sensor.
- The neutron energy spectrum at the sensor location.

This section describes the procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates.

5.3.1.6.1.1 Determination of Sensor Reaction Rates. The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires or, as in the case of U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

The irradiation history of the reactor over its operating lifetime is determined from plant power generation records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum_j \frac{\mathrm{P_j}}{\mathrm{P_{ref}}} C_j \left[1 - e^{-\lambda t_j}\right] e^{-\lambda t_d}}$$

where:

A	=	measured specific activity provided in terms of disintegrations per second per gram of target material (dps/grn).
R	=	reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} expressed in terms of reactions per second per nucleus of target

isotope (rps/nucleus).

N_0 = number of target element atoms per gram of sensor.

F = weight fraction of the target isotope in the sensor material.

Y = number of product atoms produced per reaction.

P_j = average core power level during irradiation period j (MW).

P_{ref} = maximum or reference core power level of the reactor (MW).

C_j = calculated ratio of $\varphi(E > 1.0\ MeV)$ during irradiation period j to the time weighted average $\varphi(E > 1.0\ MeV)$ over the entire irradiation period.

λ = decay constant of the product isotope (sec^{-1}).

t_j = length of irradiation period j (sec).

t_d = decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month-by-month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay-corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

5.3.1.6.1.2 Corrections to Reaction Rate Data. Prior to using the measured reaction rates in the least squares adjustment procedure discussed in paragraph 5.3.1.6.1.3, additional corrections are made to the U-238 measurements to account for the presence of U-235 impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections are also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occurring over the course of the irradiation.

5.3.1.6.1.3 Least Squares Adjustment Procedure. Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{Ri} = \sum_g \left(\sigma_{ig} \pm \delta_{\sigma_{ig}}\right)\left(\phi_g \pm \delta_{\phi g}\right)$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, Φg, through the multigroup dosimeter reaction cross-section, σ_{ig}, each with an uncertainty δ.

The use of least squares adjustment methods in light water reactor (LWR) dosimetry evaluations is not new. ASTM has addressed the use of adjustment codes in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance," and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its biannual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, bypass region and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of the calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ).

By combining the calculated results with available measurements, the uncertainties associated with the key neutron exposure parameters can be reduced. Specifically ASTM Standard E 944 states, "The algorithims of the adjustment codes tend to decrease the variances of the adjusted data compared to the corresponding input values. The least squares adjustment codes yield estimates for the output data with minimum variances, that is, the "best estimates." This is the primary reason for using these adjustment procedures." ASTM E 944 provides a comprehensive listing of available adjustment codes.

The FERRET least squares adjustment code[(1)] was initially developed at the Hanford Engineering Development Laboratory (HEDL) and has had extensive use in both the Liquid Metal Fast Breeder (LMFBR) program and the NRC Sponsored Light Water Reactor Dosimetry Improvement Program (LWR-PV-SDIP). As a result of participation in several cooperative efforts associated with the LWR-PV-SDIP, the FERRET approach was adopted by Westinghouse in the mid 1980's as the preferred approach for the evaluation of LWR surveillance dosimetry. The least squares methodology was judged superior to the previously employed spectrum averaged cross-section approach that is totally dependent on the accuracy of the shape of the calculated neutron spectrum at the measurement locations.

The FERRET code is employed to combine the results of plant specific neutron transport calculations and multiple-foil reaction-rate measurements to determine best estimate values of exposure parameters in terms of both neutron fluence greater than 1.0 MeV, (Φ(E > 1.0 MeV),

and iron atom displacements, (dpa), along with associated uncertainties in the measurement locations.

The application of the least squares methodology requires the following input:

- The calculated neutron energy spectrum and associated uncertainties at the measurement location.
- The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.
- The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant-specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. This calculation is performed using the benchmarked transport calculational methodology described in paragraph 5.3.1.6.2. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties are obtained from the SNLRML dosimetry cross-section library(2). The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)." There are no additional data or data libraries built into the FERRET code system. All of the required input is supplied externally at the time of the analysis.

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

5.3.1.6.1.4 External Neutron Monitoring System. The external neutron monitoring system also known as the external vessel neutron dosimetry system (EVNDS) provides for continuing neutron fluence measurement after sufficient specimen material exposure has been achieved and even after the last of the six internal surveillance capsules has been removed from the reactor vessel. It enables verification of fast neutron exposure distributions within the reactor vessel wall beltline region and establishes a mechanism to enable long term monitoring of this portion of the reactor vessel as required per 10 CFR 50 Appendix H. These fluence data can also support potential license renewal activities.

The EVNDS is located external to the reactor vessel, allowing for ease of dosimetry removal and replacement. It is installed in the annular air gap between the reactor vessel insulation and the primary concrete shield wall. The EVNDS is a passive system consisting of six aluminum dosimeter capsules containing radiometric monitors and four stainless steel gradient chains, which are bead chains connecting and supporting the dosimeter capsules. The bead chains are in turn supported by an arrangement of stainless steel hardware-tubular brackets on a support bar suspended by chains from bracket plate assembly, which is welded to the ventilation port liner plate under a banana cover. The bead chains are mechanically secured to the concrete floor below the reactor vessel. The system is shown on drawings 1X6AB03-00020, 1X6AB03-00021, and 1X6AB03-00022 for Unit 1 and is shown on drawings 2X6AB03-00022, 2X6AB03-00023, and 2X6AB03-00024 for Unit 2.

The EVNDS measures fluence for approximately 1/8 of the vessel wall circumference, positioned relative to well known reactor features. Neutron transport calculations then determine the fluence for the entire vessel beltline wall. The system assists in the evaluation of radiation damage to the reactor vessel beltline region by measuring the fluence to this region, which can be used to predict the shift in the reference nil ductility transition temperature (RT_{NDT}). When used in conjunction with previously removed dosimetry from the internal surveillance capsules and with the results of neutron transport calculations, the external vessel neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will help to assure that the reactor can be operated in the least restrictive mode possible with respect to:

- 10 CFR 50 Appendix G pressure / temperature limit curves for normal heatup and cooldown of the reactor coolant system,
- Emergency Response Guideline (ERG) pressure / temperature limit curves, and
- Pressurized thermal shock (PTS) RT_{NDT} screening criteria.

Comprehensive sensor sets are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition, the stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

The first replacement of irradiation dosimetry is at refueling outage 1R15 (Sept. 2009). The first replacement of irradiation dosimetry is at refueling outage 2R14 (March 2010). An irradiation interval of five fuel cycles between replacements is typical.

5.3.1.6.2 Calculation of Integrated Fast (E > 1.0 MeV) Exposure at the Irradiation Samples and Reactor Vessel Wall

Discrete ordinates transport calculations are performed on a fuel cycle-specific basis to determine the neutron and gamma ray environment within the reactor geometry. The specific methods applied have been benchmarked according to the guidelines of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001, and have been approved by the NRC staff for general application to pressurized water reactor (PWR) analysis. A description of the transport methodology along with the SER documenting NRC staff approval of the method and computer codes are provided in Reference 13.

In the application of this methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, a series of two-dimensional plant-specific transport calculations are carried out and then synthesized to generate a three-dimensional neutron flux distribution, $\Phi(r,\theta,z)$ throughout the geometry of interest using the procedures outlined in Regulatory Guide 1.190. These three-dimensional mappings of the neutron environment are completed for each operating fuel cycle and then integrated to determine the neutron fluence experienced by the surveillance test specimens and the pressure vessel wall.

In the approved analysis methodology, the transport calculations are completed using the DORT discrete ordinates code Version 3.1[(3)] and the BUGLE-96 cross-section library[(10)]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set

produced specifically for LWR application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature.

Energy- and space-dependent core power distributions, as well as system operating temperatures, are treated on a fuel cycle-specific basis. The spatial variation of the neutron source is obtained from a burnup-weighted average of the respective power distributions from individual fuel cycles including pinwise gradients for all fuel assemblies located on the periphery of the core. The energy distribution of the source is determined on a fuel assembly-specific basis and includes the effects of fissioning in both uranium and plutonium isotopes.

The results of the transport calculations are validated on a plant-specific basis by comparison with the results of surveillance capsule dosimetry developed using the procedures described in paragraph 5.3.1.6.1. These comparisons are used to demonstrate that the plant-specific application is consistent with the uncertainty evaluations provided in Reference 13 and to establish that the 20% uncertainty criterion listed in Regulatory Guide 1.190 is met. These comparisons are not used to modify or bias the results of the transport calculations.

5.3.1.6.2.1 Reference Forward Calculation. The forward transport calculation for the reactor is carried out in r,θ geometry using the DORT two-dimensional discrete ordinates code[3] and the BUGLE-96 cross-section library[10]. The BUGLE-96 library is a 47 neutron group, ENDFB-VI based, data set produced specifically for LWR applications. In these analyses, anisotropic scattering is treated with a P_3 expansion of the scattering cross-sections and the angular discretization is modeled with an S_8 order of angular quadrature. The reference forward calculation is normalized to a core midplane power density characteristic of operation at the stretch rating for the reactor.

The spatial core power distribution utilized in the reference forward calculation is derived from statistical studies of long-term operation of Westinghouse 4-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, a 2σ uncertainty derived from the statistical evaluation of plant-to-plant and cycle-to-cycle variations in peripheral power is used for the peripheral fuel assemblies.

Due to the use of this bounding spatial power distribution, the results from the reference forward calculation establish conservative exposure projections for reactors of this design operating at the stretch rating. Since it is unlikely that actual reactor operation would result in the implementation of a power distribution at the nominal +2σ level for a large number of fuel cycles and, further, because of the widespread implementation of low-leakage fuel management strategies, the fuel cycle-specific calculations for this reactor will result in exposure rates well below these conservative predictions.

5.3.1.6.2.2 Cycle-Specific Adjoint Calculations. All adjoint analyses are also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the BUGLE-96 library. Adjoint source locations are chosen at several key azimuths on the pressure vessel inner radius. In addition, adjoint calculations were carried out for sources positioned at the geometric center of all surveillance capsules. Again, these calculations are run in r,θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, $\varphi(E > 1.0\ \text{MeV})$.

The importance functions generated from these individual adjoint analyses provide the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle-specific neutron source distributions, yield absolute predictions of neutron exposure at the locations of interest for each of the operating fuel cycles and establish the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles.

Having the importance functions and appropriate core source distributions, the response of interest can be calculated as:

$$\varphi(R_o,\theta_o) = \int_r \int_\theta \int_E \ I(r,\theta,E)\, S\,(r,\theta,E)\, r\, dr\, d\theta\, dE$$

where: $\varphi(R_0,\theta_0)$ = Neutron flux (E > 1.0 MeV) at radius R_0 and azimuthal angle θ_0.

$I(r,\theta,E)$ = Adjoint importance function at radius r, azimuthal angle θ, and neutron source energy E.

$S(r,\theta,E)$ = Neutron source strength at core location r,θ and energy E.

It is important to note that the cycle-specific neutron source distributions, S(r,θ,E), utilized with the adjoint importance functions, I(r,θ,E), permit the use not only of fuel cycle-specific spatial variations of fission rates within the reactor core, but also allow for the inclusion of the effects of the differing neutron yield per fission and the variation in fission spectrum introduced by the build-in of plutonium isotopes, as the burnup of individual fuel assemblies increases.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The closure studs are fabricated of SA-540, Class 3, Grade B24. The closure stud material meets the fracture toughness requirements of the ASME Code, Section III, and 10 CFR 50, Appendix G. Conformance with Regulatory Guide 1.65, Materials and Inspections for Reactor Vessel Closure Studs, is discussed in section 1.9. Nondestructive examinations are performed in accordance with the ASME Code, Section III. Bolting material properties for Units 1 and 2 are given in tables 5.3.1-4 and 5.3.1-5, respectively.

Refueling procedures require that the reactor vessel studs, nuts, and washers are lifted part of the way out of their respective holes and a stud support collar be put in place prior to the lift of the integrated head assembly during preparation for refueling. In this way the studs are lifted with and stored on the head. An alternative method of the procedures is that the reactor vessel studs, nuts, and washers may be removed from the reactor closure and placed in storage racks during preparation for refueling. In this method, the storage racks are removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. In either case, the reactor closure studs are not exposed to the borated refueling cavity water. Additional protection against the possibility of incurring corrosion effects is ensured by the use of a manganese base phosphate surfacing treatment.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.3.1.8 References

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3. RSICC Computer Code Collection CCC-650, "DOORS 3.1 One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center, Oak Ridge National Laboratory, August 1996.

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9. Grendys, P. A., et al., "Analysis of Capsule Y From the GPC VEGP Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-14532, February 1996.

10. RSIC Data Library Collection DLC-185, "BUGLE 96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived From ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center, Oak Ridge National Laboratory, March 1996.

11. Laubham, T. J., et al., "Analysis of Capsule V from the Georgia Power Vogtle Electric Generating Plant Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15067, September 1998.

12. Laubham, T. J., et al., "Analysis of Capsule X from the Southern Nuclear Vogtle Electric Generating Plant Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-15159, March 1999.

13. J. D. Andrachek, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-A, Revision 4, May 2004.

14. RSIC Computer Code Collection PSR-145, “FERRET: Least-Squares Solution to Nuclear Data and Reactor Physics Problems,” Radiation Shielding Information Center, Oak Ridge National Laboratory, January 1980.

15. Westinghouse LTR-REA-06-136, “Alvin W. Vogtle Units 1 and 2 Ex-Vessel Neutron Dosimetry System Description and Safety Evaluation Factors,” (AX6AB03-00019).

16. NRC Letter to C. K. McCoy, “Vogtle Electric Generating Plant, Units 1 and 2 - Acceptance for Referencing of Pressure Temperature Limits Report,” February 12, 1996.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Startup and shutdown operating limitations are based on the properties of the reactor pressure vessel beltline materials. Actual material property test data are used. The methods outlined in Appendix G Section XI of the American Society of Mechanical Engineers (ASME) Code are employed for the shell regions in the analysis of protection against nonductile failure. The initial operating curves are calculated, assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the Pressure and Temperature Limits Report as required by the Technical Specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil ductility temperature, which includes a reference nil ductility temperature shift (ΔRT_{NDT}). The reference temperature, RT_{NDT}, for materials in the reactor vessel closure flange region and the beltline regions are shown in tables 5.3.1-2 and 5.3.1-3. Tables 5.3.2-2 through 5.3.2-5 give the properties for the vessel beltline materials and data for the C_v curve (energy vs. temperature).

Predicted ΔRT_{NDT} values are derived using guidance provided in Regulatory Guide 1.99 Revision 2. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the reactor coolant system (RCS) will be limiting in the analysis.

The operating curves including pressure-temperature limitations shown in the Pressure and Temperature Limits Report are calculated in accordance with 10 CFR 50, Appendices G and H, and ASME Code, Section XI, Appendix G requirements, Code Case N-640 and WCAP-16142, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2, Revision 1.”

The results of the material surveillance program described in paragraph 5.3.1.6 will be used to verify that the ΔRT_{NDT} predicted from the effects of the fluence, copper content curve is appropriate if ΔRT_{NDT} determined from the surveillance program is greater than the predicted ΔRT_{NDT}. Temperature limits for inservice leak and hydrotests will be calculated in accordance with ASME Code, Section XI, Appendix G Conformance with Regulatory Guide 1.99, Revision 2, is discussed in section 1.9.

5.3.2.2 Operating Procedures

The transient conditions that are considered in the design of the reactor vessel are presented in paragraph 3.9.1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected form a conservative basis for evaluation of the RCS to ensure the integrity of the RCS equipment.

Those transients listed as upset condition transients are given in table 3.9.N.1-1. None of these transients will result in pressure-temperature changes which exceed the heatup and cooldown limitations, as described in the Pressure and Temperature Limits Report.

5.3.3 REACTOR VESSEL INTEGRITY

5.3.3.1 Design

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, bolted, flanged, and gasketed hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff connections, one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adapters. These head adapters are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adapters contains acme threads for the assembly of control rod drive mechanisms (CRDMs) or instrumentation adapters. The seal arrangement at the upper end of these adapters consists of a welded flexible canopy seal. Mechanical assemblies may be used to fix or prevent leaks in the canopy seal weld. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the reactor coolant system equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of either an Inconel or an Inconel-stainless steel composite tube. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.125-in. minimum of stainless steel or Inconel.

The reactor vessel is designed and fabricated in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section III. Principal design parameters of the reactor vessel are given in table 5.3.3-1. The reactor vessel is shown in figure 5.3.3-1.

There are no special design features which would prohibit the in situ annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature greater than 650 °F for a maximum period of 168 h would be applied.[(4)] Various modes of heating may be used, depending on the temperature required.

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

Cyclic loads are introduced by normal power changes, reactor trips, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life[a]. Vessel analysis results in a usage factor that is less than 1.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup

[a] Metal fatigue is evaluated as a TLAA for license renewal (see subsection 19.4.2).

and cooldown rates are 100°F/h for normal operations and under abnormal or emergency conditions. This rate is reflected in the vessel design specifications.

5.3.3.2 Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in subsection 5.2.3.

5.3.3.3 Fabrication Methods

The VEGP reactor vessel manufacturer is Combustion Engineering Corporation.

The fabrication methods used in the construction of the reactor vessel are discussed in paragraph 5.3.1.2.

5.3.3.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in paragraph 5.3.1.3.

5.3.3.5 Shipment and Installation

The reactor vessel is shipped in a horizontal position on a shipping sled with a vessel-lifting truss assembly. All vessel openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the vessel. These are usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces, except for the vessel support surfaces and the top surface of the external seal ring, are painted with a heat-resistant paint before shipment.

The closure head is also shipped with a shipping cover and skid. An enclosure attached to the ventilation shroud support ring protects the control rod mechanism housings. All head openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the head. These are placed in a wire mesh basket attached to the head cover. All carbon steel surfaces are painted with heat- resistant paint before shipment. A lifting frame is provided for handling the vessel head.

5.3.3.6 Operating Conditions

Operating limitations for the reactor vessel are presented in the Pressure Temperature Limit Report (PTLR).

In addition to the analysis of primary components discussed in paragraph 3.9.1.4, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Actuation of the emergency core cooling system (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in regions of the reactor vessel which come into contact with ECCS water. Primary consideration is given to these areas, including the reactor vessel beltline region and the reactor vessel primary coolant nozzle, to ensure the integrity of the reactor vessel under this severe postulated transient. The Westinghouse Owners Group evaluated TMI Action Item II.K.2.13, and the item is satisfied upon submittal of RT_{NDT} values which are below the pressurized thermal shock (PTS) rule screening values. Additionally, new

calculations were performed based on the requirements of Generic Letter 92-01 and the results are given in tables 5.3.3-2 and 5.3.3-3.

For the beltline region, significant developments have recently occurred in order to address PTS events. On the basis of recent deterministic and probabilistic studies, taking U.S. pressurized water reactor operating experience into account, the Nuclear Regulatory Commission staff concluded that conservatively calculated screening criterion values of RT_{NDT} less than 270° for plate material and axial welds, and less than 300° for circumferential welds, present an acceptably low risk of vessel failure from PTS events. These values were chosen as the screening criterion in the PTS rule for 10 CFR 50.34 (new plants) and 10 CFR 50.61 (operating plants).[(2)] The conservative methods chosen by the NRC staff for the calculation of RT_{PTS} for the purpose of comparison with the screening criterion is presented in paragraph (b)(2) of 10 CFR 50.61. Details of the analysis method and the basis for the PTS rule can be found in SECY-82-465.[(3)]

The reactor vessel beltline materials are specified in subsection 5.3.1. The fluence of 4.76 x 10^{19} n/cm^2 which is the design basis fluence at the vessel inner radius, at 48 EFPY, at the peak location, was used for calculating all of the RT_{PTS} values. Based on the latest capsule data for each unit, the expected fluence at the vessel inner radius after 56.3 EFPY will be significantly less than the 4.76 x 10^{19} n/cm^2 design basis fluence[a]. RT_{PTS} is RT_{NDT}, the reference nilductility transition temperature as calculated by the method chosen by the NRC staff as presented in paragraph (b)(2) of 10 CFR 50.61, and the PTS rule. The PTS rule states that this method of calculating RT should be used in reporting values used to be compared to the above screening criterion set in the PTS rule. The screening criteria will not be exceeded using the method of calculation prescribed by the PTS rule for the vessel design lifetime. The material properties, initial RT_{NDT}, and end-of-life RT_{PTS} values are in tables 5.3.3-2 and 5.3.3-3. The materials identified in tables 5.3.3-2 and 5.3.3-3 are those materials that are exposed to high fluence levels at the beltline region of the reactor vessel and are, therefore, the subject of the PTS rule. These materials, therefore, are a subset of the materials identified in subsection 5.3.1.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity factor K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack), the stress intensity factor is designated as K_I and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature. Any combination of applied load, structural configuration, crack geometry, and size which yields a stress intensity factor greater than K_{IC} for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by American Society of Testing Materials (ASTM)) of

[a] Reactor vessel embrittlement is evaluated as a TLAA for license renewal (see subsection 19.4.1).

LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed approximately 2 percent of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20 percent of the crack depth. However, LEFM has been successfully used to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from the Heavy Section Steel Technology program intermediate pressure vessel tests have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the elastically calculated thermal stresses, which results in total stresses in excess of the yield strength, does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted conditions analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

Additional details on this method of analysis of reactor vessels under severe thermal transients are given in reference 1.

5.3.3.7 Inservice Surveillance

The internal and external surfaces of the reactor vessel are accessible for periodic inspection. Visual and/or nondestructive techniques are used. During refueling, the vessel cladding is capable of being inspected in certain areas between the closure flange and the primary coolant inlet nozzles, and, if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, CRDM nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle testing, and ultrasonic testing. The closure studs and nuts can be inspected periodically using visual, magnetic particle, and ultrasonic techniques.

The closure studs, nuts, washers, and the vessel flange seal surface, as well as the full-penetration welds in the following areas of the installed reactor vessel, are available for nondestructive examination:

A. Vessel shell, from the inside and outside surfaces.

B. Primary coolant nozzles, from the inside and outside surfaces.[a]

C. Closure head, from the inside and outside surfaces; bottom head, from the inside and outside surfaces.

D. Field welds between the reactor vessel nozzle safe ends and the main coolant piping, from the inside and outside surfaces.

The design considerations which have been incorporated into the system design to permit the above inspection are as follows:

[a] Only partial outside diameter coverage is provided.

A. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.

B. The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.

C. All reactor vessel studs, nuts, and washers can be removed to dry storage during refueling.

D. Access is provided to the reactor vessel nozzle safe ends. The insulation covering the nozzle-to-pipe welds may be removed.

The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME inservice inspection code. These are as follows:

A. Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to ensure an adequate cladding bond to allow later ultrasonic testing of the base metal from inside surface. The size of cladding bond defect allowed is 1/4 in. by 3/4 in. with the greater direction parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full-penetration pressure boundary weld. Unbounded areas exceeding 0.442 in.^{2} (3/4-in. diameter) in all other regions are rejected.

B. The design of the reactor vessel shell is an uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.

C. The weld-deposited clad surface on both sides of the welds to be inspected is specifically prepared to ensure meaningful ultrasonic examinations.

D. During fabrication, all full-penetration ferritic pressure boundary welds are ultrasonically examined in addition to code examinations.

E. After the shop hydrostatic testing, all full-penetration ferritic pressure boundary welds (with the exception of the closure head welds), as well as the nozzles to safe end welds, are ultrasonically examined from both the inside and outside diameters in addition to ASME Code, Section III requirements. The closure head ferritic pressure boundary welds are examined from the outside diameter only.

The vessel design and construction enables inspection in accordance with the ASME Code, Section XI. The reactor vessel inservice inspection requirements are detailed in the VEGP inservice inspection program.

The Reactor Vessel Closure Head Stud Program is credited as a license renewal aging management program (see subsection 19.2.23).

5.3.3.8 References

1. Buchalet, C., Bamford, W. H., and Chirigos, J. N., "Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, December 1975.

2. PTS Rule, Federal Register Vol. 50, No. 141, July 23, 1985, 10 CFR 50.34.

3. NRC Policy Issue, "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

4. EPRI NP 2712, "Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel," November 1982.

TABLE 5.3.1-1 (SHEET 1 OF 2)
REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	RT[(a)]	UT[(a)]	PT[(a)]	MT[(a)]
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRDM head adapter flange		Yes	Yes	
CRDM head adapter tube		Yes	Yes	
Instrumentation tube		Yes	Yes	
Main nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Plates		Yes		Yes
Weldments				
CRDM head adapter to closure head connection			Yes	
Instrumentation tube to bottom head connection			Yes	
Main nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
CRDM head adapter flange to CRDM head adapter tube	Yes		Yes	
All full-penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes
Full-penetration nonferritic pressure boundary welds accessible after hydrotest				
a. Nozzle to safe ends		Yes	Yes	
b. CRDM head adapter flange to CRDM head adapter tube			Yes	
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	

TABLE 5.3.1-1 (SHEET 2 OF 2)

a. RT - Radiographic.
UT - Ultrasonic.
PT - Dye penetrant.
MT - Magnetic particle.

NOTE:

Base metal weld repairs as a result of UT, MT, RT, and/or PT indications are cleared by the same NDE technique/procedure by which the indications were found. The repairs meet all Section III requirements.

In addition, UT examination in accordance with the inprocess/posthydro UT requirements is performed on base metal repairs in the core region and base metal repairs in the inservice inspection zone (1/2 T).

TABLE 5.3.1-2[a]

VEGP UNIT 1 REACTOR VESSEL FRACTURE TOUGHNESS PROPERTIES

Component	Code No.	Material Spec. No.	Cu (%)	Ni (%)	P (%)	NDT (°F)	RT_{NDT} (°F)	Use NMWD[b] (ft-lb)
Closure head dome	B8807-1	A533B Cl. 1	0.16	0.67	0.008	-50	15	88
Closure head torus[c]	B8808-1	A533B Cl. 1	0.14	0.56	0.010	-30	8	85
Closure head flange[c]	B8801-1	A508 Cl. 2	-	0.70	0.011	20	20	132
Vessel flange[c]	B8802-1	A508 Cl. 2	-	0.71	0.014	0	0	119
Inlet nozzle	B8809-1	A508 Cl. 2	-	0.86	0.011	-20	-20	107
Inlet nozzle	B8809-2	A508 Cl. 2	-	0.84	0.014	-10	-10	95
Inlet nozzle	B8809-3	A508 Cl. 2	-	0.82	0.013	-10	-10	117
Inlet nozzle	B8809-4	A508 Cl. 2	-	0.87	0.014	-20	-20	105
Outlet nozzle	B8810-1	A508 Cl. 2	-	0.82	0.006	-10	-10	>124
Outlet nozzle	B8810-2	A508 Cl. 2	-	0.79	0.006	-10	-10	>100
Outlet nozzle	B8810-3	A508 Cl. 2	-	0.77	0.006	-10	-10	>102
Outlet nozzle	B8810-4	A508 Cl. 2	-	0.80	0.006	-10	-10	> 75
Nozzle shell	B8804-1	A533B Cl. 1	0.14	0.62	0.011	-10	28	94
Nozzle shell	B8804-2	A533B Cl. 1	0.10	0.58	0.006	-40	15	104
Nozzle shell	B8804-3	A533B Cl. 1	0.14	0.69	0.013	-30	40	92
Intermediate shell[c]	B8805-1	A533B Cl. 1	0.08	0.59	0.004	0	0	90
Intermediate shell[c]	B8805-2	A533B Cl. 1	0.08	0.59	0.004	-10	20	100
Intermediate shell[c]	B8805-3	A533B Cl. 1	0.06	0.60	0.003	-20	30	107
Lower shell[c]	B8606-1	A533B Cl. 1	0.05	0.59	0.005	-50	20	116
Lower shell[c]	B8606-2	A533B Cl. 1	0.05	0.58	0.009	-10	20	113
Lower shell[c]	B8606-3	A533B Cl. 1	0.06	0.64	0.007	-20	10	118
Bottom head torus	B8813-1	A533B Cl. 1	0.13	0.50	0.009	-40	-10	88
Bottom head dome	B8812-1	A533B Cl. 1	0.10	0.53	0.009	-40	-28	122
Intermediate and lower[c] shell vertical weld seams and girth	G1.43	SAW	0.03	0.10	0.007	-80	-80	129

a. This table is based on initial testing of the vessel materials at the time the surveillance capsule program was developed.

b. Normal to major working direction.

c. Denotes materials in reactor vessel closure flange and beltline region.

TABLE 5.3.1-3[a]

VEGP UNIT 2 REACTOR VESSEL FRACTURE TOUGHNESS PROPERTIES

Component	Code No.	Material Spec. No.	Cu (%)	Ni (%)	P (%)	NDT (°F)	RT_{NDT} (°F)	Use NMWD[b] (ft-lb)
Closure head dome	R9-1	A533B Cl. 1	0.07	0.61	0.008	-40	-30	123
Closure head torus[c]	R10-1	A533B Cl. 1	0.07	0.64	0.010	-30	-0	84
Closure head flange[c]	R7-1	A508 Cl. 2	-	0.72	0.011	10	10	130
Vessel flange[c]	R1-1	A508 Cl. 2	-	0.87	0.011	-60	-60	115
Inlet nozzle	B9806-1	A508 Cl. 2	0.07	0.84	0.010	-50	-50	119
Inlet nozzle	B9806-2	A508 Cl. 2	0.06	0.83	0.009	-40	-40	128
Inlet nozzle	R5-1	A508 Cl. 2	0.09	0.87	0.008	-20	-20	147
Inlet nozzle	R5-2	A508 Cl. 2	0.08	0.85	0.009	-20	-20	134
Outlet nozzle	R6-3	A508 Cl. 2	-	0.69	0.011	-10	-10	122
Outlet nozzle	R6-4	A508 Cl. 2	-	0.66	0.010	-10	-10	140
Outlet nozzle	B9807-3	A508 Cl. 2	-	0.66	0.005	-30	-30	116
Outlet nozzle	B9807-4	A508 Cl. 2	-	0.64	0.010	-10	10	132
Nozzle shell	R3-1	A533B Cl. 1	0.20	0.67	0.015	0	-20	79
Nozzle shell	R3-2	A533B Cl. 1	0.20	0.67	0.015	0	-40	79
Nozzle shell	R3-3	A533B Cl. 1	0.15	0.62	0.010	-10	-60	84
Intermediate shell[c]	R4-1	A533B Cl. 1	0.06	0.64	0.009	-20	10	95
Intermediate shell[c]	R4-2	A533B Cl. 1	0.05	0.62	0.009	-10	10	104
Intermediate shell[c]	R4-3	A533B Cl. 1	0.05	0.59	0.009	0	30	84
Lower shell[c]	B8825-1	A533B Cl. 1	0.05	0.59	0.006	-20	40	83
Lower shell[c]	R8-1	A533B Cl. 1	0.06	0.62	0.007	-20	40	87
Lower shell[c]	B8628-1	A533B Cl. 1	0.05	0.59	0.007	-20	50	85
Bottom head torus	R12-1	A533B Cl. 1	0.17	0.64	0.012	-20	-20	89
Bottom head dome	R11-1	A533B Cl. 1	0.10	0.62	0.008	-30	-30	115
Intermediate and lower[c] shell vertical weld seams and girth	G1.60	SAW	0.07	0.13	0.007	-10	-10	147
Intermediate and lower[c] shell vertical weld seams and girth	E3.23	SAW	0.06	0.12	0.007	-50	-30	90

a. This table is based on initial testing of the vessel materials at the time the surveillance capsule program was developed.

b. Normal to major working direction.

c. Denotes materials in reactor vessel closure flange and beltline region.

TABLE 5.3.1-4

VEGP UNIT 1 REACTOR VESSEL CLOSURE HEAD BOLTING MATERIAL PROPERTIES

Closure Head Studs

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength (ksi)	Ultimate Tensile Strength (ksi)	Elongation (%)	Reduction in Area (%)	Energy at 10°F (ft-lb)	Lateral Expansion (mils)
82029	SA540, B24	166	148.2	162.5	17.0	55.5	53, 54, 52	31, 30, 28
82029	SA540, B24	166-1	149.5	163.0	16.5	56.0	50, 51, 50	28, 32, 27
82029	SA540, B24	167	150.2	162.5	17.0	57.3	54, 54, 52	31, 33, 30
82029	SA540, B24	167-1	148.8	162.0	17.0	56.5	50, 51, 50	29, 36, 30
82029	SA540, B24	170	142.0	158.0	17.0	56.0	51, 51, 54	28, 30, 34
82029	SA540, B24	170-1	150.5	164.0	16.5	54.7	49, 49, 50	26, 26, 29
82029	SA540, B24	172	142.0	157.5	17.0	53.8	52, 50, 52	30, 29, 33
82029	SA540, B24	172-1	151.0	164.0	16.5	55.7	49, 50, 51	27, 30, 30
82029	SA540, B24	173	151.5	164.0	16.5	56.5	50, 50, 49	29, 31, 29
82029	SA540, B24	173-1	150.2	165.0	16.0	54.7	49, 50, 52	27, 30, 31
82552	SA540, B24	197	142.5	156.0	18.0	53.8	51, 51, 52	28, 29, 30
82552	SA540, B24	197-1	141.5	155.0	17.0	53.8	50, 51, 50	28, 31, 28
82552	SA540, B24	201	146.0	158.5	15.5	51.7	48, 48, 49	27, 27, 27
82552	SA540, B24	201-1	145.5	159.5	15.5	50.6	47, 49, 47	27, 30, 27
82552	SA540, B24	207	138.0	153.0	17.0	52.5	51, 52, 51	31, 32, 31
82552	SA540, B24	207-1	138.5	153.0	16.5	51.4	51, 50, 49	28, 28, 27
82552	SA540, B24	212	141.0	155.0	17.0	53.0	52, 52, 51	32, 31, 30
82552	SA540, B24	212-1	144.0	157.0	16.0	49.8	51, 50, 49	31, 29, 29

Closure Head Nut and Washers

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength (ksi)	Ultimate Tensile Strength (ksi)	Elongation (%)	Reduction in Area (%)	Energy at 10°F (ft-lb)	Lateral Expansion (mils)
19632	SA540, B23	75	146.0	159.5	17.0	55.7	49, 52, 49	29, 30, 27
19632	SA540, B23	75-1	142.0	155.0	17.0	54.9	47, 49, 47	30, 29, 29
19632	SA540, B23	78	150.5	162.0	16.0	51.9	48, 48, 48	28, 27, 28
19632	SA540, B23	78-1	143.0	156.0	17.5	54.5	51, 51, 52	32, 31, 34

TABLE 5.3.1-5

VEGP UNIT 2 REACTOR VESSEL CLOSURE HEAD BOLTING MATERIAL PROPERTIES

Closure Head Studs

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength (ksi)	Ultimate Tensile Strength (ksi)	Elongation (%)	Reduction in Area (%)	Energy at 10˜F (ft-lb)	Lateral Expansion (mils)
83090	SA540, B24	265	146.0	159.0	16.5	52.9	50, 50, 51	31, 31, 33
83090	SA540, B24	265-1	146.0	160.0	16.5	51.8	48, 50, 51	30, 31, 33
83090	SA540, B24	271	151.5	164.0	17.5	54.7	47, 48, 47	26, 28, 27
83090	SA540, B24	271-1	150.0	162.0	16.5	51.9	50, 52, 50	31, 33, 30
83090	SA540, B24	273	151.5	163.5	17.0	55.2	47, 46, 48	28, 26, 29
83090	SA540, B24	273-1	152.8	163.0	16.5	53.0	47, 47, 46	27, 29, 26
83090	SA540, B24	278	150.5	162.5	16.0	51.9	47, 46, 48	26, 25, 29
83090	SA540, B24	278-1	152.5	165.0	16.5	51.4	47, 46, 48	27, 29, 29
83090	SA540, B24	283	150.0	163.0	16.5	52.5	48, 48, 49	28, 28, 30
83090	SA540, B24	283-1	145.0	157.0	16.0	50.0	47, 47, 47	29, 29, 27
83090	SA540, B24	286	147.0	158.5	16.5	50.8	48, 47, 48	27, 28, 29
83090	SA540, B24	286-1	142.5	155.0	16.5	52.5	50, 49, 49	32, 30, 30
Closure Head Nuts and Washers								
83294	SA540, B24	105	148.2	163.0	17.0	52.5	50, 51, 51	29, 31, 28
83294	SA540, B24	105-1	145.2	158.5	18.5	57.3	55, 55, 55	34, 32, 30
83294	SA540, B24	110	148.0	161.5	18.5	57.2	53, 53, 53	30, 31, 32
83294	SA540, B24	110-1	148.0	162.0	18.0	56.2	51, 53, 53	29, 31, 32
83294	SA540, B24	113	146.5	161.0	18.0	54.7	53, 54, 55	34, 33, 33
83294	SA540, B24	113-1	146.2	160.0	18.0	56.5	55, 57, 57	34, 34, 34
83294	SA540, B24	117	143.5	157.0	17.0	57.3	55, 55, 54	36, 34, 32
83294	SA540, B24	117-1	145.0	158.5	17.5	57.8	53, 53, 53	29, 30, 28

TABLE 5.3.1-6

DELETED

TABLE 5.3.1-7

[HISTORICAL]

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

VEGP Unit 1

	Capsules U, V, W, X, Y, and Z		
Material	*Charpy*	*Tensile*	*1/2T-CT*
Plate B8805-3 (long.)	15	3	4
Plate B8805-3 (trans.)	15	3	4
Weld metal (G-1.43)	15	3	4
HAZ	15	-	-

VEGP Unit 2

	Capsules U, V, W, X, Y, and Z		
Material	*Charpy*	*Tensile*	*1/2T-CT*
Plate B8628-1 (long.)	15	3	4
Plate B8628-1 (trans.)	15	3	4
Weld metal (E-3.23)	15	3	4
HAZ	15	-	-

TABLE 5.3.1-8

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE (UNIT 1)

Capsule Number	Vessel Location	Lead Factor[a]	Withdrawal Time EFPY[b]	Approximate Capsule Fluence (n/cm^2 , E > 1.0 MeV) [a]
U	58.5°	4.15	1.14	3.34×10^{18} [c]
Y	241°	3.99	4.85	1.16×10^{19} [c]
V	61°	3.98	8.78	1.97×10^{19} [c] [d]
X	238.5°	4.21	14.33	3.53×10^{19} [c] [e]
W	121.5°	4.17	Standby	(f)
Z	301.5°	4.17	Standby	(f)

a. Updated in Capsule X dosimetry analysis.

b. Effective Full Power Years (EFPY) from plant startup.

c. Plant-specific evaluation.

d. This capsule was withdrawn at approximately the current end-of-license (36 EFPY) peak fluence.

e. This capsule was withdrawn at approximately (60 EFPY) peak fluence.

f. To be withdrawn at a fluence that is not less than once nor greater than twice the peak EOL fluence for an additional 20-year license renewal term to 80 years. Since the lead factor for both capsule W and Z are the same, either one may be withdrawn for 80 years license renewal.

TABLE 5.3.1-9

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE (UNIT 2)

Capsule Number	Vessel Location	Lead Factor[a]	Withdrawal Time EFPY[b]	Approximate Capsule Fluence (n/cm^2 , E > 1.0 MeV) [a]
U	58.5°	4.10	1.20	3.56×10^{18} [c]
Y	241°	3.95	4.98	1.12×10^{19} [c]
X	238.5°	4.25	7.78	1.78×10^{19} [c]
W	121.5°	4.14	13.29	2.98×10^{19} [c] [d]
Z	301.5°	4.15	18.48	4.16×10^{19} [c]
V	61°	3.84	18.48[e]	-------

a. Updated in Capsule W dosimetry analysis.

b. Effective Full Power Years (EFPY) from plant startup.

c. Plant-specific evaluation.

d. This capsule was withdrawn at a fluence not less than once nor greater than twice the peak EOL fluence for a standard license term of 40 years (36 EFPY). In addition, this capsule was withdrawn at a fluence not less than once nor greater than twice the peak EOL fluence for an additional 20-year license renewal term to 60 years (54 EFPY).

e. Capsule V has been removed from the reactor vessel and placed in the spent fuel pool. No testing or analysis has been performed on this capsule. Reinsertion of this capsule may be considered in the future.

TABLE 5.3.2-1

DELETED

TABLE 5.3.2-2 (SHEET 1 OF 2)

VOGTLE UNIT 1 REACTOR VESSEL CORE BELTLINE REGION TOUGHNESS PROPERTIES

Intermediate Shell Course

Plate B8805-1				Plate B8805-2				Plate B8805-3			
Temp. (°F)	Energy (ft lb)	Lat. Exp (mils)	Shear (%)	Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)	Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	21	17	5	-40	12	9	0	-40	16	12	0
-40	16	14	0	-40	14	11	0	-40	5	6	0
-40	18	15	0	-40	7	5	0	-40	7	7	0
10	31	24	15	10	36	28	20	10	29	21	20
10	28	22	15	10	38	30	20	10	28	22	20
10	36	27	20	10	33	25	15	10	32	24	20
40	50	38	25	40	40	31	20	40	45	32	25
40	40	30	20	40	41	33	20	40	46	34	25
40	44	34	25	40	30	25	15	40	32	24	15
50	44	32	20	70	52	38	25	100	55	43	40
50	50	37	25	70	54	41	30	100	53	46	40
50	52	39	25	70	47	35	20	100	58	49	40
60	58	43	30	80	61	44	35	160	95	70	90
60	57	43	30	80	54	39	30	160	104	74	90
60	72	51	50	80	58	40	30	160	90	67	90
100	77	61	70	100	73	56	50	212	108	73	100
100	70	56	60	100	58	48	40	212	110	77	100
100	75	59	60	100	65	52	50	212	104	75	100
160	83	57	70	160	89	70	90				
160	90	70	95	160	96	73	95				
160	85	65	90	160	92	71	95				
212	80	63	100	212	104	77	100				
212	99	74	100	212	96	72	100				
212	90	68	100	212	99	75	100				

T_{NDT} = 0°F　　T_{NDT} = -10°F　　T_{NDT} = -20°F

RT_{NDT} = 0°F　　RT_{NDT} = +20°F　　RT_{NDT} = +30°F

TABLE 5.3.2-2 (SHEET 2 OF 2)

Lower Shell Course

Plate B8606-1				Plate B8606-2				Plate B8606-3			
Temp. (°F)	Energy (ft lb)	Lat. Exp (mils)	Shear (%)	Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)	Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	9	7	0	-40	8	7	0	-40	6	3	0
-40	8	6	0	-40	7	5	0	-40	7	4	0
-40	8	6	0	-40	7	6	0	-40	8	4	0
10	17	13	5	10	22	22	10	10	24	18	10
10	20	15	5	10	22	20	10	10	19	17	10
10	16	13	5	10	17	17	5	10	24	19	10
40	34	28	15	40	35	27	15	40	40	28	20
40	37	26	20	40	36	29	15	40	35	26	15
40	27	23	15	40	36	26	15	40	43	31	20
70	46	33	25	70	53	40	30	60	39	30	20
70	39	29	20	70	38	29	15	60	54	38	25
70	49	35	25	70	54	42	30	60	49	34	25
80	53	41	30	80	50	36	30	70	51	36	25
80	50	38	25	80	54	40	30	70	58	41	30
80	54	40	30	80	51	37	30	70	54	40	30
100	67	52	40	100	76	56	50	100	74	58	50
100	62	50	40	100	77	59	50	100	61	49	35
100	70	55	40	100	65	52	50	100	72	56	45
160	95	72	90	160	114	78	100	160	105	72	90
160	103	74	95	160	119	76	100	160	113	78	95
160	110	76	95	160	97	74	95	160	113	79	95
212	116	75	100	212	108	76	100	212	118	79	100
212	120	78	100	212	116	79	100	212	120	80	100
212	111	72	100	212	115	80	100	212	117	76	100

Plate B8606-1: T_{NDT} = 50°F, RT_{NDT} = 20°F

Plate B8606-2: T_{NDT} = 10°F, RT_{NDT} = 20°F

Plate B8606-3: T_{NDT} = -20°F, RT_{NDT} = 10°F

TABLE 5.3.2-3

VEGP UNIT 1 REACTOR VESSEL CORE BELTLINE
REGION WELD METAL TOUGHNESS PROPERTIES

Intermediate and Lower Shell Long, and Girth Weld Seams
Weld Code No. G1.43

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-80	6	2	0
-80	5	1	0
-80	6	1	0
-40	65	44	35
-40	10	3	0
-40	8	2	0
10	124	77	80
10	114	66	70
10	96	60	60
60	120	71	70
60	118	70	80
60	119	76	70
100	124	80	80
100	127	78	80
100	123	80	80
160	127	78	100
160	136	82	100
160	140	84	100

T_{NDT} = -80°F

RT_{NDT} = -80°F

TABLE 5.3.2-4 (SHEET 1 OF 2)

VEGP UNIT 2 REACTOR VESSEL CORE BELTLINE REGION TOUGHNESS PROPERTIES

Intermediate Shell Course

Plate R4-1

Temp. (°F)	Energy (ft lb)	Lat. Exp (mils)	Shear (%)
-40	11	6	0
-40	10	5	0
-40	12	7	0
10	30	24	15
10	27	19	10
10	33	24	15
40	43	33	25
40	45	34	25
40	40	31	20
60	52	42	35
60	51	43	35
60	46	36	30
70	61	47	40
70	72	54	50
70	54	43	40
100	80	65	80
100	86	67	90
100	91	70	90
160	97	72	100
160	95	69	100
160	93	67	100

T_{NDT} = -20°F

RT_{NDT} = 10°F

Plate R4-2

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	6	4	0
-40	7	4	0
-40	8	5	0
10	23	17	10
10	33	27	15
10	38	31	20
50	36	28	15
50	40	34	20
50	43	35	20
60	63	49	40
60	66	50	40
60	48	34	25
70	66	53	40
70	59	47	35
70	51	39	30
100	88	67	80
100	85	64	80
100	96	65	95
160	103	69	100
160	99	70	100
160	110	74	100

T_{NDT} = -10°F

RT_{NDT} = 10°F

Plate R4-3

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	9	6	0
-40	7	5	0
-40	7	4	0
10	19	16	5
10	18	15	5
10	19	18	5
60	37	30	15
60	40	36	20
60	55	43	35
80	42	35	30
80	44	38	30
80	48	41	30
90	59	51	40
90	52	46	35
90	50	41	35
100	70	51	70
100	67	50	70
100	74	59	80
160	86	68	100
160	89	66	100
160	78	61	100

T_{NDT} = 0°F

RT_{NDT} = 30°F

TABLE 5.3.2-4 (SHEET 2 OF 2)

Lower Shell Course

Plate B8825-1

Temp. (°F)	Energy (ft lb)	Lat. Exp (mils)	Shear (%)
-40	11	7	0
-40	11	8	0
-40	10	5	0
0	16	10	10
0	14	9	10
0	14	9	10
40	37	25	30
40	28	20	30
40	32	23	30
90	51	38	50
90	49	35	50
90	56	42	50
100	55	41	50
100	60	47	50
100	61	49	50
160	76	60	90
160	79	64	90
160	77	60	90
212	86	66	100
212	82	65	100
212	80	64	100

T_{NDT} = -20°F

RT_{NDT} = 40°F

Plate R8-1

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	14	7	5
-40	12	5	0
-40	11	4	0
0	22	18	20
0	22	16	20
0	20	16	20
40	33	22	25
40	34	23	25
40	35	25	25
90	49	39	40
90	62	46	60
90	62	46	60
100	67	50	65
100	68	51	65
100	69	52	65
160	85	64	95
160	86	65	95
160	90	66	95
212	83	61	100
212	86	64	100
212	93	68	100

T_{NDT} = -20°F

RT_{NDT} = 40°F

Plate B8628-1

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-40	11	4	0
-40	9	3	0
-40	11	4	0
0	15	12	10
0	15	11	10
0	15	11	10
40	26	20	30
40	25	18	30
40	22	17	30
100	55	45	40
100	43	35	30
100	48	36	35
110	61	50	70
110	61	49	70
110	58	45	70
160	76	60	95
160	70	67	95
160	68	58	95
212	76	60	100
212	70	58	100
212	70	57	100
275	76	60	100
275	82	65	100
275	96	68	100

T_{NDT} = -20°F

RT_{NDT} = 50°F

TABLE 5.3.2-5 (SHEET 1 OF 2)

VEGP UNIT 2 REACTOR VESSEL CORE BELTLINE REGION
WELD METAL TOUGHNESS PROPERTIES

Intermediate to Lower Shell Girth Weld Seam

Weld Code No. E3.23

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-80	6	4	0
-80	17	13	0
-80	12	8	0
-40	43	30	20
-40	39	28	15
-40	12	6	0
10	53	33	25
10	45	39	20
10	52	30	25
20	28	19	10
20	68	51	50
20	64	50	50
30	58	45	30
30	59	46	35
30	60	46	35
60	68	54	70
60	71	57	70
60	62	44	60
100	92	72	100
100	90	74	80
100	87	68	90
160	88	69	100
160	94	70	100
160	89	68	100

T_{NDT} = 50°F

RT_{NDT} = -30°F

TABLE 5.3.2-5 (SHEET 2 OF 2)

Intermediate and Lower Shell Long Weld Seams

Weld Code No. 1.60

Temp. (°F)	Energy (ft lb)	Lat. Exp. (mils)	Shear (%)
-30	15	10	0
-30	23	15	5
-30	24	17	5
-10	45	31	20
-10	57	41	30
-10	51	37	25
10	120	82	95
10	105	70	70
10	110	76	80
30	118	70	70
30	111	64	60
30	125	76	80
50	142	86	100
50	153	90	100
50	145	87	100
100	156	88	100
100	159	83	100
100	141	79	100

T_{NDT} = 10°F

RT_{NDT} = -10°F

TABLE 5.3.3-1

REACTOR VESSEL DESIGN PARAMETERS

Design/operating pressure (psig)	2485/2235
Design temperature (°F)	650
Overall height of vessel and closure head, bottom head outside diameter to top of control rod mechanism adapter (ft-in.)	43-10
Thickness of reactor pressure vessel head insulation, minimum (in.)	3
Number of reactor closure head studs	54
Diameter of reactor closure head/studs, minimum shank (in.)	6 13/16
Outside diameter of flange (in.)	205
Inside diameter of flange (in.)	167
Outside diameter at shell (in.)	190 1/2
Inside diameter at shell (in.)	173
Inlet nozzle inside diameter (in.)	27 1/2
Outlet nozzle inside diameter (in.)	29
Clad thickness, minimum (in.)	1/8
Lower head thickness, minimum (in.)	5 3/8
Vessel beltline thickness, minimum (in.)	8 5/8
Closure head thickness (in.)	7
Nominal water volume (ft^3)	3700

TABLE 5.3.3-2

UNIT 1 REACTOR VESSEL VALUES FOR ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS[a]

Material	Cu wt-%	Ni wt-%	Initial RT_{NDT} (°F)	10 CFR 50.61 Predicted RT_{PTS} (°F) 36 EFPY	10 CFR 50.61 Predicted RT_{PTS} (°F) 57 EFPY	Initial USE (ft-lb)	Regulatory Guide 1.99 Predicted USE (ft-lb) 36 EFPY	Regulatory Guide 1.99 Predicted USE (ft-lb) 57 EFPY
Intermediate Shell Plate, B8805-1	0.083	0.597	0	98	104	90	72	69
Intermediate Shell Plate, B8805-2	0.083	0.61	20	118	124	100	80	77
Intermediate Shell Plate, B8805-3[b]	0.062	0.598	30	110 121[d]	115 126[d]	107	86[e]	82[e]
Lower Shell Plate, B8606-1	0.053	0.593	20	94	97	116	93	89
Lower Shell Plate, B8606-2	0.057	0.60	20	97	101	113	90	87
Lower Shell Plate, B8606-3	0.067	0.623	10	95	100	118	94	91
Intermediate Shell Longitudinal Weld Seams 101-124 A, B, & C [c]	0.042	0.102	-80	3 -24[d]	11 -21[d]	134	107[e]	103[e]
Lower Shell Longitudinal Weld Seams 101-142 A, B, & C [c]	0.042	0.102	-80	3 -24[d]	11 -21[d]	134	107[e]	103[e]
Intermediate to Lower Shell Girth Weld 101-171 [c]	0.042	0.102	-80	3 -24[d]	11 -21[d]	134	107[e]	103[e]

NOTES:

a. RT_{PTS} values are based on the peak fluences at the vessel inner radius of 2.155 E19 (for 36 EFPY) and 3.485 E19 (for 57 EFPY). USE was predicted using the 1/4T fluence values based on the peak fluence at the vessel inner radius. The vessel wall thickness is 8.625 inches at the beltline region. Copper and nickel values for all materials are the latest Best Estimate values as of April 2005.
b. Limiting vessel material for Pressurized Thermal Shock event.
c. All of the core region welds were fabricated from wire heat 83653, linde 0091 flux, lot # 3536.
d. Determined using 10 CFR 50.61 with credible surveillance capsule data for the welds and noncredible surveillance capsule data for the plate.
e. Conservatively determined using Position 1.2 (without surveillance capsule data) of Regulatory Guide 1.99, Revision 2; however, surveillance data were available.

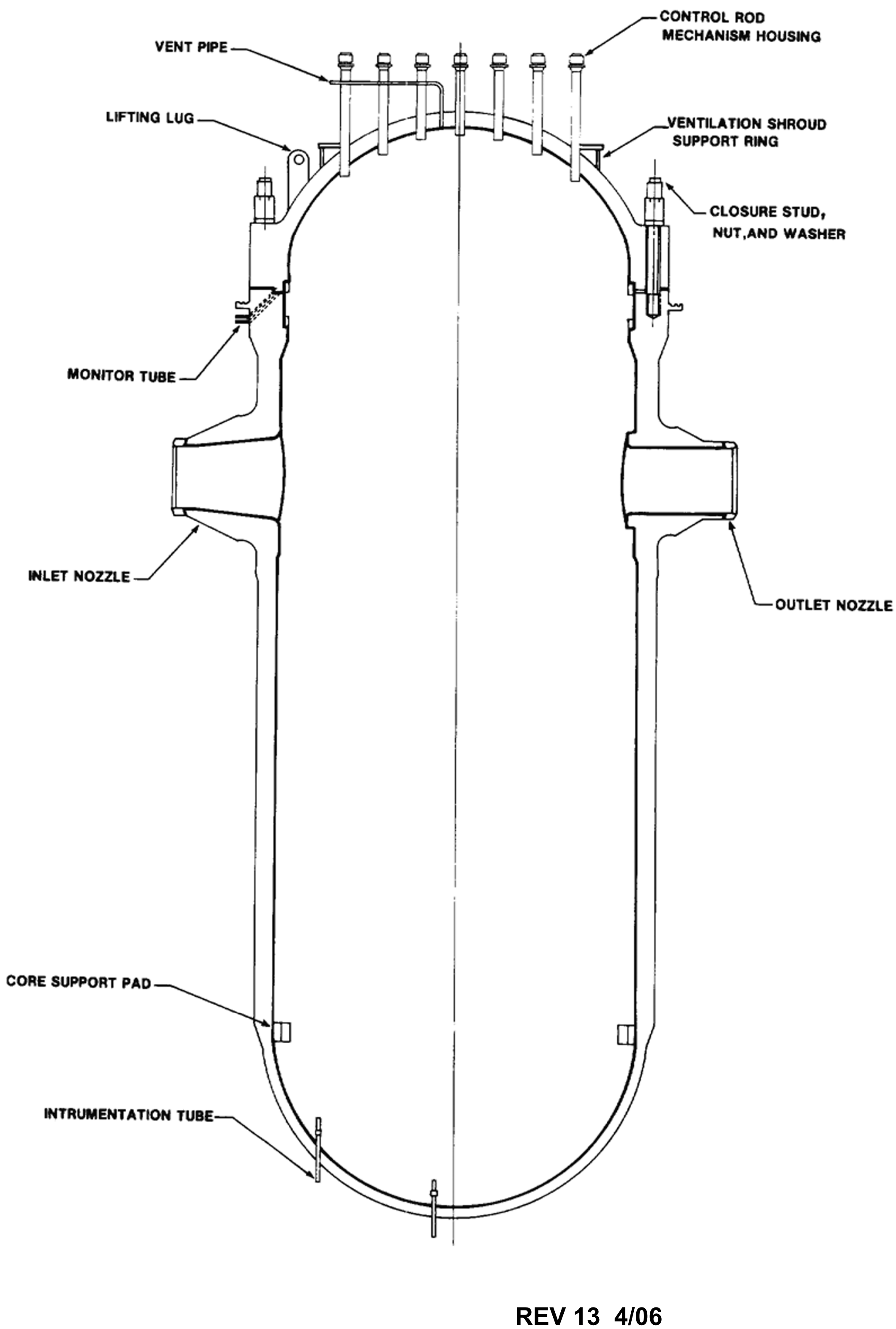
TABLE 5.3.3-3

UNIT 2 REACTOR VESSEL VALUES FOR ANALYSIS OF POTENTIAL PRESSURIZED THERMAL SHOCK EVENTS[a]

Material	Cu wt-%	Ni wt-%	Initial RT_{NDT} (°F)	10 CFR 50.61 Predicted RT_{PTS} (°F) 36 EFPY	10 CFR 50.61 Predicted RT_{PTS} (°F) 57 EFPY	Initial USE (ft-lb)	Regulatory Guide 1.99 Predicted USE (ft-lb) 36 EFPY	Regulatory Guide 1.99 Predicted USE (ft-lb) 57 EFPY
Intermediate Shell Plate, R4-1	0.07	0.63	10	95.9	101.0	95	76	74
Intermediate Shell Plate, R4-2	0.06	0.61	10	87.7	91.9	104	83	81
Intermediate Shell Plate, R4-3	0.05	0.60	30	100.6	104.1	84	67	66
Lower Shell Plate, B8825-1	0.06	0.62	40	117.7	121.9	83	66	65
Lower Shell Plate, R8-1[b]	0.07	0.63	40	125.9	131.0	87	70	68
Lower Shell Plate, B8628-1	0.05	0.59	50	120.6 91.1[d]	124.1 93.4[d]	85	79[e]	79[e]
Intermediate Shell Longitudinal Weld Seams 101-124 A, B, & C [c]	0.05	0.15	-10	92.2 40.8[d]	102.1 45.6[d]	152	140[e]	140[e]
Lower Shell Longitudinal Weld Seams 101-142 A, B, & C [c]	0.05	0.15	-10	92.2 40.8[d]	102.1 45.6[d]	152	140[e]	140[e]
Intermediate to Lower Shell Girth Weld[c]	0.05	0.15	-30	72.2 20.8[d]	82.1 25.6[d]	90	83[e]	83[e]

NOTES:

a. RT_{PTS} values are based on the peak fluence at the vessel inner radius of 1.93 E19 (for 36 EFPY) and 3.06 E19 (for 57 EFPY). USE was predicted using the 1/4T fluence values based on the peak fluence at the vessel inner radius. The vessel wall thickness is 8.625 inches at the beltline region. Copper and nickel values for all materials are the latest Best Estimate values as of April 2005.
b. Limiting vessel material for Pressurized Thermal Shock event.
c. The longitudinal welds were fabricated from wire heat 87005 linde 0091 flux, lot 0145. The girth weld was fabricated from weld wire heat 87005, linde 124 flux, lot 1061.
d. Determined using 10 CFR 50.61 with credible surveillance capsule data.
e. Determined using Position 2.2 (with credible surveillance capsule data) of Regulatory Guide 1.99, Revision 2.



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SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	REACTOR VESSEL
		FIGURE 5.3.3–1

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMP ASSEMBLY

5.4.1.1 Design Bases

The reactor coolant pump assembly ensures an adequate core cooling flowrate for sufficient heat transfer to maintain a departure from nucleate boiling ratio greater than the design basis limit within the parameters of operation. The required net positive suction head (NPSH) is by conservative pump design always less than that available by system design and operation.

Sufficient pump assembly rotation inertia is provided by a motor flywheel, motor rotor, and pump rotating parts which provide adequate flow during coastdown conditions. This forced flow following an assumed loss of offsite electrical power and the subsequent natural circulation effect provides the core with adequate cooling.

The reactor coolant pump is shown in figure 5.4.1-1. The reactor coolant pump design parameters are given in table 5.4.1-1.

The automatic trip of the reactor coolant pumps discussed in TMI Action Item II.K.3.5 has not been implemented. Generic studies[3][4] performed in response to Nuclear Regulatory Commission Generic Letters 83-10c and d[5] have demonstrated the acceptability of manual tripping of the reactor coolant pumps following accident events postulated to occur at VEGP.

The auxiliary component cooling water (ACCW) pump motors are accessible to the emergency diesel generators as described in section 8.3 and meet the intent of TMI Action Item II.K.3.25.

5.4.1.2 Pump Assembly Description

5.4.1.2.1 Design Description

The reactor coolant pump is a vertical, single-stage, controlled leakage, centrifugal pump designed to pump large volumes of reactor coolant at high temperatures and pressures.

The pump assembly consists of three major sections. They are the hydraulics, the seals, and the motor.

A. The hydraulic section consists of the casing, impeller, turning vane diffuser, and diffuser adapter.

B. The shaft seal section consists of the No. 1 controlled leakage film-riding face seal, No. 2 and No. 3 rubbing face seals, and a shutdown seal assembly (SDS). The seals are contained within the thermal barrier heat exchanger assembly and seal housing. The SDS is housed within the No. 1 seal area and is a passive device actuated by high temperature resulting from a loss of seal injection and ACCW cooling to the thermal barrier heat exchanger.

C. The motor section consists of a drip-proof squirrel cage induction motor with a vertical solid shaft, an oil-lubricated, double-acting Kingsbury type thrust bearing, upper and lower oil-lubricated radial guide bearings, and a flywheel.

Additional components of the pump are the shaft, pump radial bearing, thermal barrier heat exchanger assembly, coupling, spool piece, and motor stand.

In the event of a loss of seal injection and ACCW flow to the thermal barrier heat exchanger, the SDS will actuate only when the No. 1 seal temperature reaches 250ºF to 300ºF. SDS actuation limits leakage from the RCS through the RCP seal package. Leakage is limited when the SDS thermal actuator retracts due to intrusion of hot reactor coolant water into the seal area, which causes the SDS piston and polymer rings to constrict around the shaft.

5.4.1.2.2 Description of Operation

The reactor coolant enters the suction nozzle, is pumped by the impeller through the diffuser, and exits through the discharge nozzle. The diffuser adapter limits the leakage of reactor coolant back to the suction.

Seal injection flow, under slightly higher pressure than the reactor coolant, enters the pump through a connection on the thermal barrier flange and is directed into the plenum between the thermal barrier housing and the shaft. The flow splits, with the major portion flowing down the shaft through the radial bearing and into the RCS. The remaining seal injection flow passes up the shaft through the seals.

The ACCW is provided to the thermal barrier heat exchanger. During normal operation, the thermal barrier limits the heat transfer from hot reactor coolant to the radial bearing and to the seals. In addition, if a loss of seal injection flow should occur, the thermal barrier heat exchanger cools the reactor coolant to an acceptable level before it enters the bearing and seal area.

The reactor coolant pump motor oil-lubricated bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearing is a double-acting Kingsbury type.

Auxiliary component cooling water is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler.

The oil spillage protection system is attached to the reactor coolant pump motor and is provided to contain and channel oil to a common collection point.

The motor is a drip-proof squirrel cage induction motor with Class F thermalastic epoxy insulation, fitted with external water/air coolers. The rotor and stator are of standard construction. Six resistance temperature detectors are embedded in the stator windings to sense stator temperature. A flywheel and an antireverse rotation device are located at the top of the motor.

The internal parts of the motor are cooled by air. Integral vanes on each end of the rotor draw air in through cooling slots in the motor frame. This air passes through the motor with particular emphasis on the stator end turns. It is then routed to the external water/air coolers, which are supplied with ACCW. Each motor has two such coolers, mounted diametrically opposed to each other. Coolers are sized to maintain optimum motor-operating temperature. The air is finally exhausted to the containment environment.

Each of the reactor coolant pump assemblies is equipped for continuous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by two relative shaft probes mounted on top of the pump seal housing; the probes are located 90° apart in the same

horizontal plane and mounted near the pump shaft. Frame vibration is measured by two velocity seismoprobes located 90° apart in the same horizontal plane and mounted at the top of the motor support stand. Proximeters and converters linearize the probe output, which is displayed on monitor meters in the control building, with alarm functions available in the control room. The monitor meters automatically indicate the highest output from the relative probes and seismoprobes; manual selection allows monitoring of individual probes. Indicator lights display caution and danger limits of vibration.

The spool piece, which is a removable shaft segment, is located between the motor coupling flange and the pump coupling flange; the spool piece allows removal of the pump seals with the motor in place. The pump internals, motor, and motor stand can be removed from the casing without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings, and special parts.

5.4.1.2.3 Loss of Seal Injection

Should a loss of seal injection to the reactor coolant pumps occur, the pump radial bearing and seals are lubricated by reactor coolant flowing up through the pump. Under these conditions, the ACCW continues to provide flow to the thermal barrier heat exchanger; this heat exchanger, functioning in its backup capacity, cools the reactor coolant before it enters the pump radial bearing and the shaft seal area. The loss of seal injection flow may result in a temperature increase in the pump bearing area, a temperature increase in the seal area, and a resultant increase in the No. 1 seal leak rate; however, pump operation can be continued provided these parameters remain within the allowable limits. Under certain low seal leak-off conditions it is possible that seal temperatures may increase above allowable limits in 1 to 2 hours requiring shutdown of the pump.

5.4.1.2.4 Loss of Auxiliary Component Cooling Water

Should a loss of ACCW to the reactor coolant pumps occur, the chemical and volume control system continues to provide seal injection flow to the reactor coolant pumps; the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling. However, the loss of ACCW to the motor bearing oil coolers will result in an increase in oil temperature and a corresponding rise in motor bearing metal temperature. It has been demonstrated by testing at the Westinghouse Electromechanical Division that the reactor coolant pumps will incur no damage as a result of an ACCW flow interruption of 10 min.

Safety-related transmitters will be provided to redundantly monitor ACCW flow for the upper and lower reactor coolant pump bearings, as well as to monitor ACCW flow for the reactor coolant pump thermal barriers. These transmitters will provide flow indication and actuate low-flow alarms in the control room.

Operating procedures are provided for a loss of ACCW and seal injection to the reactor coolant pumps and/or motors. Included in these operating procedures is the provision to trip the reactor if ACCW flow, as indicated by the instrumentation discussed above, is lost to the reactor coolant pump motors and cannot be restored within 10 min. The reactor coolant pumps will also be tripped following the reactor trip.

5.4.1.3 Design Evaluation

5.4.1.3.1 Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flowrates. Initial RCS tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The estimated performance characteristic is shown in figure 5.4.1-2. The knee, at about 25% design flow, introduces no operational restrictions, since the pumps only operate at a speed which corresponds to full flow.

The reactor coolant pump motor is tested, without mechanical damage, at overspeeds up to and including 125% of normal speed. The integrity of the flywheel during a loss-of-coolant accident (LOCA) is demonstrated in reference 1, which is undergoing generic review by the NRC.

The reactor trip system ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also ensures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long-term tests were conducted on less than full-scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the No. 1 seal (seal ring) is such as to allow deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The spring rate of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the No. 1 seal entirely bypassed (full system pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for a period of time which is sufficient to secure the pump; even if the No. 1 seal fails entirely during normal operation, the No. 2 seal would maintain these small leakage rates if the proper action is taken by the operator. The plant operator is warned of No. 1 seal damage by an increase in No. 1 seal leakoff rate. Following warning of excessive seal leakage conditions, the plant operator should close the No. 1 seal leakoff line and secure the pump, as specified in the instruction manual.

Gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal leakage conditions.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the ACCW for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of offsite electrical power, so that ACCW flow and seal injection flow are automatically restored.

5.4.1.3.2 Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow following loss of offsite electrical power, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The

coastdown flow transients are provided in the figures in section 15.3. The pump/motor system is designed for the safe shutdown earthquake at the site. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of loss of offsite electrical power coincident with the safe shutdown earthquake. Core flow transients and figures are provided in subsections 15.3.1 and 15.4.4. An inadvertent actuation of the SDS on the rotating assembly will not have any measurable impact on the RCP coastdown.

5.4.1.3.3 Bearing Integrity

The design requirements for the reactor coolant pump bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. The surface bearing stresses are held at very low values and even under the most severe seismic transients do not begin to approach loads that cannot be adequately carried for short periods of time.

Because there are no established criteria for short-time, stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

Low oil levels in the motor lube oil sumps signal alarms in the control room. Each motor bearing contains embedded temperature detectors, so that initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This requires pump shutdown. If these indications are ignored and the bearing proceeds to failure, the low melting point of Babbitt metal on the pad surfaces ensures that sudden seizure of the shaft will not occur. In this event, the motor continues to operate, since it has sufficient reserve capacity to drive the pump under such conditions. However, the high torque required to drive the pump will require high current, which leads to the motor being shutdown by the electrical protection systems.

5.4.1.3.4 Locked Rotor

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss of coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, since it is still supported on a shaft with two bearings. Flow transients are provided in the figures in subsection 15.3.3 for the assumed locked rotor.

There are no other credible sources of shaft seizure other than impeller rubs. A sudden seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the antirotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are initially given by high-temperature signals from the bearing water temperature detector, by excessive No. 1 seal leakoff indications, and by offscale No. 1 seal leakoff indications. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble, and the pump is shut down for investigation. Note an inadvertent actuation of the SDS on a rotating assembly will not prevent sufficient cooling flow to the reactor core.

5.4.1.3.5 Critical Speed

The reactor coolant pump shaft is designed so that its operating speed is below its first critical speed. This shaft design, even under the most severe postulated transient, gives low values of actual stress.

5.4.1.3.6 Missile Generation

Precautionary measures taken to preclude missile formation from reactor coolant pump components ensure that the pumps will not produce missiles under any anticipated accident condition.

Appropriate components of the reactor coolant pump have been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator frame. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation are contained in reference 1.

5.4.1.3.7 Pump Cavitation

The minimum NPSH required by the reactor coolant pump at best estimate flow is approximately a 300-ft head (approximately 133 psi). In order for the controlled leakage seal to operate correctly, it is necessary to require a minimum differential pressure of approximately 200 psi across the No. 1 seal. This corresponds to a primary loop pressure at which the minimum NPSH is exceeded, and no limitation on pump operation occurs from this source.

5.4.1.3.8 Pump Overspeed Considerations

For turbine trips actuated by either the reactor trip system or the turbine protection system, the generator and reactor coolant pumps remain connected to the external network for 30 s to prevent any pump overspeed condition.

An electrical fault requiring immediate trip of the generator (with resulting turbine trip) could result in an overspeed condition. However, the turbine control system and the turbine intercept valves limit the overspeed to less than 120%. As additional backup, the turbine protection system on Unit 1 utilizes redundant overspeed protection with trip manifold assemblies using primary and emergency overspeed trips set at 110% and 110.5%, while Unit 2 has a mechanical overspeed protection trip, usually set at about 110% (of turbine speed). In case a generator trip deenergizes the pump buses, the reactor coolant pump motors are transferred from the unit auxiliary transformers to the reserve auxiliary transformers within 6 to 10 cycles. Further discussion of pump overspeed considerations is contained in reference 1.

5.4.1.3.9 Antireverse Rotation Device

Each of the reactor coolant pumps is provided with an antireverse rotation device in the motor. This antireverse mechanism consists of pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

At an approximate forward speed of 70 rpm, the pawls drop and bounce across the ratchet plate; as the motor continues to slow, the pawls drag across the ratchet plate. After the motor has slowed and come to a stop, the dropped pawls engage the ratchet plate, and as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed, the pawls are bounced into an elevated position and are held in that position by friction resulting from centrifugal forces acting upon the pawls. While the motor is running at speed, there is no contact between the pawls and ratchet plate.

Considerable plant experience with the design of the antireverse rotation device has shown high reliability of operation.

5.4.1.3.10 Shaft Seal Leakage

During normal operation, leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series such that reactor coolant leakage to the containment is essentially zero. Seal injection flow is directed to each reactor coolant pump via a seal water injection filter. It enters each pump through a connection on the thermal barrier flange. Here the flow splits; the major portion flows down the shaft to cool the bearing and enters the RCS, and the remainder flows up the shaft through the seals. This seal flow provides a backpressure on the No. 1 seal and a controlled flow through the seal. After passing through the No. 1 seal, most of the flow leaves the pump via the No. 1 seal leakoff line. Minor flow passes through the No. 2 seal to its leakoff line. A backflush injection from a head tank flows into the No. 3 seal between its double dam seal area. At this point the flow divides, with half flushing through one side of the seal and out the No. 2 seal leakoff, while the remaining half flushes through the other side and out the No. 3 seal leakoff. This arrangement ensures essentially zero leakage of reactor coolant or trapped gases to the containment.

In the event of a loss of seal injection and ACCW flow to the thermal barrier heat exchanger, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. Once the temperature of the No. 1 seal reaches 250ºF to 300ºF, the SDS actuates by retraction of a thermal actuator, causing the SDS piston and polymer rings to constrict around the shaft. SDS actuation controls shaft seal leakage and limits the loss of reactor coolant through the RCP seal package.

5.4.1.3.11 Seal Discharge Piping

The No. 1 seal reduces the leakoff pressure to that of the volume control tank. Water from each pump No. 1 seal is piped to a common manifold, through the seal water return filter, and through the seal water heat exchanger where the temperature is reduced to that of the volume control tank. The No. 2 and 3 leakoff lines route No. 2 and 3 seal leakage to the reactor coolant drain tank and the containment sump, respectively.

5.4.1.4 <u>Tests and Inspections</u>

The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, for inservice inspection of nuclear RCSs.

The pump casing is cast in one piece. Support feet are cast integrally with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for visual access to the pump casing.

The reactor coolant pump quality assurance program is given in table 5.4.1-2.

Tests and inspections performed under the following license renewal aging management programs are credited as applicable to the reactor coolant pumps and their subcomponents:

- Bolting Integrity Program (see subsection 19.2.2).
- Boric Acid Corrosion Control Program (see subsection 19.2.3).
- Inservice Inspection Program (see subsection 19.2.13).
- Water Chemistry Control Program (see subsection 19.2.28).

5.4.1.5 Pump Flywheel

The integrity of the reactor coolant pump flywheel is ensured on the basis of the following design and quality assurance procedures[a].

5.4.1.5.1 Design Basis

The calculated stresses at operating speed are based on stresses due to centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The reactor coolant pumps run at approximately 1190 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of offsite electrical power. For conservatism, however, 125% of operating speed was selected as the design speed for the reactor coolant pumps. The flywheels are given a manufacturer's test of 125% of the maximum synchronous speed of the motor.

5.4.1.5.2 Fabrication and Inspection

The flywheel consists of two thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, i.e., an electric furnace with vacuum degassing. Each plate is fabricated from SA-533, Grade B, Class 1 steel. Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the requirements of NRC Regulatory Guide 1.14.

Flywheel blanks are flame cut from the SA-533, Grade B, Class 1 plates with at least 1/2 in. of stock left on the outer surface and bore surface for machining to final dimensions. The finished machined bores, keyways, and drilled holes are subjected to magnetic particle or liquid penetrant examinations in accordance with the requirements of Section III of the ASME Code.

[a] Reactor coolant pump flywheel fatigue is evaluated as a time-limited aging analysis (TLAA) for license renewal (see paragraph 19.4.2.3).

The finished flywheels, as well as the flywheel material (rolled plate), are subjected to 100% volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The reactor coolant pump motors are designed such that, by removing the cover to provide access, the flywheel is available to allow an inservice inspection program which was originally in accordance with the recommendations of Regulatory Guide 1.14, referencing Section XI of the ASME Code for inspection scheduling purposes. Subsequently, the NRC authorized the use of an alternative flywheel inspection as addressed in Westinghouse WCAP-14535A, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination.” In this alternative inspection program, each flywheel is inspected at least once every 10 years by conducting either: (1) an in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or (2) a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

5.4.1.5.3 Material Acceptance Criteria

The reactor coolant pump motor flywheel conforms to the following material acceptance criteria:

A. The nil ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests which exhibit no-break performance at 20°F in accordance with ASTM E-208. The above drop weight tests demonstrate that the NDTT of the flywheel material is no higher than 10°F.

B. A minimum of three Charpy V-notch (C_V) impact specimens from each plate shall be tested at ambient (70°F) temperature in accordance with the ASME SA-370 specification. The Charpy V-notch (C_V) energy in both the parallel and normal orientation with respect to the final rolling direction of the flywheel plate material is at least 50 ft-lb and 35-mil lateral expansion at 70°F, and, therefore, the flywheel material has a reference nil ductility temperature (RT_{NDT}) of 10°F. An evaluation of flywheel overspeed has been performed which concludes that flywheel integrity will be maintained.[(1)]

Thus, it is concluded that flywheel plate materials are suitable for use and can meet Regulatory Guide 1.14 acceptance criteria on the bases of suppliers’ certification data. The degree of compliance with Regulatory Guide 1.14 is further discussed in section 1.9.

5.4.1.6 <u>References</u>

1. "Reactor Coolant Pump Integrity in LOCA," <u>WCAP-8163</u>, September 1973.

2. "Safety-Related Research and Development for Westinghouse Pressurized Water Reactor, Program Summaries; Winter 1977 - Summer 1978," <u>WCAP-8768</u>, Revision 2, October 1978.

3. Westinghouse Owners Group Letter OG-110 to Nuclear Regulatory Commission, December 11, 1983.

4. Westinghouse Owners Group Letter OG-117 to Nuclear Regulatory Commission, March 9, 1983.

5. "Resolution of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," Nuclear Regulatory Commission Generic Letters 83-10c and d, February 8, 1983.

6. “Use of Westinghouse Shield Passive Shutdown Seal for Flex Strategies,” TR-FSE-14-1-P, Revision 1, March 2014.
7. “PRA Model for the Generation III Westinghouse Shutdown Seal, “PWROG-14001-P, Revision 0, June 2014.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

Steam generator design data are given in table 5.4.2-1. Code classifications of the steam generator components are given in section 3.2. Although the American Society of Mechanical Engineers (ASME) classification for the secondary side is specified to be Class 2, the current philosophy is to design all pressure-retaining parts of the steam generator, and thus both the primary and secondary pressure boundaries, to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The design stress limits, transient conditions, and combined loading conditions applicable to the steam generator are discussed in subsection 3.9.1. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation and the bases for the estimates are given in chapter 11. The accident analysis of a steam generator tube rupture is discussed in chapter 15.

A design objective of the internal moisture separation equipment is that moisture carryover should not exceed 0.25% by weight under the following conditions:

A. Steady-state operation up to 100% of full-load steamflow with water at the normal operating level.

B. Loading or unloading at a rate of 5% of full-power steamflow per minute in the range from 15 to 100% of full-load steamflow.

C. A step load change of 10% of full power in the range from 15- to 100% full-load steamflow.

The water chemistry on the reactor side, selected to provide the necessary boron content for reactivity control, should minimize corrosion of reactor coolant system (RCS) surfaces. The effectiveness of the water chemistry of the steam side in affecting corrosion control is discussed in chapter 10. Compatibility of steam generator tubing with both primary and secondary coolants is discussed further in paragraph 5.4.2.4.3.

The steam generator is designed to minimize unacceptable damage from mechanical or flow-induced vibration. Tube support adequacy is discussed in paragraph 5.4.2.3.3. The tubes and tube sheet are analyzed and confirmed to withstand the maximum accident loading conditions as they are defined in subsection 3.9.1. Further consideration is given in paragraph 5.4.2.3.4 to the effect of tube-wall thinning on accident condition stresses.

5.4.2.2 Design Description

The steam generator is a Model F, vertical-shell, and U-tube evaporator, with integral moisture separating equipment. Figure 5.4.2-1 shows the model, indicating several of its design features which are described in the following paragraphs.

On the primary side, the reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles located in the hemispherical bottom head of the steam generator. The

head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tube sheet.

Steam is generated on the shell side, flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle or the auxiliary feed nozzle. Stratification and striping in the main nozzle region is reduced by utilization of an auxiliary feedwater nozzle during startup, hot standby, and power escalation. During startup, the auxiliary feedwater system supplies the steam generator via the auxiliary nozzle. This mode continues up to about 4% power. At this level the switch is made to the main feedwater pumps which continue to feed into the auxiliary nozzle. Between 12- and 20% power, the main feed pumps feed into the main feed nozzle. The flowrate and temperature of the feedwater above 12% power should be high enough to substantially reduce stratification and striping loads on the main feed nozzle. During normal operation, the bypass line isolation valve is normally open, therefore, some flow will always be directed to the auxiliary nozzle. During hot standby the steam generator is supplied by the auxiliary feed system via the auxiliary nozzle. The water entering through the main feed nozzle is distributed circumferentially around the steam generator by means of a feedwater ring and then flows through an annulus between the tube wrapper and shell. The feedwater enters the ring via a welded thermal sleeve connection and leaves it through inverted J-tubes located at the flow holes which are at the top of the ring. The J-tubes are arranged to distribute the bulk of the colder feedwater to the hot leg side of the tube bundle. The feed ring is designed to minimize conditions which can result in water hammer occurrences in the feedwater piping. Water that enters through the auxiliary feed nozzle is not distributed through a feed ring. At the bottom of the wrapper, the water is directed toward the center of the tube bundle by a flow distribution baffle. This baffle arrangement serves to minimize the tendency in the relatively low-velocity fluid for sludge deposition. Flow-blocking devices discourage the water from flowing up the bypass lane as it enters the tube bundle where it is converted to a steam-water mixture. Subsequently, the steam-water mixture from the tube bundle rises into the steam drum section, where 16 individual centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators for further moisture removal, increasing its quality to a designed minimum of 99.75%. The moisture separators divert the separated water, which is combined with entering feedwater to flow back down the annulus between the wrapper and shell for recirculation through the steam generator. The dry steam exits from the steam generator through the outlet nozzle, which is provided with a steamflow restrictor. (Refer to subsection 5.4.4.) The moisture carryover is expected to be well below the internal moisture separator design objective of 0.25 weight% for the cases with a primary T_{avg} at the high end of the proposed range of 570.7°F to 588.4°F.

With the T_{avg} at the lower end of the range, moisture carryover may exceed 0.25%. However, even for the low T_{avg} condition, moisture carryover will remain below 0.5%, which is the threshold beyond which erosion and corrosion of the piping and valves downstream of the steam generators are of a concern.

5.4.2.3 Design Evaluation

5.4.2.3.1 Forced Convection

The effective heat transfer coefficient is determined by the physical characteristics of the Model F steam generator and the fluid conditions in the primary and secondary systems for the nominal 100% design case. It includes a conservative allowance for fouling and uncertainty. A

designed heat transfer area is provided to permit the achievability of the full-design heat-removal rate.

5.4.2.3.2 Natural Circulation Flow

In the event of loss of offsite power and consequential loss of forced circulation within the RCS, natural circulation functions to remove core decay heat and permit the plant to be stabilized in the hot standby operational mode. Under this condition, pressurizer pressure is maintained by one pressurizer backup heater group, which is powered from one of the emergency electrical buses. To ensure that one backup heater group is available, assuming a single failure, the VEGP is designed with the capability for manual loading of separate backup heater groups (i.e., group A and group B, respectively) on independent emergency, electrical buses (i.e., train A and train B, respectively) within 1 h, following loss of offsite power. One heater group (485 kW Unit 1 group A, 415 kW Unit 2 group A, 485 kW Unit 1 and Unit 2 group B) loaded within 1 h is sufficient to satisfy the minimum heat capacity requirement (150 kW) for natural circulation following loss of offsite power. This minimum heat capacity requirement conservatively covers the pressurizer heat losses at or below normal operating pressure, following loss of offsite power, and permits pressurizer pressure to be stabilized and maintained at any desired value.

The pressurizer heater design is such that following loss of offsite power and assuming a single failure, sufficient heater capacity is available to stabilize pressurizer pressure and preclude boiling in the RCS. If pressurizer heaters are not available to maintain pressurizer pressure, the RCS could be cooled via secondary side steam release. This operation would prevent saturation pressure from being reached in the active portion of the RCS. Depending on the circumstances, saturation conditions could occur in the upper head of the reactor vessel which would lead to the formation of a steam bubble. This would not impede natural circulation flow, however, since any vapor that entered the hot legs and subsequently the steam generators would be condensed by heat transfer to the secondary side of the steam generators. Vapor would be condensed in this manner as long as the steam generator tube bundle remains submerged.

5.4.2.3.3 Mechanical and Flow-Induced Vibration Under Normal Operating Conditions

In the design of the steam generators, the possibility of degradation of tubes due to either mechanical or flow-induced excitation is thoroughly evaluated. This evaluation includes detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

In evaluating degradation due to vibration, consideration is given to sources of excitation such as those generated by primary fluid flowing within the tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. During normal operation, the effects of primary fluid flow within the tubes and mechanically-induced vibration are considered to be negligible and should cause little concern.

Thus, the primary source of tube vibrations is the hydrodynamic excitation by the secondary fluid on the outside of the tubes. In general, three vibration mechanisms have been identified:

- Vortex shedding.
- Fluidelastic excitation.

- Turbulence.

Vortex shedding does not provide detectable tube bundle vibration. There are several reasons why this happens.

A. Flow turbulence in the downcomer and tube bundle inletregion inhibit the formation of Von Karman's vortex train.

B. The spatial variations of crossflow velocities along the tube preclude vortex shedding at a single frequency.

C. Both axial flow and crossflow velocity components exist on the tubes. The axial flow component disrupts the Von Karman vortices.

Fluid elastic excitation was observed during the testing. The amplitudes of the vibrations were two orders of magnitude smaller than those of the turbulent flow-induced vibrations. Therefore, fluid elastic excitation is excluded from consideration as a factor in steam generator tube bundle vibrations.

Flow-induced vibrations due to flow turbulence cause stresses in the tubes that are two orders of magnitude below the endurance limit (30,000 psi) of the tube material. Therefore, the contribution to fatigue is negligible; and fatigue degradation from flow-induced vibration is not anticipated during normal operation.

Summarizing the results of analyses and tests of the steam generator for vibration, it can be stated that a check of all modes of tube vibration mechanisms has been completed. The conclusions that can be drawn are that the primary source of tube vibration is fluid turbulence and that the magnitude of the vibration is so small that, when combined with its total random nature, its contribution to tube fatigue is negligible. Therefore, fatigue degradation due to flow-induced vibration is not anticipated.

The impact of operation at a power level of 3653 MWt has been evaluated and it is concluded that significant levels of vibration will not occur from the fluid-elastic, vortex shedding, or turbulent mechanisms. The projected level of tube wear as a result of vibration will remain small and will not result in unacceptable tube wear.

5.4.2.3.4 Allowable Tube-Wall Thinning Under All Plant Conditions[a]

An analysis has been performed to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions at a power level of 3653 MWt. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The allowable tube repair limit, in accordance with Regulatory Guide 1.121, is obtained by incorporating into the structural limit a growth allowance for continued operation until the next scheduled inspection, as well as an allowance for eddy current measurement uncertainty.

Calculations have been performed to establish the tube straight leg (free span) region of the tube for degradation over an unlimited axial length, and for degradation over limited axial extent at the tube support plate (TSP), the flow distribution baffle (FDB), and antivibration bar intersections for the 3653 MWt power level conditions.

The minimum structural limit is calculated to be 60.0% allowable tube wall loss for the high / low T_{avg} straight length location. The straight leg structural limit is also applicable to the tube / AVB

[a] Loss of material from the steam generator tube walls was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in paragraph 19.4.6.3.

tangent points. These tube / AVB tangent points correspond to row 7 for the inner sets of AVBs, row 20 for the middle sets of AVBs, and row 31 for the outer sets of AVBs. The enveloping structural limits at the FDB, TSP, and anti-vibration bars at high / low T_{avg} (other than the tube / AVB tangent points) are 66.5%, 62.0%, and 72.0%, respectively.

Structural integrity performance criteria (SIPC) requirements were evaluated to address circumferential cracks at the TSP and U-bend regions.

Results are provided in% degraded area (reference 3).

5.4.2.4 Steam Generator Materials

5.4.2.4.1 Selection and Fabrication of Materials

All pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section III of the ASME Code. A general discussion of materials specifications is given in subsection 5.2.3, with types of materials listed in tables 5.2.3-1 and 5.2.3-2. Fabrication of reactor coolant pressure boundary (RCPB) materials is also discussed in subsection 5.2.3, particularly in paragraphs 5.2.3.3 and 5.2.3.4

Testing has justified the selection of corrosion-resistant Inconel-600, a nickel-chromium-iron alloy (ASME SB-163), for the steam generator tubes. The channel head divider plate is Inconel (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles, and manways are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel (ASME SFA-5.14). The tubes are then seal welded to the tube-sheet cladding. These fusion welds, performed in compliance with Sections III and IX of the ASME Code, are dye-penetrant inspected and leakproof tested before each tube is hydraulically expanded the full depth of the tube-sheet bore.

Code cases used in material selection are discussed in subsection 5.2.1. The extent of conformance with Regulatory Guides 1.84, Design and Fabrication Code Case Acceptability ASME Section III, Division 1, and 1.85, Materials Code Case Acceptability ASME Section III, Division 1, is discussed in section 1.9.

During manufacture, cleaning is performed on the primary and secondary sides for the steam generator in accordance with written procedures which follow the guidance of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, and American National Standards Institute (ANSI) Standard N45.2.1-1973, Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants. Onsite cleaning and cleanliness control also follow the guidance of Regulatory Guide 1.37 as discussed in section 1.9. Cleaning process specifications are discussed in paragraph 5.2.3.4.

The fracture toughness of the materials is discussed in paragraph 5.2.3.3. Adequate fracture toughness of ferritic materials in the RCPB is provided by compliance with 10 CFR 50, Appendix G, Fracture Toughness Requirements, and Paragraph NB-2300 of Section III of the ASME Code, and by meeting the requirements of General Design Criteria 1, 14, 15, and 31.

5.4.2.4.2 Steam Generator Design Effects on Materials

Several features have been introduced into the Model F steam generator to minimize the deposition of contaminants from the secondary-side flow. Such deposits could otherwise

produce a local environment in which adverse conditions could develop and result in material corrosion. The support plates are made of corrosion resistant stainless steel 405 alloy and incorporate a four-lobe-shaped tube hole design that provides greater flow area adjacent to the tube outer surface and eliminates the need for interstitial flow holes. The resulting increase in flow provides higher sweeping velocities at the tube/tube support plate intersections. These increased sweeping velocities reduce the potential for sludge deposition on the tube support plates. Historically, sludge removal from the tube support plates has not been performed. Figure 5.4.2-2 is an illustration of the quatrefoil broached holes. This modification in the support plate design is a major factor contributing to the increased circulation ratio. The increased circulation results in increased flow in the interior of the bundle, as well as increased horizontal velocity across the tube sheet, which reduces the tendency for sludge deposition. The effect of the increased circulation on the vibrational stability of the tube bundle has been analyzed with consideration given to flow-induced excitation frequencies. The unsupported span length of tubing in the U-bend region and the corresponding optimum number of antivibration bars has been determined. The antivibration bars are fabricated from square Inconel barstock which is then chromium treated to improve frictional characteristics. Because of the increased circulation ratio, the moisture separating equipment has been modified to maintain an adequate margin with respect to the moisture carryover. To provide added strength as well as resistance to vibration, the quatrefoil tube support plate thickness has been increased. In addition, either 8 or 12 peripheral supports provide stability to the plates so that tube fretting or wear due to flow-induced plate vibrations at the tube support contact regions is minimized. For the top tube support plate, there is a continuous ring to distribute the lateral loads.

Assurance against significant flow-induced tube vibration is provided by a combination of analysis and testing.

Combining both vortex shedding and turbulence effects in a conservative manner, the maximum predicted local tube wear depth over a 40-year operating design objective[a] remains less than 0.006 inch with the operation at a power level of 3653 MWt. This value is considerably below the plugging limit for Model F steam generators.

5.4.2.4.3 Compatibility of Steam Generator Tubing With Primary and Secondary Coolants[b]

As mentioned in paragraph 5.4.2.4.1, corrosion tests which subjected the steam generator tubing material, Inconel-600 (ASME SB-163), to simulated steam generator water chemistry have indicated that the loss due to general corrosion over the 40-year operating design objective[a] is insignificant compared to the tube-wall thickness. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel-600 has resisted general corrosion in severe operating water conditions. Many reactor years of successful operation have shown the same low general corrosion rates as indicated by the laboratory tests.

Recent operating experience, however, has revealed areas on secondary surfaces where localized corrosion rates were significantly greater than the low general corrosion rates. Both intergranular stress corrosion and tube-wall thinning were experienced in localized areas,

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[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).
[b] The Water Chemistry Control Program is credited as a license renewal aging management program (see subsection 19.2.28).

although not simultaneously at the same location or under the same environmental conditions (water chemistry, sludge composition).

The adoption of the all volatile treatment (AVT) control program minimizes the possibility for recurrence of the tube-wall thinning phenomenon. Successful AVT operation requires maintenance of low concentrations of impurities in the steam generator water, thus reducing the potential for formation of highly concentrated solutions in low-flow zones, which is the precursor of corrosion. By restriction of the total alkalinity in the steam generator and prohibition of extended operation with free alkalinity, the AVT program should minimize the possibility for recurrence of intergranular corrosion in localized areas due to excessive levels of free caustic.

Laboratory testing has shown that the Inconel-600 tubing is compatible with the AVT environment. Isothermal corrosion testing in high-purity water has shown that commercially produced Inconel-600 exhibiting normal microstructures tested at normal engineering stress levels does not suffer intergranular stress corrosion cracking in extended exposure to high-temperature water. These tests also showed that no general type corrosion occurred. A series of autoclave tests in reference secondary water with planned excursions has produced no corrosion attack after 1938 days of testing on any as-produced Inconel-600 tube samples.

Model boiler tests have been used to evaluate the AVT chemistry guidelines adopted in 1974. The guidelines appear to be adequate to preserve tube integrity with one significant alteration: operation with contaminant ingress must be limited.

Additional extensive operating data are presently being accumulated with the conversion to AVT chemistry. A comprehensive program of steam generator inspections, including the recommendations of Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes, with the exceptions as stated in section 1.9, should provide for detection of any degradation that might occur in the steam generator tubing. Action levels for secondary side water chemistry during power operation are given in the EPRI PWR Secondary Water Chemistry Guidelines.

Increased margin against stress corrosion cracking has been obtained by the use of thermally treated Inconel-600 tubing. Thermal treatment of Inconel tubes has been shown to be particularly effective in resisting caustic corrosion. Tubing used in the Model F is thermally treated in accordance with a laboratory derived treatment process.

The tube support plates used in the Model F are ferritic stainless steel, which has been shown in laboratory tests to be resistant to corrosion in the AVT environment. If corrosion of ferritic stainless steel were to occur due to concentration of contaminants, the volume of the corrosion products is essentially equivalent to the volume of the parent material consumed. This would be expected to preclude denting. The support plates are also designed with quatrefoil tube holes rather than cylindrical holes. The quatrefoil tube-hole design promotes high-velocity flow along the tube and is expected to minimize the accumulation of impurities at the support plate location.

Additional measures are incorporated in the Model F design to prevent areas of dryout in the steam generator and accumulations of sludge in low-velocity areas. Modifications to the wrapper have increased water velocities across the tube sheet. A flow distribution baffle is provided which forces the low-flow area to the center of the bundle. Increased capacity blowdown pipes have been added to enable continuous blowdown of the steam generators at a high volume. The intakes of these blowdown pipes are located below the center cutout section of the flow distribution baffle in the low-velocity region where sludge may be expected to accumulate.

The impact of operating at a power level of 3653 MWt on steam generator water chemistry has been considered. The occurrence of stress corrosion cracking (SCC) and other forms of

degradation that might occur at current and enhanced rates will be found using the nondestructive examination (NDE) techniques specified in the degradation assessment that must be completed for subsequent plant outages.

5.4.2.4.4 Cleanup of Secondary-Side Materials

Several methods are employed to clean operating steam generators of corrosion-causing secondary-side deposits. Sludge lancing, a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits, can be performed when the need is indicated by the results of steam generator tube inspection. Six 6-in. access ports are provided for sludge lancing and inspection. Three of these are located above the tube sheet and three above the flow distribution baffle. Continuous blowdown is performed to monitor water chemistry. The location of the blowdown piping suction, adjacent to the tube sheet and in a region of relatively low-flow velocity, facilitates the removal of particulate impurities to reduce the accumulation on the tube sheet.

5.4.2.5 Steam Generator Inservice Inspection[a]

The steam generator is designed to permit inspection of Class 1 and 2 parts, including individual tubes. The design includes a number of openings to provide access to both the primary and secondary sides of the steam generator. The preservice inspection of the Unit 1 steam generators was completed to the requirements of the 1980 ASME Code through Winter 1980 Addenda. Inservice inspection will comply with the requirements of 10 CFR 50.55a. The openings include four manways, two for access to both chambers of the reactor coolant channel head inlet and outlet sides and two in the steam drum for inspection and maintenance of the moisture separators, and eight 6-in. handholes, three located just above the tube-sheet secondary surface, three located just above the flow distribution baffle, and two located on the tubelane diameter between the upper tube support plate and the row 1 tubes. Additional access to the tube U-bend is provided through each of the three deck plates. For proper functioning of the steam generator, some of the deck-plate openings are covered with welded, but removable, hatch plates. Inspection/access to the primary side is provided by two 16-in. manways located in the channel head.

Regulatory Guide 1.83 provides recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. The steam generators are designed to permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. Regulatory Guide 1.121, Basis for Plugging Degraded Pressurized Water Reactor (PWR) Steam Generator Tubes, provides recommendations concerning tube plugging. The minimum requirements for inservice inspection of steam generators, including tube plugging criteria, are established as part of the Technical Specifications.

5.4.2.6 Quality Assurance

The steam generator quality assurance program is given in table 5.4.2-2.

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[a] The Inservice Inspection Program, Steam Generator Tubing Integrity Program, and Steam Generator Program for Upper Internals are credited as license renewal aging management programs (see subsections 19.2.13, 19.2.26, and 19.2.27).

Radiographic inspection and acceptance standard are in accordance with the requirements of Section III of the ASME Code.

Liquid penetrant inspection is performed on weld-deposited tube-sheet cladding, channel-head cladding, divider-plate-to-tube-sheet and to channel-head weldments, tube-to-tube-sheet weldments, and weld-deposit cladding. Liquid penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Some of the tube-to-tube sheet welds of Unit 1 steam generator number 4 were deformed by impacts from a loose part. Based on the completion of a testing and analysis program, it is concluded that tube-to-tubesheet joint integrity remains consistent with the original plant design basis. The minimum requirements for the inservice inspection of steam generator tubes, including tube plugging criteria, are established as part of the Technical Specifications and are not changed as a result of the mechanical deformation of the tube-to-tubesheet welds.

Magnetic particle inspection is performed on the tube-sheet forging, channel-head casting, nozzle forgings, and the following weldments:

A. Nozzle to shell.

B. Support brackets.

C. Instrument connection (secondary).

D. Temporary attachments after removal.

E. All accessible pressure-retaining welds after hydrostatic test.

Magnetic particle inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code.

Ultrasonic tests are performed on the tube-sheet forgings, tube-sheet cladding, secondary-shell and heat plates, and nozzle forgings.

The heat transfer tubing is subjected to eddy current testing and ultrasonic examination.

Hydrostatic tests are performed in accordance with Section III of the ASME Code.

5.4.2.7 References

1. WCAP-16543-P, Rev. 0, “Regulatory Guide 1.121 Analysis and Structural Integrity Performance Criterion Application for the Vogtle Units 1 & 2 Model F Steam Generators,” August 2006.

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Bases

The reactor coolant system (RCS) piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Stresses are maintained within the limits of Section III of the American Society of Mechanical Engineers (ASME) Code. Code and material requirements are provided in section 5.2.

Materials of construction are specified to minimize corrosion/ erosion and ensure compatibility with the operating environment.

The piping in the RCS is Safety Class 1 and is designed and fabricated in accordance with ASME Code, Section III, Class 1 requirements.

Stainless steel pipe conforms to American National Standards Institute (ANSI) B36.19 for sizes 1/2 in. through 12 in. and wall thickness schedules 40S through 80S. Stainless steel pipe outside of the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thickness of the loop piping and fittings are no less than those calculated using the ASME Code, Section III, Class 1 formula of Paragraph NB-3641.1(3) with an allowable stress value of 17,550 psi. The pipe wall thickness for the pressurizer surge line is schedule 160. The minimum pipe bend radius is 5 nominal pipe diameters, and ovality does not exceed 6%.

All butt welds, branch connection nozzle welds for 3 in. nominal pipe sizes and greater, and boss welds are of a full penetration design. Socket-welded connections could be used for branch connection nozzle welds for 2 in. nominal pipe sizes and smaller and for some thermowell connections.

Processing and minimization of sensitization are discussed in subsection 5.2.3.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

Inservice inspection is discussed in subsection 5.2.4.

5.4.3.2 Design Description

The RCS piping includes those sections of piping interconnecting the reactor vessel, steam generator, and reactor cooling pump (RCP). It also includes the following:

A. Charging line and alternate charging line from the designated check valve up to the branch connections on the reactor coolant loop (RCL).

B. Letdown line and excess letdown line from the branch connections on the RCL to the system isolation valve.

C. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

D. Residual heat removal (RHR) lines to or from the RCLs up to the designated check valve or isolation valve.

E. Safety injection lines from the designated check valve to the RCLs.

F. Accumulator lines from the designated check valve to the RCLs.

G. Deleted.

H. Loop, fill, drain, sample, and instrument[a] lines from the designated isolation valve to the RCLs.

I. Pressurizer surge line from one RCL hot leg to the pressurizer vessel surge nozzle.

J. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection[a] with scoop, reactor coolant temperature element installation boss, the temperature element well itself, and the resistance temperature detector (RTD) fast response thermowells.

K. All branch connection nozzles attached to RCLs.

L. Pressure safety and relief lines from nozzles on top of the pressurizer vessel up to and through the pressurizer power-operated relief valves and pressurizer safety valves.

M. Auxiliary spray line from the isolation valve to the main pressurizer spray line.

N. Sample lines[a] from the pressurizer to the isolation valve.

O. Vent line from the reactor vessel head to the system isolation valves.

P. Reactor vessel level instrumentation lines from the isolation valves to the reactor vessel and to two RCL hot legs.

Q. RCP seal water injection lines to or from the RCP.

R. Boron injection lines from the check valve to the RCL.

Principal design data for the reactor coolant piping are given in table 5.4.3-1.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in section 5.2.

The reactor coolant piping and fittings that make up the loops are austenitic stainless steel. Pipe and fittings are cast, seamless without longitudinal or electroslag welds, and comply with the requirements of the ASME Code, Section II (parts A and C), Section III, and Section IX. All smaller piping that is part of the RCS, such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems, are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used.

All piping connections from auxiliary systems are above the horizontal centerline of the reactor coolant piping, with the exception of:

A. RHR pump suction lines, which are 45° down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the RHRS, should this be required for maintenance.

B. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

C. The differential pressure taps for flow measurement, which are downstream of the steam generators of the first 90° elbow.

D. The pressurizer surge line, which is attached at the horizontal centerline.

E. Two of the three thermowells in each resistance temperature detector hot leg connection.

Penetrations into the coolant flow path are limited to the following:

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[a] Lines with a 3/8-in. or less flow restricting orifice qualify as Safety Class 2. In the event of a break in one of these Safety Class 2 lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

A. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the RCL flow adds to the spray driving force.

B. The RCS sample taps protrude into the main stream to obtain a representative sample of the reactor coolant.

C. The hot and cold leg thermowells for the resistance temperature detectors extend into the reactor coolant to measure the RCS temperatures. For the hot legs, the RTDs are located inside of the RTD scoops.

D. The wide-range temperature detectors are located in resistance temperature detector wells that extend into both the hot and cold legs of the reactor coolant piping.

Separate RTDs mounted in thermowells in each RCL hot and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection systems. The RTDs are contained in thermowells which extend into the hot leg flow to become exposed to a representative temperature sample of the reactor coolant. Each hot leg is instrumented with three thermowells at locations 120 degrees apart around the reactor coolant piping with one located at the top of the pipe. The temperature measured by the three RTDs is then averaged using electronic weighting to provide the temperature input to the reactor control and protection systems.

An RTD mounted in a thermowell for the cold leg temperature measurement is located downstream of each reactor coolant pump discharge. This connection is located close to the same weld connection at the pump discharge and is in the same relative position in each loop.

Signals from the temperature detectors are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature of the cold leg, T_{cold}), and an average reactor coolant temperature, T_{avg}. The ΔT and T_{avg} for each loop is indicated on the main control board.

5.4.3.3 <u>Design Evaluation</u>

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown, and seismic loads is discussed in section 3.9.N.

5.4.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications. (See subsection 5.2.3.)

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in table 5.2.3-3. Maintenance of the water quality to minimize corrosion is accomplished using the CVCS and sampling system which are described in chapter 9.

The design and installation are in compliance with the ASME Code, Section III. Pursuant to this, all pressure-containing welds out to the second valve that delineates the RCS boundary are accessible for inservice examination as required of ASME Code, Section XI, and are fitted with removable insulation.

5.4.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in subsection 5.2.3.

5.4.3.3.3 Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury, and lead is prohibited. Colloidal graphite is the only permissible thread lubricant.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg chloride/dm^2 and 0.0015 mg fluoride/dm^2.

5.4.3.3.4 Detection Of Thermal Stratification[a]

To assure that unisolable sections of piping connected to the reactor coolant system (RCS) and pressurizer surge line will not be subjected to combined cyclic and thermal stresses that could cause fatigue failure, a program has been implemented that will detect adverse temperature distributions as described below.

5.4.3.3.4.1 Unisolable Section of Piping Connected To The RCS. Unisolable sections of piping for the safety injection, normal and alternate charging, and auxiliary spray lines interconnected with the RCS are instrumented to detect adverse thermal stratification and cycling due to potential isolation valve leakage into the RCS boundary. Fluid leakage is detected by temperature measurements utilizing resistance temperature detectors (RTDs) strapped on the pipe. Temperature data is periodically recorded and evaluated for thermal stratification and cycling to determine its impact on piping structural integrity. Additionally (on Unit 2 only), two 12-in. residual heat removal (RHR) suction lines attached to the reactor coolant loop (RCL) hot leg are instrumented with RTDs.

5.4.3.3.4.2 Pressurizer Surge Line. The Unit 2 pressurizer surge line was instrumented to detect temperature distribution and thermal movement. The instruments consisted of RTDs and displacement transducers. The temperature distribution and thermal movements were monitored and recorded through the first refueling cycle. Analysis utilizing the leak-before-break methodology and the monitoring data taken from the Unit 2 surge line, combined with Unit 1 surge line support design changes and operating procedure changes, are an acceptable program to address the effects of thermal stratification on the pressurizer surge line. Therefore, no further monitoring is required.

5.4.3.4 Tests and Inspections

The RCS piping quality assurance program is given in table 5.4.3-2.

Volumetric examination is performed throughout 100% of the wall volume of each pipe and fitting in accordance with the applicable requirements of Section III of the ASME Code for all

[a] The Fatigue Monitoring Program is credited as a license renewal aging management program (see subsection 19.3.2).

pipe 27 1/2 in. and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on all accessible surfaces of each finished fitting, in accordance with the criteria of the ASME Code, Section III. Acceptance standards are in accordance with the applicable requirements of the ASME Code, Section III.

The pressurizer surge line conforms to SA-376, Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests) and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100% of the piping wall volume.

The end of pipe sections, branch ends, and fittings are machined back to provide a smooth weld transition adjacent to the weld path.

Tests and inspections performed under the following license renewal aging management programs are credited as applicable to various portions of RCS piping:

- Boric Acid Corrosion Control Program (see subsection 19.2.3).
- CASS RCS Fitting Evaluation Program (see subsection 19.2.5).
- External Surfaces Monitoring Program (see subsection 19.2.8).
- Inservice Inspection Program (see subsection 19.2.13).
- Oil Analysis Program (see subsection 19.2.16).
- One-Time Inspection Program (see subsection 19.2.17).
- One-Time Inspection Program for ASME Class 1 Small Bore Piping (see subsection 19.2.18).
- Water Chemistry Control Program (see subsection 19.2.28).
- Fatigue Monitoring Program (see subsection 19.3.2).

5.4.4 MAIN STEAM LINE FLOW RESTRICTIONS

5.4.4.1 Design Bases

The outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steamflow in the unlikely event of a break in the main steam line. A large increase in steamflow will create a backpressure which limits further increase in flow. The flow restrictor performs the following functions:

- Rapid rise in containment pressure is limited.
- The rate of heat removal from the reactor is such as to keep the cooldown rate within acceptable limits.
- Thrust forces on the main steam line piping are reduced.

- Stresses on internal steam generator components, particularly the tube sheet and tubes, are limited.

The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven Inconel ASME SB-163 venturi inserts which are installed in holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle and the other six equally spaced around it. After insertion into the nozzle forging holes, the Inconel venturi inserts are welded to the Inconel cladding on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restrictor design has been analyzed to determine its structural adequacy. The equivalent throat diameter of the steam generator outlet is 16 in. and the resultant pressure drop through the restrictor at 100% steamflow is approximately 2.78 psig. This is based on a design flowrate of 3.78×10^6 lb/h. Materials of construction and manufacturing of the flow restrictor are in accordance with Code Class 1 Section III of the ASME Code. The method for seismic analysis is dynamic.

5.4.4.4 Inspections

Since the restrictor is not part of the steam system boundary, no inspections beyond those performed during fabrication are anticipated.

5.4.5 MATERIALS AND INSPECTIONS

This subsection is not applicable to VEGP.

5.4.6 REACTOR VESSEL DESIGN DATA

This subsection is not applicable to VEGP.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

The residual heat removal system (RHRS) transfers heat from the reactor coolant system (RCS) to the nuclear service cooling water system via the component cooling water (CCW) system to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown and maintains this temperature until the plant is started up again.

Refer to subsection 9.2.2 for a description of the CCW system. Parts of the RHRS also serve as parts of the emergency core cooling system (ECCS) for accident mitigation (section 6.3).

In addition, the RHRS is used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

Nuclear plants employing the same RHRS design as the VEGP are given in section 1.3.

5.4.7.1 Design Bases

The RHRS design parameters are listed in table 5.4.7-1.

The following establishes a design bases supporting a 50°F/h cooling rate. The RHR system and support systems have been reviewed and evaluated to document the acceptability of an operational cooling rate of up to 100°F/h not to exceed 100°F in any 1 h period.

The RHRS is placed in operation approximately 2 to 4 hours after reactor shutdown, when the temperature and pressure of the RCS are approximately 350°F and 365 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with 5000 gal/min CCW initially at 105°F, the RHRS is designed to reduce the temperature of the reactor coolant to 140°F within 20 h following reactor shutdown. Under these conditions, the time required to reduce the reactor coolant temperature from 350°F to 200°F is approximately 3 h. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 h following reactor shutdown from an extended run at full power.

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with CCW at 5000 gal/min and initially at 105°F, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within approximately 30 h. The time required under these conditions to reduce reactor coolant temperature from 350°F to 212°F is approximately 20 h. The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor-operated valves in series on each suction line. Each motor-operated valve is interlocked to prevent its opening if RCS pressure is greater than approximately 365 psig. The valves have a control room alarm which alerts the operators if one or both of the valves is not fully closed and the RCS pressure exceeds 420 psig. The RHRS is isolated from the RCS on the discharge side by two check valves in each return line. Also provided on the discharge side is a normally open motor-operated valve downstream of each RHRS heat exchanger. (These check valves and motor-operated valves are not considered part of the RHRS; they are shown as part of the ECCS. See drawing 1X4DB121.)

Each inlet line to the RHRS is equipped with a pressure relief valve designed to prevent RHRS overpressurization assuming the most severe overpressure transients. These relief valves protect the system from inadvertent overpressurization during plant startup, shutdown, and cold shutdown decay heat-removal operations.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible backleakage through the valves isolating the RHRS from the RCS. These valves are considered part of the ECCS, as depicted in drawing 1X4DB120. Relief capacity of these valves is given in table 6.3.2-2.

The RHRS is designed for a single nuclear power unit and is not shared between units.

The RHRS is designed to be fully operable from the control room for normal operation except as described in paragraph 5.4.7.2.7. By nature of its redundant design, the RHRS is designed to accept all major component single failures, with the only effect being an extension in the

required cooldown time. There are no motor-operated valves in the RHRS that are subject to flooding following a secondary side break or a LOCA. Although considered to be of low probability, spurious operation of a single motor-operated valve can be accepted without loss of function as a result of the redundant two-train design.

Missile protection, protection against dynamic effects associated with the postulated rupture of piping, and seismic design are discussed in section 3.5 and subsections 3.6.2 and 3.7.N.2, respectively.

5.4.7.2 System Design

5.4.7.2.1 Schematic Piping and Instrumentation Diagrams

The RHRS, as shown in drawings 1X4DB122 and 2X4DB122 and figure 5.4.7-1, consists of two residual heat exchangers, two RHR pumps, and the associated piping, valves, and instrumentation necessary for operational control. Notes to figure 5.4.7-1 identify modes of operation, valve alignments, and process conditions at various points in the system. The inlet lines to the RHRS are connected to the hot legs of two reactor coolant loops (RCLs), while the return lines are connected to the cold legs of each of the RCLs. These return lines are also the ECCS low-head injection lines (drawings 1X4DB119, 2X4DB119, 1X4DB120, and 1X4DB121). The RHRS suction lines are isolated from the RCS by two motor-operated valves in series located inside the containment. Each discharge line is isolated from the RCS by two check valves located inside the containment and by a normally open motor-operated valve, with power lockout capability, located outside the containment. (The check valves and the motor-operated valve on each discharge line are not part of the RHRS; these valves are shown as part of the ECCS (drawings 1X4DB119, 2X4DB119, 1X4DB120, and 1X4DB121).

During RHRS operation, reactor coolant flows from the RCS to the RHR pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the CCW circulating through the shell side of the residual heat exchangers.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the chemical and volume control system (CVCS) low-pressure isolation letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirements of the reactor vessel and by the No. 1 seal differential pressure and net positive suction head requirements of the reactor coolant pumps.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. The flow control valve in the bypass line around each residual heat exchanger automatically maintains a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature, and total flow.

The RHRS is also used for filling the refueling cavity before refueling. After refueling operations, the RHRS is used to pump the water back to the refueling water storage tank until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling canal.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the process sampling system to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each RHR train between the pump and heat exchanger.

The RHRS also functions, in conjunction with the high-head portion of the ECCS, to provide injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss-of-coolant accident (LOCA).

In its capacity as the low-head portion of the ECCS, the RHRS provides long-term recirculation capability for core cooling following the injection phase of the LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps and safety injection pumps.

The use of the RHRS as part of the ECCS is more completely described in section 6.3.

The RHR pumps, in order to perform their ECCS function, are interlocked to start automatically on receipt of a safety injection signal (section 6.3).

The RHRS suction isolation valves in each inlet line from the RCS are separately interlocked to prevent both of them from being opened when RCS pressure is greater than approximately 365 psig, and the valves have a control room alarm, which alerts the operators if one or both of the valves is not fully closed and the RCS pressure exceeds 420 psig. These interlocks are described in more detail in paragraph 5.4.7.2.4 and subsection 7.6.2.

The RHRS suction isolation valves are also interlocked to prevent their being opened unless the isolation valves in the following lines are closed:

A. Recirculation lines from the residual heat exchanger outlets to the suctions of the safety injection pumps and centrifugal charging pumps.

B. RHR pump suction line from the refueling water storage tank.

C. RHR pump suction line from the containment sump.

The motor-operated valves in the RHRS miniflow bypass lines are interlocked to open when the RHRS pump discharge flow is less than the open setpoint (824 gpm at 350 °F, 780 gpm at 100 °F) and to close when the flow exceeds the closed setpoint (1944 gpm at 350 °F, 1841 gpm at 100 °F).

5.4.7.2.2 Equipment and Component Descriptions

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material. Component parameters are given in table 5.4.7-2.

5.4.7.2.2.1 Residual Heat Removal Pumps. Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the RHR heat exchangers to meet the plant cooldown requirements. The use of two separate RHR trains ensures that cooling capacity is only partially lost should one pump become inoperative.

The RHR pumps are protected from overheating and loss of suction flow by miniflow bypass lines that ensure flow to the pump suction. A valve located in each miniflow line is regulated by a signal from the flow transmitters located in each pump discharge header. The control valves open when the residual pump discharge flow is less than the open setpoint (824 gpm at 350 °F, 780 gpm at 100 °F) and close when the flow exceeds the closed setpoint (1944 gpm at 350 °F, 1841 gpm at 100 °F).

Although both valves could be closed by operator error, the design of the miniflow system would preclude any pump damage. The miniflow bypass valve is a fast operating motor-operated gate valve which will open or close in 10 s or less. The residual heat removal pump can operate safely without damage during this period with no flow.

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high-pressure alarm is also actuated by the pressure sensor.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The RHR pumps also function as the low-head safety injection pumps in the ECCS (section 6.3).

A pump performance curve is provided in figure 5.4.7-2.

5.4.7.2.2.2 Residual Heat Exchangers. Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and CCW existing 20 h after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate and independent RHR trains ensures that the heat-removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while CCW circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (section 6.3).

5.4.7.2.2.3 RHRS Valves. Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to a local equipment drain.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions. The RHR discharge cross-connect valves are provided with bonnet vents to the RHR pump side of the valves. RHR loop suction valves 1/2HV8701A, 1/2HV8702A, 1/2HV8702B, and 1HV8701B are provided with bonnet vents to the upstream side of the valves. The RHR loop suction valve 2HV8701B bonnet vent to the upstream side of the valve has been removed.

5.4.7.2.3 System Operation

5.4.7.2.3.1 Plant Startup. Generally, while in the cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At the beginning of plant startup, at least one RHR pump is operating, and a portion of the discharge flow may be directed to the CVCS.

This arrangement augments RCS pressure control during startup. When the reactor coolant pumps are started, the RHR pump is stopped. The thermal input of the reactor coolant pumps heats the reactor coolant inventory. Once the pressurizer steam bubble formation is complete, the RHRS is isolated from the RCS and aligned for operation as part of the ECCS.

5.4.7.2.3.2 Power Generation and Hot Standby Operation. During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

5.4.7.2.3.3 Plant Shutdown. Plant shutdown is defined as the operation that brings the plant from no-load temperature and pressure to cold conditions.

5.4.7.2.3.4 Normal Cold Shutdown. The initial phase of plant shutdown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 365 psig, approximately 2 to 4 hours after reactor shutdown, the second phase of cooldown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period, during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the residual heat exchangers. By adjusting these control valves downstream of the residual heat exchangers, the mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment of flow through the heat exchangers, each heat exchanger bypass valve is automatically regulated to give the required total flow.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits and the operating temperature limits of the CCW system. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube side outlet line.

Should both RHR heat exchanger outlet and bypass flow control valves fail simultaneously (i.e., loss of instrument air), then the maximum cooldown rate may be increased. The maximum cooldown rate depends on many factors, including the time of failure, the RHR flowrate, the CCW flowrates and temperatures, and other heat loads on the CCW system. One of the key factors is the RCS water temperature, since the cooldown rate depends upon the temperature difference between the RHR (RCS) flow and the CCW flow in the RHR heat exchanger. Even with the maximum flow through the RHR heat exchangers, it is typically impossible to maintain a cooldown rate as high as the technical specification design rate of 100°F/h when the RCS temperature is less than 250°F. The operator can significantly limit the cooldown rate by merely stopping one of the RHR pumps.

During plant shutdown, pressurizer steam bubble operation is maximized to control RCS pressure. When the RHRS is in operation, RCS control is augmented by regulating the charging flowrate and the rate of letdown from the RHRS to the CVCS.

After the reactor coolant pressure is reduced and the temperature is 140°F or lower, the RCS may be opened for refueling or maintenance.

A failure modes and effects analysis for normal cooldown operations is provided in table 5.4.7-3.

5.4.7.2.3.5 Safety-Grade Cold Shutdown. It is expected that the systems normally used for cold shutdown will be available anytime the operator chooses to perform a reactor cooldown. However, to ensure that the plant can be taken to cold shutdown at anytime, the safety-grade cold shutdown design enables the RCS to be taken from no-load temperature and pressure to cold conditions using only safety-grade systems, with only onsite or offsite power available, and assuming the most limiting single failure.

Should portions of normal shutdown systems be unavailable, the operator will maintain the plant in a hot standby condition while making those normal systems functional. Local manual actions are performed as described in table 5.4.7-4. Appropriate procedures are provided for the use of safety-grade backups contingent upon the inability to make normal systems available. The operator should use any of the normal systems that are available in combination with the safety-grade backups for the systems that cannot be made operable. The safety-grade provisions are to be used only upon the inability to make available the equipment normally used for the given function.

The safety-grade cold shutdown design enables the operator to maintain the plant in hot standby for approximately 4 h. Since it is assumed that the reactor coolant pumps are not available, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and the steam generators as the heat sink. Heat removal is accomplished via the steam generator power-operated relief valves and auxiliary feedwater system.

The boration of the RCS is initiated prior to cooling the RCS. The charging pumps are used to provide borated water to the RCS at a rate of approximately 0 to 50 gal/min. The borated water is delivered to the RCS via the safety-grade charging line or the high-head safety injection lines. Both flow paths have provisions for flow control. Reactor coolant pump seal injection is also maintained. To accommodate this addition to RCS inventory, continuous letdown is discharged from the reactor vessel head letdown line to the pressurizer relief tank.

During boration to cold shutdown concentration, the safety-grade cooldown is accomplished by increasing the steam dump from the steam generator power-operated relief valves to attain a rate of primary side cooling of approximately 35°F/h. In conjunction with this portion of the cooldown, the charging pumps are used to deliver borated water to makeup for primary contraction due to cooling. Makeup is also provided for the RCS inventory discharged when the reactor vessel head letdown path is periodically cycled to provide head cooling. Upon approaching the end of this phase of cooldown, the RCS is depressurized by venting or letdown through the reactor vessel head vent valves to the pressurizer relief tank.

To ensure that the accumulators do not repressurize the RCS, the accumulator discharge valves are closed prior to the RCS pressure dropping below the accumulator discharge pressure. Additionally, each accumulator is provided with two Class 1E solenoid-operated valves in parallel to ensure that the accumulator may be vented should it fail to be isolated from the RCS.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 365 psig, respectively, and the RCS borated to cold shutdown concentration, the second phase of cooldown starts with the RHRS being placed in operation.

As safety-grade cooldown continues, the reactor vessel head letdown line is periodically opened to increase head cooling and to accommodate any additional input to the RCS, such as reactor

coolant pump seal injection. Since loss of nonsafety-grade equipment results in a loss of the air supply to the flow control valves that are normally used to limit the initial RHRS cooldown rate, the operator may choose to use only one of the RHR subsystems. Should a single failure, such as that of an RHRS component or of an emergency power train (when only onsite power is available), limit operation to one of the RHR subsystems, the operator would open the series isolation valves in the suction of only the operable RHR subsystem. In this case, the operator would also close the cross-connect isolation valves between the subsystems. RHR would continue under these conditions until the redundant subsystem could be made available.

A failure modes and effects analysis for safety-grade cold shutdown operations is provided in table 5.4.7-4.

5.4.7.2.3.6 Refueling. The RHRS is utilized during refueling to transfer borated water from the refueling water storage tank to the refueling cavity. During this operation, one RHR train is selected for fill, the isolation valves in the suction lines from the RCS are closed, the isolation valves from the refueling water storage tank are opened and the RHR pump may be started or the refueling water storage tank (RWST) water is allowed to gravity drain.

The refueling cavity is prepared for flooding and the vessel head is removed to its storage pedestal using the containment polar crane. The refueling water is then transferred into the reactor vessel through the RHRS hot-leg or cold-leg return lines and into the refueling cavity through the open reactor vessel. After the water level reaches the normal refueling level, the RHR pump is stopped, the refueling water storage tank supply valves are closed, and the suction isolation valves from the RCS are opened.

During refueling, the RHRS is maintained in service with the number of pumps and heat exchangers in operation as required by the heat load. Also during refueling, an RHR pump not being applied for shutdown cooling can be used to satisfy the Technical Requirements Manual requirement that a pump be available for boration when the refueling cavity water level is ≥ 23 feet above the flange. The flowpath associated with this application of the RHR pump is the pump taking suction from the RWST and injecting into the RCS cold legs.

Following refueling, the RHRS is used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the refueling water storage tank. The vessel head is then replaced and the normal RHRS flow path reestablished. The remainder of the water is removed from the refueling canal via a drain connection in the bottom of the canal.

An alternative methodology may be used for raising the vessel head. In this method, the reactor vessel head is lifted slightly. Then the cavity is filled as discussed above, while reactor vessel head is gradually raised as the water level in the refueling cavity increases. The filling of the cavity is also terminated as discussed above. Following refueling, placement of the head on the vessel is accomplished by maintaining the vessel head just above the water in the cavity, as the level lowers while the refueling water is being pumped from the cavity to the RWST by the RHR pumps.

5.4.7.2.3.7 Mid-loop and Drain Down Operations. The RHR system is used to provide core cooling when the RCS must be partially drained to allow maintenance or inspection of the reactor head, steam generators, or reactor coolant pump seals.

The level in the primary system is lowered to near the mid-line of the hot and cold legs. At this water level the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs of loops 1 and 4, and care must be taken to avoid air entrainment into the RHR

pump suction. Air ingestion by an RHR pump can cause loss of pump function, creating the potential for loss of residual heat removal.

Instrumentation has been provided to assist the operator in safely maintaining adequate level in the RCS hot legs during mid-loop and drain down operations. This instrumentation is shown in drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113. Instrumentation has also been provided to assist the operator in quickly identifying air ingestion in the RHR pumps.

Two differential pressure transmitters are connected to the RCS to provide independent level indications in the main control room. One transmitter is connected to the RCS loop 1 hot leg and provides narrow range indication. This narrow range indicator spans from the bottom of the hot leg upward 8 feet. This instrument loop also provides annunciation of the low hot leg level. The other transmitter is connected to the RCS loop 4 hot leg and provides wide range indication. This wide range indicator spans from the bottom of the hot leg upward 20 feet. The instrument loops are powered from separate breakers to maximize the availability of the indication.

Local RCS level indication is available via a sight glass located in the containment building. The sight glass spans from the top of the pressurizer heaters down to just below the bottom of the RCS hot legs. The piping for this sight glass is connected to the RCS as required during modes 5 and 6. The tubing used for the sight glass is vacuum resistant.

Current transducers monitor the 4,160 V power feeders to each RHR pump. The output of these transducers is routed to the plant computer. Historical traces of the pump motor current can be obtained at any plant computer terminal. The logic associated with modes 5 and 6 core cooling critical safety function status trees provides a visual alarm at the plant computer safety parameter display system (SPDS) terminal in the main control room, if the motor current becomes unstable. The critical safety function status trees alarm the control room annunciator when any adverse condition occurs.

5.4.7.2.4 Control

Each inlet line to the RHRS is equipped with a relief valve to prevent RHRS overpressurization during plant startup, shutdown, and cold shutdown decay heat-removal operation. Each valve has a relief capacity of 900 gal/min at a set pressure of 450 psig. An analysis has been conducted to confirm the capability of the RHRS relief valve to prevent overpressurization in the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences. The analysis confirmed that one relief valve has the capability to maintain the RHRS maximum pressure within code limits. The above capacities of the RHRS suction line relief valves are adequate to provide relief protection necessary for the RHRS and the RCS as part of the cold overpressure mitigating system. For a discussion of the cold overpressure mitigating system and the overpressure events examined, refer to WCAP-10529.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gal/min at a set pressure of 600 psig. These relief valves are located in the ECCS (drawing 1X4DB121).

The fluid discharged by the suction-side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the recycle holdup tank.

The design of the RHRS includes two motor-operated gate isolation valves in series on each inlet line between the high-pressure RCS and the lower pressure RHRS. They are closed during normal operation and are opened only for RHR during a plant shutdown after the RCS

pressure is reduced to approximately 365 psig or lower and RCS temperature is reduced to approximately 350°F. During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 425 psig. These isolation valves are provided with independent and diverse "prevent-open" interlocks and main annunciator alarms which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure. The two inlet isolation valves in each RHR subsystem are independently interlocked with diverse pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately 365 psig. Additionally, during plant startup, the valves are interlocked such that they alarm in the main control room when RCS pressure increases to 420 psig to alert the operators that one or both of the valves is not fully closed.

The use of two independently powered motor-operated valves in each of the two inlet lines, with an independently powered pressure transmitter for each valve, along with two independent and diverse interlock signals for each function, ensures a design which meets applicable single-failure criteria. These protective interlock designs, in combination with plant operating procedures, provide the means of accomplishing the protective function. For further information on the instrumentation and control features, see subsection 7.6.2.

The RHR inlet isolation valves are provided with control switches with integral red-green position indicator lights on the main control board and an alarm window on a main control board annunciator panel.

Isolation of the low-pressure RHRS from the high-pressure RCS is provided on the discharge side by two check valves in series. These check valves are located in the ECCS, and their testing is described in paragraph 6.3.4.2.

5.4.7.2.5 Applicable Codes and Classifications

The entire RHRS is designed as Nuclear Safety Class 2 with the exception of the suction isolation valves, which are Safety Class 1. Component codes and classifications are given in section 3.2.

5.4.7.2.6 System Reliability Considerations

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety-grade systems are provided in the plant design, both nuclear steam supply system scope and balance of plant scope, to perform this function. The safety-grade systems that perform this function for all plant conditions except a LOCA are the RCS and steam generators, which operate in conjunction with the auxiliary feedwater system and the steam generator safety and power-operated relief valves, and the RHRS, which operates in conjunction with the CCW system and the nuclear service cooling water (NSCW) system. For LOCA conditions, the safety-grade system which performs the function of removing residual heat from the reactor core is the ECCS, which operates in conjunction with the CCW system and the NSCW system.

The auxiliary feedwater system, along with the steam generator safety and power-operated relief valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHRS when RCS temperature is less than 350°F. The auxiliary feedwater system is capable of performing this function for an extended period of time following plant shutdown.

The RHRS is provided with two RHR pumps and two residual heat exchangers arranged in two separate, independent flow paths. To ensure reliability, each RHR pump is connected to a different vital bus. Each RHR train is isolated from the RCS on the suction side by two motor-operated valves in series, with each valve receiving power via a separate motor control center and from a different vital bus. The suction isolation valves prevent exposure of the RHRS to the normal operating pressure of the RCS (paragraph 5.4.7.2.4).

The RHRS operation for normal conditions, even with a major failure, is accomplished completely from the control room except as described in paragraph 5.4.7.2.7. The redundancy in the RHRS design provides the system with the capability to maintain its cooling function, even with a major single failure such as failure of a pump, valve, or heat exchanger, since the redundant train can be used for continued heat removal.

Status-indicating lights are provided at the control board for the RHR pump, the RHRS suction isolation valves, and the miniflow isolation valves.

The major portion of the RHRS is located in the auxiliary building. Leakages resulting from a passive failure of the RHRS piping are collected by the floor drain system. See subsection 9.3.3 for a discussion of this system and the alarms and instrumentation provided to detect any radioactive leaks, should they occur.

5.4.7.2.7 Manual Actions

The RHRS is designed to be fully operable from the control room for normal operation except for restoring power to the suction isolation valves, and for restoring air to the heat exchanger flow control valves prior to RHR initiation. Manual operations required of the control room operator are restoring power to and opening the suction isolation valves, restoring air to and positioning the flow control valves downstream of the residual heat exchangers, and starting the RHR pumps. The power lockout of the suction isolation valves is discussed in subsection 7.6.2. The air is isolated from the RHR heat exchanger flow control valves by closing the air supply isolation valve. During normal cooldown, there is adequate time and accessibility to perform these actions. Refer to paragraph 6.3.2.8 for the manual actions required during ECCS operations.

5.4.7.3 <u>Performance Evaluation</u>

The performance of the RHRS in reducing reactor coolant temperature is evaluated through the use of heat balance calculations on the RCS, RHRS, and CCW system at stepped intervals following the initiation of removal operation. Heat removal through the RHR and component cooling heat exchangers is calculated at each interval by use of standard water-to-water heat exchanger performance correlations; the resultant fluid temperatures for the RCS, RHRS, and CCW system are calculated and used as input to the next interval's heat balance calculation.

Assumptions utilized in the series of heat balance calculations describing plant RHRS cooldown are as follows:

A. RHR operation is initiated 4 h after reactor shutdown.[a]

B. RHR operation begins at a reactor coolant temperature of 350°F.

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[a] This value is used in design bases criteria and analyses to support the plant maximum cold shutdown condition. The RHR system and support systems have been reviewed and evaluated to document the acceptability of an operational cooling rate up to 100°F/h.

C. Thermal equilibrium is maintained throughout the RCS during the cooldown (i.e., one reactor coolant pump in operation whenever the reactor coolant temperature is above 160°F).

D. CCW temperature during cooldown is limited to a maximum of 120°F.

E. RCS cooldown rates of 50°F/h are not exceeded.[b]

Cooldown curves calculated using this method are provided in figure 5.4.7-3.

5.4.7.4 Preoperational Testing

Preoperational testing of the RHRS is addressed in chapter 14.

5.4.7.5 Reliability Tests and Inspections

As the RHRS functions as part of the ECCS, periodic tests and inspections are conducted in conjunction with those conducted on ECCS components. (See paragraph 6.3.4.2 for a discussion of the tests and inspections. See the Technical Specifications for the selection of test frequency, acceptability of testing, and measured parameters. A description of the inservice inspection program is included in section 6.6.)

5.4.8 REACTOR WATER CLEANUP SYSTEM

This subsection is not applicable to VEGP.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam and feedwater systems of VEGP are not part of the reactor coolant system. The main steam system, including the main steam piping, is described in section 10.3. The feedwater system, including piping, is described in subsection 10.4.7.

5.4.10 PRESSURIZER

5.4.10.1 Design Bases

The pressurizer provides a point in the reactor coolant system (RCS) where liquid and vapor are maintained in equilibrium under saturated conditions for control of pressure of the RCS during steady-state operations and transients.

The volume of the pressurizer is equal to, or greater than, the minimum volume of steam, water, or total of the two which satisfies all of the following requirements:

A. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

B. The water volume is sufficient to prevent the heaters from being uncovered during a step-load increase of 10% at full power.

C. The steam volume is large enough to accommodate the surge resulting from a 50% reduction of full load with automatic reactor control and a 40% steam dump without the water level reaching the high level reactor trip point.

D. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip, without reactor control or steam dump.

E. The pressurizer will not empty following reactor trip and turbine trip.

F. A low pressurizer pressure safety injection signal will not be activated because of a reactor trip and turbine trip.

The surge line is sized to maintain the pressure drop between the RCS and the safety valves within allowable limits during a design discharge flow from the safety valves.

The surge line is designed to withstand the thermal stresses resulting from volume surges occurring during operation. In accordance with TMI Action Item II.K.3.9, the derivative action setting of the proportional integral derivative controller has been set to zero to prevent spurious power-operated relief valves (PORV) operation.

5.4.10.2 Design Description

5.4.10.2.1 Pressurizer and Surge Line

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all internal surfaces exposed to the reactor coolant. The surge line is constructed of stainless steel.

The general configuration of the pressurizer is shown in figure 5.4.10-1. The design data for the pressurizer are given in table 5.4.10-1. A comparison of the design basis operating conditions determined that the conditions originally qualified for the pressurizer will envelop the operation at a power level of 3653 MWt. Codes and material requirements are provided in section 5.2.

The pressurizer surge line connects the pressurizer to one reactor coolant hot leg, thus enabling continuous coolant volume and pressure adjustments between the RCS and the pressurizer.

The surge line nozzle and electric heaters are located in the bottom head of the pressurizer. The heaters are designed to be removable for maintenance or replacement.

The pressurizer surge line nozzle diameter is given in table 5.4.10-1, and the pressurizer surge line size is shown in drawing 1X4DB112.

A retaining screen is located above the nozzle to prevent passage of any foreign matter from the pressurizer to the RCS. Baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and also assist in mixing.

The spray line nozzles and the relief and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually from the control room.

A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the pressurizer liquid and the reactor coolant and to prevent excessive cooling of the spray piping.

During an outsurge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the low-pressure reactor trip setpoint. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the pressurizer to prevent the pressurizer pressure from reaching the setpoint of the PORVs. The heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop. The skirt-type support is attached to the lower head and extends for a full 360° around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes for securing the vessel to its foundation. The support is provided with ventilation holes to ensure free convection of ambient cooling air past the heater and connector ends.

Material specifications are provided in table 5.2.3-1 for the pressurizer and the surge line. Design transients for the components of the RCS are discussed in subsection 3.9.N.1.

5.4.10.2.2 Pressurizer Spray and Relief Line Instrumentation

Refer to chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

Temperatures in the spray lines from the cold legs of two loops are measured and indicated. Alarms from these signals are actuated to warn the operator of low spray water temperature or to indicate insufficient flow in the spray lines.

Temperatures in the pressurizer safety and relief valves discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve. High temperature alarms are initiated if the leakage is abnormal.

5.4.10.3 <u>Design Evaluation</u>

5.4.10.3.1 System Pressure Control

The RCS pressure is controlled by the pressurizer whenever a steam volume is present in the pressurizer.

A design basis safety limit has been set such that the RCS pressure does not exceed the maximum transient value allowed under the American Society of Mechanical Engineers (ASME) Code, Section III. Evaluation of plant conditions of operation considered for design indicates that this safety limit is not reached.

During startup and shutdown, the rate of temperature change in the RCS is controlled by the operator. Heatup rate is controlled by energy input from the reactor coolant pumps and by the pressurizer electrical heating capacity.

The pressurizer heaters are powered from four electrical panels: one panel for each heater group. Two groups of heaters can be administratively loaded onto the non-Class 1E emergency buses (drawings 1X3D-AA-A01A and 2X3D-AA-A01A). The Class 1E 4160-V breakers supplying the non-Class 1E buses are automatically opened upon a safety injection signal. The non-Class 1E buses can be manually reenergized under administrative procedure. Actions associated with these two groups of heaters can be controlled from either the control room or the shutdown panels. Engineered safety features loads need not be shed to manually load a pressurizer heater group.

Pressurizer heater controls are described in paragraph 7.7.1.D.

Analysis performed for a four-loop plant with an 1800-ft^3 pressurizer similar to VEGP indicates that a heater capacity of 150 kW is adequate to maintain subcooled conditions in the RCS during natural circulation. Each VEGP heater panel has a heater capacity of 483 kW. Furthermore, the analysis demonstrates that the RCS sensible heat capacity is such that adequate subcooling can be maintained in the RCS for 4 h without heat input from the pressurizer heaters. Thus the recommendations of Action Item II.E.3.1 of NUREG-0737 are satisfied.

When the pressurizer is filled with water, i.e., during initial system heatup or near the end of the second phase of plant cooldown, RCS pressure is controlled by the letdown flowrate via the residual heat removal system.

5.4.10.3.2 Pressurizer Level Control

The normal operating water volume at full-load conditions is approximately 60% of the free internal vessel volume. Under part-load conditions the water volume in the pressurizer is reduced proportionally with reductions in plant load to approximately 25% of the free internal vessel volume at the zero-power condition.

5.4.10.3.3 Pressure Setpoints

The RCS design and operating pressure, together with the safety valve setpoints, PORV setpoints, pressurizer spray valve setpoints, and the protection system pressure setpoints, are listed in table 5.4.10-2. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The low pressurizer pressure reactor trip does not require a coincident low water level signal. This is in accordance with the recommendations of Action Item II.K.1.17 of NUREG-0660.

Temperature changes which may affect the relief valve setpoints have been considered. Normal ambient air temperature variations have no significant effects. However, cold valves relieving hot fluid may show reduced setpoints; therefore, this has been considered in the design of such valves.

5.4.10.3.4 Pressurizer Spray

Two separate automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping routed to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The design spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the PORVs during a step reduction in power level of 10% of full load.

The pressurizer spray lines and valves are large enough to provide the required spray flowrate under the driving force of the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop in order to utilize the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray valves and spray line connections are arranged so that the spray operates, although at a reduced capacity when one reactor coolant pump is not operating. The spray line also assists in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flowpath from the chemical and volume control system to the pressurizer spray line is also provided. This path provides auxiliary spray to the vapor space of the pressurizer during cooldown when the reactor coolant pumps are not operating. The pressurizer spray connection and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.4.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with the ASME Code, Section III.

To implement the requirements of the ASME Code, Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The weld surface is ground smooth for ultrasonic inspection.

- Support skirt to the pressurizer lower head.
- Surge nozzle to the lower head.
- Safety, relief, and spray nozzles to the upper head.
- Nozzle to safe end attachment welds.
- All girth and longitudinal full-penetration welds.
- Manway attachment welds.

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in table 5.4.10-3.

Tests and inspections performed under the following license renewal aging management programs are credited as applicable to the pressurizer and its subcomponents:

- Bolting Integrity Program (see subsection 19.2.2).
- Boric Acid Corrosion Control Program (see subsection 19.2.3).
- Inservice Inspection Program (see subsection 19.2.13).
- Nickel Alloy Management Program for Nonreactor Vessel Closure Head Penetration Locations (see subsection 19.2.14).

- Water Chemistry Control Program (see subsection 19.2.28).
- Fatigue Monitoring Program (see subsection 19.3.2).

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

5.4.11.1 Design Bases

The pressurizer relief discharge system collects, cools, and directs the steam and water discharged from various safety and relief valves in the containment for processing. The system consists of the pressurizer relief tank (PRT), the pressurizer safety and relief valve discharge piping, the relief tank internal spray header and associated piping, the tank nitrogen supply, and the drain to the liquid waste processing system.

The system design, including the PRT design volume, is based on the requirement to condense and cool a discharge of steam equivalent to 110% of the full-power pressurizer steam volume, without exceeding a pressure/temperature condition of 50 psig/200°F in the PRT. These values are well below the PRT design conditions of 100 psig and 340°F. Additional design data for the tanks are given in table 5.4.11-1.

The minimum volume of water in the PRT is determined by the energy content of the steam to be condensed and cooled, by the assumed initial temperature of the water, and by the desired final temperature of the water volume. The initial water temperature is assumed to be 120°F, which corresponds to the design maximum expected containment temperature for normal conditions. Provision is made to permit cooling of the water in the tank should the water temperature rise above 120°F during plant operation. The design final temperature, following a design discharge to the tank, is 200°F, which allows the contents of the tank to be drained directly to the liquid waste processing system without cooling.

The PRT saddle supports and anchor bolt arrangement are designed to withstand the loadings resulting from the vessel seismic, static, and nozzle loadings.

The pressurizer safety and relief valve piping and support arrangement is designed such that the effect of thrust forces on the piping system from valve operations is minimized. The piping analysis is discussed in section 3.9.N.

The design and location of the PRT rupture disks are such that they do not pose a missile threat to any safety-related equipment.

5.4.11.2 System Description

The piping and instrumentation diagram for the pressurizer relief discharge system is given in drawing 1X4DB112.

The steam and reactor grade water discharged from the various safety and relief valves inside the containment is routed to the PRT. Table 5.4.11-2 provides an itemized list of the discharges to the tank, together with references to the corresponding piping and instrumentation diagrams.

The pressurizer safety and relief valve piping and support arrangement in figure 5.4.11-2 shows the valve discharge piping, as well as the piping upstream of the safety and relief valves. The

piping upstream of the valves, which is not considered part of the pressurizer relief discharge system, includes the following:

A. Three lines with loop seal arrangements connecting the pressurizer nozzles to the three safety valves.

B. A line from the pressurizer relief nozzle branching to the two power-operated relief valves (PORVs), which have individual water seals and motor-operated isolation valves.

The pressurizer safety and relief valve discharge piping consists of:

A. A common piping manifold (supported over the top of the pressurizer) into which the safety and relief valves discharge.

B. Safety valve discharge lines to the manifold.

C. Relief valve discharge lines to the manifold.

D. A manifold downcomer discharge pipe.

E. Piping to the PRT.

The main support structure for the safety and relief valve piping consists of four column members equally spaced around the common manifold coupled to the valve support brackets on the pressurizer. No welding to the pressurizer is required. To increase the natural frequency of the system, auxiliary crossmembers are provided from the common manifold to the main support columns. The safety valves are provided with a bottom saddle type support coupled to the auxiliary crossmembers. The relief valves are positioned above the manifold, and the relief valve lines are supported at various points along the manifold.

The pressurizer safety and relief valve piping is constructed of austenitic stainless steel. Design data for the pressurizer safety and relief valve piping are given in table 5.4.3-1.

The piping upstream of the safety and relief valves is part of the reactor coolant system (RCS) and is designed and fabricated in accordance with American Society of Mechanical Engineers (ASME) Code, Section III, Class 1 requirements. The piping between these valves and the downcomer tee connection is nonnuclear safety related but is designed and fabricated to ASME Code, Section III, Class 2, to the extent practical. The support structure for the piping from the pressurizer to the downcomer tee connection is designed and fabricated to ASME Code, Section III, Subsection NF.

The piping from the pressurizer to the PRT is designed to Seismic Category 1 requirements. The principal design codes are indicated in table 3.2.2-1.

The general configuration of the PRT is shown in figure 5.4.11-1. The tank is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is overpressure protected by means of two safety heads with stainless steel rupture discs. Also shown in figure 5.4.11-1 are the flanged connection for the pressurizer safety and relief valve discharge line, the spray water inlet, the bottom drain connection, the gas vent connection, and the vessel supports. Although the tank is classified as nonnuclear safety related, it is designed and fabricated to Section III, Division 1, Class 3 of the ASME Code.

The tank normally contains water and a predominantly nitrogen atmosphere. In order to obtain effective condensing and cooling of the discharged steam, the tank is installed horizontally so that the steam can be discharged through a sparger pipe located near the bottom, under the water level. The sparger holes are designed to ensure good mixing of the discharged steam with the water initially in the tank.

A nitrogen gas blanket is used to control the atmosphere in the tank and to allow room for the expansion of the original water, plus the condensed steam discharge. The tank gas volume is sized such that the pressure following a design basis steam discharge does not exceed 50 psig, assuming an initial pressure of 3 psig. This pressure is low enough to prevent opening of the rupture discs. Provisions are made to permit the gas in the tank to be periodically analyzed to determine the concentration of hydrogen and/or oxygen.

The internal spray and bottom drain on the PRT function to cool the water when the temperature exceeds 120°F, as in the case following a steam discharge. The contents are cooled by a feed-and-bleed process, with cold reactor makeup water entering the tank through the spray water inlet and the warm mixture draining to the reactor coolant drain tank (RCDT). The contents may also be cooled by recirculation through the RCDT heat exchanger of the liquid waste processing system.

5.4.11.3 <u>Safety Evaluation</u>

The pressurizer relief discharge system does not constitute part of the reactor coolant pressure boundary in accordance with 10 CFR 50.2, since all of its components are downstream of the RCS safety and relief valves; thus, General Design Criteria 14 and 15 are not applicable. Furthermore, complete failure of the auxiliary systems serving the PRT will not impair the capability for safe plant shutdown.

The design of the system piping layout and piping restraints is consistent with the hazards protection requirements discussed in section 3.6 and appendix 3F. The safety and relief valve discharge piping is restrained so that the integrity and operability of the valves are maintained in the event of a rupture. Regulatory Guide 1.67 is not applicable since the system is not an open discharge system.

The pressurizer relief discharge system is capable of handling the design discharge of steam without exceeding the design pressure and temperature. The volume of nitrogen in the PRT is that required to limit the maximum pressure accompanying the design basis discharge to 50 psig, half the design pressure of the tank. The volume of water in the PRT is capable of absorbing the heat from the assumed discharge while maintaining the water temperature below 200°F.

If a discharge results in a pressure that exceeds the design, the rupture discs on the tank would pass the discharge through the tank to the containment. The rupture discs on the relief tank have a relief capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse, if the contents cool following a discharge without nitrogen being added.

The discharge piping from the pressurizer safety and relief valves to the PRT is sufficiently large to prevent backpressure at the safety valves from exceeding 20% of the setpoint pressure at full flow.

The recommendations of NUREG-0737, Action Items II.G.1 and II.K.3.1, are met as discussed below. The pressurizer is equipped with two Class 1E PORVs (solenoid operated) and two Class 1E PORV block valves (motor operated). The PORV and associated block valve on one line are supplied with control and motive power from train A, while the other PORV and associated block valve on the other line are powered from train B (drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113).

The PORV block valves 1HV-8000A and 1HV-8000B are powered from Class 1E 480-V buses. These buses are normally supplied from offsite power. In the event of a loss of offsite power,

these buses are automatically loaded onto the diesels (drawings 1X3D-AA-A01A and 2X3D-AA-A01A). PORVs 455A and 456A are Class 1E dc solenoid valves and are powered from redundant Class 1E 125-V dc trains A and B, respectively. The train assignment for power to the PORVs and block valves is based on:

A. The ability to open one of the parallel pressurizer vent paths in conjunction with a single failure.

B. The ability to close both parallel paths in conjunction with a single failure. (Capability to isolate both parallel paths in conjunction with a single failure is based upon the fact that the solenoid-operated PORVs are qualified, dc powered, and designed to fail closed.)

Pressurizer pressure is interlocked with the PORV block valves, which provides automatic closure of the block valves upon low pressurizer pressure.

5.4.11.4 Instrumentation Requirements

The following instrumentation is provided on the main control board:

A. The PRT pressure transmitter provides a signal to an indicator. An alarm is provided to indicate high tank pressure.

B. The PRT level transmitter supplies a signal to an indicator. High- and low-level alarms are also provided.

C. The temperature of the water in the PRT is displayed by an indicator. An alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

D. The temperature of the safety and relief valve discharge lines is displayed by indicators. Alarms actuated by high temperature notify the operator of steam discharge due to either leakage or valve actuation.

5.4.11.5 Inspection and Testing Requirements

The nondestructive examinations performed during fabrication of the forged piping from the pressurizer to the downcomer tee connection are identified in table 5.4.11-3.

The PRT is subject to nondestructive and hydrostatic testing during construction and after installation in accordance with Section III, Division 1, Class 3 of the ASME Code.

The downcomer piping to the PRT is subject to nondestructive and hydrostatic testing during construction.

Periodic visual inspections and preventive maintenance are conducted on the system components according to normal industrial practice.

5.4.12 VALVES

5.4.12.1 Design Bases

As noted in section 5.2, all valves out to and including the second valve that is normally closed or capable of automatic or remote closure, larger than 3/4 in., are American Society of Mechanical Engineers (ASME) Code, Section III, Class 1 valves. Valves 3/4-in. or smaller in lines connected to the reactor coolant system (RCS) are Class 1, unless the Class 1 piping interface is provided with suitable flow limiting orificing, then the valves are Class 2.

For a check valve to qualify as part of the RCS, it must be located inside the containment system. When the second of two normally closed check valves is considered part of the RCS (as defined in section 5.1), means are provided to periodically assess backflow leakage of the first valve when closed.

To ensure that the valves will meet the design objectives, the materials of construction minimize corrosion/erosion and are compatible with the environment. Leakage is minimized to the extent practicable by design.

5.4.12.2 Design Description

All manual- and motor-operated valves of the RCS which are 3 in. and larger are provided with double-packed stuffing boxes and intermediate lantern ring leakoff connections or a reduced packing configuration with the valve stem leakoff line removed. Throttling control valves over 2 in. are provided with double-packed stuffing boxes and stem leakoff connections. In general, RCS leakoff connections are piped to a closed collection system. Leakage to the atmosphere is essentially zero for these valves.

Gate valves at the engineered safety features interface are wedge design and are essentially straight through. The wedges are flex-wedge or solid. All gate valves have backseats.

Globe valves are "T" and "Y" styles of outside screw and yoke construction.

Check valves are swing type for sizes 2 1/2 in. and larger. All check valves which contain radioactive fluid are stainless steel and do not have body penetrations other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet. All operating parts are contained within the valve body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.4.12.3 Design Evaluation

The design requirements for Class 1 valves, as discussed in section 5.2, limit stresses to levels which ensure the structural integrity of the valves. In addition, the testing programs described in section 3.10.N demonstrate the ability of the valves to operate, as required, during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are specified in the design specifications to ensure the compatibility of valve construction materials with the reactor coolant. To ensure that the reactor coolant continues to meet these parameters, the chemical composition of the coolant will be analyzed periodically, as discussed in the Technical Requirements Manual.

The above requirements and procedures, coupled with the previously described design features for minimizing leakage, ensure that the valves will perform their intended functions.

5.4.12.4 Tests and Inspections

Technical Specification 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage, requires leakage from reactor coolant pressure isolation valves to be within limits. The valves to which Technical specification 3.4.14 applies and which are required to satisfy leakage limits are stated in FSAR table 5.4.12-3.

FSAR table 5.4.12-3 shows the following:

- The valves to which Technical Specification 3.4.14 applies and the associated function.
- The size of each RCS pressure isolation valve, and
- The maximum allowable leakage for each RCS pressure isolation valve.

Hydrostatic shell test and seat leakage and functional tests are performed on all RCS valves. The tests and inspections discussed in section 3.10.N are performed to ensure the operability of the active valves.

There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practical. Plant layout configurations determine the degree of inspectability. The valve nondestructive examination program is given in table 5.4.12-2. Inservice inspection is discussed in subsection 5.2.4.

Tests and inspections performed under the following license renewal aging management programs are credited as applicable to various RCS valves:

- Bolting Integrity Program (see subsection 19.2.2).
- Boric Acid Corrosion Control Program (see subsection 19.2.3).
- External Surfaces Monitoring Program (see subsection 19.2.8).
- Inservice Inspection Program (see subsection 19.2.13).
- Oil Analysis Program (subsection 19.2.16).
- One-Time Inspection Program (see subsection 19.2.17).
- Water Chemistry Control Program (see subsection 19.2.28).

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Design Bases

The combined capacity of the pressurizer safety valves can accommodate the maximum pressurizer surge resulting from complete loss of load. Sizing of the pressurizer safety valves is discussed in subsection 5.2.2.

The pressurizer power-operated relief valves (PORVs) are designed to limit pressurizer pressure to a value below the high pressure reactor trip setpoint. They are designed to fail in the closed position on loss of actuating power.

5.4.13.2 Design Description

The pressurizer safety valves are of the pop type. The valves are spring loaded and self-actuated by direct fluid pressure and have backpressure compensation features.

The pipe connecting each pressurizer nozzle to its safety valve is shaped in the form of a loop seal. Condensate resulting from normal heat losses accumulates in the loop. This loop seal minimizes any leakage of hydrogen gas or steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they start lifting, and the water from the seal discharges during the actuation period.

The pressurizer PORVs are solenoid-operated valves which respond to a signal from a pressure-sensing system or to manual control. Remotely operated block valves are provided to isolate the inlets to the PORVs if excessive leakage develops.

The PORVs and their associated block valves are interlocked by a pressurizer low-pressure interlock. Actuation of the interlock prevents the relief valves from opening and closes the block valves. Manual control may override this interlock.

In the event that a pressurizer PORV open signal actuation is sent due to a failure in a pressure channel associated with normal PORV operation, the interlock is provided to close the PORV as pressure decreases below the interlock pressure setpoint. The pressure signal associated with the interlock originates in the narrow range pressurizer pressure instrumentation. This signal and interlock operate separately from the cold overpressure pressure control signal, which originates in the wide range pressure instrumentation in the reactor coolant system loops. The overpressure protection system would not become disabled in the event of a single failure. The logic diagrams for the PORV interlocks are shown in drawing 1X6AA02-235.

In accordance with the requirements of NUREG-0737, TMI Action Item II.D.3, positive position indication is provided for the primary safety and relief valves.

Position indication on the PORV is accomplished through electrical reed switches. A magnetic rod, actuated by the valve plug, is located inside a projection above the top face of the bonnet and operates the reed switches contained in a switch assembly mounted externally on the bonnet. Safety valve indication is also accomplished through reed switches.

Temperatures in the pressurizer safety and relief valve discharge lines are measured, and an indication and a high alarm are provided on the main control board. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve.

The PORVs provide the safety-related means for reactor coolant system depressurization to achieve cold shutdown. For a discussion of the use of these valves to achieve safety-grade cold shutdown, see subsection 5.4.7.

Design parameters for the pressurizer safety valves and power relief valves are given in table 5.4.13-1.

Relief and safety valve failures to close will promptly be reported to the Nuclear Regulatory Commission. This will meet the requirements of NUREG-0737, Action Item II.K.3.3.

5.4.13.3 Design Evaluation

The pressurizer safety valves prevent reactor coolant system pressure from exceeding 110% of system design pressure, in compliance with the ASME Code, Section III.

The slight time delay associated with the discharge of water from the loop seal piping configuration is accounted for in the limiting case analyses discussed in chapter 15.

The limiting pressure transient is the turbine trip event as described in subsection 15.2.3 and in the ASME Code Overpressure Protection Report. In the event, the pressurizer safety valves are assumed to open following a 2% tolerance in the set pressure, plus a 1% shift resulting from the presence of the loop seal, plus the time delay required to purge the loop seal. After these effects have been accounted for, the safety valves are assumed to relieve at their full capacity. As described in subsection 15.2.3, there is margin to the 110% of design pressure limit of 2748.5 psia. The results from the limiting case from subsection 15.2.3 are also presented in figure 5.4.13-1.

The pressurizer PORVs prevent actuation of the reactor high-pressure trip for all design transients up to and including the design step-load decreases with steam dump. The relief valves also limit undesirable opening of the spring-loaded safety valves.

5.4.13.4 Tests and Inspections

All safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections, as required. For safety valves that are required to function during a faulted condition, additional tests are performed. These tests are described in section 3.10.N. There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection. Refer to subsection 5.4.12 for the list of license renewal aging management programs credited to manage aging of various RCS valves.

Safety and relief valves similar to those at VEGP have been tested within the Electric Power Research Institute safety and relief test program and have been found adequate for steamflow and waterflow. The completion of this program addresses the requirements of Action Item II.D.1 of NUREG-0737 as related to valve testing.

5.4.14 COMPONENT SUPPORTS

5.4.14.1 Design Bases

Component supports allow unrestrained lateral thermal movement of the loop during plant operation and provide restraint to the loops and components during accident and seismic conditions. The loading combinations and design stress limits are discussed in paragraph 3.9.B.3.[a] Support design is in accordance with the American Society of Mechanical Engineers

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[a] Reference 1 provides the original criteria for postulating breaks in the reactor coolant loop. The basis for eliminating eight of these postulated large pipe breaks in the reactor coolant loop is provided in reference 2. The RCL component support design configuration of Unit 1 is unchanged from that provided in reference 1. However, the elimination of the large pipe breaks led to partial installation of the primary loop whip restraints on Unit 1 and the removal of the primary loop whip restraints on Unit 2 and the reduction from five large bore hydraulic snubbers to two in the steam generator upper support assemblies

(ASME) Code, Section III, Subsection NF. The design maintains the integrity of the RCS boundary for normal, seismic, and accident conditions and satisfies the requirements of the piping code. The results of piping and supports stress evaluation are presented in section 3.9.

Conformance with Regulatory Guides 1.124 and 1.130 is discussed in section 1.9.

5.4.14.2 Description

The support structures are welded structural steel sections. Linear-type structures (tension and compression struts, columns, and beams) are used in all cases except for the reactor vessel supports, which are plate-type structures. Attachments to the supported equipment are nonintegral type that are bolted to or bear against the components. The supports-to-concrete attachments are either anchor bolts or embedded fabricated assemblies.

The supports permit virtually unrestrained thermal growth of the supported systems but restrain vertical, lateral, and rotational movement resulting from seismic and pipe break loadings. This is accomplished using spherical bushings in the columns for vertical support and girders, bumper pedestals, hydraulic snubbers, and tie rods for lateral support.

Because of manufacturing and construction tolerances, ample adjustment in the support structures is provided to ensure proper erection alignment and fit-up. This is accomplished by shimming or grouting at the supports-to-concrete interface and by shimming at the supports-to-equipment interface.

The supports for the various components are described in the following paragraphs.

5.4.14.2.1 Reactor Pressure Vessel

Supports for the reactor vessel (figure 5.4.14-1) are individual, air-cooled, rectangular box structures beneath the vessel nozzles bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate that transfers the loads to the primary shield wall concrete, and connecting vertical plates. The supports are air-cooled to maintain the supporting concrete temperature within acceptable levels.

5.4.14.2.2 Steam Generator

As shown in figure 5.4.14-2, the steam generator supports consist of the following elements:

A. Vertical Support

   Four individual columns provide vertical support for each steam generator. These are bolted at the top to the steam generator and at the bottom to the concrete structure. Spherical ball bushings at the top and bottom of each column allow unrestrained lateral movement of the steam generator during heatup and cooldown. The column base design permits both horizontal and vertical adjustment of the steam generator for erection and adjustment of the system.

B. Lower Lateral Support

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for Unit 2. The structural analysis of the RCL component supports for Units 1 and 2 considers the elimination of loads due to RCL breaks.

Lateral support is provided at the generator tube sheet by fabricated steel girders and struts. These are bolted to the compartment walls and include bumpers that bear against the steam generator but permit unrestrained movement of the steam generator during changes in system temperature. Stresses in the beams caused by wall displacements during compartment pressurization are considered in the design.

C. Upper Lateral Support

Upper lateral support of the steam generator is provided by a builtup ring plate girder at the operating deck. The 2-way acting snubbers restrain sudden seismic or blowdown-induced motion but permit the normal thermal movement of the steam generator.

Movement perpendicular to the thermal growth direction of the steam generator is prevented by struts.

5.4.14.2.3 Reactor Coolant Pump

Three individual columns, similar to those used for the steam generator, provide the vertical support for each pump. Lateral support for seismic and blowdown loading is provided by three lateral tension tie bars. The pump supports are shown in figure 5.4.14-3.

5.4.14.2.4 Pressurizer

The supports for the pressurizer, as shown in figure 5.4.14-4, consist of:

A. A steel ring plate between the pressurizer skirt and the supporting structure. The ring serves as leveling and adjusting member for the pressurizer.

B. The upper lateral support consists of struts cantilevered off the compartment walls that bear against the lugs provided on the pressurizer.

5.4.14.2.5 Control Rod Drive Mechanism (CRDM) Supports

The support system for the CRDM provides lateral restraint to limit CRDM deflections due to seismic or pipe break loadings. The CRDM support system consists of the following:

A. A support platform extends across the top of the control rod drive housing columns. The CRDM rod travel housing extensions protrude through holes in this platform, thus limiting lateral deflection of the CRDM housing.

B. Vertical support of the support platform is provided by three reactor vessel head lifting legs. The lifting legs are vertical columns which are pinned to the reactor vessel head.

C. Horizontal support of the support platform is provided by lateral tension tie rods which are pinned to the refueling cavity wall.

5.4.14.3 Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or loss-of-coolant accident conditions. Loads which the system is expected to encounter often during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values. The modeling and analysis methods are discussed in paragraph 3.9.N.1.4

5.4.14.4 Tests and Inspections

Nondestructive examinations are performed in accordance with the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF. The Inservice Inspection Program is credited as a license renewal aging management program for RCS primary equipment supports (see subsection 19.2.13). The Structural Monitoring Program is also credited as a license renewal aging management program for ASME piping and component supports (see subsection 19.2.32).

5.4.14.5 References

1. "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082-P-A (proprietary) and WCAP-8172-A (nonproprietary), January 1975.
2. Federal Register, Vo. 50, No. 27, February 8, 1985.

5.4.14.6 Bibliography

DeRosa, P., et al., "Evaluation of Steam Generator Tube, Tube Sheet and Divider Plant Under Combined LOCA Plus SSE Conditions," WCAP-7832, December 1973.

Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactor, Program Summaries-Winter 1976," WCAP-8768, Revision 1, June 1977.

"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September l973.

5.4.15 REACTOR VESSEL HEAD VENT SYSTEM

The reactor vessel head vent system (RVHVS) (figure 5.1.2-1) removes noncondensable gases or steam from the reactor vessel head. This system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases in the reactor coolant system (RCS). The design of the RVHVS is in accordance with the requirements of TMI action plan item II.B.1 of NUREG 0737 as discussed below.

5.4.15.1 Design Bases

The RVHVS is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the control room. The system discharges to the pressurizer relief tank. Additionally, a letdown flow path is provided from the reactor vessel head vent to the excess letdown heat exchanger in the chemical and volume control system. The RVHVS is designed to vent a volume of hydrogen at system design pressure and temperature approximately equivalent to one-half of the RCS volume in 1 h.

The system provides for venting the reactor vessel head by using only safety grade equipment. The RVHVS satisfies applicable requirements and industry standards, including ASME Code classification, safety classification, single-failure criteria, and environmental qualification.

All piping and equipment from the vessel head vent up to and including the second isolation valve in each flow path are designed and fabricated in accordance with ASME Section III, Class 1 requirements. The piping and equipment in the flow paths from the isolation valve to the modulating valves and from the isolation valves to the excess letdown heat exchanger are designed and fabricated in accordance with ASME Section III, Class 2 requirements. The remainder of the piping and equipment is Seismic Category 1, nonnuclear safety.

All supports and support structures conform with the requirements of the ASME Code.

The analysis of the reactor vessel head vent piping is based on the following plant operating conditions defined in the ASME Code, Section III:

A. Normal Condition

Pressure, deadweight, and thermal expansion analysis of the vent piping during:

1. Normal reactor operation with the vent isolation valves closed.
2. Post-refueling venting.

B. Upset Condition

Loads generated by the operating basis earthquake (OBE).

C. Faulted Condition

Loads generated by the safe shutdown earthquake (SSE). Loads generated by valve thrust during venting. In accordance with ASME III, faulted conditions are not included in fatigue evaluations.

The Class 1 piping used for the reactor vessel head vent is 1 in. schedule 160 and, therefore, in accordance with ASME III, is analyzed following the procedures of NC-3600 for Class 2 piping.

For all plant operating conditions listed above, the piping stresses are shown to meet the requirements of equations 8, 9, and 10 or 11 of ASME III, NC-3600, with a design temperature of 650°F and a design pressure of 2,485 psig.

5.4.15.2 System Description

The RVHVS consists of a single active failure proof flow path with redundant isolation valves. The equipment design parameters are listed in table 5.4.15-1.

The active portion of the system consists of four 1-in. open/close solenoid-operated isolation valves connected to the existing 1-in. vent pipe, which is located near the center of the reactor vessel head. The system design with two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The isolation valves in one flow path

are powered by one vital power supply and the valves in the second flow path are powered by a second vital power supply. The isolation valves are fail closed normally closed valves. The valves are included in the valve operability program and will be qualified to IEEE-323-1975, -344-A75 and 382-1972. The control valves are also normally closed, fail closed.

The vent system piping is supported to ensure that the resulting loads and stresses on the piping and on the vent connection to vessel head are acceptable.

5.4.15.3 Safety Evaluation

If one single active failure prevents a venting operation through one flow path, the redundant path is available for venting. The two isolation valves in each flow path provide a similar method of isolating the venting system. With two valves in series, the failure of any one valve or power supply will not inadvertently open a vent path or prevent opening and closing a flow path. Thus, the combination of safety grade train assignments and valve failure modes will not prevent vessel head venting nor venting isolation with any single active failure.

The RVHVS has two normally deenergized valves in series in each flow path. This arrangement eliminates the possibility of an opened flow path due to the spurious movement of one valve. As such, power lockout to any valve is not considered necessary.

A break of the RVHVS line would result in a small loss-of-coolant accident (LOCA) of not greater than 1-in. diameter. Such a break is similar to those analyzed in WCAP-9600 (1979). Since a break in the head vent line would behave similarly to the hot leg break case presented in WCAP-9600, the results presented therein are applicable to a RVHVS line break. Therefore, this postulated vent line break results in no calculated core uncovery.

5.4.15.4 Inspection and Testing Requirements

Inservice inspection is conducted in accordance with section 6.6.

5.4.15.5 Instrumentation Requirements

The system is operated from the control room or the shutdown panels. The isolation valves have stem position switches. The position indication from each valve is monitored at the control room by status lights.

TABLE 5.4.1-1 (SHEET 1 OF 2)

REACTOR COOLANT PUMP DESIGN PARAMETERS

Unit design pressure (psig)	2485
Unit design temperature (°F)	650[a]
Unit overall height (ft)	27.4
Seal water injection (gal/min)	8
Seal water return (gal/min)	3
Component cooling waterflow (gal/min)	596
Maximum continuous component cooling water inlet temperature (°F)	105
Total weight, dry (lb)	201,300
Pump	
Design flow (gal/min)	100,600
Developed head (ft)	288
NPSH required (ft)	Figure 5.4.1-2
Suction temperature, thermal design (°F)	558.2
Pump discharge nozzle, inside diameter (in.)	27-1/2
Pump suction nozzle, inside diameter (in.)	31
Speed (rpm)	1187
Water volume (ft^3)	80[b]
Motor	
Type	Drip-proof squirrel cage induction, with water/air coolers
Power (hp)	7000
Voltage (V)	13,200
Phase	3
Frequency (Hz)	60
Insulation class	Class F thermalastic epoxy insulation
Current (amp)	
Starting	1750 at 13,200 V
Nominal input, hot reactor coolant	253
Nominal input, cold reactor coolant	336

TABLE 5.4.1-1 (SHEET 2 OF 2)

Pump moment of inertia, maximum (lb/ft^2)	
Flywheel	70,000
Motor	22,500
Shaft	520
Impeller	1980

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a. Design temperature of pressure-retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high-pressure side of the controlled leakage seal is that temperature determined for the parts for a reactor coolant loop temperature of 650°F.

b. Composed of reactor coolant in the casing and of seal injection and cooling water in the thermal barrier.

TABLE 5.4.1-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Castings	Yes		Yes	
Forgings				
Main shaft		Yes	Yes	
Main studs		Yes		Yes
Plate				
Flywheel		Yes	Yes[b]	Yes[b]
Weldments				
Circumferential	Yes		Yes	
Instrument connections			Yes	

a. RT - Radiographic.
UT - Ultrasonic.
PT - Dye penetrant.
MT - Magnetic particle.

b. Of machined bores keyways and drilled holes (either PT or MT).

TABLE 5.4.2-1

STEAM GENERATOR DESIGN DATA

Design pressure, reactor coolant side (psig)	2485
Design pressure, steam side (psig)	1185
Design pressure, primary to secondary (psi)	1600
Design temperature, reactor coolant side (°F)	650
Design temperature, steam side (°F)	600
Design temperature, primary to secondary (°F)	650
Total heat transfer surface area (ft^2)	55,000
Maximum moisture carryover (weight percent)	0.25
Overall height (ft-in.)	67-8
Number of U-tubes	5626
U-tube nominal diameter (in.)	0.688
Tube-wall nominal thickness (in.)	0.040
Number of manways	4
Inside diameter of manways (in.)	16
Number of handholes	8
Design fouling factor (ft^2-h-°F/Btu)	0.00006
Steamflow (lb/h)	4.055×10^6 to 4.080×10^6

TABLE 5.4.2-2 (SHEET 1 OF 2)

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]	ET[a]
Tube Sheet					
Forging		Yes		Yes	
Cladding		Yes[b]	Yes		
Channel Head (if fabricated)					
Fabrication	Yes[c]	Yes[d]		Yes	
Cladding			Yes		
Secondary Shell and Head					
Plates		Yes			
Tubes		Yes			Yes
Nozzles (Forgings)		Yes		Yes	
Weldments					
Shell, longitudinal	Yes			Yes	
Shell, circumferential	Yes			Yes	
Cladding (channel head-tube sheet joint cladding restoration)			Yes		
Primary nozzles to fab head	Yes			Yes	
Manways to fab head	Yes			Yes	
Steam and feedwater nozzles to shell	Yes			Yes	
Support brackets			Yes		
Tube to tube sheet			Yes		
Instrument connections (primary and secondary)				Yes	
Temporary attachments after removal				Yes	

TABLE 5.4.2-2 (SHEET 2 OF 2)

	RT[a]	UT[a]	PT[a]	MT[a]	ET[a]
After hydrostatic test (all major pressure boundary welds and complete cast channel head - where accessible)				Yes	
					Yes
Nozzle safe ends (if weld deposit)	Yes		Yes		

a. RT - Radiographic.
UT - Ultrasonic.
PT - Dye penetrant.
MT - Magnetic particle.
ET - Eddy current.

b. Flat surfaces only.

c. Weld deposit.

d. Base material only.

TABLE 5.4.3-1

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor Inlet Piping Inside diameter (ID) (in.)	27 1/2
Reactor Inlet Piping Nominal wall thickness (in.)	2.32
Reactor Outlet Piping ID (in.)	29
Reactor Outlet Piping Nominal wall thickness (in.)	2.45
Coolant Pump Suction Piping ID (in.)	31
Coolant Pump Suction Piping Nominal wall thickness (in.)	2.60
Pressurizer Surge Line Piping Nominal pipe size (in.)	16 reduced to 14
Pressurizer Surge Line Piping	
Nominal wall thickness, for 16 in. nominal pipe size, schedule 160 (in.)	1.594
Nominal wall thickness, for 14 in. nominal pipe size, schedule 160 (in.)	1.406
Nominal Water Volume, all four loops including surge line (ft^3)	1200-1300
RCL Piping	
Design/operating pressure (psig)	2485/2235
Design temperature (°F)	650
Pressurizer Surge Line	
Design pressure (psig)	2485
Design temperature (°F)	680
Pressurizer Safety Valve Inlet Line	
Design pressure (psig)	2485
Design temperature (°F)	680
Pressurizer Power-Operated Relief Valve Inlet Line	
Design pressure (psig)	2485
Design temperature (°F)	680

TABLE 5.4.3-2

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]
Fittings and Pipe (Castings)	Yes		Yes
Fittings and Pipe (Forgings)		Yes	Yes
Weldments			
Circumferential	Yes		Yes
Nozzle to runpipe (except no RT for nozzles less than 6 in.)	Yes		Yes
Instrument connections			Yes
Castings	Yes		Yes (after finishing)
Forgings			Yes (after finishing)

a. RT - Radiographic; UT - ultrasonic; PT - dye penetrant.

TABLE 5.4.7-1

DESIGN BASES FOR RHRS OPERATION

RHRS initiation, hours after reactor shutdown	~4
RCS initial pressure (psig)	~365
RCS initial temperature (°F)	~350
CCW design temperature (°F)	105[(a)]
Cooldown time, hours after initiation of RHRS operation (two trains in service)	~16[(b)]
RCS temperature at end of cooldown (°F)	140[(b)]
Decay heat generation at 20 h after reactor shutdown (Btu/h)	80.4×10^6
Residual heat removal heat exchanger UA	2.5×10^6 Btu/h°F
Fouling factor	0.0003 ft^2h°F/Btu

a. The maximum CCW temperature during cooldown with offsite power available is 120°F.

b. The design bases for cold shutdown are 32 h (one train) to 200°F.

TABLE 5.4.7-2

RHRS COMPONENT DATA

RHR Pumps		
Number		2
Design pressure (psig)		600
Design temperature (°F)		400
Design flow (gal/min)		3000
Design head (ft)		375
Material		Austenitic stainless steel
Residual Heat Exchangers		
Number		2
Design heat removal capacity (Btu/h)		32.8×10^6
Estimated UAF_{LMTD} (Btu/h)		2.5×10^6
	Tube Side	Shell Side
Design pressure (psig)	600	150
Design temperature (°F)	400	200
Design flow (lb/h)	1.48×10^6	2.48×10^6
Inlet temperature (°F)	140	105
Outlet temperature (°F)	117.8	118.2
Material	Austenitic stainless steel	Carbon steel
Fluid	Reactor coolant	CCW

TABLE 5.4.7-3 (SHEET 1 OF 5)

FAILURE MODES AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM
ACTIVE COMPONENTS - NORMAL COOLDOWN OPERATION

Component	Failure Mode	Effect on System Operation[(a)]	Failure Detection Method[(b)]	Remarks
1. Motor-operated gate valve HV-8701A (HV-8701B analogous).	a. Fails to open on demand (open manual mode CB switch selection).	Failure blocks reactor coolant flow from hot leg of RC loop 1 through train A of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop 4 flowing through train B of RHRS. However, time required to reduce RCS temperature will be extended.	Valve position indication (closed to open position change) at CB; RC loop 1 or 4 hot leg pressure indication at CB; RHR train A discharge flow indication; and RHR pump discharge pressure indication at CB.	Valve is electrically interlocked with the containment sump suction valve HV-8811A, with RWST isolation valve HV-8812A, with RHR to charging pump suction isolation valve HV-8804A and with a "prevent-open" pressure interlock PT-438 (PT-408). The valve cannot be opened remotely from the CB if one of the indicated isolation valves is open or if RC loop pressure exceeds 365 psig. If both trains of RHRS are unavailable for plant cooldown due to multiple component failures, the auxiliary feedwater system and SG power-operated relief valves can be used to perform the safety function of removing residual heat.
2. Motor-operated gate valve HV-8702A (HV-8702B analogous).	Same failure modes as those stated for item 1.	Same effect on system operation as that stated for item 1.	Same methods of detection as those stated for item 1.	Same remarks as those stated for item 1, except for pressure interlock PT-418 (PT-428) control.

TABLE 5.4.7-3 (SHEET 2 OF 5)

Component	Failure Mode	Effect on System Operation	Failure Detection Method	Remarks
3. RHR pump 1, (RHR pump 2 analogous).	Fails to deliver working fluid.	Failure results in loss of reactor coolant flow from hot leg of RC loop 1 through train A of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements are met by reactor coolant flow from hot leg or RC loop 4 flowing through train B of RHRS. However, time required to reduce RCS temperature will be extended.	Open pump switchgear circuit breaker indication at CB; circuit breaker close position monitor light for group monitoring of components at CB; common breaker trip alarm at CB; RC loop 1 hot leg pressure indication at CB; RHR train A discharge flow indication and low flow alarm at CB; and pump discharge pressure indication at CB.	The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program. (See subsection 6.3.4.) Pump failure may also be detected during ECCS testing.
4. Motor-operated globe valve FV-610 (FV-611 analogous).	a. Fails to open on demand (open manual mode CB switch selection).	Failure blocks miniflow line to suction of RHR pump 1 during cooldown operation. No effect on safety for system operation. Operator may establish miniflow for RHR pump 1 operation by opening of CVCS letdown control valve HCV-128 and manual valve 1205-U4-021 to allow flow to CVCS.	Valve position indication (closed to open position change) at CB.	Valve is automatically controlled to open when pump discharge is less than the open setpoint (824 gpm at 350°F, 780 gpm at 100°F) and close when the discharge exceeds the closed setpoint (1944 gpm at 350°F, 1841 gpm at 100°F). The valve protects the pump from deadheading during ECCS operation CB switch set to "Auto" position for automatic control of valve positioning.
	b. Fails to close on demand (auto mode CB switch selection).	Failure allows for a portion of RHR heat exchanger discharge flow to be bypassed to suction of RHR pump 1. RHRS train A is degraded for the regulation of coolant temperature by RHR heat exchanger 1. No effect on safety for system operation. Cooldown of RCS within established specification cooldown rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train B.	Valve position indication (open to closed position change) and RHRS train A discharge flow indication at CB.	

TABLE 5.4.7-3 (SHEET 3 OF 5)

Component	Failure Mode	Effect on System Operation	Failure Detection Method	Remarks
5. Air diaphragm-operated butterfly valve FCV-618 (FCV-619 analogous).	a. Fails to open on demand (Auto mode CB switch selection).	Failure prevents coolant discharged from RHR pump 1 from bypassing RHR heat exchanger 1resulting in mixed mean temperature of coolant flow to RCS being low. RHRS train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train B.	RHR pump 1 discharge flow temperature and RHRS train A discharge to RCS cold leg flow temperature recorded on the plant computer; and RHRS train A discharge to RCS cold leg flow indication at CB.	Valve is designed to fail closed and is electrically wired so that electrical solenoid of the air diaphragm operator is energized to open the valve. Valve is normally closed to align RHRS for ECCS operation during plant power operation and load follow.
	b. Fails to close on demand (Auto mode CB switch selection).	Failure allows coolant discharge from RHR pump 1 to bypass RHR heat exchanger 1, resulting in mixed mean temperature of coolant flow to RCS being high. RHRS train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train B. However, cooldown time will be extended.	Same methods of detection as those stated for item 5.a.	

TABLE 5.4.7-3 (SHEET 4 OF 5)

Component	Failure Mode	Effect on System Operation	Failure Detection Method	Remarks
6. Air diaphragm-operated butterfly valve HCV-606 (HCV-607 analogous).	a. Fails to close on demand for flow reduction.	Failure prevents control of coolant discharge flow from RHR heat exchanger 1, resulting in loss of mixed mean temperature coolant flow adjustment to RCS. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished by operator action of controlling cooldown with redundant RHRS train B.	Same methods of detection as those stated for item 5. In addition, monitor light and alarm (valve closed) for group monitoring of components at CB.	Valve is designed to fail open. Valve is normally open to align RHRS for ECCS operation during plant power operation and load follow.
	b. Fails to open on demand for increased flow.	Same effect on system operation as that stated for item 6.a.	Same methods of detection as those stated for item 6.a.	
7. Motor-operated gate valve HV-8812A (HV-8812B analogous).	Fails to close on demand.	Failure reduces the redundancy of isolation valves provided to isolate RHRS train A from RWST. No effect on safety for system operation. Check valve in series with motor-operated valve provides the primary isolation against the bypass of RCS coolant flow from the suction of RHR pump A to RWST.	Valve position indication (open to closed position change) at CB and valve (closed) monitor light and alarm at CB.	Valve is a component of the ECCS that performs an RHR function during plant cooldown. Valve is normally open to align RHRS for ECCS operation during plant power operation and load follow.

TABLE 5.4.7-3 (SHEET 5 OF 5)

Component	Failure Mode	Effect on System Operation	Failure Detection Method	Remarks
8. Motor-operated gate valve HV-8716A (HV-8716B analogous).	Fails to close on demand.	Failure reduces the redundancy of isolating, train A from train B for single train operation of RHRS. No effect on safety for system operation. Isolation will be provided by closing valve HV-8716B (HV-8716A).	Same as item 7.	

a. List of acronyms and abbreviations.

Auto	-	Automatic.
CB	-	Main control board.
CVCS	-	Chemical and volume control system.
ECCS	-	Emergency core cooling system.
RC	-	Reactor coolant.
RCS	-	Reactor coolant system.
RHR	-	Residual heat removal.
RHRS	-	Residual heat removal system.
RWST	-	Refueling water storage tank.
SG	-	Steam generator.

b. As part of plant operation; periodic tests, surveillance inspections, and instrument calibrations are conducted to monitor equipment and performance. Failures may be detected during such monitoring of equipment, in addition to detection methods noted.

TABLE 5.4.7-4 (SHEET 1 OF 5)

RESIDUAL HEAT REMOVAL SYSTEM - SAFETY GRADE COLD SHUTDOWN OPERATIONS - FAILURE MODES AND EFFECTS ANALYSIS

Component[a]	Failure Mode	Function[b]	Effect on System Operation	Failure Detection Methods[c]	Remarks
1. Motor-operated gate valve HV-8701A (HV-8701B analogous).	a. Fails to open on demand.	Provides isolation of fluid flow from the RCS to the suction of RHR pump 1.	a. Failure blocks reactor coolant flow from hot leg of RC loop 1 through train A of RHRS. Failure reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements are met by reactor coolant flow from hot leg of RC loop 4 flowing through train B of RHRS, however time required to reduce RCS temperature is extended.	a. Valve open/close position indication at CB; RC loop 1 or 4 hot leg pressure indication at CB; RHR train A discharge flow indication, and RHR pump 1 discharge pressure indication at CB.	Valve is electrically interlocked with a containment suction valve HV-8811A RWST to RHR suction line isolation valve HV-8812A, with RHR to charging pump suction line isolation valve HV-8804A and with a "prevent-open" pressure interlock PT-438 (PT-408). The valve can not be opened remotely from the CB if one of the indicated isolation valves is open or if RC loop pressure exceeds 365 psig. The valve can be manually opened.
	b. Once the valves are open the main control board annunciator alarm fails and RCS pressure exceeds 420 psig.		b. Failure reduces redundancy of main control board annunciator alarm at 420 psig. No effect on safety system operation. Plant operating procedures require that operators close both valves prior to an RCS pressure of 420 psig. Alternate alarm is provided by valve HV-8701B (HV-8701A).	b. Valve is interlocked with pressure interlock PT-438 (PT-408) to alarm on the main control board annunciator panel if one or both of the valves is not fully closed and RCS pressure exceeds 420 psig.	

TABLE 5.4.7-4 (SHEET 2 OF 5)

Component	Failure Mode	Function	Effect on System Operation	Failure Detection Methods	Remarks
2. Motor-operated gate valve HV-8702A (HV-8702B analogous).	Same as item 1.	Same as item 1.	Same as item 1.	Same as item 1.	Same as item 1, except for pressure interlock PT418 (PT428) control.
3. RHR pump 1 (RHR pump 2 analogous).	Fails to deliver working fluid.	Provides fluid flow of reactor coolant through RHR heat exchanger 1 to reduce RCS temperature during cooldown operation.	Failure results in loss of reactor coolant flow from hot leg of RC loop 1 through train A of RHRS. Failure reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements are met by reactor coolant flow from hot leg of RC loop 4 flowing through train B of RHRS, however, time required to reduce RCS temperature is extended.	Open pump switchgear circuit breaker indication at CB; circuit breaker close position monitor light for group monitoring of components at CB; common breaker trip alarm at CB; RC loop 1 hot leg pressure indication at CB; RHR train A discharge flow indication and low flow alarm at CB; and pump discharge pressure indication at CB.	The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program. (See subsection 6.3.4.)
4. Motor-operated globe valve FV-610 (FV-611 analogous).	a. Fails closed.	Provides regulation of fluid flow through mini-flow bypass line to suction of RHR pump 1 to protect against overheating of the pump and loss of discharge flow from the pump.	a. Failure blocks mini-flow line to suction of RHR pump 1 during cooldown operation. No effect on safety for system operation. Plant cooldown requirements are met by reactor coolant flow from hot leg of RC loop 4 flowing through train B of RHRS. However, time required to reduce RCS temperature is extended.	a. Valve open/close position indication at CB; and RHRS train A discharge flow indication at CB.	Valve is automatically controlled to open when pump discharge is less than the open setpoint (824 gpm at 350°F, 780 gpm at 100°F) and close when the discharge exceeds the closed setpoint (1944 gpm at 350°F, 1841 gpm at 100°F).

TABLE 5.4.7-4 (SHEET 3 OF 5)

Component	Failure Mode	Function	Effect on System Operation	Failure Detection Methods	Remarks
	b. Fails open.		b. Failure allows for a portion of RHR heat exchanger 1 discharge flow to be bypassed to suction of RHR pump 1. RHRS train A is degraded for the regulation of coolant temperature by RHR heat exchanger 1. No effect on safety for system operation. Cool-down of RCS remains within established specification cooldown rate.	Same as item 4.a.	
5. Air - diaphragm-operated - butterfly valve FCV-618 (FCV-619 analogous).	a. Fails to open on demand for flow increase ("Auto" mode CB switch selection).	Controls rate of fluid flow bypassed around RHR heat exchanger 1 during cooldown operation.	a. Failure prevents cool ant discharged from RHR pump 1 from bypassing RHR heat exchanger 1 resulting in mixed mean temperature of coolant flow to RCS being low. RHRS train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train B.	a. RHR pump 1 discharge flow temperature and RHRS train A discharge to RCS cold leg flow temperature recorded on the plant computer; and RHRS train A discharge to RCS cold leg flow indication at CB.	Valve is designed to fail closed and is electrically wired so that electrical solenoid of the air diaphragm operator is energized to open the valve. Valve is normally closed to align RHRS for ECCS operation during plant power operation and load follow. Valve is designed for normal plant cooldown operation. It is not required for safety grade cold shutdown operations.

TABLE 5.4.7-4 (SHEET 4 OF 5)

Component	Failure Mode	Function	Effect on System Operation	Failure Detection Methods	Remarks
	b. Fails to close on demand for flow reduction (Auto mode CB switch selection).		b. Failure allows coolant discharged from RHR pump 1 to bypass RHR heat exchanger 1 resulting in mixed mean temperature of coolant flow to RCS being high. RHRS train A is degraded for the regulation controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve HCV-606 and controlling cooldown with redundant RHRS train B; however, cooldown time is extended.	b. Same as item 5.a.	
6. Air diaphragm-operated butterfly valve HCV-606 (HCV-607 analogous).	a. Fails to close on demand for flow reduction.	Controls rate of fluid flow through RHR heat exchanger 1 during cool-down operation.	a. Failure prevents control of coolant discharge flow from RHR heat exchanger 1 resulting in loss of mixed mean temperature coolant flow adjustment to RCS. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished by operator action of controlling cooldown with redundant RHS train B.	a. Same methods of detections as those stated for item 5.a. addition, monitor light and alarm (valve closed) for group monitoring of components at CB.	Valve is designed to fail open. Valve is normally open to align RHRS for ECCS operation during plant power operation and load follow.
	b. Fails to open on demand for flow increase.		b. Same as item 6.a.	b. Same as item 6.a.	

TABLE 5.4.7-4 (SHEET 5 OF 5)

Component	Failure Mode	Function	Effect on System Operation	Failure Detection Methods	Remarks
7. Motor-operated gate valve HV-8812A (HV-8812B analogous).	Fails to close on demand.	Provides isolation of fluid from the RWST to suction of RHR pump 1 during cooldown operation.	No effect on safety for system operation. Plant cooldown requirements are met by reactor coolant flow from hot leg loop 4 flowing through train B of RHRS; however, time required to reduce RCS temperature is extended.	Valve open/closed position indication at CB and valve (closed) monitor light and alarm at CB.	Valve is normally open to align RHRS for ECCS operation during plant power operation and load follow. Valve must be closed during plant cooldown to satisfy electrical interlock to permit valves HV-8701A and B (HV-8702A, B) to be opened.
8. Motor-operated gate valve HV-8716A (HV-8716B analogous).	Fails to close on demand.	Provides separation between the two RHR trains during cooldown operation.	Failure reduces the redundancy for isolating RHR trains during cooldown. Negligible effect on system operation. Isolation valve HV-8716B (HV-8716A) provides backup isolation between the two RHR trains.	Same as item 7.	

a. Component 7 is a component of the ECCS that performs a safety-grade cold shutdown function.

b. List of acronyms and abbreviations.

Auto - Automatic.
BAT - Boric acid tank.
BIT - Boron injection tank (Unit 1 only).
CB - Main control board.
CVCS - Chemical and volume control system.
ECCS - Emergency core cooling system.
HELB - High-energy line break.
MELB - Moderate-energy line break.
PRT - Pressurizer relief tank.
RC - Reactor coolant.
RCS - Reactor coolant system.
RHR - Residual heat removal.
RHRS - Residual heat removal system.
RWST - Refueling water storage tank.
RV - Reactor vessel.
SI - Safety injection.
VCT - Volume control tank.

c. As part of plant operation, periodic tests, surveillance inspections, and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection methods noted.

TABLE 5.4.10-1

PRESSURIZER DESIGN DATA

Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle diameter (in.)	14
Heatup rate of pressurizer using heaters only (°F/h)	55
Internal volume (ft^3)	1800
Nominal conditions at 100-percent rated load	
Steam volume (ft^3)	720
Water volume (ft^3)	1080

TABLE 5.4.10-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Psig
Hydrostatic test pressure	3106
Design pressure	2485
Safety valves (begin to open)	2460
High pressure reactor trip	2385
High pressure alarm	2310
Power-operated relief valves	
PV-0455A	2345[(a)]
PV-0456A	2335[(b)]
Pressurizer spray valves (full open)	2310
Pressurizer spray valves (begin to open)	2260
Proportional heaters (begin to operate)	2250
Operating pressure	2235
Proportional heater (full operation)	2220
Backup heaters on	2210
Low pressure alarm	2210
Pressurizer power-operated relief valve interlock	2185
Low pressure reactor trip	1870

a. At 2345 psi, a pressure signal initiates actuation (opening) of this valve. Remote manual control is also provided.

b. At 2335 psi, a pressure signal initiates actuation (opening) of this valve. Remote manual control is also provided.

TABLE 5.4.10-3

PRESSURIZER QUALITY ASSURANCE PROGRAM

	RT[a]	UT[a]	PT[a]	MT[a]
Heads				
Plates		Yes		
Cladding			Yes	
Shell				
Plates		Yes		
Cladding			Yes	
Heaters				
Tubing		Yes[b]	Yes	
Centering of element	Yes			
Nozzle (Forgings)		Yes	Yes[c]	Yes[c]
Weldments				
Shell,longitudinal	Yes			Yes
Shell, circumferential	Yes			Yes
Cladding			Yes	
Nozzle safe end	Yes		Yes	
Instrument connection			Yes	
Support skirt, longitudinal seam	Yes			Yes
Support skirt to lower head		Yes		Yes
Temporary attachments (after removal)				Yes
All external pressure boundary welds after shop hydrostatic tests				Yes

a. RT - Radiographic.
UT - Ultrasonic.
PT - Dye Penetrant.
MT - Magnetic Particle.

b. Eddy current testing can be used in lieu of UT.

c. MT or PT.

TABLE 5.4.11-1

PRESSURIZER RELIEF TANK DESIGN DATA

Design pressure (psig)	100
Normal operating pressure (psig)	3
Final operating pressure (psig) [(a)]	50
Rupture disc release pressure (psig)	
Nominal	91
Range	86 to 100
Normal water volume (ft^3)	1350
Normal gas volume (ft^3)	450
Design temperature (°F)	340
Initial operating water temperature (°F) [(a)]	120
Final operating water temperature (°F) [(a)]	200
Total rupture disc relief capacity at 100 psig (lb/h)	1.6×10^6
Cooling time required following maximum discharge, approximately (h)	
Spray feed and bleed	1
Utilizing RCDT heat exchanger	8

a. For the design basis pressurizer steam discharge.

TABLE 5.4.11-2

DISCHARGES TO THE PRESSURIZER RELIEF TANK

Reactor coolant system (drawings 1X4DB111, 2X4DB111, 1X4DB112, 2X4DB112, and 1X4DB113)

Two pressurizer PORVs
Three pressurizer safety valves
One reactor vessel head vent

Residual heat removal system (drawing 1X4DB122)

Two suction line relief valves from the RCS hot legs

Chemical and volume control system (drawing 1X4DB114)

One seal water return line relief valve
One RCS letdown line relief valve

TABLE 5.4.11-3

PRESSURIZER RELIEF DISCHARGE SYSTEM NONDESTRUCTIVE TESTING PROGRAM

Components	Radiographic	Ultrasonic	Dye Penetrant
Fittings and pipe (castings)	Yes		Yes
Fittings and pipe (forgings)		Yes	Yes
Weldments			
Circumferential	Yes		Yes
Nozzle to runpipe (except no radiographic for nozzles less than 4 in.)	Yes		Yes
Instrument connections			Yes

TABLE 5.4.12-1

REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design pressure (psig)	2485
Preoperational plant hydrotest (psig)	3106
Design temperature (°F)	650

TABLE 5.4.12-2

REACTOR COOLANT SYSTEM VALVES NONDESTRUCTIVE EXAMINATION PROGRAM

	RT[a]	UT[a]	PT[a]
Castings			
Larger than 4 in.	Yes		Yes
2 to 4 in.	Yes[b]		Yes
Forgings			
Larger than 4 in.	[c]	[c]	Yes
2 to 4 in.			Yes

a. RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant

b. Weld ends only.

c. Either RT or UT.

TABLE 5.4.12-3 (SHEET 1 OF 2)

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

	VALVE NUMBER	VALVE SIZE (in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE(gpm)
1.	HV-8701A	12	RHR Suction (gate valve)	5.0
2.	HV-8701B	12	RHR Suction (gate valve)	5.0
3.	HV-8702A	12	RHR Suction (gate valve)	5.0
4.	HV-8702B	12	RHR Suction (gate valve)	5.0
5.	1204-U4-120	2	SI-Hot Leg 2nd Isolation Valve	1.0
6.	1204-U4-121	2	SI-Hot Leg 2nd Isolation Valve	1.0
7.	1204-U4-122	2	SI-Hot Leg 2nd Isolation Valve	1.0
8.	1204-U4-123	2	SI-Hot Leg 2nd Isolation Valve	1.0
9.	1204-U6-079	10	Accumulator 2nd Isolation Valve	5.0
10.	1204-U6-080	10	Accumulator 2nd Isolation Valve	5.0
11.	1204-U6-081	10	Accumulator 2nd Isolation Valve	5.0
12.	1204-U6-082	10	Accumulator 2nd Isolation Valve	5.0
13.	1204-U6-083	10	Injection Line 1st Isolation Valve	5.0
14.	1204-U6-084	10	Injection Line 1st Isolation Valve	5.0
15.	1204-U6-085	10	Injection Line 1st Isolation Valve	5.0

TABLE 5.4.12-3 (SHEET 2 OF 2)

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

	VALVE NUMBER	VALVE SIZE (in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE(gpm)
16.	1204-U6-086	10	Injection Line 1st Isolation Valve	5.0
17.	1204-U6-124	6	SI-Hot Leg 1st Isolation Valve	3.0
18.	1204-U6-125	6	SI-Hot Leg 1st Isolation Valve	3.0
19.	1204-U6-126	6	SI-Hot Leg 1st Isolation Valve	3.0
20.	1204-U6-127	6	SI-Hot Leg 1st Isolation Valve	3.0
21.	1204-U6-128	8	RHR-Hot Leg 2nd Isolation Valve	4.0
22.	1204-U6-129	8	RHR-Hot Leg 2nd Isolation Valve	4.0
23.	1204-U4-143	2	SI-Cold Leg 2nd Isolation Valve	1.0
24.	1204-U4-144	2	SI-Cold Leg 2nd Isolation Valve	1.0
25.	1204-U4-145	2	SI-Cold Leg 2nd Isolation Valve	1.0
26.	1204-U4-146	2	SI-Cold Leg 2nd Isolation Valve	1.0
27.	1204-U6-147	6	RHR Cold Leg 2nd Isolation Valve	3.0
28.	1204-U6-148	6	RHR Cold Leg 2nd Isolation Valve	3.0
29.	1204-U6-149	6	RHR Cold Leg 2nd Isolation Valve	3.0
30.	1204-U6-150	6	RHR Cold Leg 2nd Isolation Valve	3.0

TABLE 5.4.13-1

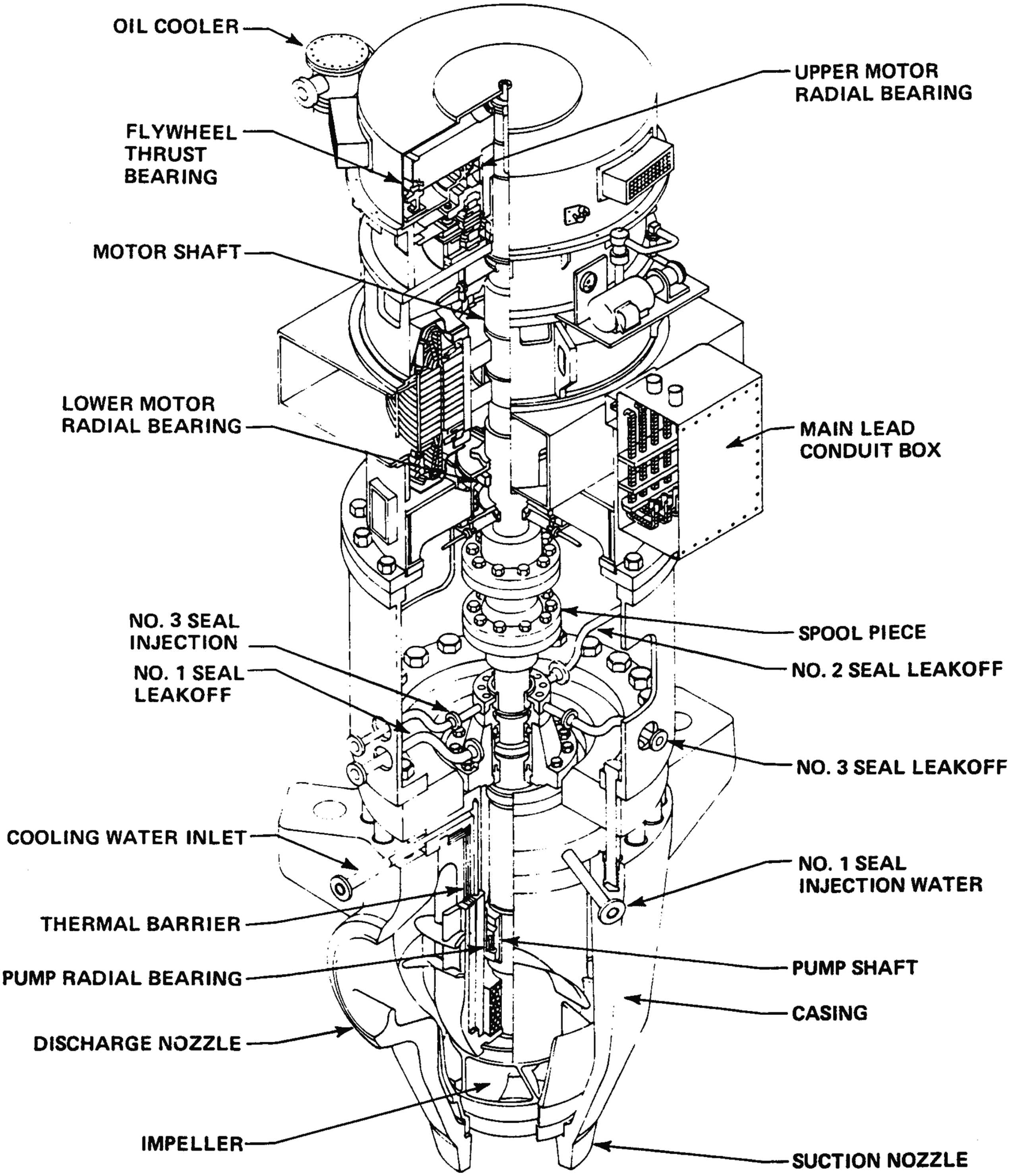
PRESSURIZER SAFETY AND RELIEF VALVES DESIGN PARAMETERS

Pressurizer safety valves	
Number	3
Minimum relieving capacity at 2560 psig per valve, ASME flowrate (lb/h)	420,000
Set pressure (psig)	2460
Design temperature (°F)	650
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100
Pressurizer power-operated relief valves	
Number	2
Design pressure (psig)	2485
Design temperature (°F)	650
Saturated steam-relieving capacity at 2385 psig, per valve (lb/h)	210,000
Saturated water-relieving capacity at 2485 psig, per valve (gal/min)	230

TABLE 5.4.15-1

REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT
DESIGN PARAMETERS

<u>Valves</u>	
Number (includes one manual valve)	7
Design pressure, psig	2485
Design temperatures, °F	650
<u>Piping</u>	
Vent line, nominal diameter, in.	1
Design pressure, psig	2485
Design temperature, °F	650
Maximum operating temperature	620



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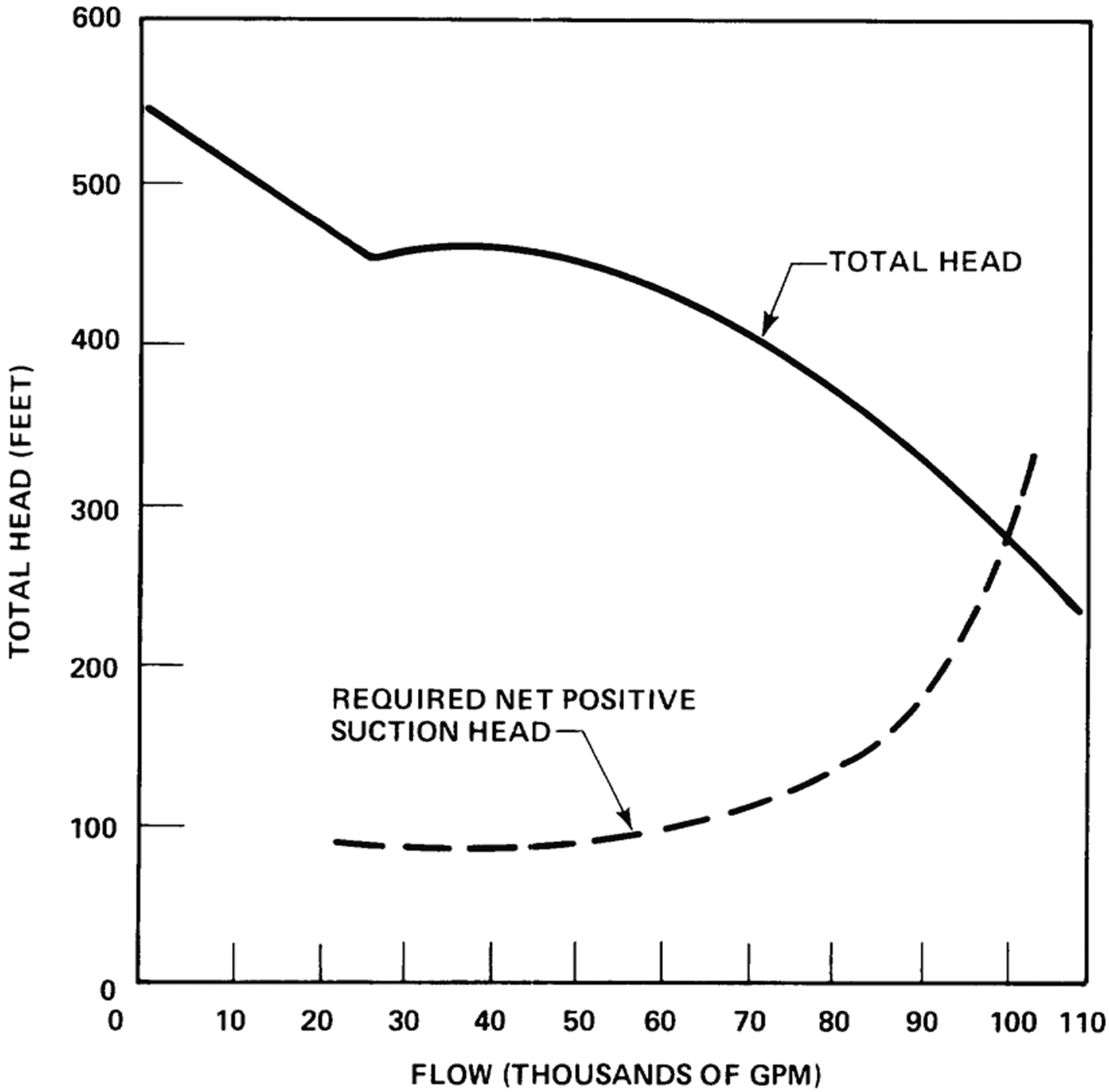
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR COOLANT PUMP FIGURE 5.4.1–1



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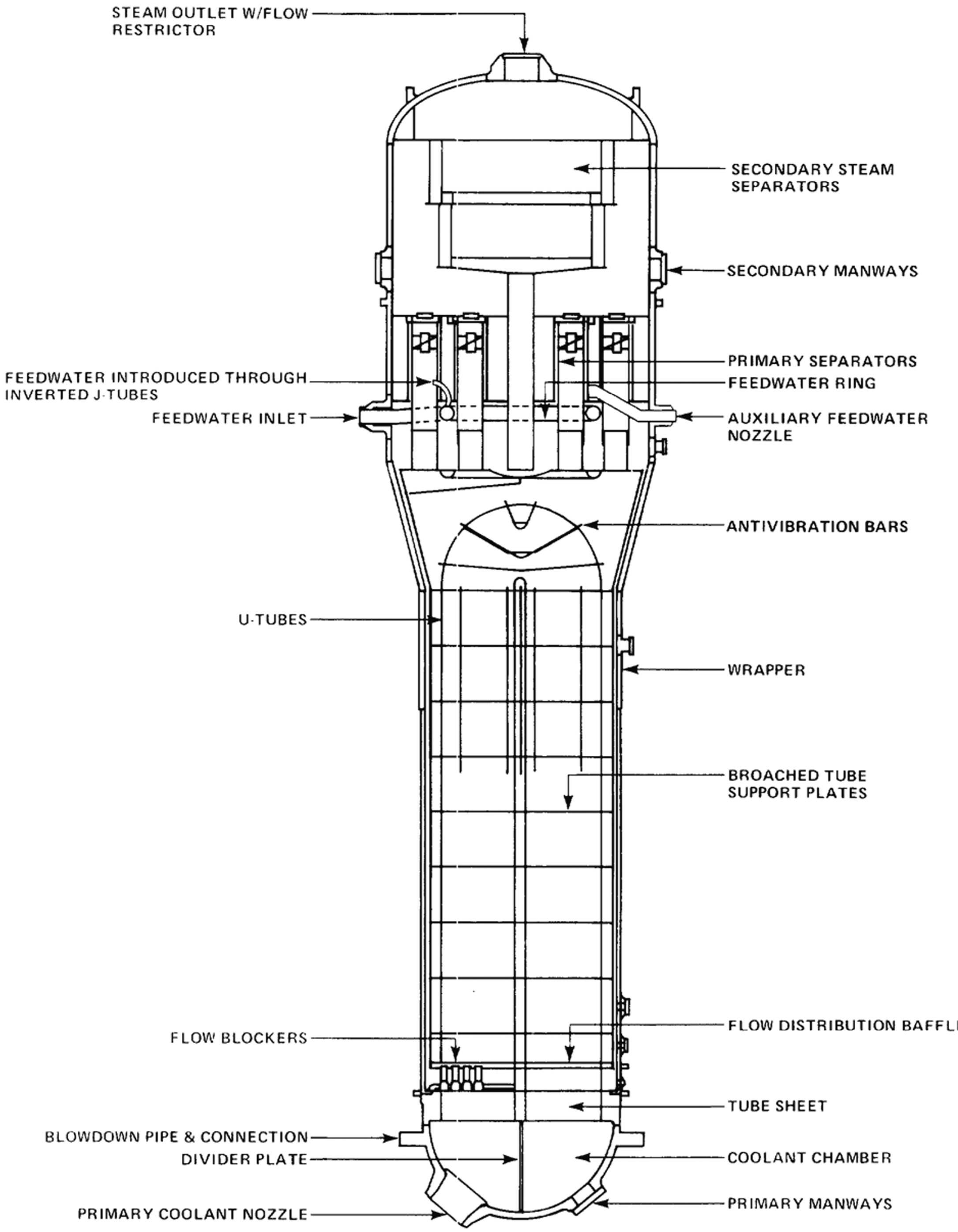
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTIC

FIGURE 5.4.1–2



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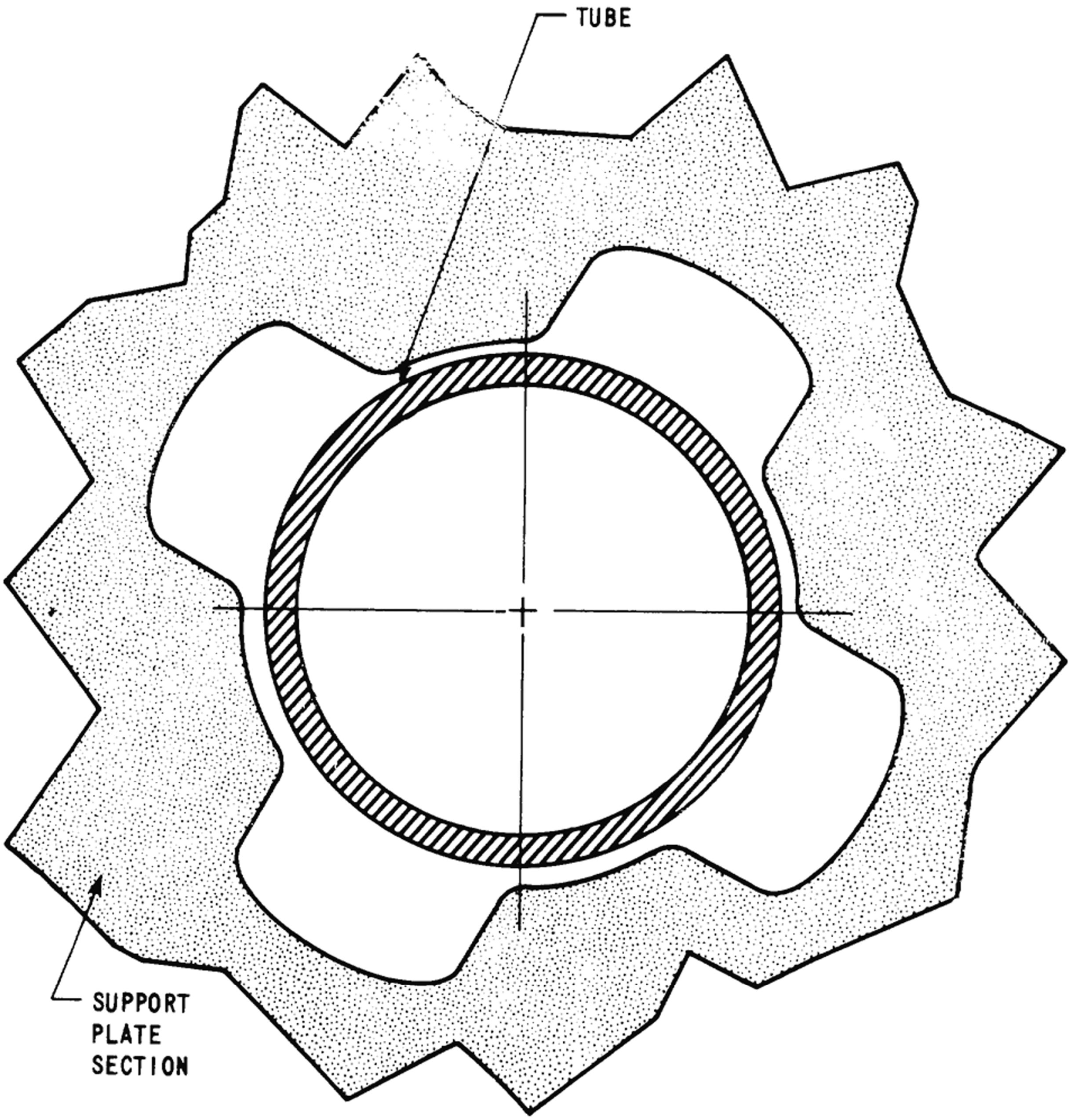
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	MODEL F STEAM GENERATOR FIGURE 5.4.2–1



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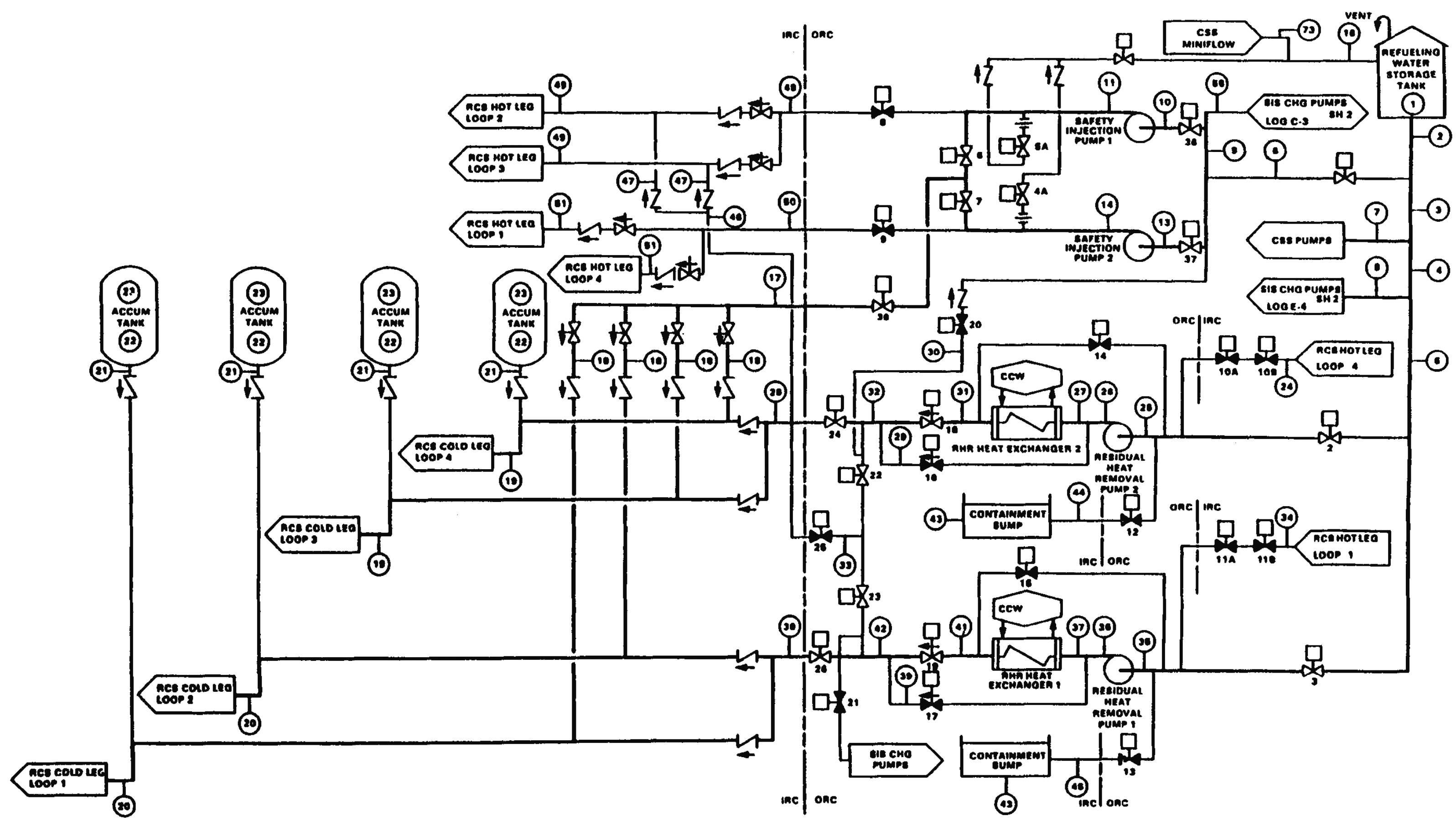
UNIT 1 AND UNIT 2

QUATREFOIL TUBE SUPPORT PLATES

FIGURE 5.4.2–2



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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

RHRS PROCESS FLOW DIAGRAM

FIGURE 5.4.7–1 (SHEET 1 OF 5)

MODES OF OPERATION

MODE A INITIATION OF RHR OPERATION

When the reactor coolant temperature and pressure are reduced to 350°F and 365 psig, approximately 2 to 4 hours after reactor shutdown, the second phase of plant cooldown starts with the RHRS being placed in operation. Before starting the pumps, the inlet isolation valves are opened, the heat exchanger flow control valves are set at minimum flow, and the outlet valves are verified open. The automatic miniflow valves are open and remain so until the pump flow exceeds the closed setpoint (1944 gpm at 350°F, 1841 gpm at 100°F), at which time they trip closed. Should the pump flow drop below the open setpoint (824 gpm at 350°F, 780 gpm at 100°F), the miniflow valves open automatically.

Startup of the RHRS includes a warmup period, during which reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS. The rate of heat removal from the reactor coolant is controlled manually by regulating the reactor coolant flow through the residual heat exchangers. The total flow is regulated automatically by control valves in the heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited to 100°F/h based on equipment stress limits and a 120°F maximum component cooling water temperature.

MODE B END CONDITIONS OF A NORMAL COOLDOWN

This situation characterizes most of the RHRS operation. As the reactor coolant temperature decreases, the flow through the residual heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

NOTE

For the safeguards functions performed by the RHRS, refer to section 6.3.

VALVE ALIGNMENT CHART

Valve No.	Flow Diagram Valve No.	Operational Mode[a] A	Operational Mode[a] B
HV-8812B	2	C	C
HV-8812A	3	C	C
HV-8702A/B	10	O	O
HV-8701A/B	11	O	O
HV-8811B	12	C	C
HV-8811A	13	C	C
FV-0611	14	C	C
FV-0610	15	C	C
FV-0619	16	P	C
FV-0618	17	P	C
HV-0607	18	P	P
HV-0606	19	P	P
HV-8804B	20	C	C
HV-8804A	21	C	C
HV-8716B	22	C	C
HV-8716A	23	C	C
HV-8809B	24	O	O
HV-8809A	26	O	O

a. O = Open.
C = Closed.
P = Partially open.

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SOUTHERN COMPANY Energy to Serve Your World® VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RHRS PROCESS FLOW DIAGRAM
	FIGURE 5.4.7–1 (SHEET 3 OF 5)

MODE A INITIATION OF RHR OPERATION

Location	Fluid	Pressure (psig)	Temperature (°F)	Flow (gal/min)[a]	Flow (lb/h × 10^6)
24	RC	400	350	3000	1.340
25	RC	417	350	3000	1.340
26	RC	565	350	3000	1.340
27	RC	541	350	1657	0.740
31	RC	536	140	1657	0.740
29	RC	492	350	1343	0.600
32	RC	492	234	3000	1.340
28	RC	432	234	3000	1.340
19 Loop 4	RC	400	234	1552	0.680
19 Loop 3	RC	400	234	1478	0.660
34	RC	400	350	3000	1.340
35	RC	418	350	3000	1.340
36	RC	567	350	3000	1.340
37	RC	543	350	1657	0.740
41	RC	536	140	1657	0.740
39	RC	492	350	1343	0.600
42	RC	492	234	3000	1.340
38	RC	435	234	3000	1.340
20 Loop 1	RC	400	234	1487	0.664
20 Loop 2	RC	400	234	1513	0.676

a. At reference conditions 350°F and 400 psig.

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UNIT 1 AND UNIT 2

RHRS PROCESS FLOW DIAGRAM

FIGURE 5.4.7–1 (SHEET 4 OF 5)

MODE B END CONDITIONS OF A NORMAL COOLDOWN

Location	Fluid	Pressure (psig)	Temperature (°F)	Flow (gal/min)[a]	Flow (lb/h x 10^6)
24	RC	0	140	3000	1.480
25	RC	19	140	3000	1.480
26	RC	181	140	3000	1.480
27	RC	152	140	3000	1.480
31	RC	134	115	3000	1.480
29	RC	105	115	0	0
32	RC	105	115	3000	1.480
28	RC	38	115	3000	1.480
19 Loop 4	RC	0	115	1570	0.750
19 Loop 3	RC	0	115	1480	0.730
34	RC	0	140	3000	1.480
35	RC	19	140	3000	1.480
36	RC	184	140	3000	1.480
37	RC	154	140	3000	1.480
41	RC	136	115	3000	1.480
39	RC	105	115	0	0
42	RC	105	115	3000	1.480
38	RC	42	115	3000	1.480
20 Loop 1	RC	0	115	1486	0.733
20 Loop 2	RC	0	115	1514	0.747

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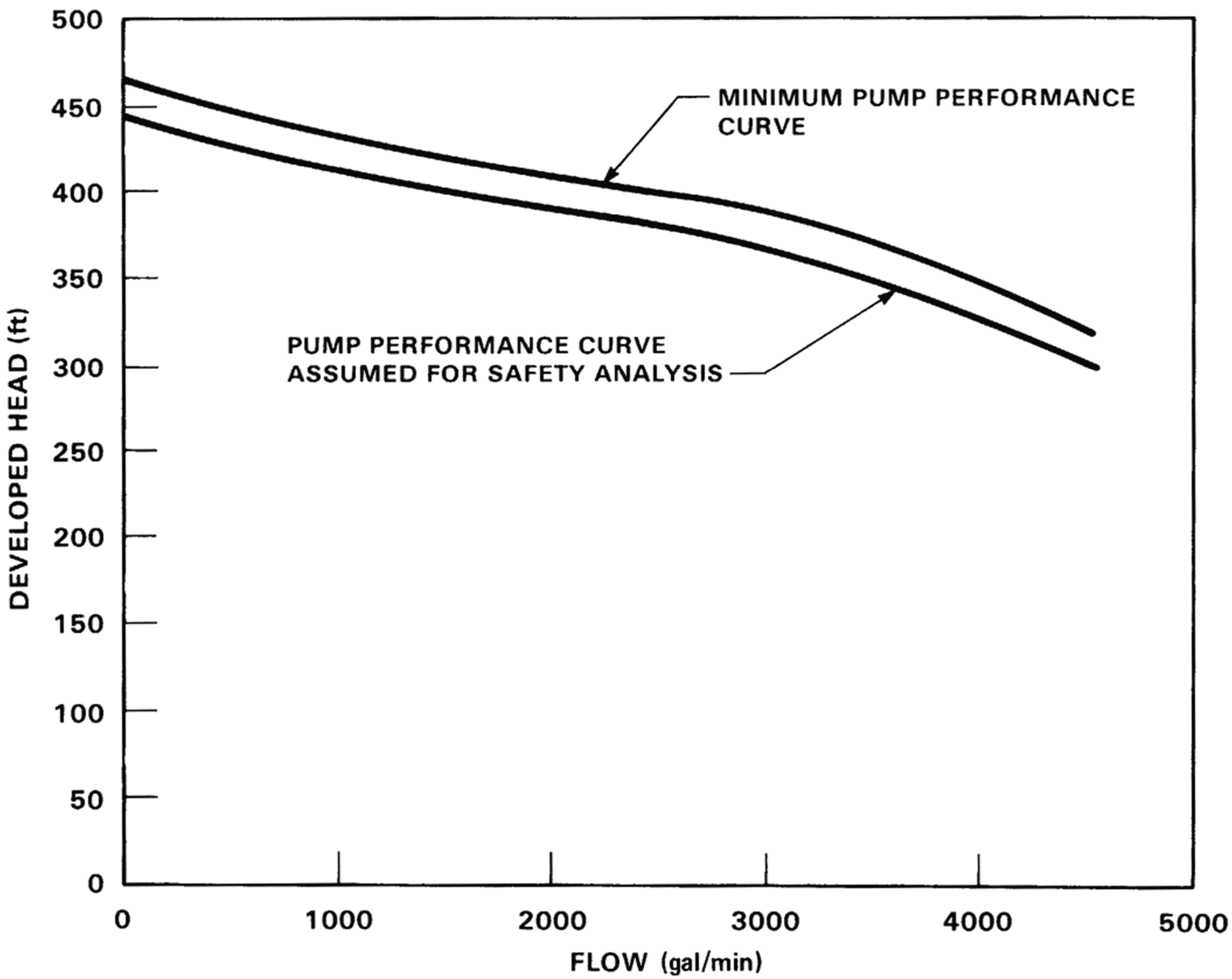

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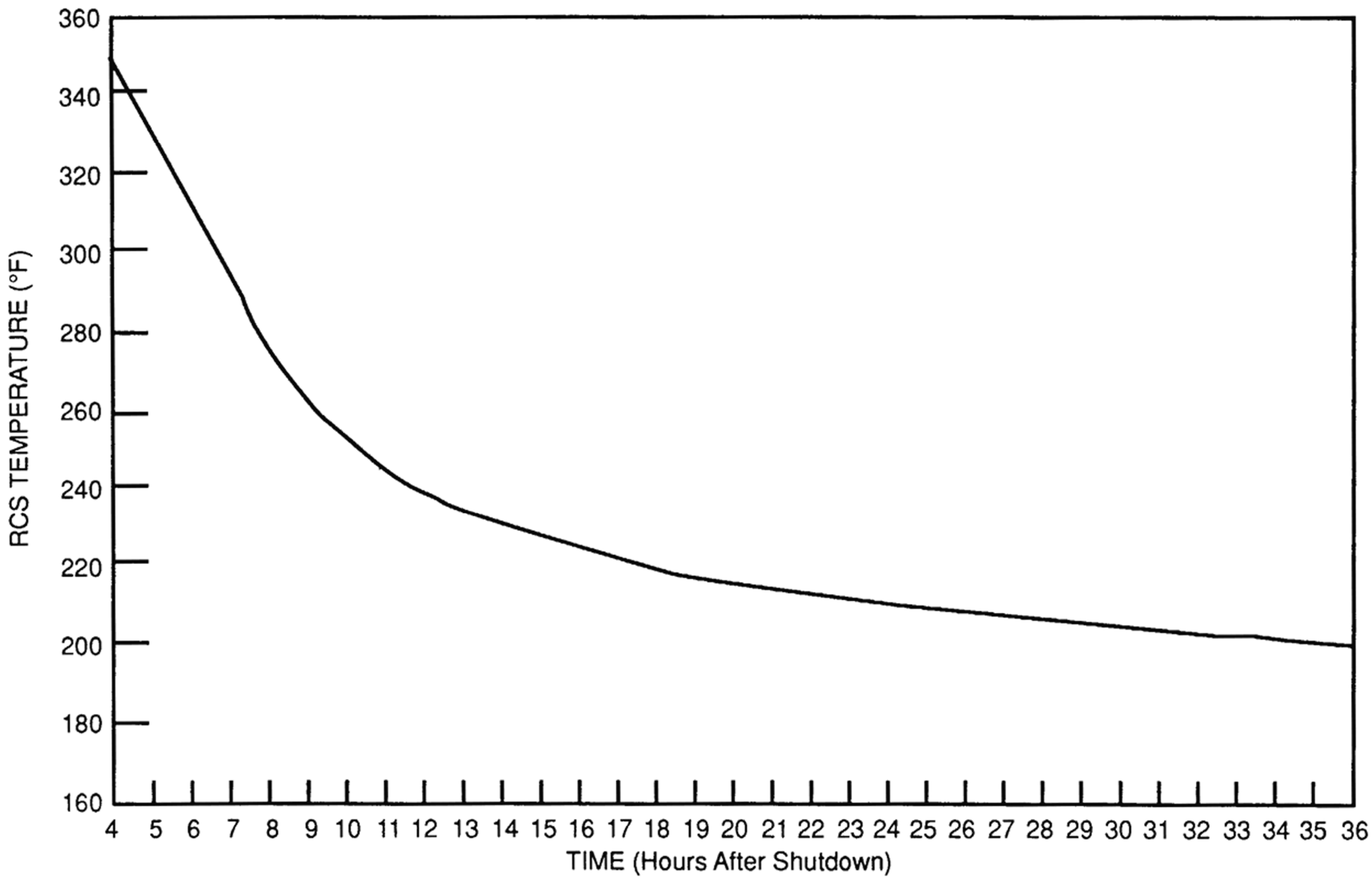
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RHRS PUMP PERFORMANCE CURVE
	FIGURE 5.4.7–2



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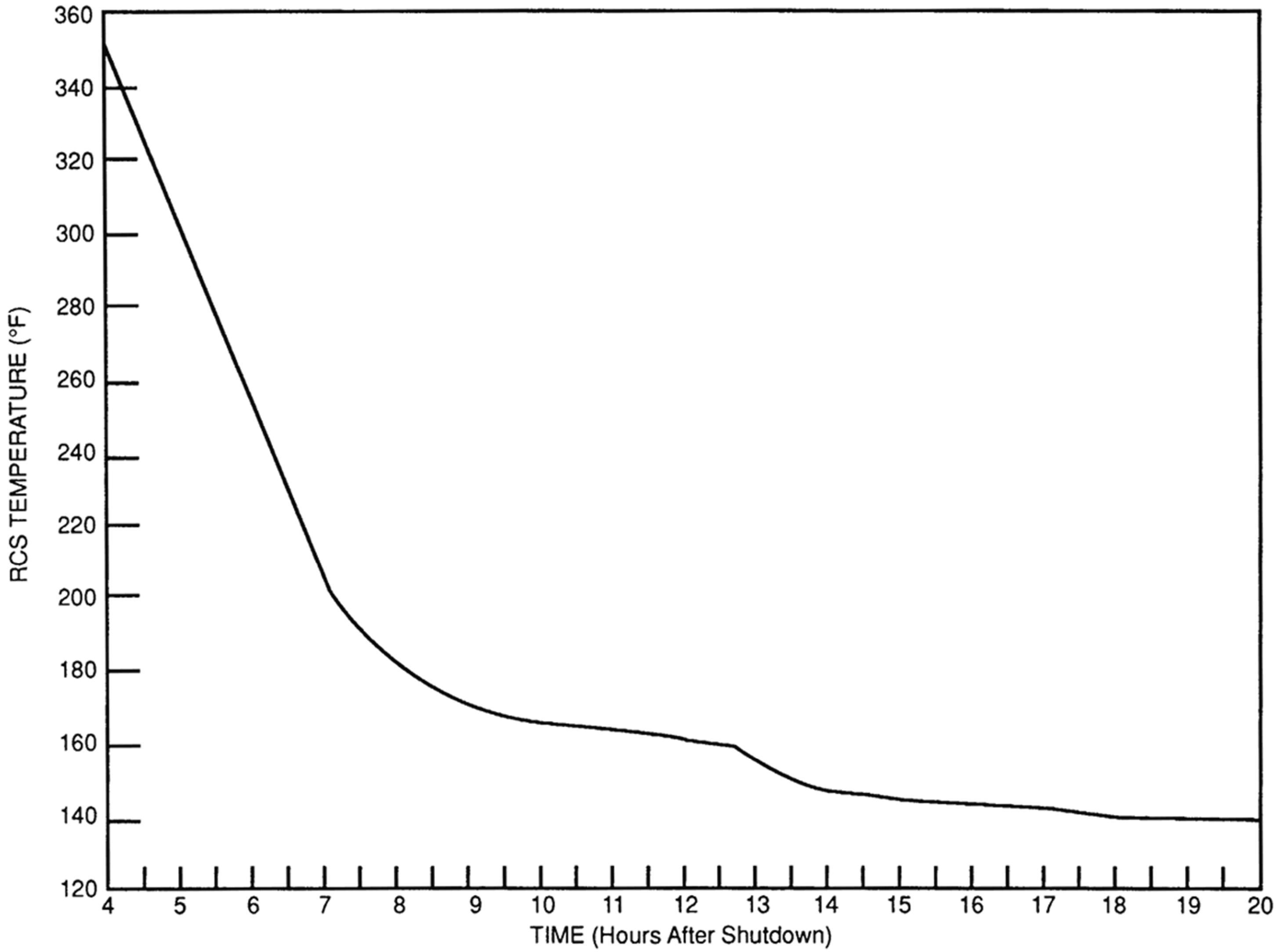
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE** **ELECTRIC GENERATING PLANT** **UNIT 1 AND UNIT 2**	**RHRS COOLDOWN CURVE** **(ONE TRAIN)** FIGURE 5.4.7–3 (SHEET 1 OF 2)



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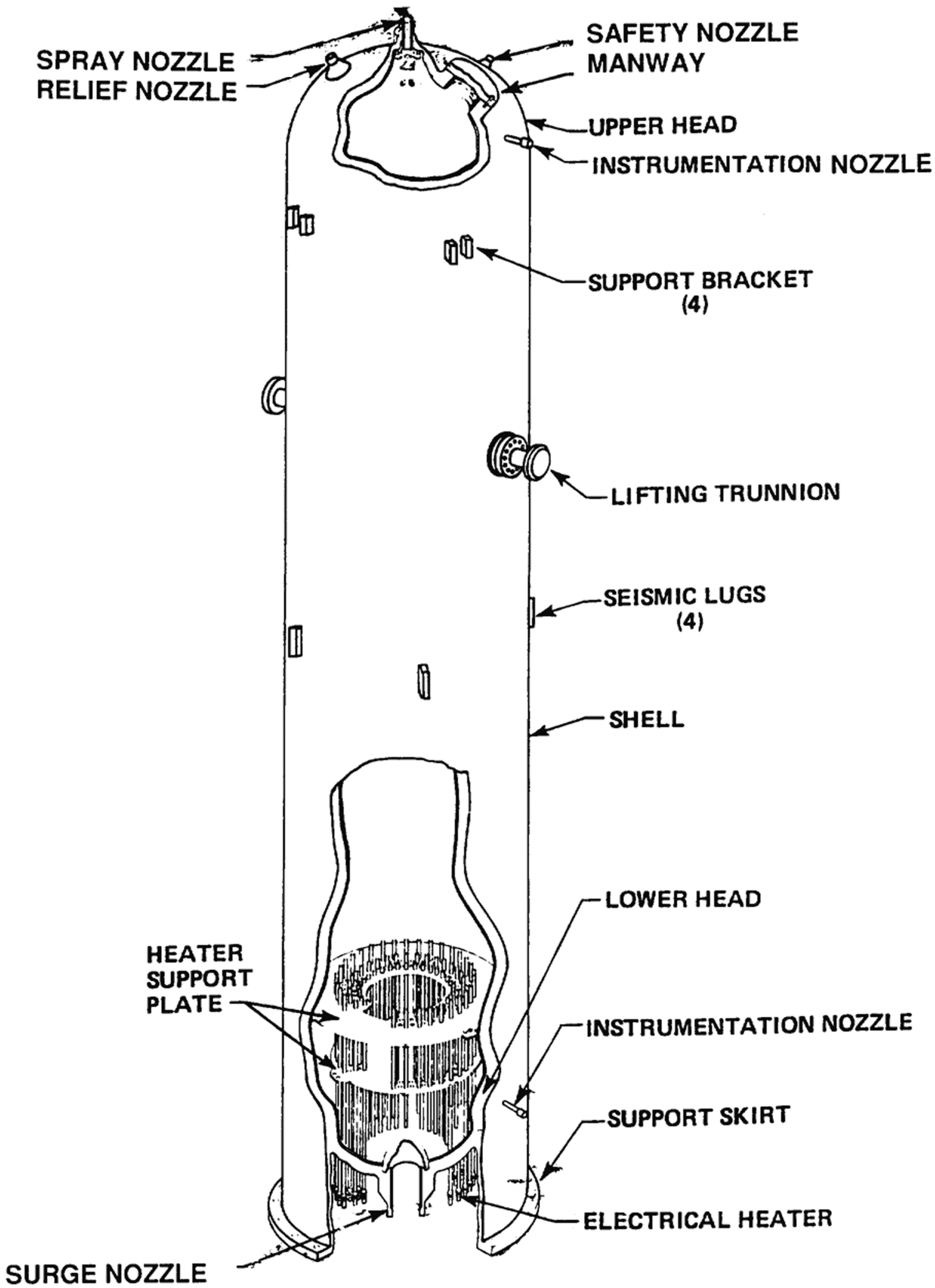
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RHRS COOLDOWN CURVE (TWO TRAIN)
		FIGURE 5.4.7–3 (SHEET 2 OF 2)



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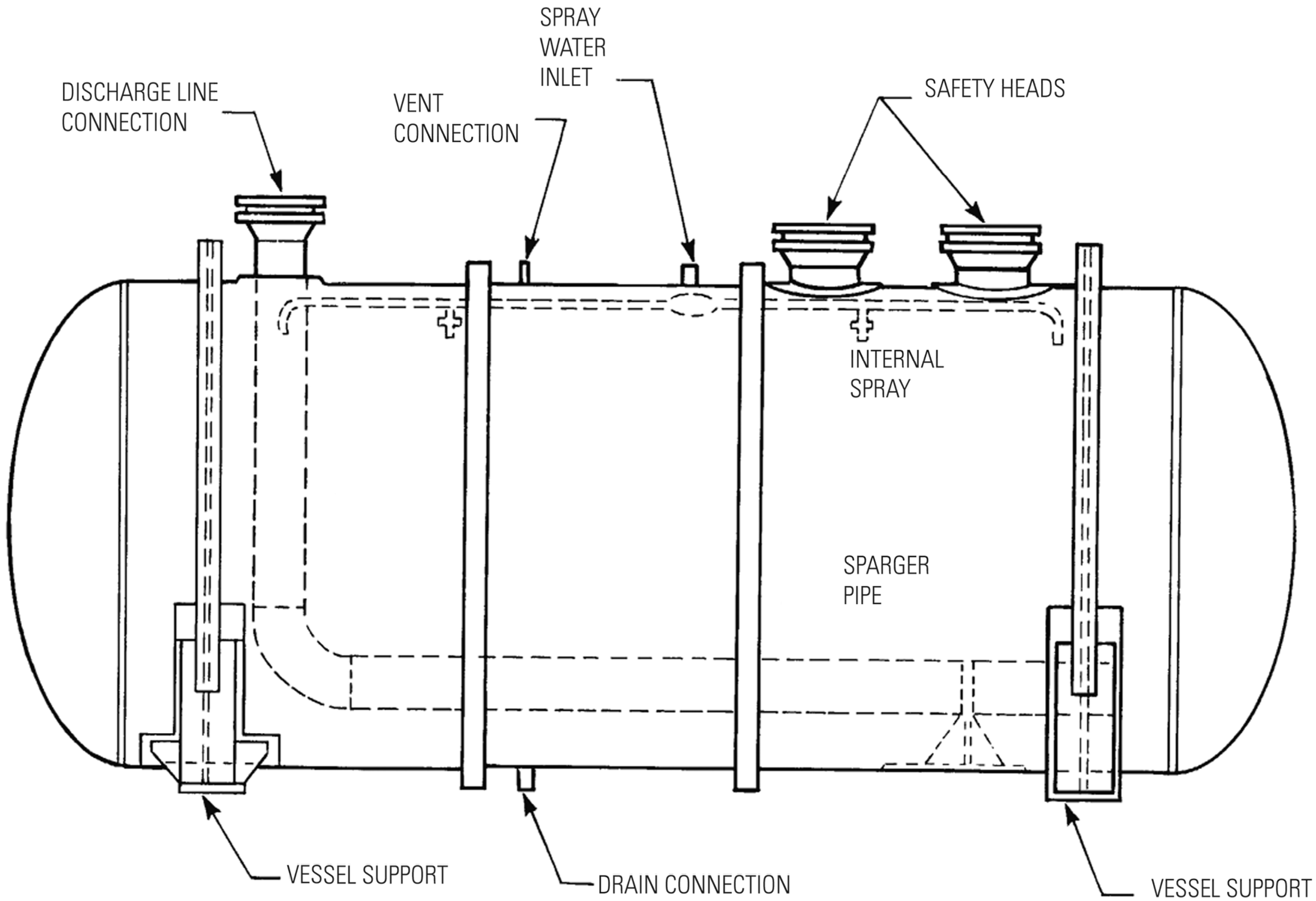
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURIZER FIGURE 5.4.10–1



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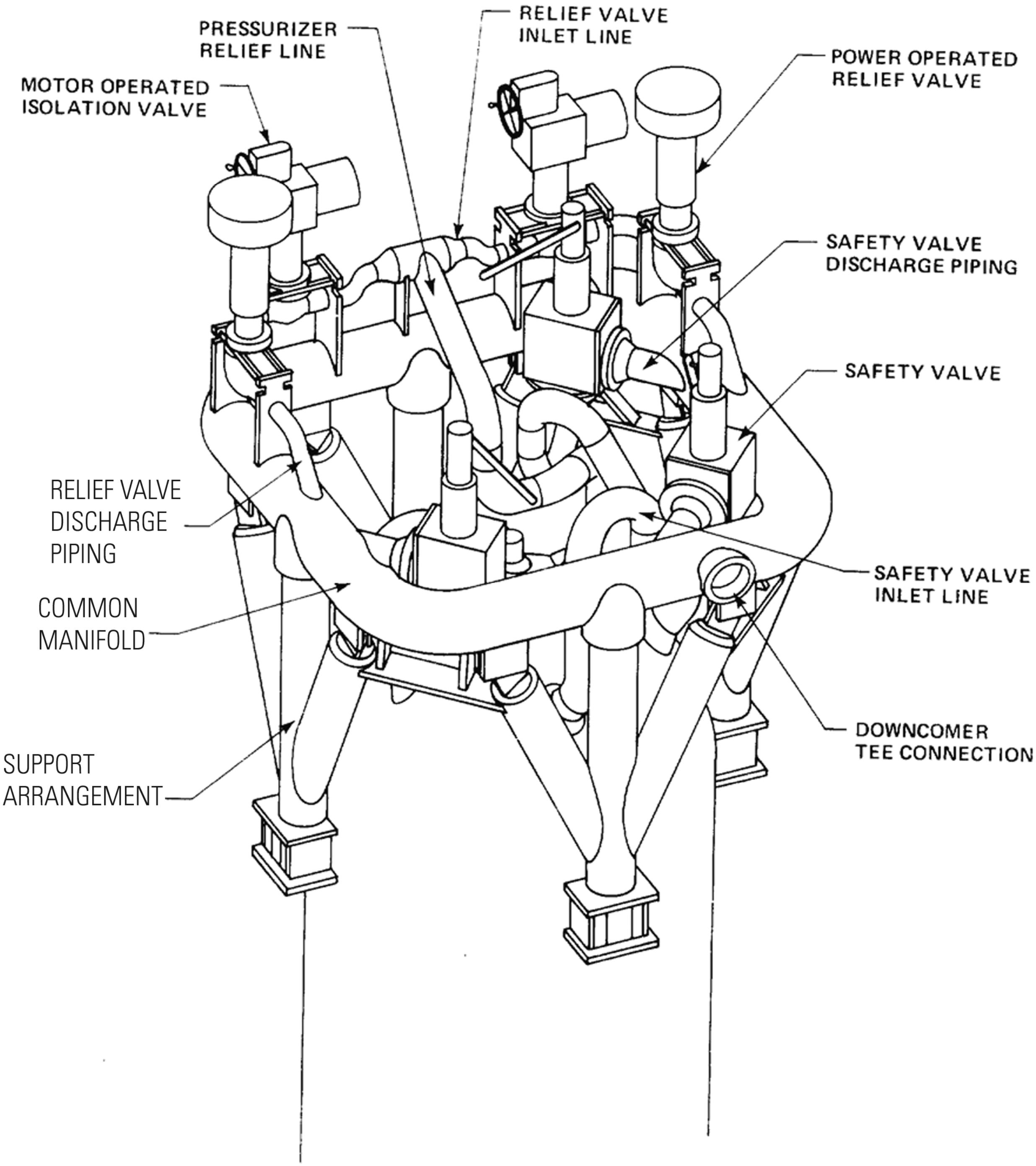
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	PRESSURIZER RELIEF TANK FIGURE 5.4.11–1



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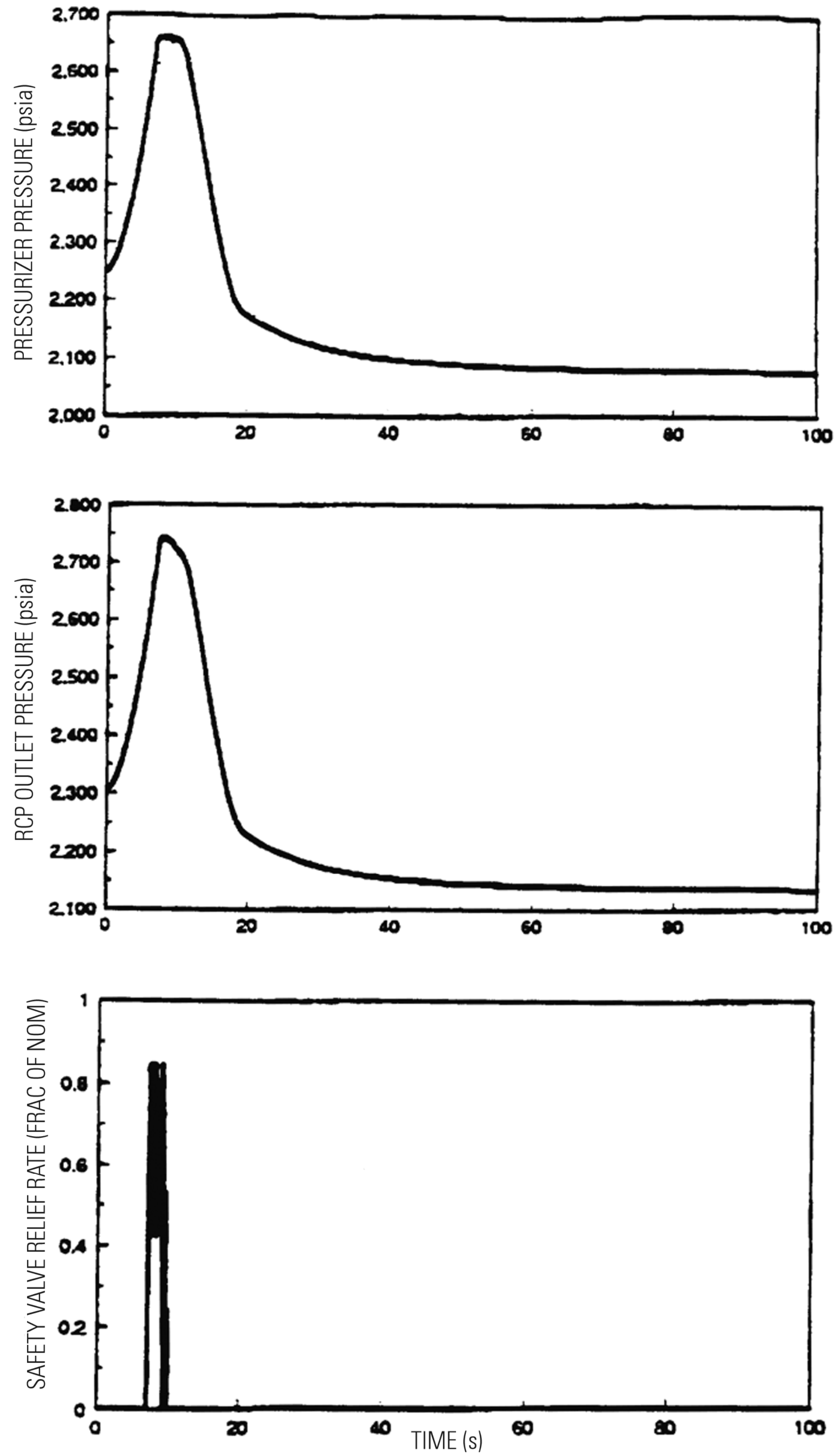
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORT ARRANGEMENT
	FIGURE 5.4.11–2



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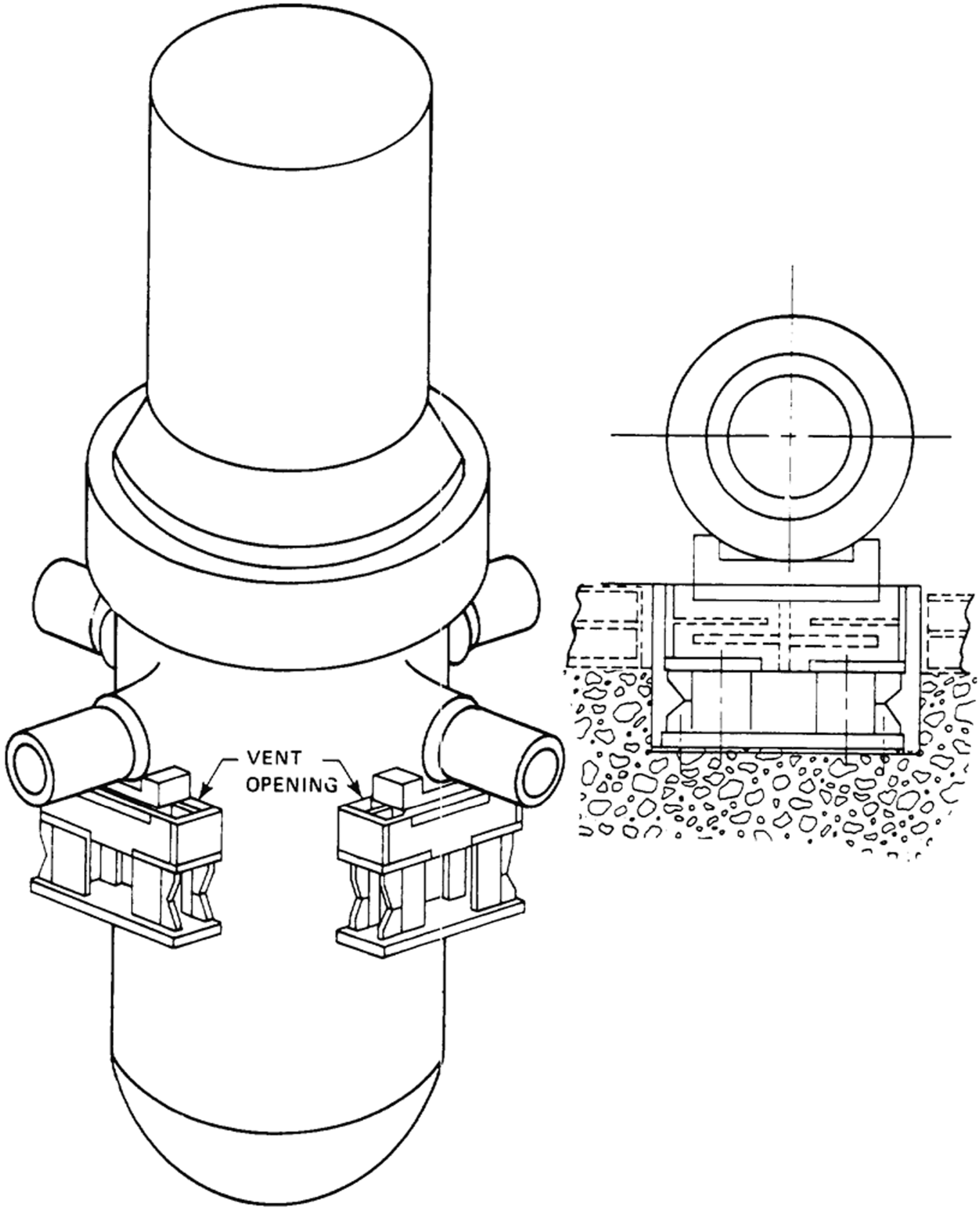
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE** **ELECTRIC GENERATING PLANT** **UNIT 1 AND UNIT 2**	PRESSURIZER SAFETY VALVES RELIEF RATE FIGURE 5.4.13–1



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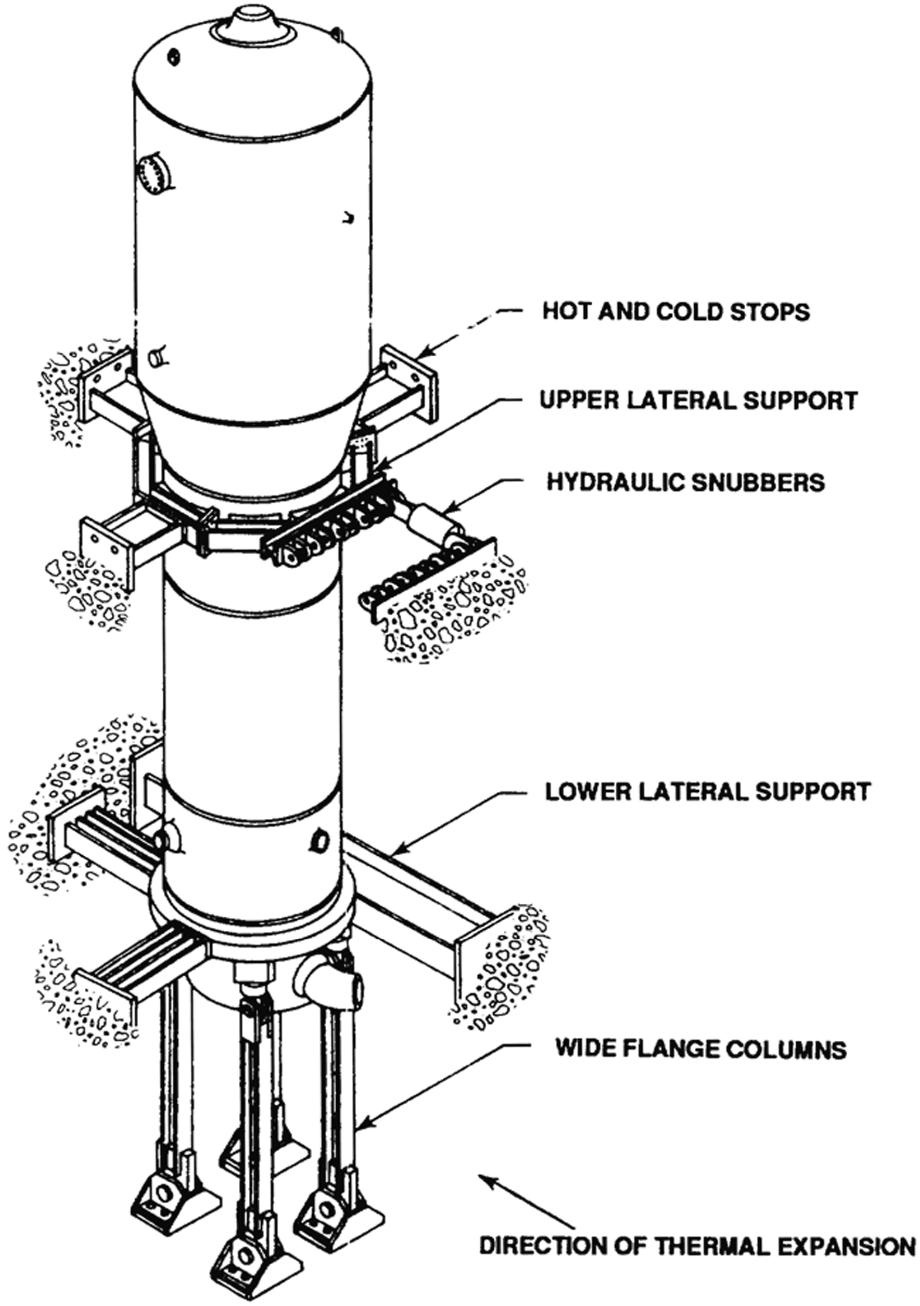
UNIT 1 AND UNIT 2

REACTOR VESSEL SUPPORTS

FIGURE 5.4.14–1



*SYMMETRICALLY INSTALLED WITH RESPECT TO THE CENTER LINE.

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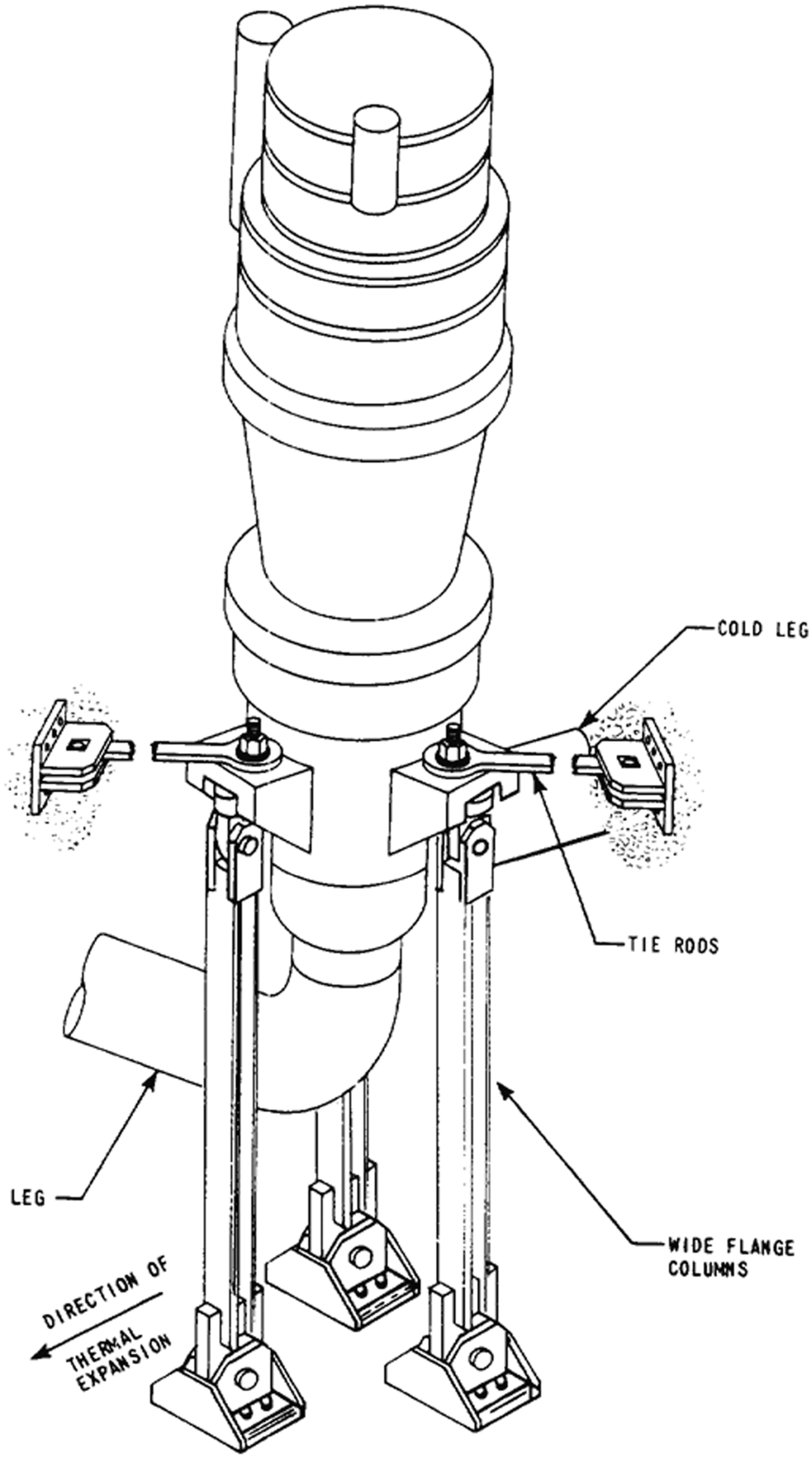
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	STEAM GENERATOR SUPPORTS FIGURE 5.4.14–2



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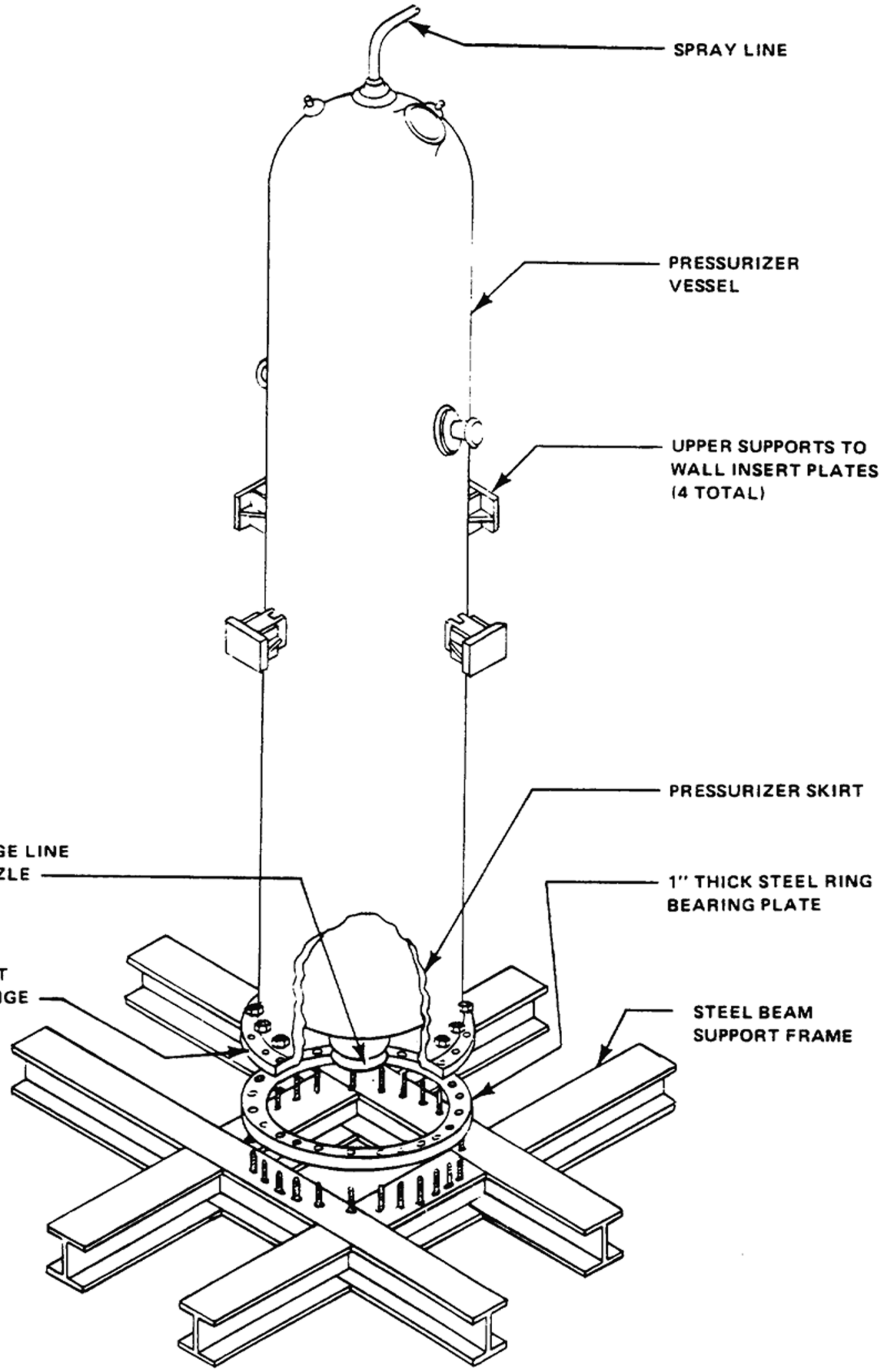
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REACTOR COOLANT PUMPS SUPPORTS

FIGURE 5.4.14–3



PRESSURIZER

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SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURIZER SUPPORTS FIGURE 5.4.14–4

6.0 ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) systems protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system (RCS), particularly as the result of a loss-of-coolant accident (LOCA). The safety features function to localize, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines (e.g., 10 CFR 100). The following systems are defined as ESF systems:

A. Containment Building (subsection 6.2.1)

The containment building is a steel-lined, reinforced, prestressed concrete cylinder with a hemispherical dome and flat circular basemat. The building houses the nuclear steam supply system (NSSS) and is designed to minimize radioactive fission product release from the NSSS to the environs subsequent to postulated design basis accidents (DBAs).

B. Containment Spray System-Iodine Removal System (subsections 6.2.2 and 6.5.2)

The containment spray system provides borated water spray for post-accident containment heat removal and pressure reduction. The iodine removal system provides chemical additives to the containment spray and/or recirculation system for post-accident iodine removal from the containment atmosphere.

C. Containment Fan Cooler System (subsection 6.2.2)

The containment fan cooler system consists of redundant fan cooler units provided for post-accident containment atmosphere heat removal and pressure reduction.

D. Containment Isolation System (subsection 6.2.4)

The containment isolation system provides for automatic containment isolation upon receipt of a containment isolation actuation signal. This system precludes the release of the containment atmosphere to the plant and the surroundings.

E. Combustible Gas Control System (subsection 6.2.5)

The combustible gas control system consists of two hydrogen recombiner subsystems and two hydrogen monitoring subsystems. A post-LOCA cavity purge system is designed to prevent hydrogen pocketing in the reactor cavity. A manually initiated, non-ESF hydrogen purge subsystem is provided as a backup to the ESF subsystems. The system functions to maintain post-LOCA hydrogen concentrations below the combustible limit.

F. The Emergency Core Cooling System (ECCS) (section 6.3)

The ECCS described in section 6.3 injects borated water into the RCS. This provides post-accident cooling of the core to limit core damage and fission product release and ensures adequate shutdown margin. The system also provides continuous long-term, post-accident cooling of the core by recirculation of borated water from the containment sump through the residual heat exchanger and back to the reactor core.

G. Habitability Systems (section 6.4 and subsection 9.4.1)

The control room heating, ventilation, and air-conditioning (HVAC) system is provided to protect control room personnel from post-accident airborne radioactivity.

H. ESF Filter Systems (subsection 6.5.1)

The fuel handling building post-accident exhaust system and the piping penetration filter exhaust system control fission product release resulting from postulated accidents. The control room HVAC system reduces radiation exposures to operating personnel in the control room.

I. Auxiliary Feedwater System (subsection 10.4.9)

The auxiliary feedwater system is provided to automatically supply feedwater to the steam generators for heat removal from the RCS during emergency conditions.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

This section provides a discussion of the materials used in the fabrication of ESF components and of the material interactions that could potentially impair the operation of the ESF.

6.1.1 METALLIC MATERIALS

6.1.1.1 Materials Selection and Fabrication

Information on the selection and fabrication of the materials in the ESF of the plant, such as the ECCS, the containment heat removal systems, the combustible gas control system, and the containment spray systems is provided below. Materials for use in ESF are selected for their compatibility with the reactor coolant system (RCS) and containment spray solutions as described in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Articles NC-2160 and NC-3120.

6.1.1.1.1 Specifications for Principal Pressure-Retaining Materials

All pressure-retaining materials in ESF system components comply with the corresponding material specification permitted by ASME Section III, Division 1. The material specifications for pressure-retaining materials in each component of an ESF system meet the requirements of Article NC-2000 of ASME Section III, Class 2, for Quality Group B and Article ND-2000 of ASME Section III, Class 3, for Quality Group C components. Materials produced under American Society of Testing Materials (ASTM) designation are acceptable as complying with the corresponding ASME specification, provided the ASME specification is designated as being identical with the ASTM specification for the grade, class, or type produced and that the material is confirmed as complying with the ASME specification by a certified material test report or certification from the material manufacturer (Subarticle NA/NCA-1220). Containment penetration materials meet the requirements of Articles NC-2000 and NE-2000 of ASME Section III, Division 1. The quality groups assigned to each component are given in table

3.2.2-1. Principal pressure-retaining materials are indicated in table 6.1.1-1. Material specifications for equipment within the NSSS scope are provided in table 5.2.3-1.

6.1.1.1.2 Engineered Safety Features Construction Materials

The welding materials used for joining the ferritic base materials of the ESF conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14.

The welding materials used for joining the austenitic stainless steel base materials conform to ASME Material Specifications SFA 5.4 and 5.9. These materials are qualified to the requirements of the ASME Code, Section III and Section IX, and are used in procedures which have been qualified to these same rules. The methods utilized to control delta ferrite content in austenitic stainless steel weldments are discussed in section 1.9 and subsection 5.2.3.

Components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion-resistant material. The integrity of the safety-related components of the ESF is maintained during all stages of component manufacture. Austenitic stainless steel is utilized in the final heat-treated condition as required by the respective ASME Code, Section II, material specification for the particular type or grade of alloy. Furthermore, austenitic stainless steel materials used in the ESF components are handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. These controls are stipulated in specifications which are discussed in subsection 5.2.3. Additional information concerning austenitic stainless steel, including the avoidance of sensitization and the prevention of intergranular attack, can be found in subsection 5.2.3. No cold-worked austenitic stainless steels having yield strengths greater than 90,000 psi are used for components of the ESF.

Materials utilized in ESF components within the containment that would be exposed to core cooling water and containment sprays in the event of a LOCA are listed in table 6.1.1-2. These components are manufactured primarily of stainless steel or other corrosion-resistant material. Protective coatings are applied on carbon steel equipment located inside the containment. (See subsection 6.1.2.)

To limit the generation of hydrogen within the containment, restrictions are placed on the use of aluminum, zinc, and mercury in the containment:

A. Aluminum is severely attacked by the alkaline containment spray solution, which results in the generation of gaseous hydrogen and the possible loss of structural integrity. The amount of aluminum present inside the containment is restricted to an essential minimum.

B. Boric acid spray reacts with zinc, oxidizing it and liberating hydrogen gas. The use of zinc in the containment is minimized to reduce generation of hydrogen.

   Table 6.2.5-6 contains a list of the amounts of aluminum and zinc which are expected to be present in the containment and which could potentially be exposed to a corrosive environment. These materials are listed by the system or component in which they are used, and an estimate of their expected corrosion rate is given. The use of mercury and mercuric compounds is prohibited inside the containment. Temporary use of fluorescent and high-pressure sodium lamps

is permitted during refueling outages/plant shutdowns during Modes 5 and 6 only. Usage during these times is administratively controlled.

6.1.1.1.3 Integrity of Safety-Related Components

The integrity of the materials of construction for ESF equipment when exposed to post-DBA conditions have been evaluated. Post-DBA conditions were conservatively represented by test conditions. The test program[(1)] considered spray and core cooling solutions of the design chemical compositions, as well as the design chemical compositions contaminated with corrosion and deterioration products which may be transferred to the solution during recirculation. The effects of sodium (free caustic), chlorine (chloride), and fluorine (fluoride) on austenitic stainless steels were considered. Based on the results of this investigation, as well as testing by Oak Ridge National Laboratory and others, the behavior of austenitic stainless steels in the post-DBA environment is acceptable. No cracking is anticipated on any equipment even in the presence of postulated levels of contaminants, provided the core cooling and spray solution Ph is maintained at an adequate level. The inhibitive properties of alkalinity (hydroxyl ion) against chloride cracking and the inhibitive characteristic of boric acid on fluoride cracking have been demonstrated.

The selection, procurement, testing, storage, and installation of all nonmetallic thermal insulation ensures that the leachable concentrations of chloride, fluoride, sodium, and silicate are in conformance with Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel.

Conformance with Regulatory Guide 1.36 is summarized in section 1.9.

Information is provided in section 1.9 concerning the degree of conformance with the following Regulatory Guides:

A. 1.31, Control of Ferrite Content in Stainless Steel Weld Metal.

B. 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel.

C. 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

D. 1.44, Control of the Use of Sensitized Stainless Steel.

6.1.1.2 Composition, Compatibility, and Stability of Containment and Core Spray Coolants

The information given below is provided on the composition, compatibility, and stability of the core cooling water and the containment sprays of the ESF.

Supply for the containment sprays and ECCS is drawn from the refueling water storage tank. As described in sections 3.8 and 6.3, the refueling water storage tank is a stainless steel-lined concrete tank not subject to significant corrosive attack by the tank's contents. Trisodium phosphate for recirculation fluid pH adjustment is stored in baskets located in the containment.

The accumulator tanks, which store boric acid solution (1900-2600 ppm) for the accumulator portion of the safety injection system, are made of carbon steel and clad with stainless steel to ensure corrosion resistance.

The boron injection tank (Unit 1 only) is stainless steel. Because of the corrosion resistance of this material, significant corrosive attack on the vessel is not expected. The boron injection tank contains a boron concentration, as boric acid, of 0-2600 ppm.

6.1.1.3 Reference

1. Picone, L. F., and Whyte, D. D., "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss-of-Coolant Environment," WCAP-7798-L (Proprietary), November 1971, and WCAP-7803 (Nonproprietary), December 1971.

6.1.2 ORGANIC MATERIALS

6.1.2.1 Protective Coatings

Certain coatings, which are in common industrial use, may deteriorate in the post-accident environment and may contribute substantial quantities of foreign solids and residue to the containment sump. Consequently, protective coatings used inside the containment, excluding components limited by size and/or exposed surface area, are demonstrated to withstand the design basis accident (DBA) conditions (subsection 3.11.B.1) and meet the intent of American National Standards Institute (ANSI) N101.2 (l972), Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities, as well as the recommendations of Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants. Information regarding conformance with Regulatory Guide 1.54 is provided in table 6.1.2-1 and further conformance information for nuclear steam supply system (NSSS) equipment has been submitted to the Nuclear Regulatory Commission (NRC) for review via reference 1 and accepted via reference 2.

A. Regulatory Guide 1.54 is imposed for items located within the containment building as follows:

   1. For shop priming of liner plate, structural steel, and fabricated shapes.
   2. For shop priming of fabricated pipes, tanks, heating, ventilation, and air-conditioning (HVAC) ducts, and equipment.
   3. For field finish painting of steel where called for in drawings and specifications.
   4. For surfacing of concrete where indicated in drawings and specifications.

B. Regulatory Guide 1.54 is implemented by requirements as follows:

   1. Use of specific coatings systems which are prequalified to ANSI N101.2.
   2. Surface preparation standards.
   3. Surface profile requirements.
   4. Application of the coating systems in accordance with instructions approved by the paint manufacturer.
   5. Inspections and nondestructive examinations.

6. Identification of all nonconformances. Coatings which do not conform with the VEGP position on Regulatory Guide 1.54 are limited in use and are evaluated on a case basis relative to impact on plant safety.
7. Certifications of compliance and/or documentation procedures to satisfy project requirements.
8. The vendor's procedures are subject to review prior to application, and the vendor's implementation of the specification requirements is monitored.
9. An inventory of unqualified coatings is maintained to ensure appropriate control of coatings inside containment.

C. Regulatory Guide 1.54 is not imposed for the following:

1. Surfaces to be insulated.
2. Surfaces "contained" within a cabinet or enclosure; for example, the interior surfaces of ducts.
3. Field repair to any small areas previously coated with a qualified coating system such as:
    a. Bolt heads, nuts, and miscellaneous fasteners.
    b. Damage resulting from spot, tack, or stud welding.

Field touchup and repair of large areas shall be in accordance with Regulatory Guide 1.54.

4. Small "production line" items such as small motors, handwheels, pipe supports, snubbers, electrical cabinets, control panels, loudspeakers, etc., where special painting requirements would be impracticable.
5. Stainless steel or galvanized surfaces.
6. Coating used for the banding of piping.
7. Concrete designated to receive a sealer coat only.

D. The majority of the coatings specified for use inside the containment are the inorganic type (ethyl silicate inorganic zinc). The mode of failure of inorganic zinc is powdering rather than blistering and delamination. This failure mode minimizes the accumulation of solid debris in the containment sumps. Any particles of appreciable size that do occur either settle out prior to reaching the sump screens or are trapped by the sump filter screens. The screen opening size (3/32 in.) for the Emergency Containment Cooling System (ECCS) Residual Heat Removal (RHR) Systems is smaller than the line piping, the RHR heat exchanger tubes, the branch line needle valves (used for throttling), pump running clearances, and clearances in the reactor core so particles that could potentially block the system are filtered out. (Refer to section 6.2 for a discussion of the sump design and consideration given to screen clogging.)

The screen opening size is 3/32-in. diameter by design for the Containment Spray (CS) System and the 3/32-in. diameter is smaller than the line piping, the spray nozzles, and pump running clearances, so particles that could potentially block the system are filtered out. (Refer to section 6.2 for a discussion of the sump design and consideration given to screen clogging.) After the new screens were installed, it was discovered that 124 holes (0.002% of total screen surface

area) in the plates had a hole > 3/32-in. diameter. No round holes were found to be > ¼-in. diameter in the screens and all hole areas had < ¼-in. length in the major axis through the minor axis. The CS nozzles have a 3/8-in. diameter hole; therefore, blockage in the system will not occur.

E. "N" AREAS

Areas within the containment are identified as "N" areas. Coatings used inside the containment, where required, are the prequalified coating systems. These coatings are prequalified to the intent of ANSI N101.2 and applicable portions of ANSI N5.12. Quality assurance and documentation requirements of ASME NQA-1-1994, as described in the SNC Quality Assurance Topical Report (QATR) and ANSI N101.4 (Class I) are enforced for both coating materials and applications procedures as discussed in table 6.1.2-1.

F. "D" AREAS

Areas outside the containments, but with potential contamination from radioactive sources, are identified as "D" areas. Coating materials specified for "D" areas are either the same or similar materials which are used in "N" areas, except the quality assurance program and the documentation requirements of ANSI do not apply (Class II).

G. "C" AREAS

Areas outside the containments, and not subject to potential contamination from radioactive sources are identified as "C" areas. "C" area coatings are standard commercial coatings formulated to withstand exposure to industrial environment and require minimum maintenance (Class II).

A coating schedule for items inside the containment is given in tables 6.1.2-2 and 6.1.2-3. Approximate paint film thickness and exposed surface area for major components and structures inside the containment are also provided. The painted areas of valve operators, miscellaneous parts on the reactor coolant pump drives, and instrumentation are considered insignificant. Exposed concrete in the containment is coated as indicated in table 6.1.2-2. The containment temperature profile used for the specification of coatings used inside containment is shown in figure 6.1.2-1.

Protective coatings for use on NSSS components in the reactor containment have been evaluated as to their suitability in post-DBA conditions. Tests have shown that the inorganic zinc, epoxy, and modified phenolic systems are the most desirable of the generic types evaluated for use inside containment. This evaluation[3] considers resistance to high temperature and chemical conditions anticipated during a loss-of-coolant accident, as well as high radiation resistance.

6.1.2.2 Other Organic Materials

A listing of other organic materials in the containment is included in table 6.1.2-4. The materials listed are not protective coatings applied to surfaces of nuclear facilities.

6.1.2.3 References

1. Letter NS-CE-1352, C. Eicheldinger (Westinghouse) to C. J. Heltemes, Jr. (NRC), dated February 1, 1977.

2. Letter, C. J. Heltemes, Jr. (NRC) to C. Eicheldinger (Westinghouse), dated April 27, 1977.

3. Picone, L. F., "Evaluation of Protective Coatings for use in Reactor Containment," WCAP-7198-L (Proprietary), April 1968, and WCAP-7825 (Nonproprietary), December 1971.

6.1.3 POST-ACCIDENT CHEMISTRY

Following a main steam line break or design basis loss-of-coolant accident, trisodium phosphate and boric acid solutions will be present in the containment sumps. Subsection 6.5.2 indicates the quantities of trisodium phosphate and boric acid that will be present in the containment after an accident. The pH control reduces the probability of chloride stress corrosion cracking of stainless steel and maximizes iodine retention in the sump solution.

TABLE 6.1.1-1 (SHEET 1 OF 2)

PRINCIPAL ESF PRESSURE-RETAINING MATERIALS

Component	Material
Piping/tubing	SA-53 Gr. B SA-106 Gr. B and C SA-155 Gr. 70 Class 1 and Gr. KC70 Class 1 SA-213, TP 304, 304L and 316 SA-249, TP 304L SA-312, TP 304 and 304L SA-333 Gr. 1 and 6 SA-335 Gr. P11 and P22 SA-376, TP 304 and 316 SB-111 Gr. CDA 706 SB-466 Gr. CDA 706
Fittings/flanges	SA-105 N SA-181 Gr. I and II SA-182, TP F304, F304L, F316, F316L SA-234 Gr. WPB, WPC, WPBW, and WPCW SA-403 WP 304, 304L, 304W, and 304LW SA-420 Gr. WPL6 SA-479, TP 304, 304L and 316
Plate	SA-240, TP 304, 304L and 316L SA-283 Gr. C SA-285 Gr. A and C SA-515 Gr. 70 SA-516 Gr. 70 SA-537 Class 1 SB-171 Gr. CDA 706
Bolting/nuts/studs	SA-193 Gr. B6, B7, B8, and B8M SA-194, Gr 2H, 4, 6, 7, 8H, 8M, and B8 SA-307 Gr. B SA-320 Gr. L7 SA-453 Gr. 660A and 660B SA-564 Gr. 630

TABLE 6.1.1-1 (SHEET 2 OF 2)

Component	Material
Castings	SA-216 Gr. WCB and WCC SA-217 Gr. WC9 SA-351 Gr. CF8M and CF3M SA-487 Gr. CA6NM SB-61 SB-62 Gr. CDA 836 SB-148 Gr. CA 952 ASTM-A276 TP 410
Forgings	SA-105 SA-182, TP F304, F304L, F316, and F316L; Gr. F11 and F22 SA-240 TP 304 and 316 SA-350 Gr. LF1 and LF2 SA-479, TP 304, 304L and 316
Bars	SA-479, TP 304, 316 and 410 Gr 316L and F316 SA-564 Gr. 630
Weld rod	SFA 5.1, E 6010 and E 7018 SFA 5.4, E 308-16, E 308L-16 and E 309 SFA 5.9, ER 308, ER 308L, and ER 309 SFA 5.17, EM 12K SFA 5.18, E 70S-2, E 70S-3, E 70S-4, E70S-6, and E70S-1B SFA 5.20, E 70T-1 and 70T-5

TABLE 6.1.1-2 (SHEET 1 OF 2)

PRINCIPAL ESF MATERIALS EXPOSED TO REACTOR COOLANT
OR CONTAINMENT SPRAY

Component	Material
Piping/tubing	SA-106 Gr. B and C SA-155 Gr. KC70 Class 1 and 70 Class 1 SA-213, TP 304, 304L, and 316 SA-249, TP 304L SA-312, TP 304 and 304L SA-333 Gr. 1 and 6 SA-376, TP 304 and 316 SB-111 Gr. CDA 706 SB-466 Gr. CDA 706
Fittings/flanges	SA-105 N SA-181 Gr. I and II SA-182, TP F304, F304L, F316, and F316L SA-234 Gr. WPB, WPBW, WPCW, and WPC SA-403, WP 304, 304L, 304W, and 304LW SA-420 Gr. WPL6 SA-479, TP 304, 304L, and 316
Plate	SA-240, TP 304, 304L, and 316L SA-285 Gr. A and C SA-515 Gr. 70 SA-516 Gr. 70 SA-537 Class 1 SB-171 Gr. CDA 706 ASTM A 515 Gr. 70
Shapes	SA-36 ASTM A-36 ASTM A-500 Gr. B (Code Case N-71-10)
Bolts/nuts/studs/pins	SA-193 Gr. B6, B7, B8 and B8M SA-194 Gr. 2H, 8H, 8M, 7, 4, 6 and B8 SA-307 Gr. B SA-320 Gr. L7 SA-325 Type 1 SA-453 Gr. 660A and 660B

TABLE 6.1.1-2 (SHEET 2 OF 2)

Component	Material
	SA-564 Gr. 630 ASTM A 193 Gr. B7 ASTM A 194 Gr. 7 ASTM A 307 ASTM A 354 ASTM A 490
Bars	SA-479, TP 304, 316, 410, 316L, and F316 SA-564 Gr. 630 ASTM A 108 Gr. 1018 CW (Code Case N-71-5)
Forgings	SA-105 SA-182, TP F304, F304L, F316 and F316L; Gr F11 and F22 SA-240, TP 304 and 316 SA-350 Gr. LF1 and LF2 SA-479, TP 304, 304L and 316 ASTM A 668 Class C (Code Case N-71-5)
Castings	SA-216 Gr. WCB and WCC SA-217 Gr. WC9 SA-351 Gr. CF8M and CF3M SA-487 Gr. CA6NM SB-61 SB-62 Gr. CDA 836 SB-148 Gr. CA 952 ASTM A 276 TP 410 ASTM A-216 Gr. WCB
Cooling coil fins	SB-152 Gr. CDA 122

TABLE 6.1.2-1 (SHEET 1 OF 3)

REGULATORY GUIDE 1.54, REVISION 0, JUNE 1973, QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.54 Position	Position on Non-NSSS Components	Position on NSSS Components
The requirements and guidelines included in ANSI N101.4-1972, Quality Assurance for Protective Coatings Applied to Nuclear Facilities, for protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of water-cooled nuclear power plants are generally acceptable and provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR 50 subject to the following:		Westinghouse has developed an alternate approach to ANSI N101.4 for satisfying Reg. Guide 1.54 for the NSSS components inside containment. Stringent requirements are specified for the painting of major components in Westinghouse Process Specifications that are imposed on vendors by procurement documents. Large equipment must have coating systems qualified to meet ANSI N101.2 and requirements are defined for surface preparation, use of undercoating, and where applicable, inspection. Other major equipment is either fabricated from stainless steel or covered by insulation. For small items of equipment, conventional industry practices are applied. Details of this approach follow:
A. ANSI N101.4-1972 should be used in conjunction with ASME NQA-1-1994, Quality Assurance Requirements for Nuclear Facility Applications, as described in the SNC QATR.	A. The requirements of VEGP are that practical and adequate corrosion protection shall be provided for all surfaces being painted. All surfaces, however, are not coated with coating materials tested and accepted under ANSI N101.2 or ANSI N5.12 criteria, nor are all coatings documented as outlined under ANSI N101.4-1972. Coating materials used for items located outside of the containment are not documented, since removal of the coating from such items does not affect the safe shutdown of the facility. Coating materials used for items located within the containment shall meet, whenever possible, the requirements of ANSI N101.2 and selected portions of ANSI N5.12 and are documented in accordance with ANSI N101.4.	NSSS equipment located in the containment building is separated into four categories to identify the applicability of this regulatory guide to various types of equipment. These categories of equipment are as follows: Category 1 - Large equipment Category 2 - Intermediate equipment Category 3 - Small equipment Category 4 - Insulated/stainless steel equipment A discussion of each equipment category follows: Category 1 - Large Equipment The Category 1 equipment consists of the following: • Reactor coolant system supports. • Reactor coolant pumps (motor and motor stand). • Accumulator tanks. • Refueling machine. Since this equipment has a large surface area and is procured from only a few vendors, it is

TABLE 6.1.2-1 (SHEET 2 OF 3)

Regulatory Guide 1.54 Position

B. Subdivision 2.7 of ANSI N101.4-1972, states that when references are made to other standards, these references shall imply the most recent or current editions of the referenced standards. The specific applicability or acceptability of referenced standards will be covered separately in other regulatory guides, where appropriate.

C. Subdivision 1.1.2 of ANSI N101.4-1972 states that quality assurance, as covered by this standard, comprises all those planned and systematic actions necessary to provide specified documentation and adequate confidence that shop or field coating work for nuclear facilities will perform satisfactorily in service. This statement should not be interpreted as implying that the end product of quality assurance actions is the production of specified documentation. The term "quality assurance" as used in ANSI N101.4-1972 should be considered to comprise all those planned and systematic actions necessary to provide adequate confidence that shop or field coating work for nuclear facilities will perform satisfactorily in service.

In this connection it is emphasized that records and documents listed in subdivisions 7.4 through 7.8, and included in the standard, are suggested forms only. Alternate documentation consistent with the requirements of Appendix B to 10 CFR 50 is also considered acceptable.

Position on Non-NSSS Components

B. VEGP follows Steel Structures Painting Council (SSPC-1963 and 1971), ANSI N101.2-1972, and sections, where applicable, of ANSI N5.12-1974

C. Conform. Forms used by the project are similar to those of ANSI N101.4. Alternate means of documentation are also used to ensure the coating work is performed satisfactorily in service.

The planned and systematic actions which are necessary to provide adequate confidence that shop or field coating work for nuclear facilities will perform satisfactorily in service are as follows.

Regulatory Guide 1.54 is implemented as follows:

1. The use of specific coatings systems which are prequalified to ANSI N101.2 are specified.
2. Surface preparation standards are used which illustrate the surface preparation procedures used in ANSI N101.2.
3. Surface profile requirements are met.
4. Application of the coating systems are made in accordance with instructions approved by the paint manufacturer.
5. Inspections and nondestructive testing are performed.
6. Nonconformances are identified and evaluated as discussed in paragraph 6.1.2.1.B.
7. Certifications of compliance and/or documentation procedures are furnished to satisfy project requirements

Position on NSSS Components

possible to implement tight controls over these items. Stringent requirements are specified for protective coatings on this equipment through the use of a painting specification in the procurement documents. This specification defines requirements for:

1. Preparation of vendor procedures.
2. Use of specific coatings systems which are qualified to ANSI N101.2.
3. Surface preparation.
4. Application of the coating systems in accordance with the paint manufacturer's instructions.
5. Inspections and nondestructive examinations.
6. Exclusive of certain materials.
7. Identification of all nonconformance.
8. Certifications of compliance.

The vendor's procedures are subject to review by engineering personnel, and the vendor's implementation of the specification requirements is monitored during quality assurance surveillance activities.

This system of controls provides assurance that the protective coatings will properly adhere to the base metal during prolonged exposure to a post-accident environment present within the containment building.

The Category 2 equipment consists of the following:

- Seismic platform and tie rods.
- Reactor internals lifting rig.
- Head lifting rig.

Since these items are procured from a large number of vendors, and individually have very small surface areas, it is not practical to enforce the complete set of stringent requirements which are applied to Category 1 items. Another painting specification is used in these procurement documents. This specification defines to the vendors the requirements for:

1. Use of specific coating systems which are qualified to ANSI N101.2.
2. Surface preparation.
3. Application of the coating systems in accordance with the paint manufacturer's instructions.

TABLE 6.1.2-1 (SHEET 3 OF 3)

Regulatory Guide 1.54 Position	Position on Non-NSSS Components	Position on NSSS Components
D. Sections 3 and 4 of ANSI N101.4-1972 delineate quality assurance requirements for coating materials and surface preparation of substrates. Coatings and cleaning materials used with stainless steel should not be compounded from or treated with chemical compounds containing elements that could contribute to corrosion, intergranular cracking, or stress corrosion cracking. Examples of such chemical compounds are those containing chlorides, fluorides, lead, zinc, copper, sulfur, or mercury where such elements are leachable or where they could be released by breakdown of the chemical compounds under expected environmental conditions (e.g., by radiation). This limitation is not intended to prohibit the use of trichlorotrifluoroethane which meets the requirements of Military Specification MIL-C-81302b for cleaning or degreasing of austenitic stainless steel, provided adequate removal is ensured prior to painting.	D. Conform.	The vendor's compliance with the requirements is also checked during quality assurance surveillance activities in the vendor's plant. These measures of degree of assurance that the protective coatings will control provide a high adhere properly to the base metal and withstand the postulated accident environment within the containment building. Category 3 - Small Equipment Category 3 equipment consists of the following: • Transmitters. • Alarm horns. • Small instruments. • Valves. • Heat exchanger supports. These items are procured from several different vendors and are painted by the vendor in accordance with conventional industry practices. Because the total exposed surface area is very small, Westinghouse does not specify further requirements. Category 4 - Insulated or Stainless Steel Equipment Category 4 equipment consists of the following: • Steam generators - covered with wrapped insulation. • Pressurizer - covered with wrapped insulation. • Reactor pressure vessel - covered with rigid reflective insulation. • Reactor cooling piping - stainless steel. • Reactor coolant pump casings - stainless steel. Since Category 4 equipment is insulated or is stainless steel, no painted surface areas are exposed within the containment. Therefore, this regulatory guide is not applicable for Category 4 equipment.

TABLE 6.1.2-2 (SHEET 1 OF 10)

CONTAINMENT COMPONENTS - COATING SCHEDULE

Item	Material[(a)]	Thickness (in.)	Estimated Surface Area (ft^2)[(b)]	Tolerance Surface Area(%)	1st Coat[(c)]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
Containment Liner Plate System[(a]												
Dome	CS	0.25	30,800(1)	±5	Inorganic zinc	2.5	Epoxy-polyamide	3.0				
Cylinder shell	CS	0.25	68,612(1)	±5	Inorganic zinc	2.5						
Sumps	SS	0.19	220(1)	±5	NC							
Refueling canal walls and bottom	SS	0.25	8,115(1)	±5	NC							
Basemat	CS	0.25	14,423(2)	±5	Inorganic zinc	2.5						
Reactor cavity walls	CS	0.25	3,700(1)	±5	Inorganic zinc	2.5						
Reactor cavity bottom	CS	0.25	787(2)	±5	Inorganic zinc	2.5						
Primary Shield[(a)]												
Cylinder wall	CS	0.25	805(1)	±5	Inorganic zinc	2.5						
Cylinder wall and nozzles	CS	1.00	1,287(1)	±5	Inorganic zinc	2.5						
Nozzles	CS	0.50	54(1)	±5	Inorganic zinc	2.5						
Upper cone and nozzles	CS	0.38	71(1)	±5	Inorganic zinc	2.5						
Containment Locks and Hatch[(a)]												
Equipment Hatch												
Spherical head	CS	1.50	174(3)	+15 -5	Inorganic zinc	2.5						
Exposed hatch ring	CS	3.00	172(3)	+15 -5	Inorganic zinc	2.5						
Hatch structure (CBI details)	CS	0.50	3,250(4)	+15 -5	Inorganic zinc	2.5						

TABLE 6.1.2-2 (SHEET 2 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft^2)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
Personnel Lock												
Barrel	CS	0.5	25(4)	+15 -5	Inorganic zinc	2.5						
Sleeve	CS	2.0	52(1)	+15 -5	Inorganic zinc	2.5						
Escape Lock												
Barrel	CS	0.5	25(4)	+15 -5	Inorganic zinc	2.5						
Sleeve	CS	2.0	33(1)	+15 -5	Inorganic zinc	2.5						
Containment Concrete[a]												
Dome												
Buttresses to 50° above horizontal	Concrete	72.0	2278(5)	±10	NC							
Buttresses from 50° to vertical	Concrete	48.0	1822(5)	±10	NC							
Dome area less buttress area	Concrete	45.0	26,700(5)	±10	NC							
Shell												
Buttresses	Concrete	72.0	5920(5)	±10	NC							
Shell	Concrete	45.0	62,692(5)	±10	NC							
Internal Structural Concrete[a]												
Basemat	Concrete	126.0	14,423(5)	±10	NC							
Reactor cavity walls	Concrete	96.0	3818(6)	±10	NC							
Reactor cavity slab	Concrete	96.0	787(6)	±10	NC							
Exterior refueling canal walls												

TABLE 6.1.2-2 (SHEET 3 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft²)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
5-ft 0-in. thick walls	Concrete	60.0	258(6)	±10	Epoxy[a] (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
7-ft 0-in. thick walls	Concrete	84.0	584(6)	±10	Epoxy[a] (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
4-ft 0-in. thick walls	Concrete	48.0	6067(6)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Refueling canal slab	Concrete	48.0	1394(6)	±10	NC							
Filler slab at 171 ft 9 in.	Concrete	33.0	13,715(7)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
Reactor cavity filler slab	Concrete	12.0	787(1)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
Primary shield	Concrete	108.0	2014(8)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Cavity access walls	Concrete	24.0	830(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
North/south beams at 199 ft 0 in.	Concrete	72.0	1849(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Pressurizer walls	Concrete	30.0	4394(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Secondary shield walls	Concrete	36.0	29,722(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Air Shaft Concrete Walls												
Walls, Nos. 1, 2, and 3												
Below 181 ft 0 in.	Concrete	30.0	1114(9)	±10	Epoxy (sealer)	0.5						
Above 185 ft 0 in.	Concrete	36.0	3045(9)	±10	Epoxy (sealer)	0.5						
No. 4 airshaft	Concrete	36.0	1055(9)	±10	Epoxy (sealer)	0.5						
No. 1 and 2 ceiling slab	Concrete	48.0	229(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0

TABLE 6.1.2-2 (SHEET 4 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft^2)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
No. 3 ceiling slab	Concrete	48.0	216(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
Not used												
Operating deck slab - 220 ft 0 in.												
2-ft 0-in. slab, 180° to 360°	Concrete	24.0	1610(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
2-ft 9-in. slab at 90°	Concrete	33.0	625(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
3-ft 0-in. slab at 90° and 260°	Concrete	36.0	210(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epxoy (finish)	3.0
5-ft 0-in. slab at 0° and 180°	Concrete	60.0	940(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
Miscellaneous Walls												
1-ft 6-in. wall at 90°	Concrete	18.0	580(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
3-ft 0-in. wall	Concrete	36.0	1089(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
3-ft 0-in. wall - below R.F. canal	Concrete	36.0	229(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Mass concrete	Concrete	176.0	678(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Walls under R.F. canal	Concrete	36.0	315(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0

TABLE 6.1.2-2 (SHEET 5 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft²)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
Instrument walls	Concrete	30.0	1307(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Stairs and Elevator Shaft Walls												
Stair No. 1												
Walls	Concrete	12.0	2206(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Ceiling	Concrete	18.0	137(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Stair No. 2												
1-ft 0-in. wall	Concrete	12.0	1400(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
1-ft 6-in. wall and ceiling	Concrete	18.0	578(9)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	10.0	Epoxy (finish)	3.0
Elevator												
Walls	Concrete	12.0	1391(9)	±10	Epoxy (sealer)	0.5						
Ceiling	Concrete	18.0	63(9)	±10	Epoxy (sealer)	0.5						
Miscellaneous pads	Concrete	24.0	200(4)	±10	Epoxy (sealer)	0.5	Epoxy (filler)	As needed to fill holes	Epoxy (surfacer)	15.0	Epoxy (finish)	3.0
Structural Steel												
Platform grating	CS	0.19	86,630(4)	±15	Galvanized							
Ladders and stairways	CS	0.22	1328(4)	±15	Inorganic zinc	2.5						

TABLE 6.1.2-2 (SHEET 6 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft^2)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
Pipe whip restraints												
Light and medium capacity	CS	1.5	340(4)	±20	Inorganic zinc	2.5						
Heavy and extra heavy capacity	CS	3.0	15,670(4)	±20	Inorganic zinc	2.5						
Pressurizer support steel												
Pressurizer supports	CS	1.0	914(4)	±10	Inorganic zinc	2.5						
Grating	CS	0.19	5060(4)	±10	Galvanized							
Platform beams	CS	0.5	2588(4)	±10	Inorganic zinc	2.5						
Crossover leg support	CS	2.5	840(4)	±10	Inorganic zinc	2.5						
Platform at steam generator												
Grating	CS	0.19	3250(4)	+100 -20	Galvanized							
Support beams	CS	0.5	30,000(4)	+100 -20	Inorganic zinc	2.5						
Polar crane runway												
Brackets - 37	CS	1.25	2714(4)	±10	Inorganic zinc	2.5						
Rail	CS	1.5	980(4)	±10	Inorganic zinc	2.5						
Girders - 37	CS	1.25	14,950(4)	±10	Inorganic zinc	2.5						
Internal platform structure												
Columns	CS	1.2	14,625(4)	±10	Inorganic zinc	2.5						
Beams	CS	0.38	6194(4)	±15	Inorganic	2.5						

TABLE 6.1.2-2 (SHEET 7 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft^2)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
					zinc							
	CS	0.5	13,096(4)	±15	Inorganic zinc	2.5						
	CS	0.62	37,991(4)	±15	Inorganic zinc	2.5						
	CS	0.75	13,012(4)	±15	Inorganic zinc	2.5						
	CS	1.0	12,741(4)	±15	Inorganic zinc	2.5						
	CS	1.3	3141(4)	±15	Inorganic zinc	2.5						
Cable trays	CS	0.0613	20,697	-	Galvanized							
Cable tray support	CS	0.38	7900(4)	±20	Galvanized							
	CS	0.19	3500(4)	±20	Galvanized							
Conduit	CS	0.15	12,779	-	Galvanized							
Conduit Boxes	CS	0.105	6141	-	Galvanized							
HVAC ducting	SS	0.1	12,000	-	NC							
HVAC ducting bracing and hangers	CS	0.38	2000(4)	±20	Inorganic zinc	2.5						
Pipe supports (steel members)	CS	0.25	51,787	± 5	Inorganic zinc	2.5						
Pipe racks	CS	1.25	13,750	± 2	Inorganic zinc	2.5						
Snubbers	CS	0.5	2970	± 5	Inorganic zinc	2.5						
Spring hangers	CS	0.5	386	± 5	Inorganic zinc	2.5						
Uninsulated piping												
24 in.	SS	0.375	182	± 5	NC							
14 in.	SS	0.312	205	± 5	NC							
12 in.	SS	1.312	194	± 5	NC							
12 in.	SS	1.125	53	± 5								
12 in.	SS	0.688	387	± 5	NC							

TABLE 6.1.2-2 (SHEET 8 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft^2)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
12 in.	SS	0.375	367	± 5	NC							
10 in.	CS	0.365	206	± 5	Inorganic zinc	2.5						
10 in.	SS	0.365	1509	± 5	NC							
8 in.	CS	0.322	832	± 5	Inorganic zinc	2.5						
8 in.	SS	0.322	4769	± 5	NC							
8 in.	SS	0.906	286	± 5	NC							
6 in.	SS	0.280	5752	± 5	NC							
6 in.	CS	0.280	1139	± 5	Inorganic zinc	2.5						
4 in.	CS	0.237	2530	± 5	Inorganic zinc	2.5						
4 in.	SS	0.337	9	± 5	Inorganic zinc	2.5						
4 in.	SS	0.12	2806	± 5	NC							
4 in.	SS	0.237	1289	± 5	NC							
4 in.	SS	0.531	210	± 5	NC							
3 in.	CS	0.216	119	± 5	Inorganic zinc	2.5						
3 in.	CS	0.438	203	± 5	Inorganic zinc	2.5						
3 in.	SS	0.12	58	± 5	NC							
3 in.	SS	0.216	1146	± 5	NC							
3 in.	SS	0.438	388	± 5	NC							
2 1/2 in.	CS	0.203	451	± 5	Inorganic zinc	2.5						
2 in.	CS	0.218	267	± 5	Inorganic zinc	2.5						
2 in.	CS	0.344	71	± 5	Inorganic zinc	2.5						
2 in.	CS	0.154	12	± 5	Inorganic zinc	2.5						
2 in.	SS	0.154	733	± 5	NC							

TABLE 6.1.2-2 (SHEET 9 OF 10)

Item	Material[a]	Thickness (in.)	Estimated Surface Area (ft²)[b]	Tolerance Surface Area(%)	1st Coat[c]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
2 in.	SS	0.344	512	± 5	NC							
2 in.	SS	0.218	25	± 5	NC							
2 in.	SS	0.109	21	± 5	NC							
1 1/2 in.	CS	0.145	36	± 5	Inorganic zinc	2.5						
1 1/2 in.	CS	0.2	421	± 5	Inorganic zinc	2.5						
1 1/2 in.	SS	0.2	48	± 5	NC							
1 1/2 in.	SS	0.281	463	± 5	NC							
1 1/2 in.	SS	0.145	627	± 5	NC							
1 in.	CS	0.133	35	± 5	Inorganic zinc	2.5						
1 in.	CS	0.179	783	± 5	Inorganic zinc	2.5						
1 in.	CS	0.25	14	± 5	Inorganic zinc	2.5						
1 in.	SS	0.133	443	± 5	NC							
1 in.	SS	0.179	72	± 5	NC							
1 in.	SS	0.25	190	± 5	NC							
3/4 in.	SS	0.154	303	± 5	NC							
3/4 in.	SS	0.218	407	± 5	NC							
3/4 in.	SS	0.113	956	± 5								
1/2 in.	CS	0.188	1	± 5	Inorganic zinc	2.5						
1/2 in.	SS	0.147	140	± 5	NC							
1/2 in.	SS	0.109	15	± 5	NC							
1/2 in.	SS	0.065	278	± 5	NC							
1/2 in.	SS	0.049	123	± 5	NC							

TABLE 6.1.2-2 (SHEET 10 OF 10)

Item	Material[(a)]	Thickness (in.)	Estimated Surface Area (ft^2)[(b)]	Tolerance Surface Area(%)	1st Coat[(c)]	Minimum Thickness (mils of) 1st Coat	2nd Coat	Minimum Thickness (mils) of 2nd Coat	3rd Coat	Minimum Thickness (mils) of 3rd Coat	4th Coat	Minimum Thickness (mils) of 4th Coat
3/8 in.	SS	0.065	38	± 5	NC							
Containment coolers	CS	0.25	20,500	+10	Inorganic zinc	2.5	Epoxy	3.0				
Containment auxiliary coolers	CS	0.25	4400	+10	Inorganic zinc	2.5	Epoxy	3.0				
Cavity cooling coils	CS	0.25	2200	+10	Inorganic zinc	2.5	Epoxy	3.0				
ESF fans	CS	0.25	200	+10	Inorganic zinc	2.5	Epoxy	3.0				

a. CS - carbon steel
SS - stainless steel

b. (1) Interior surface (liner plate) backed by concrete. Only one side directly exposed to containment environment. Surface area of exposed side given.
(2) Interior surface (liner plate) backed by concrete and covered with filler slab (concrete). Not directly exposed to containment environment. Surface area of one side given.
(3) Direct boundary between containment interior and exterior environment. Interior surface area given.
(4) Specified member (beams, walls, slabs, etc.) completely exposed (all sides) to the containment environment. Total exposed surface area given (e.g., for walls both sides are given; for wide flange shapes both sides of flanges and web are given).
(5) Exterior surface. One side directly exposed to exterior environment. Surface area of exposed side given.
(6) Interior surface covered with steel liner plate. Not directly exposed to containment environment. Interior surface area given.
(7) Interior surface. One side directly exposed to containment environment with basemat steel liner plate on opposite side. Exposed surface area given.
(8) Interior surface. One side directly exposed to containment environment with steel liner on opposite side (liner in turn exposed to containment environment). Surface area of directly exposed side given.
(9) Outer surface directly exposed to general containment environment with inner area almost entirely self-enclosed. Directly exposed outer surface area given.

c. NC denotes no coating.

d. This epoxy is applied as a wainscot which covers containment interior walls from elevations 171 ft 9 in. to 181 ft 9 in. and elevations 220 ft to 228 ft. The remainder of concrete walls are coated with 0.5 mils of epoxy only.

e. Although actual areas for individual categories may differ slightly from the values shown, the total surface areas are considered to be within the indicated tolerances.

TABLE 6.1.2-3

PROTECTIVE COATINGS ON WESTINGHOUSE-SUPPLIED EQUIPMENT INSIDE CONTAINMENT

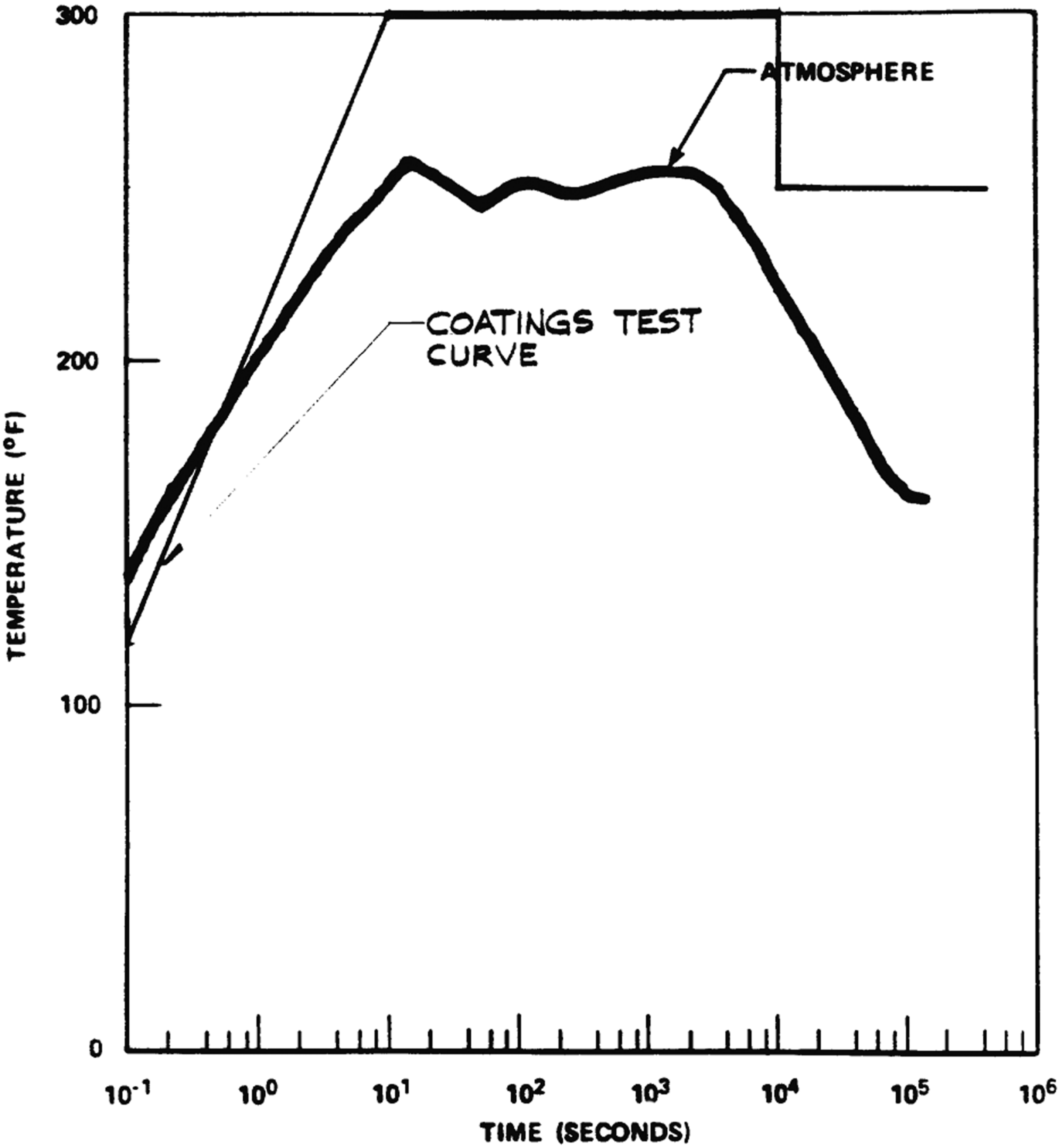
Component	Estimated Coated Surface Area (ft^2)
Reactor coolant system component supports	11,000[a]
Reactor coolant pump assemblies	5200
Accumulator tanks	5400
Refueling machine	3925
Other refueling equipment	2125
Remaining equipment (such as valves, auxiliary tanks and heat exchanger supports, transmitters, alarm horns, small instruments, etc.)	<1300

a. Primer only.

TABLE 6.1.2-4

OTHER ORGANIC MATERIALS

Item	Material	Quantity Enclosed (lb)	Quantity Exposed (lb)
Cable insulation	Ethylene propylene rubber and chlorosulphonated polyethylene	16,100	24,800
Heat shrink tubing	Raychem WCSF-N	650	-
Lug insulation	AMP special industries type PVF	200	-
Cable ties	Thomas and Bets Tefzel	100	500
Terminal blocks	Diallyl-phthalate long glass fiber fill	400	-



REV 14 10/07

SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT TEMPERATURE PROFILE FOR POST–LOCA COATING TEST
		FIGURE 6.1.2–1

6.2 CONTAINMENT SYSTEMS

The containment systems include the containment, the containment heat removal systems, the containment isolation system, and the containment combustible gas control system.

The design basis accident (DBA) is defined as the most severe of a spectrum of hypothetical loss-of-coolant accidents (LOCA) and high-energy line breaks within the containment. The ability of the containment systems to mitigate the consequences of a DBA depends upon the high reliability of these systems. This section provides the design criteria and evaluations to demonstrate that these systems function within the specified limits throughout the unit operating lifetime[a].

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The containment system-is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure is below the design-pressure with an adequate margin, as presented in table 6.2.1-1.

The analyses presented in this section are based on assumptions that are conservative with respect to the containment and its heat removal systems; i.e., minimum heat removal, maximum containment pressure.

This capability is maintained by the containment system even assuming the worst single active failure affecting the operation of the emergency core cooling system (ECCS), containment spray system, and reactor containment fan coolers during the injection phase and the worst active or passive single failure during the recirculation phase. For primary system breaks, loss-of offsite power (LOSP) is assumed. For secondary system breaks, LOSP is not assumed, since this would reduce releases to the containment.

Paragraphs 6.2.1.3 and 6.2.1.4 present the mass and energy releases used in the design evaluation.

An evaluation was conducted to determine the impact of the uprating/T_{hot} reduction program conditions on the short term LOCA mass and energy releases used for the subcompartment analyses. The evaluation is discussed in paragraph 6.2.1.2.3.2.1.

6.2.1.1.2 Design Features

Containment design features are further described in the following:

---

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

A. A secondary shield wall constructed of a minimum of 3-ft-thick reinforced concrete, extends from the basement filler slab (171 ft 9 in.) to the operating deck (220 ft). The shield wall encloses the reactor coolant pumps, reactor vessel and its primary shield wall, steam generators up to 238 ft, the pressurizer up to 268 ft, and the refueling cavity. The secondary shield wall supports the operating deck, which together with the shield wall, prevents the containment liner from being impacted by potential internal missiles and the effects of-pipe whip. Refer to sections 3.5 and 3.6 for more detailed discussion.

B. The reactor containment is designed and constructed in accordance with the codes and standards discussed in subsections 3.8.1 and 3.8.3.

C. The VEGP design does not use a pressure-suppression-type containment.

D. Inasmuch as the reactor containment fan coolers are utilized during normal operation, inadvertent changes to the accident mode produce no significant effect upon containment internal pressure.

E. The only location inside the containment where water may be trapped and prevented from returning to the containment emergency sumps is the reactor cavity. Water can enter the cavity by flowing down through the ventilation openings surrounding the reactor cavity seal ring. Approximately 113,200 gal of water could collect in the reactor cavity. This would cause the static head for the residual heat removal (RHR) and spray pumps to decrease by about 1.31 ft. This decrease would not impair the operation of these pumps. Most of the water entering the refueling canal is returned to the containment emergency sumps via two, normally open, 12-in. drain lines.

F. The containment and subcompartment atmospheres are maintained during normal operation within prescribed pressure, temperature, and humidity limits by means of the nuclear service cooling water (NSCW) system which delivers 90°F water to the cooling coils within each containment fan cooler. Containment ventilation systems, such as the control rod drive mechanism booster fans and cooling fans, are used during normal operation and require no periodic testing to ensure functional capability.

6.2.1.1.3 Design Evaluation

The short-term pressure subcompartment analysis considers a LOSP. Consideration of single active failures is of no consequence, since none of the safety equipment functions during the initial seconds of the post-accident transient. The maximum calculated differential pressure in the steam generator compartment is 23.50 psid resulting from a 436 in.^{2} in area guillotine rupture in the steam generator outlet nozzle. The maximum calculated differential pressure in the upper pressurizer cubical is 5.84 psid resulting from a spray line double-ended break. The maximum calculated differential pressure in the lower pressurizer cubicle is 20.7 psid from a surge line double-ended break. The containment subcompartment differential pressure analysis is described in detail in paragraph 6.2.1.2.

An evaluation was conducted to determine the impact of the uprating/T_{hot} reduction program conditions on the short term LOCA mass and energy releases used for the subcompartment analyses. The evaluation is discussed in paragraph 6.2.1.2.3.2.1.

The results of the pressure transient analysis of the containment for the LOCA are shown in figures 6.2.1-1 through 6.2.1-3. Containment temperature curves are presented in figures 6.2.1-

4 through 6.2.1-6. The cases examined in this analysis determine the effects of the full range of large reactor coolant break sizes up to and including a double-ended rupture. Cases illustrating the sensitivity to break location are also shown. All of these cases show that the containment pressure will remain below design pressure with margin. After the peak pressure is attained, the operation of the safeguards system reduces the containment pressure. At the end of the first day following the accident, the containment pressure has been reduced to a low value. The peak pressures are shown in table 6.2.1-1.

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code COCO.[(1)] The COCO code has been used and found acceptable to calculate containment pressure transients for the H. B. Robinson (docket number 50-261) and Zion (docket number 50-295) plants. Transient phenomena within the reactor coolant system (RCS) affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into two systems. The first system consists of the air-steam phase; the second is the water phase. Sufficient relationships to describe the transients are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. Since thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code. The following are the major assumptions made in the analysis.

A. Discharge mass and energy flowrates through the RCS break are established from the analysis in paragraph 6.2.1.3.

B. For the steam line break analysis and the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

C. Homogeneous mixing is assumed. The steam-air mixture and the water phase each have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and water phase.

D. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

E. For large steam line ruptures the saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

Paragraphs 6.2.1.3 and 6.2.1.4 present the mass and energy releases used for the analysis.

6.2.1.1.3.1 Initial Conditions. An analysis of containment response to the rupture of the RCS must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis.

Also, values for the temperature of the NSCW and refueling water storage tank solution are assumed, along with the initial water inventory of the refueling water storage tank. All of these values are chosen conservatively, as shown in table 6.2.1-2.

Assumptions for containment analysis are shown in table 6.2.1-3. The assumed spray flowrate is based on one of two trains operating.

6.2.1.1.3.2 Heat Removal. The significant heat removal source during the early portion of the transient is structural heat removal. Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for transient conduction into and out of the node and temperature rise of the node. Tables 6.2.1-4 and 6.2.1-5 are summaries of the containment structural heat sinks used in the analysis.

The heat transfer coefficient to the containment structure is calculated by the code based primarily on the work of Tagami.(2) From this work, it was determined that the value of the heat transfer coefficient increases parabolically to peak value at the end of blowdown for LOCA and increases parabolically to peak at the time of steam line isolation. The value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam to air weight ratio.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as follows:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) (time interval to peak pressure}}$$

From this the maximum of h steel is calculated:

$$h_{max} = 75 \left(\frac{E}{t_{\rho} V} \right)^{0.6} \tag{1}$$

where:

h_{max}	=	Maximum value of h (Btu/h-ft^2-°F).
t_ρ	=	Time from start of accident to end of blowdown for LOCA and steam line isolation for secondary breaks (s).
V	=	Containment volume (ft^3).
E	=	Coolant energy--discharge (Btu).

The parabolic increase to the peak value is given by:

$$h_s = h_{max} \left(\frac{t}{t_\rho} \right)^{0.5} \quad (0 \le t \le t_\rho) \tag{2}$$

where:

h_s	=	Heat transfer coefficient for steel (Btu/h-ft^2-°F).
t	=	Time from start of accident (s).

For concrete, the heat transfer coefficient is taken as 40% of the value calculated for steel.

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\; e^{-0.05(t - t_\rho)} \quad (t > t_\rho)$$

where:

h_{stag} = $2 + 50\ X \quad (0 \leq X \leq 1.4)$.

h_{stag} = h for stagnant conditions (Btu/h-ft^2 °F).

x = Steam to air weight ratio in containment.

For a large break the safety features are quickly brought into operation. Because of the brief period of time required to depressurize the RCS, the safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low long-term pressure. Also, although the containment structure is not as effective a heat sink as during the RCS blowdown, it still contributes significantly as a form of heat removal during the long-term cooling period.

During the injection phase of post-accident operation, the ECCS pumps water from the refueling water storage tank into the reactor vessel. Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it can absorb heat from the core until saturation temperature is reached. During the recirculation phase of operation, water is taken from the containment sump and cooled in the residual heat exchanger.

The cooled water is then pumped back to the reactor vessel to absorb more decay heat. The heat is removed from the residual heat exchanger by component cooling water.

Another containment heat removal system is the containment spray. During the injection phase of operation, the containment spray pumps draw water from the refueling water storage tank and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the refueling water storage tank, the entire heat capacity of the spray from the refueling water storage tank temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase of post-accident operation, water is drawn from the sump and sprayed into the containment atmosphere.

When a spray drop enters the hot, saturated, steam-air containment environment following a LOCA, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the drop will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling drop are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q \tag{3}$$

$$\frac{d}{dt}(Mu) = m \tag{4}$$

where:

$q = h_c A\ (T_s - T)$.

$m = k_g A\ (P_s - P_v)$.

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, $\underline{Nu}'$ and the Nusselt number for mass transfer, $\underline{Nu}'$.

Both $\underline{Nu}$ and $\underline{Nu}'$ may be calculated from the equations of Ranz and Marshall[3].

$$\underline{Nu} \quad 2 + 0.6\,\underline{(Re)}^{1/2}\,\underline{(Pr)}^{1/3} \tag{5}$$

$$\underline{Nu'} \quad 2 + 0.6\,\underline{(Re)}^{1/2}\,\underline{(Sc)}^{1/3} \tag{6}$$

Thus, equations 3 and 4 can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean drop produced by the spray nozzles rises to a value within 99% of the bulk containment temperature in less than 2 s.

Drops of this size will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height.

Detailed calculations of the heatup of spray drops in post-accident containment atmospheres by Parsly[4] show that drops of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment.

These results confirm the assumption that the containment spray will be 100% effective in removing heat from the atmosphere.

Nomenclature relevant to the above discussion is listed below:

A	=	Area.
h_c	=	Coefficient of heat transfer.
k_g	=	Coefficient of mass transfer.
h_g	=	Steam enthalpy.
M	=	Droplet mass.
m	=	Diffusion rate.
$\underline{Nu}$	=	Nusselt number for heat transfer.
$\underline{Nu}'$	=	Nusselt number for mass transfer.
P_s	=	Steam partial pressure.
P_v	=	Droplet vapor pressure.
$\underline{Pr}$	=	Prandtl number.
q	=	Heat flowrate.
$\underline{Re}$	=	Reynolds number
$\underline{Sc}$	=	Schmidt number.
T_s	=	Droplet temperature.
T	=	Steam temperature.
t	=	Time.
u	=	Internal energy.

The reactor containment fan coolers are a final means of heat removal. The main aspects of a fan cooler from the heat removal standpoint are the fan and the banks of cooling coils. The fans draw the dense atmosphere through banks of finned cooling coils and mix the cooled steam-air mixture with the rest of the containment atmosphere. The coils are kept at a low temperature by

a constant flow of cooling water. Since this system does not use water from the refueling water storage tank, the mode of operation remains the same both before and after the spray system and ECCS change to the recirculation mode. Fan cooler heat removal performance is shown in figure 6.2.1-7.

6.2.1.1.3.3 Inadvertent Spray Actuation. In the event of inadvertent spray, the containment will depressurize until the air temperature is approximately equal to the spray temperature or the operator takes action to terminate the spray.

The COCO computer code was used to calculate the minimum pressure inside the containment. The following assumptions were made:

A. The spray flowrate is 6748 gal/min and is at 40°F.

B. The containment is initially at 120°F, 14.093 psi, and has 100% humidity.

C. Operator action to stop the spray occurs at 10 min.

The containment pressure reduces to 11.77 psia at 10 min into the transient. Thus, the peak differential pressure is 2.93 psi across the containment shell.

6.2.1.1.3.4 Inadvertent Purge Actuation. In the event of inadvertent containment purge exhaust actuation, the maximum differential pressure that can be drawn on the containment is 0.9 psi. The following assumptions were made:

A. The preaccess purge fan is assumed to be actuated because it has a larger capacity than the minipurge fan (subsection 9.4.6).

B. The preaccess purge fan is operating during mode 5 or 6. During modes 1 through 4, the preaccess purge penetrations are sealed closed.

C. There is no air supply or air leakage into the containment.

D. The containment is initially at 14.7 psia.

Consequently, the maximum differential pressure at the onset of a LOCA would be less than 0.9 psi.

In the VEGP minimum containment pressure analysis, an initial pressure of 14.7 psia (0.0 psig) has been assumed. This assumption is valid because it represents the containment pressure condition consistent with normal, full power operation of the plant. As a rule, parameters input to a 10 CFR 50.46 ECCS performance analysis correspond with expected, nominal plant conditions at 100% power operation. The large degree of conservatism present in 10 CFR 50.46 evaluation model analyses (as compared to historical best estimate results) was intended and does in fact cover uncertainties in the prevailing plant operating parameters at the time a postulated LOCA occurs. Therefore, utilization of atmospheric pressure, which is representative of the normal, full power containment condition, is appropriate for Vogtle and is consistent with the LOCA evaluation model philosophy.

Furthermore, at 100% power, the reactor coolant system is a significant heat source which heats and pressurizes air present in the containment. In the event that such pressurization occurs at Vogtle, containment purging will be conducted. In paragraph 6.2.1.5.8, a specific evaluation documents the penalty in calculated ECCS performance which is incurred when a LOCA occurs coincident with containment purging. This penalty is applied to the VEGP limiting case result. Since the very operation of the containment purge system is predicated upon a

high prevailing containment pressure, the consideration of containment purging being conducted at the time of a large break LOCA together with the assumption of 0 psig containment initial pressure accommodates any need for conservatism in the Vogtle ECCS performance analysis containment pressure computation.

Technical Specification limits on containment internal pressure cover operation through Mode 4. As discussed above, use of 0 psig is appropriate at the 100% power operation condition at which the VEGP ECCS performance analysis is conducted.

The 0 psig value of initial containment pressure is traditionally applied in 10 CFR 50.46 ECCS performance analyses.

In paragraph 6.2.1.5.8 is included an evaluation of purge operation during a double-ended cold leg guillotine break.

6.2.1.1.3.5 Accident Chronology. The accident chronology for the limiting LOCA is given in table 6.2.1-6.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

Subcompartments within containment (principally, the reactor cavity, the steam generator compartments, and the pressurizer compartment) are designed to withstand the transient differential pressures and jet impingement forces of a postulated pipe break. Venting of these chambers is employed to keep the differential pressures within structural limits. In addition, restraints on the coolant pipes, reactor vessel, steam generators, etc., are designed to limit pipe whip effects and forces transmitted through component supports to ensure the integrity of subcompartments and the containment structure.

6.2.1.2.1.1 Summary of Subcompartment Pipe Break Analyses. The postulated breaks in the high-energy lines that are analyzed to determine the maximum differential pressure across the subcompartment walls are tabulated in table 6.2.1-8. The characteristics of the main coolant pipe ruptures are determined in accordance with the methods and criteria of subsection 3.6.2.

6.2.1.2.1.2 Reactor Vessel Cavity. The pipe restraints provided for the RCS limit the size of a break in the cavity region to 144 in.^{2} The blowdown of coolant from a cold leg results in the highest pressure loadings on the reactor cavity walls and the reactor vessel. Accordingly, the analysis of the reactor cavity break assumes that the break occurs in a cold leg pipe.

A multi-compartment nodal model of the reactor cavity around the reactor vessel, reactor coolant piping nozzles, and connected volumes was developed in accordance with the guidelines presented in NUREG-0609. The subcompartment boundaries are set at planes of natural obstructions to flow.

The nodal network and the break mass/energy release rate (supplied by the nuclear steam supply system vendor) are input to the computer program COPDA, NE-699-D2, which performs

a stepwise calculation of the subcompartment pressures and temperatures as a function of time following the line break. Subcompartment pressures are used to calculate pressure differentials across the cavity structural members and force loadings on the reactor vessel.

In the development of the reactor cavity model, the following assumptions are made regarding the reactor vessel thermal insulation.

A. For the purpose of node volume calculation, all thermal insulation (uniform 4 in. thick) remains in place.

B. The upper reactor seal ring is completely plugged.

C. Fifty% of the annular space between the vessel and cavity wall is plugged along the plane perpendicular to the break location.

D. Fifty% of the remaining flow area in the instrument tunnel, after considering the steel of the first bottom mounted instrument support is plugged.

E. Access ports to annular inspection cavity are closed.

F. The nozzle penetrations into the cavity on the unaffected loops are completely plugged. The nozzle penetration into the cavity on the affected loop is open.

G. The nozzle penetrations to the steam generator compartments are open along the unaffected loops and plugged along the affected loop.

Figure 6.2.1-15 shows the resultant pressures in the central volumes shown in the nodal model in figure 6.2.1-16. The flow model is given in tables 6.2.1-9 and 6.2.1-10.

A summary of the forces and moments acting on the reactor vessel is included in table 6.2.1-73. Nodes 1 through 48 (figure 6.2.1-16) are those that contribute to F_x, F_g, F_{total}, M_X, and M . The magnitude of these loads is given at the axial centroids of the node groups. Uplift forces and the moment about the vessel axis are provided in table 6.2.1-74.

6.2.1.2.1.3 Steam Generator Compartment. For the analysis of the pressure transient in the steam generator compartment following a line break, the flow model, including control volumes, intercompartment flowpaths, and corresponding flow coefficients) is illustrated in figures 6.2.1-16A and 17 and tables 6.2.1-11, 11A through 6.2.1-22. The piping restraints on the reactor coolant loop restrict the motion of the piping following a pipe rupture in the primary loop. This restriction limits the size of the effective blowdown areas for postulated breaks in the steam generator compartments to an area less than, or equal to, a single-ended flow area of the pipe. The calculated steam generator compartment pressure response is shown in figures 6.2.1-17A through 6.2.1-23.

6.2.1.2.1.4 Pressurizer Compartment. For the analysis of the pressure transient following a pressurizer surge line or a spray (or relief) line break, the flow model of control volumes, intercompartment flowpaths, and corresponding flow coefficients are illustrated in figure 6.2.1-24 and tables 6.2.1-23 through 6.2.1-25. The calculated compartment pressure response is shown on figures 6.2.1-25 and 6.2.1-25a for the surge line break and the spray line break at the pressurizer nozzle, respectively.

6.2.1.2.2 Design Features

The effects of high-energy line breaks in the reactor cavity, steam generator compartments, and pressurizer compartment were analyzed to establish criteria for the structural design of the compartment walls. Plans and sections of these compartments are shown in drawings 1X2D01A001 and 1X2D01J015. The models of these compartments as used in the analyses are shown in figures 6.2.1-16, 6.2.1-16A, 17, and 6.2.1-24. The node volumes, flowpaths, flowpath areas, and flow coefficients for each of the compartments are tabulated in tables 6.2.1-9 through 6.2.1-25.

6.2.1.2.3 Design Evaluations

6.2.1.2.3.1 Analytical Model for Subcompartment Analyses. A digital computer program was used to perform the short-term compartment pressure transient analysis. This program is capable of handling up to 100 control volumes with a maximum of five flowpaths out of any compartment. The calculational methods used in this computer program are described in detail in BN-TOP-4, Rev. 1.

6.2.1.2.3.2 Analytical Model for Short Term Mass and Energy Releases. The computer models, which were used to develop the mass and energy release transients for the subcompartment pressurization analysis, are described in reference 9. Tables 6.2.1-26 through 6.2.1-28a provide tabulations of the mass and energy release rates versus time for the spectrum of breaks which were analyzed.

For the reactor vessel cavity, the design basis break is a double-ended rupture of the reactor vessel inlet pipe which is restrained to 144 in.^{2} (table 6.2.1-26).

For the steam generator compartment, the following breaks are considered:

A. A double-ended reactor coolant pump outlet nozzle break restrained to 236 in.^{2} (table 6.2.1-27, sheets 1 through 6).

B. A double-ended generator inlet nozzle break restrained to 306 in.^{2} (table 6.2.1-27, sheets 7 through 12).

C. A double-ended break at the loop closure weld restrained to 336 in.^{2} (table. 6.2.1-27, sheets 13 through 18).

D. A double-ended reactor coolant pump inlet nozzle break restrained to 336 in.^{2} (table 6.2.1-27, sheets 19 through 24).

E. A double-ended steam generator outlet nozzle break restrained to 436 in.^{2} (table 6.2.1-27, sheets 25 through 30).

F. A steam-generator inlet elbow split break with a break area equal to 763 in.^{2} (table 6.2.1-27, sheets 31 through 36).

G. A double-ended main feedwater line break (table 6.2.1-26A).

For the pressurizer compartment, two break locations are considered:

A. A double-ended surge line break (table 6.2.1-28).

B. A double-ended spray line rupture (table 6.2.1-28a).

6.2.1.2.3.2.1 Effects of the Uprating/T_{hot} Reduction Programs on the LOCA Short Term Mass and Energy Releases. The short term mass and energy releases to containment, as a result of a LOCA, are described in paragraph 6.2.1.2.3.2. The blowdown mass and energy release rates are affected by the initial RCS temperature conditions. Since short term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short term LOCA releases will increase due to the reductions in RCS coolant conditions associated with the uprating/T_{hot} reduction programs.

Releases currently described in paragraph 6.2.1.2.3.2 for the steam generator compartment have been evaluated for the uprating/T_{hot} reduction conditions. Based upon RCS temperature decreases of 15.0°F for the vessel outlet (from 618.2 to 603.2°F), 20.5°F for the vessel/core inlet (from 558.8 to 538.3°F), and 20.6°F for the steam generator outlet (from 558.6°F to 538.0 °F), the releases in table 6.2.1-27, sheets 1 through 36 can increase by as much as 12%. Even though the current analysis includes 10% margin to account for uncertainties in the analytical model, noding effects, and initial reactor coolant conditions, the 10% margin may not be adequate to offset the 12% increase. However, per reference 14, VEGP has been granted an exemption from a portion of the requirements of GDC 4. This exemption eliminates the need to consider the dynamic effects associated with postulated pipe breaks in the primary loop. Therefore, the releases currently in table 6.2.1-27, sheets 1 through 36 remain bounding for smaller nozzles attached to the RCS, whenever uprating/T_{hot} reduction effects are considered.

For the reactor vessel cavity, based upon RCS temperature decreases of 28.0°F for the vessel inlet, which is the maximum difference between the current data used in the paragraph 6.2.1.2.3.2 analysis and the uprating/T_{hot} reduction data, the releases in table 6.2.1-26 can increase by as much as 6%. The current analysis includes 10% margin to account for uncertainties, which is adequate to offset the 6% increase. Additionally, per reference 15, the average break opening area for the inlet break is 58 in.^{2}. The current releases, which are based upon a 144 in.^{2} break size, remain bounding considering uprating/T_{hot} reduction effects, whenever the actual break size based upon the structural evaluation is considered. Furthermore, per reference 14, VEGP has been granted an exemption from a portion of the requirements of GDC 4. This exemption eliminates the need to consider the dynamic effects associated with postulated pipe breaks in the primary loop.

For the pressurizer compartment, there are two break locations of importance; namely, the surge line and the spray line. For the surge line, based upon an RCS temperature decrease in the analysis of 20.7°F, the releases in table 6.2.1-28 can increase by as much as 16%. The current analysis, described in paragraph 6.2.1.2.3.2, includes 10% margin to account for uncertainties, which may not be adequate to offset this 16% increase. However, per references 16 and 17, VEGP has been granted an exemption from a portion of the requirements of GDC 4. This exemption eliminates the need to consider the dynamic effects associated with postulated rupture in the pressurizer surge line. For the spray line, the current releases are also expected to go up by 16%. However, upon closer inspection of the actual pipe sizes for the Vogtle spray line, and comparing to the current analysis break areas, it has been determined that the impact of the uprating/T_{hot} reduction is more than offset by the available margin in the break size, without the additional consideration of the 10% margin included in the current releases. Therefore, the releases in table 6.2.1-28a remain bounding for the spray line break.

In summary, an evaluation of the current short term LOCA mass and energy releases described in paragraph 6.2.1.2.3.2, utilized for the steam generator compartment, the reactor vessel cavity, and the pressurizer subcompartment, was conducted. It has been determined that the current short term LOCA mass and energy releases described in the previous section remain bounding for the uprating/T_{hot} reduction conditions whenever the margin in the current calculations are considered and whenever leak-before-break is appropriately credited.

6.2.1.2.3.2.2 Evaluation for MUR Power Uprate - LOCA Short-Term Mass and Energy Releases. Short-term LOCA mass and energy release calculations are performed to support the reactor cavity and loop subcompartment pressurization analyses (which includes the steam generator compartment and pressurizer compartment).

The analysis inputs that may potentially change with the uprate are the initial reactor coolant system (RCS) fluid temperatures. Since this event lasts for approximately 3 seconds, the single effect of power is not significant. The approved methodology for the short-term LOCA mass and energy analysis is documented in reference 9. The critical flow calculation employs appropriately defined critical flow correlations applied for fluid conditions at the break location. Most short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. The Zaloudek correlation was used to conservatively evaluate the impact of the changes in the RCS inlet and outlet temperatures for the 2% uprate relative to those used in the current analysis of record.

The use of the lower temperatures maximizes the critical mass flux in the Zaloudek correlation. The analysis uses the minimum composite RCS T_{HOT}, T_{COLD} and steam generator outlet temperatures that are calculated for the MUR conditions. The critical flow correlation used in the mass and energy releases for this analysis will provide an increase in the mass and energy release for a slightly lower fluid temperature.

Unit 1 and Unit 2 have been approved for leak before-break (LBB) methods (reference 14).[a] This exemption eliminates the need to consider the dynamic effects associated with postulated RCS pipe breaks in the primary loop, eliminating the postulated primary system large pipe break from the subcompartment design basis. In addition, references 16 and 17 provide further justification for LBB application to eliminate surge line from the subcompartment design basis. Note, for Unit 2 only, reference 16 also approved LBB methods to eliminate, as a design basis, the accumulator and RHR piping. Therefore, the only break locations that need to be considered are the pressurizer spray line from the cold leg, the RHR line from the hot leg, and the smaller nozzles attached to the RCS.

Steam Generator Compartment

The licensing basis releases currently discussed in paragraph 6.2.1.2.3.2 for the steam generator compartment have been evaluated for the MUR uprate conditions. Based upon a comparison of the minimum hot full power RCS cold leg (T_{COLD}) temperature and the steam generator outlet temperature at the MUR uprate conditions, the temperature may be 0.7°F lower (537.6°F versus 538.3°F and 537.3°F versus 538.0°F) respectively, than that for the current licensing basis analysis. The minimum vessel outlet (T_{HOT}) temperature at the MUR uprate conditions is 603.8°F, which is 0.6°F higher (603.8°F versus 603.2°F). The differences in the temperatures presented are negligible and will have little effect on the mass and energy releases. In addition, based on the LBB exemption (reference 14), this exemption eliminates the need to consider the dynamic effects associated with postulated pipe break in the primary loop. In summary, the benefits of the decrease in mass and energy releases associated with the smaller primary system nozzle breaks, as compared to the larger RCS and larger RCS nozzle pipe breaks, more than offsets any penalty associated with possible increased releases which will result from decreased RCS coolant temperature. Therefore, the current licensing

[a] The leak-before-break analyses have been evaluated as time-limited aging analyses (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in paragraph 19.4.6.1.

basis (mass and energy releases in table 6.2.1-27) remains bounding for the smaller nozzles attached to the RCS whenever the MUR uprate effects are considered.

Reactor Vessel Cavity

For the reactor vessel cavity, a RCS temperature decrease of 0.7°F for the vessel inlet will be seen with the MUR uprate conditions. This is the maximum difference between the basis for the current licensing basis and the MUR uprate data. This decrease in temperature is considered negligible. However, the current analysis includes 10% margin to account for residual uncertainties in the analytical model, noding effects, and initial reactor coolant conditions. It has been determined that the impact of the MUR is more than offset by the available 10% margin included in the current releases to account for residual uncertainties. Additionally, per reference 15, the average break opening area for the inlet break is 58 in.^{2}. The current releases for the reactor cavity break, which are based upon a 144 in.^{2} break size, remain bounding for the subcompartment pressurization analysis considering the MUR uprate effects whenever the actual break size based upon the structural evaluation of the reactor pressure vessel is considered. Furthermore, per reference 14, Units 1 and 2 have been granted an exemption from a portion of the requirements of GDC 4 relative to LBB. This exemption eliminates the need to consider the dynamic effects associated with postulated pipe breaks in the primary loop. Therefore, due to the conservatisms in the calculations for table 6.2.1-26 releases, it can be concluded that the releases presented in table 6.2.1-26 remain bounding when the MUR conditions are considered.

Pressurizer Compartment

For the pressurizer compartment, there are two break locations of importance: the surge line and the spray line. For the surge line, the RCS temperature will increase by 0.6°F as a result of the MUR, which results in a decrease in releases. In addition, per references 16 and 17, Units 1 and 2 have been granted an exemption from a portion of the requirements of GDC 4. Based on the LBB exemption, this exemption eliminates the need to consider the dynamic effects associated with postulated rupture in the pressurizer surge line. For the spray line, the RCS temperature reduction was determined to be 0.7°F, which is also considered negligible. The current analysis includes 10% margin to account for uncertainties in the analytical model, noding effects, and initial reactor coolant conditions. It has been determined that the impact of the MUR is more than offset by the available 10% margin included in the current releases to account for residual uncertainties. In addition, comparing the pipe size assumed in the current analysis of record versus the as-built piping; therefore, it has been determined that the impact of the reduction in temperature is more than offset by the available margin in the break size assumed in the analysis. In summary, the releases in table 6.2.1-28A remain bounding for the spray line break for application to the subcompartment pressurization analysis.

6.2.1.2.3.3 Initial Conditions for Subcompartment Pressure Analyses. The conditions tabulated in tables 6.2.1-9, 6.2.1-11 through 6.2.1-16, and 6.2.1-25 and the assumptions in paragraph 6.2.1.2.3.2 are used as the initial conditions for the subcompartment pressure analyses.

6.2.1.2.3.4 Flow Equation. The flow equations used for calculating the sonic and subsonic flow between nodes are fully described in BN-TOP-4, Rev 1. These flow equations, as

applied in the program, are based on the homogeneous equilibrium model. In addition, the application of the equations to the Nuclear Regulatory Commission (NRC) benchmark problems and to the evaluation of the Batelle Frankfurt experiments has developed conservative differential pressure between nodes.

6.2.1.2.3.5 Piping Systems. RCS, main steam, and main feedwater piping were considered in the containment subcompartment analysis; however, the limiting transient for each subcompartment resulted from a break in the encompassed RCS piping except that the main feedwater line break is the limiting transient for steam generator subcompartments above el 220 ft. There are no flow restrictions in the RCS piping. The dimensions of the pipe are tabulated in table 5.4.3-1, and the break areas are tabulated in table 6.2.1-8. For the reactor cavity and pressurizer compartments, the breaks are at the pipe to vessel nozzle weld. The break location in the cold leg pipe does not impact the analysis of the steam generator compartment node pressures.

6.2.1.2.3.6 Node Selection. The nodalization for the reactor cavity, steam generator, and pressurizer compartments is performed so that nodal boundaries are at the location of flow obstructions or geometry changes within the compartment. These discontinuities create pressure differentials across nodal boundaries; i.e., between adjacent nodes. Within each node, there are no discontinuities and hence negligible pressure gradients.

The COPDA computer code, consistent with BN-TOP-4, Rev 1, assumes stagnation within each node at every calculational time step, and requires that nodal boundaries be taken at significant flow restrictions. This, as well as the other guidelines and recommendations of section 3.2 of NUREG-0609, is followed. In light of this, and the fact that COPDA is an NRC approved code, no sensitivity studies were performed. This is consistent with section 3.2.1 of NUREG-0609. Furthermore, sensitivity studies would require the addition of nodes that are not at discontinuities which violates the requirements of the COPDA code and would lead to meaningless results.

Assumptions on insulation behavior are given in paragraph 6.2.1.2.1.2 for the reactor cavity analysis. This is in accordance with section 3.2.2.3 of NUREG-0609. The steam generator and pressurizer compartment analyses include a reduction in flow area and subcompartment volume due to insulation around piping and vessels. Subcompartment nodalization is shown in figures 6.2.1-16, 6.2.1-17, and 6.2.1-24. The COPDA computer code input data is summarized in tables 6.2.1-9 through 6.2.1-28.

6.2.1.2.3.7 Compartment Time Dependent Pressures. The time dependent pressures for those nodes that determine the structural design of the compartment walls are shown in the graphs of figures 6.2.1-15, 6.2.1-18, 6.2.1-19, 6.2.1-20, 6.2.1-21, 6.2.1-22, 6.2.1-23, and 6.2.1-25.

6.2.1.2.3.8 Vent Flowpath Flow Conditions. The node and flow characteristics for each of the subcompartments are tabulated in tables 6.2.1-9 through 6.2.1-25. The time dependent mass and energy flow conditions are provided in tables 6.2.1-26, 6.2.1-27, and 6.2.1-28.

6.2.1.2.3.9 Vent Flowpath Flow Coefficients. There are two orifice coefficients, C_v and C_g: C_v is a viscous loss coefficient, and C_g is the ratio of vena contract area to aperture area. The quantity C discussed below is obtained by multiplying C_v and C_g. The orifice coefficient is a function of the Reynolds number; however, for sufficiently high Reynolds numbers (greater than 100,000), C becomes independent of the Reynolds number.

The definition of the Reynolds number is:

$$\underline{Re} = \frac{SZ}{\gamma}$$

where:

s	=	Velocity (ft/s).
Z	=	Characteristic length; i.e., diameter (ft).
γ	=	Kinematic viscosity (ft^2/s).

Kinematic viscosity (γ) can be expected to have values of around 10^{-4}. Typical diameters of apertures are 10 to 30 ft. Velocities are 100 to 1000 ft/s. Therefore, Re will be very large and C will be independent of Re.

It is assumed throughout this section that flow coefficients obtained for single-phase flow are applicable for our two-phase situation and that the flow coefficients are constant with time.

The classification of openings and the development of head loss and flow coefficients are described in detail in BN-TOP-4, Rev 1.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

6.2.1.3.1 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture of the RCS. These releases continue through blowdown and post-blowdown.

The LOCA transient is typically divided into four phases:

A. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS pressure reaches initial equilibrium with containment.

B. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

C. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

D. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

6.2.1.3.2 Break Size and Location

Generic studies have been performed with respect to the effect on the LOCA mass and energy releases relative to postulated break size. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and post-reflood phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

- Hot leg (between vessel and steam generator).
- Cold leg (between pump and vessel).
- Pump suction (between steam generator and pump).

The break location analyzed and described herein is the double-ended pump suction guillotine break (10.48 ft^2). Pump suction break mass and energy release have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA.

The following information provides a discussion on each break location. The double ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore the reflood (and subsequent post-reflood) releases are not calculated for a hot leg break. The mass and energy releases for the hot leg break blowdown phase have been included in the scope of this containment integrity analysis.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not usually performed.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flowrates during the post-blowdown period by including all of the available energy of the reactor coolant system in calculating the releases to containment. This break location has been determined to be the limiting break for typical dry

containment plants. The choice of this break location for VEGP as the limiting break is consistent with other dry containment plants for the post blowdown phase of the event.

The analysis of the limiting break location for a dry containment has been performed. The double-ended pump suction guillotine break has historically been considered to be the limiting break location for the post blowdown phase of the event, by virtue of its consideration of all energy sources present in the RCS. The analyses support the conclusions of the double-ended pump suction (DEPS) as the limiting break case for the post-blowdown period, considering both the minimum and maximum safety injection cases. This break location provides a mechanism for the release of the available energy in the reactor coolant system, including both the broken and intact loop steam generators.

6.2.1.3.3 Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the DEPS break.

For the DEPS results an inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators required to power the safety injection system. This is not an issue for the blowdown period which is limited by the double-ended hot leg (DEHL) break.

Two cases have been analyzed for the effects of a single failure. The DEPS case with both minimum and maximum safety injection for the 3579 MWt rerated conditions was analyzed.

An evaluation has been performed to support the MUR power uprate.

The analysis of record presently assumes a NSSS power of 3650 MWt (102% of 3579 MWt), which includes a bounding allowance for the MUR power uprate in conjunction with a reduced calorimetric uncertainty.

The limiting case for the VEGP is the minimum safeguards case. This was determined by prior generic and specific VEGP analyses. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train and the containment safeguards components on that diesel, thereby minimizing the safety injection flow. The analysis further considers the safety injection pump head curves to be degraded by 5%. (See Westinghouse Letter GP-15580 for discussion of flow margins.) This results in the greatest reduction possible for the emergency core cooling system (ECCS) components. For the case analyzing maximum safety injection, a conservative assumption was made due to the availability of only the diesel train failure criteria. The maximum safety injection flows were modeled with the minimum containment safeguard components available. This applies to heat exchangers, fan coolers, and the containment spray system. This assumption would provide a bounding assessment in maximizing the mass release but minimizing the heat removal capability.

6.2.1.3.4 Mass and Energy Release Data

6.2.1.3.4.1 Significant Modeling Assumptions. The following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

A. Maximum expected operating temperature of the reactor coolant system.

B. Allowance in temperature for instrument error and dead band (+6.0°F).

C. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty).

D. NSSS power of 3650 MWt (102% of 3579 MWt), which includes a bounding allowance for the MUR power uprate in conjunction with a reduced calorimetric uncertainty.

E. Allowance for calorimetric error is included in item D.

F. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer).

G. Allowance in core stored energy effect of fuel densification.

H. Margin in core stored energy (+15%).

I. Allowance for RCS pressure uncertainty (+50 psi).

6.2.1.3.4.2 Blowdown Mass and Energy Release Data. The SATAN-VI code is used for computing the blowdown transient and is the same as that used for the ECCS calculation in reference 11. The methodology for the use of this model is described in reference 10.

Tables 6.2.1-29 and 6.2.1-30 present the calculated mass and energy releases for the blowdown phase of the break analyzed for the DEPS and DEHL breaks, respectively. The mass and energy release for the DEPS break and the DEHL break, given in tables 6.2.1-29 and 6.2.1-30, terminate 22.0 and 25.0 seconds, respectively, after the initiation of the postulated accident.

6.2.1.3.4.3 Reflood Mass and Energy Release Data. The WREFLOOD code is used for computing the reflood transient and is a modified version of that used in the ECCS calculation in reference 11. The methodology for the use of this model is described in reference 10.

An exception to the mass and energy evaluation model described in reference 10 is taken, in that steam/water mixing in the broken loop has been included in this analysis. This assumption is justified, supported by test data, and summarized as follows:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the mass and energy of the steam released is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump).

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (reference 12), which are the largest scale data available and thus most closely simulate the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and

discussed in detail in reference 10. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis during the post-blowdown phase is the DEPS break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam, which is not condensed by ECCS injection in the intact RCS loops, passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs, and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in references 10 and 12.

The methodology previously discussed and described in reference 10 has been utilized and approved on the dockets for Catawba Units 1 and 2, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, Millstone Unit 3, Beaver Valley Unit 2, and Surry Units 1 and 2.

Tables 6.2.1-31 and 6.2.1-32 present the calculated mass and energy release for the reflood phase of the DEPS break with minimum and maximum safety injection, respectively. A significantly higher discharge occurs during the period the accumulators are injecting (from 28.1 to 54.3 seconds for the minimum safety injection case and 28.1 to 54.1 seconds for maximum safety injection, as illustrated in tables 6.2.1-31 and 6.2.1-32).

The transient of the principal parameters during reflood is given in tables 6.2.1-33 and 6.2.1-34 for the minimum and maximum safety injection DEPS break cases.

6.2.1.3.4.4 Post-Reflood Mass and Energy Release Data. The FROTH code (reference 9) is used for computing the post-reflood transient. The methodology for the use of this model is described in reference 10. The mass and energy release rates calculated by the FROTH code are used in the containment analysis until the time of containment depressurization.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in reference 13, and the following input:

A. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

B. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

C. Fission rate is constant over the operating history of maximum power level.

D. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANSI/ANS 5.1-1979.

E. Operation time before shutdown is 3 years.

F. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

G. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Tables 6.2.1-35 and 6.2.1-36 present the two phase post-reflood mass and energy release data for the double-ended pump suction break minimum and maximum safety injection cases.

6.2.1.3.5 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in tables 6.2.1-37, 6.2.1-38, and 6.2.1-39. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in tables 6.2.1-40, 6.2.1-41, and 6.2.1-42. The energy sources include:

- Reactor coolant system water.
- Accumulator water.
- Pumped injection water.
- Decay heat.
- Core stored energy.
- Reactor coolant system metal.
- Steam generator metal.
- Steam generator secondary energy.
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary).

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

System parameters needed to perform confirmatory analyses are provided in table 6.2.1-43.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions).
- End of blowdown time.
- End of refill time.
- End of reflood time.
- Time of full depressurizations.

- End of analysis.

The methods and assumptions used to release the various energy sources are given in reference 10, except as noted in paragraph 6.2.1.3.4.3, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steam line rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make a reasonable determination of the single absolute worst case for both containment pressure and temperature evaluations following a steam break difficult. This section describes the methods used in determining the containment responses to a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size. The spectrum of breaks analyzed is listed in table 6.2.1-60.

6.2.1.4.1 Significant Parameters Affecting Steam Line Break Mass and Energy Releases

There are four major factors that influence the release of mass and energy following a steam line break: steam generator fluid inventory, primary-to-secondary heat transfer, protective system operation, and the state of the secondary fluid blowdown. The following is a list of those plant variables that determine the influence of each of these factors:

A. Plant power level.

B. Main feedwater system design.

C. Auxiliary feedwater system design.

D. Postulated break type, size, and location.

E. Availability of offsite power.

F. Safety system failures.

G. Steam generator reverse heat transfer and RCS metal heat capacity.

The following is a discussion of each of these variables.

6.2.1.4.1.1 Plant Power Level. Steam line breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power generally result in a greater total mass release to the plant containment. However, because of increased energy storage in the primary plant, increased heat transfer in the steam generators, and the additional energy generation in the nuclear fuel, the energy release to the containment from breaks postulated to occur during power operation may be greater than for breaks occurring with the plant in a hot shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power and have significant influence on both the

rate of blowdown and the amount of moisture entrained in the fluid leaving the break following a steam break event.

Because of the opposing effects of changing power level on steam line break releases, no single power level can be identified as a worst case initial condition for a steam line break event. Therefore, several different power levels spanning the operating range as well as the hot shutdown condition have been analyzed.

6.2.1.4.1.2 Main Feedwater System Design. The rapid depressurization which occurs following a rupture may result in large amounts of water being added to the steam generators through the main feedwater system. Rapid closing isolation valves are provided in the main feedwater lines to limit this effect. Also, the piping layout downstream of the isolation valves affects the volume in the feedwater lines that cannot be isolated from the steam generators. As the steam generator pressure decreases, some of the fluid in this volume will flash into the steam generator, providing additional secondary fluid which may exit out the rupture.

The feedwater addition which occurs prior to closing of the feedwater line isolation valves influences the steam generator blowdown in several ways. First, the rapid addition increases the amount of entrained water in large-break cases by lowering the bulk quantity of the steam generator inventory. Secondly, because the water entering the steam generator is subcooled, it lowers the steam pressure, thereby reducing the flowrate out of the break. Finally, the increased flowrate causes an increase in the heat transfer rate from the primary-to-secondary system, resulting in greater energy being released out the break. Since these are competing effects on the total mass and energy release, no worst case feedwater transient can be defined for all plant conditions. In the results presented, the worst effects of each variable have been used. For example, moisture entrainment for each break is calculated assuming conservatively small feedwater additions so that the entrained water is minimized. Determination of total steam generator inventory, however, is based on conservatively large feedwater additions, as explained in paragraph 6.2.1.4.3.2.

The unisolated feedwater line volumes between the steam generator and the isolation valves serve as a source for additional high-energy fluid to be discharged through the pipe break. This volume is accounted for in the mass and energy release data presented in paragraph 6.2.1.4.3.2.

6.2.1.4.1.3 Auxiliary Feedwater System Design. Within the first minute following a steam line break, the auxiliary feed system is initiated on any one of several protection system signals. The addition of auxiliary feedwater to the steam generators increases the secondary mass available for release to the containment, as well as increases the heat transferred to the secondary fluid.

The effects of the steam generator mass are realistically and conservatively modelled in the calculation described in paragraph 6.2.1.4.3.2 by assuming full auxiliary feedwater flow to the faulted steam generator starting at time zero and continuing until manually stopped by the plant operator for the small break cases and zero power cases. The split break cases assume full auxiliary feedwater (AFW) flow to the faulted loop starting at the time of Hi-1 containment signal and the large double-ended rupture case assumes full AFW flow to the faulted steam generator starting at the time a safety injection or low steam generator mass reactor trip signal is received.

6.2.1.4.1.4 Postulated Break Type, Size, and Location.

A. Postulated Break Type

Two types of postulated pipe ruptures are considered in evaluating steam line breaks.

First is a split rupture in which a hole opens at some point on the side of the steam pipe or steam header but does not-result in a complete severance of the pipe. A single, distinct break area is fed uniformly by all steam generators until steam line isolation occurs. The blowdown flowrates from the individual steam generators are interdependent, since fluid coupling exists between all steam lines. Because flow limiting orifices are provided in each steam generator, the largest possible split rupture can have an effective area prior to isolation that is no greater than the throat area of the flow restrictor times the number of plant primary coolant loops. Following isolation, the effective break area for the steam generator with the broken line can be no greater than the flow restrictor throat area.

The second break type is the double-ended guillotine rupture in which the steam pipe is completely severed and the ends of the break displace from each other. Guillotine ruptures are characterized by two distinct break locations, each of equal area, but being fed by different steam generators. The largest possible guillotine rupture can have an effective area per steam generator no greater than the throat area of one steam line flow restrictor.

The type of break influences the mass and energy releases to containment by altering both the nature of the steam blowdown from the piping in the steam plant and the effective break area fed by each steam generator prior to steam line isolation. For example, a double-ended rupture in a pipe having a cross-sectional area of 2.4 ft^2 would appear as a 1.4-ft^2 rupture to a single steam generator feeding one end of the break but would appear as a 0.8-ft^2 rupture to each of steam generators feeding the other end of the break.

B. Postulated Size

Break area is also important when evaluating steam line breaks. It controls the rate of releases to the containment, as well as exerts significant influence on the steam pressure decay and the amount of entrained water in the blowdown flow. The data presented in this section include releases for four break areas at each of four initial power levels. Included are three double-ended and one split rupture, as follows.

1. A full double-ended pipe rupture downstream of the steam line flow restrictor. For this case, the actual break area equals the cross-sectional area of the steam line, but the blowdown from the steam generator with the broken line is controlled by the flow restrictor throat area (1.4 ft^2). The reverse flow from the intact steam generators is controlled by the smaller of the pipe cross section, the steam stop valve seat area, or the total flow restrictor throat area in the intact loops. The reverse flow has been conservatively assumed to be controlled by the flow restrictors in each of the intact loop steam generators. Actually, the combined flow from the three steam generators must pass through an 18-in. (1.42-ft^2) line, which would greatly restrict the flow.

2. A small double-ended rupture at the steam generator nozzle having an area just larger than the area at which water entrainment occurs (i.e., with entrainment). Entrainment is assumed in the forward direction only. Dry steam blowdown is assumed to occur in the reverse direction.

3. A split break that represents the largest break which can neither generate a steam line isolation signal from the primary protection equipment nor result in moisture entrainment. Steam and feedwater line isolation signals are generated by high containment pressure signals for these cases. Being a split rupture, the effective area seen by the faulted steam generator increases by a factor of four, following steam line isolation. Conceivably, moisture entrainment could occur at that time. However, since steam line isolation for these breaks generally does not occur before 20 to 60 s, it is conservatively assumed that the pressure has decreased sufficiently in the affected steam generator to preclude any moisture carryover.

4. A small double-ended rupture at the steam generator nozzle having an area just smaller than that in which water entrainment occurs (i.e., without entrainment).

C. Postulated Break Location

Break location affects steam line blowdowns by virtue of the pressure losses which would occur in the length of piping between the steam generator and the break. The effect of the pressure loss is to reduce the effective break area- seen by the steam generator. Although this would reduce the rate of blowdown, it would not significantly change the total release of energy to the containment. Therefore, piping loss effects have been conservatively ignored in all blowdown results, except in the small double-ended ruptures in which moisture entrainment occurs. The effects of pipe friction are conservatively assumed to be sufficiently large in this case to prevent moisture entrainment in the reverse flow, thus minimizing water relief to the containment.

6.2.1.4.1.5 Availability of Offsite Power. The effects of the assumption of the availability of offsite power have been enveloped in the analysis. LOSP has been assumed where it delays the actuation of the containment heat removal systems (i.e., containment sprays and containment air coolers) due to the time required to start the emergency diesel generators. In these cases a 12-s diesel start time has been assumed. This includes the diesel initial sequencer loading step.

Offsite power has been assumed to be available where it maximizes the mass and energy released from the break due to:

A. The continued operation of the reactor coolant pumps, which maximizes the energy transferred from the RCS to the steam generator.

B. Continued operation of the feedwater pumps and actuation of the auxiliary feedwater system, which maximizes the steam generator inventories available for release.

6.2.1.4.1.6 Safety System Failures. In addition to assuming a LOSP, the following single active failures were considered.

- Loss of one emergency diesel.
- Failure of one main feedwater isolation valve.

The loss of one diesel results in the loss of one train of each of the containment heat removal systems. The analysis model conservatively accounted for the effects of the single failures by combining the failures into one bounding set of analyses. Two fast acting MSIVs are provided in each steam line. No more than one steam generator would experience an uncontrolled blowdown even if one of the MSIVs fails to close.

6.2.1.4.1.7 Steam Generator Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity. Once steam line isolation is complete, those steam generators in the intact steam loops become sources of energy that can be transferred to the steam generator with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact units, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line.

Similarly, the heat stored in the metal of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps is transferred to the primary coolant as the plant cooldown progresses. This energy also is available to be transferred to the steam generator with the broken line.

The effects of both the RCS metal and the reverse steam generator heat transfer are included in the results presented in this document.

6.2.1.4.2 Description of Blowdown Model

A description of the blowdown model used is provided in reference 6. This reference is the basis for the data shown in tables 6.2.1-61 and 6.2.1-62.

6.2.1.4.3 Containment Response Analysis

The COCO computer code (reference 8), which is discussed in paragraph 6.2.1.1.3, was used to determine the containment responses following the postulated main steam line breaks (MSLB). The following assumptions were made to obtain these responses.

6.2.1.4.3.1 Initial Conditions. The initial containment conditions are the same as those used in the containment response analysis for the postulated RCS pipe ruptures. (See table 6.2.1-2.)

6.2.1.4.3.2 Mass and Energy Release Data. Using references 6 and 7 as a basis, mass and energy release data were developed to determine the containment pressure-temperature response for the spectrum of breaks analyzed. Tables 6.2.1-61 and 6.2.1-62 provide the mass and energy release data for the cases which result in the highest temperature and pressure. Table 6.2.1-64 provides specific plant data used for each case.

The rate of auxiliary feedwater addition represents the maximum runout flowrate to a fully depressurized steam generator. The value given for mass added by feedwater pumping assumes that no reduction in feedwater pump turbine speed occurs following a MSLB and prior to main feedwater isolation. Determination of feedwater flowrates prior to isolation assumed that the feedwater regulating valve in the broken loop goes wide open while those in the intact loops remain in their prebreak positions. Actual isolation is dependent on signals generated by the primary protection system. Feedwater isolation for the split breaks was based on the time required to reach the containment pressure setpoint that generates the isolation signal. For the split breaks and small double-ended rupture, feedwater was conservatively assumed to match the steam flow until the time of feedwater isolation.

6.2.1.4.3.3 Containment Pressure-Temperature Results. Figures 6.2.1-26 through 6.2.1-29 provide curves of the resultant containment pressure-temperature analyses for the cases producing the highest peak containment pressure and temperature. Table 6.2.1-65 summarizes the results of all the cases analyzed and indicates the times at which dryout occurs and the various containment pressure setpoints are reached. The sequence of events following a postulated MSLB is listed in tables 6.2.1-66 and 6.2.1-67 for worst pressure and temperature cases.

As illustrated in figure 6.2.1-26, case 16 results in a peak pressure of 36.5 psig. This case represents the peak calculated containment pressure for the spectrum of breaks analyzed. The containment vapor temperature profile versus time for this case is provided in figure 6.2.1-27.

It is important to note that the peak calculated pressure is coincident with the completion of the blowdown of the contents of the affected steam generator. In all cases, the peak calculated containment pressure demonstrates considerable margin below the containment design pressure.

As illustrated in figure 6.2.1-29, case 13 results in a peak vapor temperature of 303.1°F. This case represents the peak calculated containment vapor temperature for the spectrum of breaks analyzed. The containment pressure profile versus time for this case is provided in figure 6.2.1-28.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

The containment backpressure and temperature and the containment wall-condensing heat transfer coefficient, used for the limiting case CD = 0.6 double-ended cold leg guillotine break for the (ECCS) analysis found in subsection 15.6.5, are presented in figures 6.2.1-30, 31, and 32. The containment backpressure is calculated using the methods and assumptions described in Westinghouse-Emergency Core Cooling System Evaluation Model -Summary, WCAP-8339, Appendix A. Input parameters including the containment initial conditions, net free containment volume, passive heat sink materials, thicknesses, surface areas, starting time, and number of containment cooling systems used in the analysis are described below.

6.2.1.5.1 Mass and Energy Release Data

The break mass and energy releases to containment during the blowdown and reflood portions of the limiting break transient are presented in tables 6.2.1-68 and 6.2.1-69, respectively.

The mathematical models which calculate the mass and energy releases to the containment are described in subsection 15.6.5. Since the requirements of Appendix K of 10 CFR 50 are very specific in regard to the modeling of the RCS during blowdown and since the models used are in conformance with Appendix K, no alterations to those models have been made in regard to the mass and energy releases. A break spectrum analysis is performed. (See the double-ended cold leg guillotines which affect the mass and energy released to the containment.) This effect is considered for each case analyzed. During refill, the mass and energy released to the containment is assumed to be zero, which minimizes the containment pressure. During reflood, the effect of steam-water mixing between the safety injection water and the steam flowing through the RCS intact loops reduces the available energy released to the containment vapor space and, therefore, tends to minimize containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

The following initial values were used in the analysis:

Containment pressure (psia)	14.7
Containment temperature (°F)	90
Refueling water storage tank temperature (°F)	40
NSCW temperature (°F)	40
Outside temperature (°F)	17

The containment initial conditions of 90°F and 14.7 psia are representatively low values anticipated during normal full-power operation. The initial relative humidity was conservatively assumed to be 99%.

6.2.1.5.3 Containment Volume

The volume used in the analysis is 2.95×10^6 ft^3 plus an additional amount to incorporate the effect of containment purge, resulting in a total containment volume of 3.20×10^6 ft^3.

6.2.1.5.4 Active Heat Sinks

The containment spray system and the containment fan coolers operate to remove heat from the containment. Pertinent data for these systems which were used in the analysis are presented in table 6.2.1-71. The heat removal capability of each fan cooler is presented in table 6.2.2-2.

Because the fan coolers use nuclear service cooling water (NSCW), the lowest normal NSCW temperature (40°F) was used in the analysis.

The containment sump temperature was not used in the analysis because the maximum peak cladding temperature occurs prior to initiation of the recirculation mode for the containment spray system. In addition, heat transfer between the sump water and the containment vapor space was not considered in the analysis.

6.2.1.5.5 Steam-Water Mixing

Water spillage rates from the broken loop accumulator are presented in table 6.2.1-70.

6.2.1.5.6 Passive Heat Sinks

The passive heat sinks used in the analysis, with their thermophysical properties, are given in table 6.2.1-72. The passive heat sinks and thermophysical properties were divided in compliance with Branch Technical Position CSB6-1, Minimum Containment Pressure Model for Pressurized-Water Reactor (PWR) ECCS Performance Evaluation.

6.2.1.5.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients used for heat transfer to the steel containment structures are given in figure 6.2.1-32 for the limiting break. The containment temperature transient for the limiting break is shown in figure 6.2.1-31.

6.2.1.5.8 Other Parameters

Operating containment purge at the onset of the double-ended cold leg guillotine break was shown to cause a drop in containment pressure of less than 0.19 psi. An increase in the containment volume was incorporated into the analysis, as discussed in paragraph 6.2.1.5.3. No other parameters have a substantial effect on the minimum containment pressure analysis.

6.2.1.5.9 Standard Review Plan Evaluation

The VEGP does not employ the heat transfer coefficients supplied in the Standard Review Plan.

The heat transfer coefficients were calculated in conformance with WCAP-8339, Appendix A, which has received Nuclear Regulatory Commission approval. This reference is for the COCO code and predates CSB6-1. Results using the heat transfer coefficients have been found acceptable in other plant applications.

6.2.1.6 <u>Instrumentation Requirements</u>

Adequate instrumentation is provided to monitor the conditions inside the containment and actuate the appropriate engineered safety features, should those conditions exceed the predetermined levels. The instruments measure the containment pressure, containment atmosphere radioactivity, purge exhaust effluent radioactivity, and containment hydrogen concentration.

The containment pressure is measured by four independent Q-class pressure transmitters and fed into the engineered safety features actuation system (ESFAS) as described in subsection 7.3.1. Upon detection of excessively high pressure inside the containment, the appropriate safety actuation signals are generated which automatically activate the necessary safety systems. These physically separated pressure transmitters are located outside the containment and connected to their sensors by filled and sealed hydraulic lines. Refer to section 7.3 for a detailed description.

The containment atmosphere radiation level is monitored by four independent Q-class area monitors located at the operating deck inside the containment building. The measurements of monitors RE-0002 and RE-0003 are continuously fed into the ESFAS logic and, in modes 1 through 4, cause the actuation of containment ventilation isolation (CVI) safety signals, should the measured radiation levels exceed their setpoints. In mode 6 during core alterations or movement of irradiated fuel assemblies in containment, two channels of radiation monitors are required operable to provide input to control room alarms to ensure prompt operator action to manually close the containment purge and exhaust valves. In addition, the containment purge exhaust air radiation is measured by a non-Q-class, three-channel airborne effluent monitor located outside the containment in the purge exhaust duct. Its isolated output is also fed into the ESFAS logic and, in modes 1 through 4, causes actuation of the CVI signal when the monitored radiation level exceeds the setpoint. In mode 6 during core alterations or movement of irradiated fuel assemblies, only operable radiation monitors are required to alert the operators of the need for containment ventilation isolation. Manual isolation using individual valve hand switches following a radiation alarm is the means for isolating containment in the event of a fuel handling accident during shutdown. For detailed information on the containment area radiation monitors, effluent monitor, and the ESFAS operation, refer to subsections 12.3.4 and 11.5.2 and section 7.3, respectively.

The containment hydrogen concentration is measured by two redundant non-Q-class hydrogen monitors as described in subsection 6.2.5. The readouts and alarms are provided in the control room to facilitate manual actuation of safety-related hydrogen control systems by the operator, should it become necessary.

6.2.1.7 References

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4. Parsly, L. F., "Spray Tests at the Nuclear Safety Pilot Plant," in "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1971," ORNL 4647, p. 182, Oak Ridge National Laboratory, 1971.
5. Shepard, R. M., et al. "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-PA, June 1975 (Proprietary) and WCAP-8312-A, Revision June 1975 (Nonproprietary).
6. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822, (Proprietary) and WCAP-8860 (Nonproprietary), September 1976.
7. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.
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9. Shepard, R. M., et al., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, June 1975 (Proprietary), and WCAP-8312-A, Revision 2, August 1975 (Nonproprietary).

10. WCAP-10325-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," April 1979.

11. "Westinghouse ECCS Evaluation Model - 1981 Version," WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev. 1.

12. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam 1/3 Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

13. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

14. NUREG-1137, "Safety Evaluation Report related to the Operation of Vogtle Electric Generating Plant, Units 1 and 2," Docket Nos. 50-424 and 50-425, Page 3-25, June 1985.

15. WCAP-10033, "Dynamic Analysis of Reactor Pressure Vessel For Postulated Loss-of-Coolant Accidents: Alvin W. Vogtle Nuclear Plant Units 1 and 2," pages 3-22, 3-23, 4-4, and 5-1, December 1981.

16. NUREG-1137, Supplement 7, "Safety Evaluation Report related to the Operation of Vogtle Electric Generating Plant, Units 1 and 2," Docket Nos. 50-424 and 50-425, January 1988.

17. Letter from T. A. Reed, of the NRC, to W. G. Hairston III, of the Georgia Power Company, "Vogtle Unit 1 Safety Evaluation on Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11) (TAC 72178)," April 12, 1990.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The functional performance objective of the containment heat removal system as an engineered safety features (ESF) system is to reduce the containment temperature and pressure following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) accident inside the containment by removing thermal energy from the containment atmosphere. These cooling systems also serve to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for leakage of fission products from the containment to the atmosphere. The containment heat removal systems include the containment cooling system and the containment spray system. The containment cooling system described here also functions during normal operation to maintain a suitable atmosphere for the equipment located within the containment.

6.2.2.1 Containment Cooling System

6.2.2.1.1 Design Bases

6.2.2.1.1.1 Safety Design Bases.

A. The containment cooling system is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods.

B. The containment cooling system is automatically placed in operation on receipt of a safety injection (SI) signal following a LOCA or MSLB accident.

C. The containment cooling system is designed so that a single failure of any active component, assuming loss of offsite power, cannot impair the capability of the system to perform its safety function.

The capability of isolating components, systems, or piping is provided, if required, so that the system's safety function will not be compromised.

D. Active components of the containment cooling system are capable of being tested during plant operation. Provisions are made for inspection of major components at appropriate times specified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI.

E. The containment cooling system components required to mitigate the consequences of an accident are designed to remain functional in the accident environment and to withstand the dynamic effect of the accident.

F. The containment cooling system in conjunction with the containment spray system is capable of removing sufficient thermal energy and subsequent decay heat from the containment atmosphere following the postulated LOCA or MSLB accident to maintain the containment pressure below design values.

G. The containment cooling system is designed and fabricated to codes consistent with Regulatory Guide 1.26 as described in table 3.2.2-1 and Seismic Category 1 in accordance with Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

6.2.2.1.1.2 Power Generation Design Bases. The containment cooling system is designed to limit the ambient containment air temperature during normal plant operation to 120°F with any four of the eight coolers operating. Normal operation and the power generation design bases are discussed in section 9.4.

6.2.2.1.2 System Design

6.2.2.1.2.1 General Description. The containment cooling units are designed to the codes and standards identified in table 3.2.2-1; flood design is discussed in section 3.4; missile protection is discussed in section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6. Environmental design and equipment qualification is discussed in section 3.11. The actuation system is discussed in section 7.3.

6.2.2.1.2.2 System Description. The containment cooling system consists of eight separate 25% fan cooler units inside the containment. The system piping and instrumentation diagram is given in drawing 1X4DB212. The location and arrangement of the fan coolers are indicated in drawings 1X4DJ4103, 1X4DJ4113, 1X4DJ4123 and 1X4DJ4133. The system is designed to permit periodic testing and inspection as discussed in paragraph 6.2.2.1.4. Table 6.2.2-1 provides a tabulation of the design and performance data for each containment cooling unit. The containment fan coolers reject heat to the nuclear service cooling water (NSCW) system, which is described in subsection 9.2.1.

6.2.2.1.2.3 Component Description. Each 25%-capacity fan cooling unit consists of a vane axial fan, a fan motor, copper-nickel cooling coils with copper fins, a carbon steel housing, round metal ducting, and a concrete discharge duct. The 8 units are located at el 238 ft in the containment building and discharge to the bottom of the containment building through four concrete ducts. Each fan unit has two speeds of operation, high speed for normal operation and low speed for post-accident operation. For purposes of heat removal from the containment atmosphere, any combination of four fans in slow speed is sufficient.

The containment cooling system is separated into two trains consisting of four fan cooler units each. The two trains are supplied cooling water and electrical power from the corresponding train of the NSCW system and the Class 1E electrical power system.

Each concrete air supply duct contains three large outlets located above the maximum calculated containment flood elevation to supply cool air to the containment. The air cooling unit enclosures and concrete air supply duct remain intact following a design basis accident (DBA). There is a backdraft damper located at the inlet of each concrete air supply duct to prevent overpressurizing the ducting and coolers. The capability to remain intact is discussed in relation to the hydrogen mixing function of the containment cooling system in paragraph 6.2.5.2.1. Plan and elevation drawings of the containment showing the routing of airflow guidance ductwork are given in drawings 1X4DJ4103, 1X4DJ4113, 1X4DJ4123 and 1X4DJ4133. The fans and motors are designed to operate in the containment post-accident environment. The heat removal capability of the containment fan coolers versus containment temperature for the maximum NSCW inlet temperature is provided in table 6.2.2-2.

6.2.2.1.2.4 System Operation. The containment cooling system is an ESF system that is in use during normal plant operation with four fans operating at high speed. The containment isolation valves on the NSCW line to the containment air cooling units are normally open. System ESF operation is initiated automatically upon receipt of an SI signal. Upon receipt of an SI signal, all fans are restarted at low speed. The basis for the setpoint for the automatic initiation of the cooling system is the diesel start time (12 + 0.5 s) and the load sequencer for the fan coolers (30.5 s). The containment cooling system is fully operational within 40.5 s following receipt of an SI signal. The containment air cooling units can be stopped and started from the control room and from the shutdown panels.

6.2.2.1.3 Safety Evaluation

A. The safety-related portions of the containment cooling system are located in the containment building. This building is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods.

B. Operation of the containment cooling system is initiated automatically following the receipt of an SI signal. Use of this signal provides a reliable indication that a LOCA or a MSLB accident has occurred inside containment. Operation of the containment air coolers may also be manually initiated from the control room and from the shutdown panels. A detailed description of the actuation system is contained in section 7.3.

C. Two trains, each containing four air cooling units, are supplied from redundant emergency power sources and redundant NSCW trains. Failure of any component in one train will not affect the operability of the other train. The containment analyses of LOCA and MSLB accidents were performed in

subsection 6.2.1, assuming the availability of one of two containment air cooling trains. A failure modes and effects analysis of the containment cooling system is presented in table 6.2.2-3.

D. Capability is provided to periodically test the entire startup sequence of the containment air cooling system. Active components can be tested periodically during plant operation to verify operability. The entire system can be inspected during unit shutdown. Additional information is contained in section 3.1, paragraph 6.2.2.2.4, and the Technical Specifications.

E. The containment air cooling units are tested and demonstrated to perform in a simulated MSLB and LOCA environment. The units are located in a manner to minimize the effects of jet impingement and pipe whip in case of a high-energy line break.

F. The analyses show that the containment cooling system in conjunction with the containment spray system is capable of removing sufficient heat energy and subsequent decay heat from the containment atmosphere to ensure the accident peak pressure is below the containment design pressure. Accident analyses assume the occurrence of a single failure that results in the loss of one air cooling train and one containment spray train. Containment cooler heat removal capacity is provided in table 6.2.2-2.

   Containment accident analysis assumes a constant NSCW temperature equal to the highest anticipated system temperature (95°F) to maximize the calculated containment peak pressure. The assumptions used in calculating this temperature are discussed in subsections 9.2.1 and 9.2.5.

   Curves showing heat removal rates of the containment air cooling system, and containment total pressure, and temperatures as a function of time for minimum ESF performance are given in the figures of subsection 6.2.1.

   Location of the containment air cooling units in different quadrants of the containment, the difference in elevation between suction and discharge points, and the significant flowrate developed (table 6.2.2-1) ensure adequate circulation in the containment following a LOCA which prevents the formation of localized high temperature air pockets or areas of high combustible gas concentration. The mixing capability of this system is supplemented by the containment spray system and natural convection. Additional information is contained in subsection 6.2.5 and section 9.4.

G. The containment cooling system is designed to Seismic Category 1 requirements as specified in section 3.2.

6.2.2.1.4 Testing and Inspection

Fans are tested and rated by the manufacturer in accordance with the standards of the Air Moving and Conditioning Association.

The containment air cooling units are tested and/or analyzed by the manufacturer to ensure operation in the post-accident environmental conditions indicated in section 3.11 and following a safe shutdown earthquake (SSE).

The preoperational testing of the containment air cooling system, as well as its components, demonstrates the initial capability of the equipment. Written test procedures establish minimum acceptance values for all tests. See section 14.2 for details of the preoperational test program.

Following completion of the preoperational integrated leakage rate tests, fans are operated at reduced speed, and the motor currents are monitored with temporarily attached ammeters to verify satisfactory operation in denser than normal atmospheres.

The fans are run and monitored during plant shutdown in accordance with the planned maintenance program.

6.2.2.1.5 Instrumentation Requirements

Containment cooling system controls and instrumentation are discussed in sections 7.3 and 7.5, respectively. The system is designed to function automatically following receipt of an SI signal. Fans and cooling water supply can also be controlled remotely from the control room and from the shutdown panels.

The status of the containment cooling system is displayed in the control room. The NSCW and emergency power supplies are discussed in subsection 9.2.1 and section 8.3, respectively.

6.2.2.2 Containment Spray System

6.2.2.2.1 Design Bases

6.2.2.2.1.1 Safety Design Bases.

A. The containment spray system is designed to withstand the effects of natural phenomena such as earthquakes.

B. The containment spray system is automatically placed in operation on receipt of two out of four containment pressure (high-3) signals.

C. The containment spray system is designed so that a single failure of any active component, assuming loss of offsite power, cannot impair the capability of the system to perform its safety function during the injection phase.

   A single active or passive failure cannot impair the capability of the system to perform its safety function during the recirculation phase.

D. Active components of the containment spray system are capable of being tested during plant operation. Provisions are made for inspection of major components at appropriate times specified in ASME Boiler and Pressure Vessel Code, Section XI.

E. The containment spray system components are designed to remain functional during the accident environment and to withstand the dynamic effect of the accident.

F. The containment spray system in conjunction with the containment cooling system is capable of removing sufficient thermal energy and subsequent decay

heat from the containment atmosphere following the postulated LOCA or MSLB accident to maintain the containment pressure below design values.

G. The containment spray system is designed and fabricated to codes consistent with Regulatory Guide 1.26 as described in table 3.2.2-1 and Seismic Category 1 in accordance with Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

6.2.2.2.1.2 Power Generation Design Bases. The containment spray system has no power generation design bases.

6.2.2.2.2 System Design

6.2.2.2.2.1 General Description. The containment spray system is designed to the codes and standards identified in table 3.2.2-1; flood design is discussed in section 3.4; missile protection is discussed in section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6. Environmental design and equipment qualification is discussed in section 3.11. The actuation system is discussed in section 7.3.

6.2.2.2.2.2 System Description. The containment spray system, shown schematically in drawing 1X4DB131, consists of two pumps, spray ring headers and spray nozzles, valves, and connecting piping. Initially, water from the refueling water storage tank (RWST) is used for the containment spray followed by water recirculated from the containment emergency sump. The recirculated spray is mixed with trisodium phosphate in the containment sump region.

At the RWST empty level alarm the operator should initiate manual switchover of the containment spray pumps to the recirculation mode of operation. See table 6.3.2-7 for a summary of the necessary manual actions for switchover. Adequate transfer allowance is provided to allow the operator to perform the switchover sequence without securing the containment spray pumps. The total amount of borated refueling water injected into the containment by the charging, safety injection, residual heat removal, and containment spray pumps will provide a sump pH of 7.5 or above when mixed with the contents of the trisodium phosphate baskets.

A single failure can occur which may prevent the switchover of one of the two trains. However, a single failure cannot prevent the switchover of both trains simultaneously. The containment pressure transient analysis shows that only one of the two redundant spray trains is necessary to prevent containment pressure from reaching the containment design point. Thus, even if one train is not available following the switchover, the remaining operating train is sufficient to control containment pressure, assuming that four of the eight containment fan coolers are also in operation.

A gas accumulation monitoring and trending process for the Vogtle Unit 1 and 2 ECCS and containment spray systems has been established to meet the requirements of NRC Generic Letter 2008-01.

6.2.2.2.2.3 Component Description. The mechanical components of the containment spray system are described in this section. Component design parameters are given in table 6.2.2-4. Parts of the system in contact with borated water are stainless steel or an equivalent corrosion-resistant material.

Corrosion tests have been performed on the materials that the spray would come in contact with, e.g., the paint on the inside of the containment structure. (Tests are detailed in WCAP-7825 and NUREG CR-3803.) These tests have shown that no significant amount of corrosion products is produced. Those corrosion products or any chemical precipitation of appreciable size that does occur is trapped by the sump filter screen. The screen size is smaller than the line piping, residual heat removal heat exchanger tubes, and the spray nozzles, so that particles which could potentially block the system will be filtered out. The spray nozzle material (stainless steel, SA351) was chosen for its resistance to corrosion. Tests have been performed on this material in the same type of environment that the nozzle would see during spray actuation. (Corrosion tests of austenitic stainless steel are detailed in WCAP-7803 and WCAP-11611.) The resulting corrosion levels were very low.

Stress corrosion does not present a problem since it only becomes a factor under a combination of conditions of high stress levels at high temperatures for extended periods of time. The stress levels in the pumps during operation are relatively low; the working temperature of the pumps is less than 212°F.

The pumps are located outside the containment. The external surfaces of the pumps, as well as the pump motors, are not subjected to the corrosive atmosphere of the spray solution. Only the internal stainless steel surfaces of the pumps are exposed to a corrosive atmosphere.

6.2.2.2.2.3.1 Refueling Water Storage Tank. This tank serves as a source of emergency borated cooling water for injection and containment spray. It is normally used to fill the refueling canal for refueling operations. During all other plant operating periods, it is aligned to the suction of the emergency core cooling pumps and the containment spray pumps. The tank is a concrete tank lined with type 304 stainless steel plates. The nominal tank volume is 715,000 gal. The contents of this tank are protected from freezing by a sludge mixing system which includes an electric circulation heater. Lines and appurtenances to the RWST serving a safety-related function are heat traced as necessary to prevent freezing.

6.2.2.2.2.3.2 Containment Spray Pumps. The containment spray pumps are of the horizontal centrifugal type, driven by electric motors powered from the emergency buses.

The design head of the pumps is sufficient to continue at rated capacity with a minimum level in the RWST against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, line losses, and nozzle pressure losses. The containment spray system is designed so that adequate net positive suction head (NPSH) is provided to the containment spray pumps, in accordance with Regulatory Guide 1.1.

To demonstrate that adequate NPSH is provided for the containment spray pumps, it is only necessary to demonstrate that the NPSH is adequate under the worst limiting conditions. NPSH for the containment spray pumps is evaluated for both the injection and recirculation modes of operation for the DBA. The recirculation mode of operation gives the limiting NPSH requirement for the containment spray pumps, and the NPSH available is determined from the following equation:

$$NPSH_{actual} = (h)_{containment\ pressure} - (h)_{vapor\ pressure} + (h)_{static\ head} - (h)_{loss}$$

To evaluate the adequacy of the available NPSH, the debris generation from a high energy line break within the containment and the resultant impact on the containment spray pump performance was evaluated using the following.

A. The minimum flood level inside the containment based on the RWST water discharged during injection and switchover to recirculation, plus the water volume of 3 accumulator tanks, is at 177 ft 0 in. The minimum flood level also takes into account the flooding of the reactor cavity/incore instrument tunnel via the penetrations in the primary shield.

B. The containment spray pump suction elevation is 121 ft 5 in.

C. The calculated maximum line losses which include losses through pipe fittings, valves, entrance and exit are 10.3 ft at 3200 gal/min, containment spray pump design flow.

D. The quantity of insulation debris generated by the double-ended rupture of the RCS hot leg at its connection to the steam generator, the limiting case, is 432 ft^3.

E. Hydraulic model studies performed on a scale model of the VEGP containment emergency sump configuration showed that the approach velocities to the four sumps during two train operation was essentially the same. Therefore, it is assumed that the volume of debris transported and deposited on one sump screen is one quarter of the total debris generated by the postulated pipe break event. For one RHR and CS pump train operating the volume of debris transported and deposited on one CS sump screen is 41.6% of the total debris.

   However, the volume of debris transported to the sumps is also evaluated as a pump flow-weighted ratio. The most limiting case is reflected in paragraph 6.2.2.2.2.3.2.

F. The maximum containment spray emergency sump screen head loss, assuming 41.6% of the total debris generated is evenly deposited on the sump screen, is less than 16.65 ft.

G. The vapor pressure of the pumped liquid is assumed to be in equilibrium with the containment ambient pressure (i.e., no credit is taken for subcooling of the sump fluid) for sump fluid temperatures of equal to or greater than 211°F:

$$h_{containment\ ambient\ pressure} = h_{vapor\ pressure}$$

The NPSH equation for the recirculation mode when suction is taken from the containment emergency sump becomes:

$$NPSH_{available} = h_{static\ head} - h_{line\ loss} - h_{sump\ screen\ loss}$$

Using the above values, the calculated containment spray pump available NPSH is greater than 36 ft. The required NPSH at 3200 gal/min is 19.5 ft. Therefore, adequate available NPSH margin is provided for proper containment spray pump operation.

Design parameters for these pumps are presented in table 6.2.2-4.

6.2.2.2.2.3.3 Spray Nozzles. The hollow cone spray nozzles are not subject to clogging by particles less than 1/4 in. in size and produce a drop size spectrum with a mean diameter of less than 700 mm at 40 psi differential pressure. During spray recirculation operation, the water is screened through perforated plates with 3/32-in. diameter holes (a small%age – 124 holes – are larger than

3/32-in. diameter but < ¼-in. diameter – see section 6.1.2.) before leaving the containment emergency sump. With the spray pump operating at design conditions and the containment at design pressure, the pressure drop provided across the nozzles exceeds 40 psi. The spray nozzles used in the construction of the containment spray system are designed to withstand differential pressures in excess of those expected to occur as a result of an accident.

The spray nozzles are stainless steel and have a 3/8-in.-diameter orifice.

6.2.2.2.2.3.4 Spray Additive Tank. The spray additive tank has been abandoned in place.

6.2.2.2.2.3.5 Spray Additive Eductors. The spray additive eductors serve only to maintain the spray system pressure boundary integrity.

6.2.2.2.2.3.6 Piping, Valves, and Containment Sumps. The piping used in the construction of the containment spray system is designed to withstand differential pressures in excess of those expected to occur as a result of an accident. The containment sump recirculation lines that run from the containment sump to the containment spray pumps are enclosed within guard pipes from the containment emergency sump floor to the first valve outside the containment. The pipe and guard pipe are capable of withstanding containment pressure and temperature. The guard pipe prevents leakage from the containment if the recirculation line ruptures.

The two containment ECCS sumps are designed in accordance with the requirements of Regulatory Guide 1.82. These sumps are located in a manner that protects them from the effects of high-energy line breaks, and they are separated from each other. The elevation of the containment emergency sumps are selected to allow optimum use of the available coolant. The sump intakes are protected by vertically stacked disk screens. The size of the opening in the screens is based on the minimum flow area through the components that receive coolant from the emergency sumps.

Analyses were conducted to ensure that effects such as reduction of NPSH and screen blockage will not result in degraded pump or system performance. The screens are installed in a manner to facilitate inspection of the structures and pump suction intakes.

There are two Containment Spray (CS) sumps. A screen is installed on each sump. The CS screens are composed of four stacks of 14 disks that are 30-in. long by 30-in. wide by ~40-in. high, four of which provide 590 ft^2 of perforated plate area and 133 ft^2 of circumscribed surface area per sump. Each typical screen disk is a welded assembly of two perforated plates and their structural support components. The screens are designed to withstand the loading for the largest postulated debris pieces, types, and amount. The plate-hole (perforation) diameter of the screen is 3/32 in. (a small%age - 124 - holes are larger than 3/32-in.diameter but none are larger than ¼-in. diameter – see section 6.1.2.) The screen is mounted over the containment sump. Because of these dimensions, it is not considered credible that a screen can plug sufficiently to impede pump suction. In the remote case that particles did traverse the screen, they would pass through the piping pumps and valves as well as the 3/8-in. diameter containment spray nozzle openings without difficulty. The screens bolt to the floor and may be removed by unbolting individual screen sections for inspection during shutdown periods, or the dedicated inspection port may be used.

Water sprayed in the refueling canal from the containment sprays may escape back to the elevation of the emergency sump through two 12-in. drain pipes located at the lowest point of

the refueling canal. The water passes from the canal to a passageway on the containment floor. Spray water from 33 out of 171 nozzles during 1-train operation or a maximum of 66 out of a total of 342 nozzles from 2 containment spray trains in operation falls into the refueling canal for a maximum canal fill rate of approximately 500 gal/min and 1000 gal/min, respectively. This represents less than 20% of the total spray rate in either case.

The drain layout is such that each 12-in. drain line is capable of passing approximately 2000 gal/min; therefore, there is no danger of starving the sump via the refueling canal. The drain piping is isolated during refueling and left open during normal reactor operation. Plant refueling procedures ensure that these drain pipes are opened after refueling prior to plant startup.

One containment spray pump is provided for each train. A single failure therefore leaves one of the two trains in service. The containment spray pumps are located within compartments sealed by watertight doors; a postulated rupture in one train cannot flood the other.

The first of the two motor-operated valves in series on the recirculation lines are totally enclosed in protective chambers, ensuring that all liquid escaping from a damaged or leaky valve does not escape to the outer environment. All materials that can come in contact with recirculation fluid are austenitic stainless steel. The spray headers are located in the proximity of the containment liner of the dome. The spray headers are anchored to the concrete through the liner plate. Each system is designed for SSE and appropriate thermal and dead weight loading conditions. The spray system is designed for maximum coverage in the containment, with the nozzles located and oriented so that the spray will not be blocked by structures or equipment.

6.2.2.2.2.3.7 Motors for Pumps and Valves. The motors for the containment spray system components will be designed in accordance with specifications discussed for the motors in the SI system. (See section 6.3.)

6.2.2.2.2.3.8 Electrical Supply. Details of the emergency bus power sources are discussed in chapter 8.

6.2.2.2.2.4 System Operation. The spray system is actuated by a signal initiated manually from the control room or automatically on coincidence of two of four containment pressure (high-3) signals. These signals start the containment spray pumps and open the discharge valves to the spray headers.

During all modes of operation except refueling, the suction of the pumps is normally aligned to the RWST. The spray pumps continue to draw a suction on the RWST until the later stages of the injection phase. After the ECCS is realigned from injection to recirculation, and when the RWST level reaches empty, the spray pump suction is remote-manually shifted to the containment emergency sumps.

6.2.2.2.2.5 pH Control. Containment spray could be operated for a short period of time and recirculation will never occur. This is during the secondary line break or inadvertent containment spray actuation events. Since the containment spray can be operated only in the injection phase without any recirculation, the spray will be acidic (4.5 pH). If containment spray is actuated and terminated prior to recirculation, a controlled cleanup and inspection of equipment in containment should begin within 5 days of the event. This is necessary to assess and recover from the effects of the acidic containment sprays on equipment in containment prior to reaching unacceptable conditions.

The initial containment spray pH will be approximately 4.5 during the injection phase which can last up to approximately 4.5 hours. After switchover to the recirculation phase, the spray pH will be between 7.5 and 10.5. If the containment spray system is actuated during a LOCA, its operation should continue in the injection phase and through switchover and into the recirculation phase. Operation of the containment spray system in the recirculation mode is necessary to ensure that the containment is resprayed with the solution (7.5 to 10.5 pH) from the emergency sumps. Once this respray occurs, then the containment sprays can be terminated, provided that containment pressures have already decreased to acceptable levels.

The pH of the sump solution will be adjusted to greater than 7.5 within 8 hours of the accident (LOCA) initiation to counteract the buildup of chloride concentrations to critical levels. This is considered a conservatively short period in which to make the pH adjustment, even with the potential for rapidly increasing sump chloride concentrations. The materials in containment are qualified for long term exposure to a high pH solution and will not be adversely affected by short term exposure to a low pH solution.

A summary of procedural requirements for containment spray operation follows:

A. If containment spray is actuated and terminated prior to recirculation, a controlled cleanup and inspection of equipment in containment should begin within 5 days of the event.

B. If containment spray was actuated and has been operated in the recirculation mode, then the operation in the recirculation mode is required for a duration of the 1.5 hours prior to the termination of containment spray to ensure desired pH of the spray solution is reached.

C. If containment spray is actuated and a primary LOCA is indicated, continuous spray for a minimum duration of 2 hours is required. This should include or be followed by operation in the recirculation mode for at least 1.5 hours.

6.2.2.2.3 Safety Evaluation

A. The components of the containment spray system are located inside the Category 1 auxiliary and containment buildings, except for the RWST, which is a Category 1 concrete tank. These buildings and this tank are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, or floods.

B. Operation of the containment spray system is automatically initiated by a coincident two-out-of-four containment (high-3) pressure signal. Use of this safety-related signal provides a reliable indication of a DBA occurring inside the containment. Operation of the spray system may also be manually initiated from the control room. A detailed description of the actuation system is contained in section 7.3.

C. Two trains, each containing a spray pump, spray eductor, and spray header, are supplied borated water from the RWST. The spray pump and motor-operated valves in each train are powered from the corresponding train of onsite emergency power. Failure of any active component in one train will not affect the operability of the redundant train. The containment analyses of LOCA and MSLB accidents performed in subsection 6.2.1 assume the availability of only one train of the containment spray system. The results of a failure modes and effects analysis are provided in table 6.2.2-5.

D. The containment spray system components are qualified to operate in the LOCA and MSLB environments through testing or analysis. Sections 3.9, 3.10, and 3.11 discuss the qualification of mechanical and electrical components.

E. The analyses described in subsection 6.2.1 show that the spray system in conjunction with the containment coolers is capable of removing sufficient heat energy and subsequent decay heat from the containment atmosphere to ensure the accident peak pressure is below the containment design pressure. The analyses assume the operation of only one train of containment coolers and containment spray.

F. The containment spray system is designed to Category 1 and Quality Group B requirements as shown in table 3.2.2-1.

As stated in 1.9.26.2, Westinghouse classifies components according to ANSI N 18.2A-1975. The spray additive tanks have been abandoned in place.

The minimum fall path of water droplets is conservatively assumed to be the distance from the lowest spray ring to the operating deck. Heat transfer calculations presented in chapter 15 show that essentially all spray droplets reach thermal equilibrium at containment design temperature and pressure in a distance considerably less than the minimum fall path. For a detailed description of the analytical methods and models used to assess the performance capability of the containment heat removal systems, refer to the containment integrity analysis presented in chapter 15.

6.2.2.2.4 Testing and Inspection

6.2.2.2.4.1 Containment Spray System.

6.2.2.2.4.1.1 Inspections. The containment spray system is designed to permit periodic determination of proper functioning to demonstrate system readiness as specified in the Technical Specifications.

The pressure containing systems are inspected for leaks from pump seals, valve packings, flange joints, and safety valves during system testing. The components of the system outside the containment are accessible for leaktightness inspection during periodic flow tests.

Examinations/inspections of the pressure retaining piping welds should be performed in accordance with the requirements of applicable codes and standards.

6.2.2.2.4.1.2 Preoperational Testing. The objective of preoperational testing is to:

A. Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections.

B. Verify that the proper sequencing of valves and pumps occurs on initiating the containment spray signal, and demonstrate the proper operation of remotely operated valves.

C. Verify the operation of the spray pumps; each pump is run at minimum flow, and the flow is directed through the normal path back to the RWST. During this time, the miniflow is measured to verify required flow.

6.2.2.2.4.1.3 Operational Testing. The objective of the periodic testing is to:

A. Verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal, and demonstrate the proper operation of remotely operated valves.

B. Verify the operation of the spray pumps; each pump is run at minimum flow, and the flow is directed through normal path back to the RWST.

C. Automatic transfer of powered components to the emergency diesel generators can be demonstrated during an integrated system test conducted when the plant is cooled down and the residual heat removal system is in operation.

6.2.2.2.4.2 Containment Spray Additive Subsystem (Abandoned in place).

6.2.2.2.5 Instrumentation Application

The status of the containment spray system is displayed in the control room. Using a combination of alarms and monitor lights, the operator is alerted to a maloperation of this equipment both during normal operation and post-accident conditions.

Operation of the containment spray system is demonstrated by monitoring spray water temperature and RWST level. Locally mounted pressure indicators are provided at the spray pumps' suction and discharge to verify pump performance.

The activation signal generating equipment fully meets Institute of Electrical and Electronic Engineers Standard 279 as to operation, diversity, separation of power supplies (diesels, instrument power), etc.; details are discussed in chapter 7.

6.2.2.3 References

1. Deleted.
2. Deleted.

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

Based on the performance of the fission product removal and control systems discussed in section 6.5 and the acceptable radiological consequences following a loss-of-coolant accident or fuel handling accident inside the containment presented in chapter 15.0, a secondary containment is not required for VEGP.

6.2.4 CONTAINMENT ISOLATION SYSTEM

The containment isolation system consists of the piping, valves, and actuators required to isolate the containment following a loss-of-coolant accident (LOCA), steam line rupture, fuel handling accident inside the containment, small breaks in the reactor coolant system (RCS), or releases of radioactivity from systems within the containment. The design of the containment isolation system satisfies the requirements of TMI Action Plan Task II.E.4.2 as described in the following paragraphs.

6.2.4.1 Design Bases

Protection of the containment isolation system from wind and tornado effects is discussed in section 3.3. Flood design is discussed in section 3.4. Missile protection is discussed in section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6. Environmental design is discussed in section 3.11.

6.2.4.1.1 Safety Design Bases

A. In the event of a LOCA, the containment isolation system provides isolation of lines penetrating the containment which are not required for operation of the engineered safety features (ESF) systems to minimize the release of radioactive materials to the atmosphere.

B. Upon failure of a main steam line, the containment isolation system isolates the steam generators as required to prevent excessive cooldown of the RCS or overpressurization of the containment.

C. To control the release of radioactivity to the outside atmosphere, the containment isolation system isolates the containment atmosphere following a fuel handling accident inside the containment.

   During refueling operations, manual containment ventilation isolation (CVI) is permitted, along with open personnel and emergency air lock doors. Manual CVI capability, using individual valve hand switches, is performed during these refueling operations as described in paragraph 6.2.4.5.

D. The containment isolation system is designed in accordance with 10 CFR 50, Appendix A, General Design Criterion 54.

E. Each line which penetrates the containment and which either is a part of the reactor coolant pressure boundary (RCPB) or connects directly to the containment atmosphere or does not meet the requirements for a closed system as defined in item F below, except instrument sensing lines, is provided with containment isolation valves in accordance with 10 CFR 50, Appendix A, General Design Criteria 55 and 56.

F. Each line which penetrates the containment and is neither part of the RCPB nor connected directly to the atmosphere of the containment and which satisfies the requirements of a closed system is provided a containment isolation valve in accordance with 10 CFR 50, Appendix A, General Design Criterion 57. A closed system is not a part of the RCPB nor connected directly to the atmosphere of the containment and meets the following additional requirements:

   1. The system is protected against missiles and the effects of high energy line break.
   2. The system is designed to Seismic Category 1 requirements.
   3. The system is designed to American Society of Mechanical Engineers Section III, Class 2 requirements.
   4. The system is designed to withstand temperatures at least equal to the containment design temperature.
   5. The system is designed to withstand the external pressure from the containment structural acceptance test.

6. The system is designed to withstand the design basis accident transient and environment.

G. The containment pressure transmitters and reactor vessel level instrumentation system (RVLIS) are designed in accordance with Nuclear Regulatory Commission (NRC) Regulatory Guide 1.141. Six containment pressure sensors are provided as sealed systems with bellow seals inside the containment, liquid filled capillaries between the seals, and the sensing element outside containment. RVLIS consists of six level sensors and has bellow seals inside the containment, liquid filled capillaries between the seals, and a secondary isolator seal outside the containment between the containment penetration and the transmitter. These instrument lines are closed systems both inside and outside containment, are designed to withstand the containment pressure and temperature conditions following a loss of coolant accident, and are designed to withstand dynamic effects.

H. The containment isolation system is designed to remain functional following a safe shutdown earthquake (SSE).

6.2.4.1.2 Power Generation Design Basis

The containment isolation system as a whole has no power generation design basis. Power generation design bases associated with individual components of the containment isolation system are discussed in the section describing the system of which they are an integral part.

6.2.4.2 <u>System Description</u>

6.2.4.2.1 General Description

Each piping system which penetrates the containment is provided with containment isolation features which serve to minimize the release of fission products following a design basis accident. Provisions are made to allow for passage of emergency fluid through the boundary following a postulated accident. Figure 6.2.4-1 provides the arrangement for each piping penetration. NRC Standard Review Plan 6.2.4 and Regulatory Guide 1.141 provide acceptable alternative arrangements to the explicit arrangements given in General Design Criteria 55, 56, and 57. Each penetration is designed so that in the event that a single failure is postulated, the containment integrity is maintained. Table 6.2.4-1 lists each penetration and provides a summary of the containment penetration/isolation valve information.

For those systems which have automatic isolation valves or for which remote-manual isolation is provided, paragraph 6.2.4.5 describes the power supply and associated actuation system. Power-operated (air, motor, electrohydraulic, or solenoid) containment isolation valves have position indication in the control room.

Two modes of valve actuation are considered in table 6.2.4-1. The actuation signal which occurs directly as a result of the event initiating containment isolation is designated as the primary actuation signal. The post-accident valve position is a consequence of the primary actuation signal. If a change in valve position is required at any time following primary actuation, a secondary actuation signal is generated which places the valve in an alternative position. The closure times for automatic containment isolation valves are provided in table 6.2.4-1. Containment isolation valves required to be operable by Technical Specification

3.6.3, Containment Isolation Valves, are demonstrated operable with isolation times as shown in FSAR table 6.2.4-2.

The containment purge system is designed in accordance with Branch Technical Position CSB 6-4 as described in table 9.4.6-4. As described in subsection 9.4.6, the 14-in. minipurge lines may be open during normal plant operation and are provided with isolation valves capable of 5-s closure against the peak calculated containment pressure following a LOCA. The 24-in. purge lines are open only during a cold shutdown condition and are provided with an isolation valve capable of 10-s closure. An analysis of the radiological consequences and the effect on the containment backpressure due to the release of containment atmosphere are discussed in chapter 15 and paragraph 6.2.1.5, respectively.

In the event of a LOCA, the secondary shield wall and other protective features prevent any missiles or high energy line break effects from damaging or degrading the performance capability of the containment isolation system. Sections 3.5 and 3.6 discuss in detail the missiles and pipe break effects, and section 3.8 discusses the internal structures, including the secondary shield wall. The actuators for power-operated containment isolation valves inside the containment are located above the maximum anticipated containment water level. In addition, lines associated with those penetrations which are considered closed systems inside the containment are protected from the effects of a LOCA.

Provisions are made to ensure that closure of the containment isolation valves is not inhibited by entrapped debris in the valve body. For the majority of the systems, the fluid is demineralized water; thus, process fluid quality does not affect valve operation. For containment minipurge lines, screens are provided in the lines inboard of the isolation valves. For the containment sump lines, including the containment emergency sump, screens are provided to prevent large debris from entering the system.

Other defined bases for containment isolation are provided in NRC Standard Review Plan 6.2.4 and Regulatory Guide 1.141. Conformance with Regulatory Guide 1.141 is provided to the extent specified in this section and in subsections 6.2.5 and 6.2.6. For the emergency core cooling system (ECCS) and containment spray system penetrations, the acceptability of the alternative arrangement relies upon provisions for the detection of possible leakage from these lines outside the containment. Subsection 9.3.3 describes the leak detection provisions that have been made in the plant drainage system. Other provisions, such as containment water level and system flow, temperature, and pressure instrumentation may be used by the operator.

The containment penetrations associated with the secondary side of the steam generators are not subject to General Design Criterion 57. The valves associated with these penetrations do not receive a containment isolation signal and are not credited with effecting containment isolation in the safety analyses. The barriers against fission product release to the environment are the steam generator tubes and the piping associated with the steam generators.

In addition to containment penetration isolation, table 6.2.4-1 also contains systems which are required for post-LOCA mitigation. Since these systems, such as the ECCS, perform additional safety-related functions, they are associated with ESF and are so indicated in table 6.2.4-1.

6.2.4.2.2 Component Description

Codes and standards applicable to the piping and valves associated with containment isolation are listed in table 3.2.2-1. Containment penetrations are classified as Quality Group B and Seismic Category 1.

Section 3.11 provides the post-LOCA environment that is used to qualify the operability of power-operated isolation valves located inside the containment.

The containment penetrations are designed to meet the stress requirements of NRC Branch Technical Position MEB 3-1 and the classification and inspection requirements of NRC Branch Technical Position APCSB 3-1, as described in section 3.6. Section 3.8 discusses the interface between the piping system and the containment liner.

6.2.4.2.3 System Operation

During normal operation, many penetrations are not isolated. Lines which are not required for the passage of emergency fluids are automatically isolated upon receipt of isolation signals, as discussed in paragraphs 6.2.4.3 and 6.2.4.5 and chapter 7. Essential lines which penetrate the containment are closed loops within the containment or provide flow paths into or out of the RCS and can be isolated by remote-manual operation when dictated by the emergency system functional requirements. Lines not in use during power operation are normally closed and remain closed under administrative control during reactor operation.

Upon detection of abnormal radioactivity levels indicative of a fuel handling accident during refueling or other release, the isolation valves in the containment purge system are closed to minimize any fission product release to the environment.

6.2.4.3 Design Evaluation

Safety evaluations are lettered to correspond to the safety design bases.

A. Containment isolation signals automatically isolate process lines which are nonessential as identified in table 6.2.4-1. Nonessential lines are those lines which are not required to mitigate or limit an accident and which, if required at all, would be required for long term recovery only; e.g., days or weeks following an accident.

   Lines which are required to mitigate an accident or which, if unavailable, could increase the magnitude of the event are designated as essential lines. Table 6.2.4-1 identifies the associated line as essential or nonessential and shows the automatic isolation signal for each penetration, if applicable.

   The containment isolation system utilizes diversity in the parameters sensed for the initiation of containment isolation. The two redundant train-oriented containment isolation phase A signals (CIA-A, CIA-B) are initiated on receipt of any of the following signals:

   1. Any signal initiating a safety injection:

      - Manual safety injection actuation.
      - High containment pressure (high-1).
      - Low steam line pressure.
      - Low pressurizer pressure.

   2. Manual containment isolation actuation.

B. Upon failure of a main steam line, the steam generators are isolated to prevent excessive cooldown of the RCS or overpressurization of the containment.

The two redundant train-oriented steam line isolation signals (SLI-A, SLI-B) are initiated upon receipt of any of the following signals:

1. High steam line pressure rate.
2. Low steam line pressure.
3. Containment high-2 pressure.
4. Manual actuation.

For main steam line breaks resulting in a high steam line pressure rate or containment high-2 pressure signal, only the main steam line isolation valves (MSIVs) and MSIV bypass valves are shut to prevent excessive cooldown of the RCS. When the main steam line break causes a low steam line pressure signal, a safety injection signal (followed by containment isolation) is generated as well as the steam line isolation signal.

The main steam line isolation valves, MSIV bypass valves, and piping are designed to prevent uncontrolled blowdown from more than one steam generator. The main steam line isolation valves and MSIV bypass valves will shut fully within 5 s after SLI is initiated. The blowdown rate is restricted by steam flow restrictors located within the steam generator outlet steam nozzles in each blowdown path. For main steam line breaks upstream of an isolation valve, uncontrolled blowdown from more than one steam generator is prevented by the isolation valves in the unaffected steam lines and by the isolation valve in the affected line. For main steam line breaks downstream of an isolation valve, blowdown from more than one steam generator is prevented by the main steam isolation valves on each main steam line.

Failure of any one of the above components relied upon to prevent uncontrolled blowdown of more than one steam generator will not permit a second steam generator blowdown to occur. Piping restraints and pipe whip barriers between the main steam lines prevent a rupture in one line from causing a blowdown from more than one steam generator. No single active component failure will result in the failure of more than one main steam isolation valve to operate. Redundant main steam isolation signals, described in section 7.3, are fed to redundant parallel activation cylinder vent valves and redundant series actuation cylinder air supply valves to ensure isolation valve closure in the event of a single isolation signal failure.

The effect on the RCS after a steam line break resulting in single steam generator blowdown and the offsite radiation exposure after a steam line break outside containment are discussed in detail in chapter 15. The containment pressure transient following a main steam line break inside containment is discussed in section 6.2.

C. The containment purge system is automatically isolated following an abnormal release of radioactivity in the containment by either of two redundant train-oriented containment ventilation isolation signals (CVI-A, CVI-B) generated upon receipt of any of the following:

1. Any signal resulting in a safety injection.
2. Containment high area radiation.
3. Containment high radioactive air particulate.

4. Containment high radioactive gas.
5. Containment high iodine concentration.
6. Manual actuation of either containment spray or containment isolation phase A.

The preaccess purge supply and exhaust valves in the 24-in. lines, which are only open in the cold shutdown condition, are designed to shut in less than 10 s. The minipurge line isolation valves, which may be open during normal operation, shut in less than 5 s.

D. The containment isolation system is designed in accordance with 10 CFR 50, Appendix A, General Design Criterion 54. Leakage detection capabilities and the leakage detection test program are discussed in subsection 6.2.6. Valve operability tests are also discussed in subsection 3.9.6. Redundancy of valves and reliability of the isolation system are ensured by conformance with the other safety design bases stated in section 6.2. Redundancy and reliability of the actuation system are covered in section 7.3.

The use of motor-operated valves which fail as is upon loss of actuating power in lines penetrating the containment is based upon the consideration of what valve position ensures the greatest plant safety. Furthermore, each of these valves that fails as is provided with redundant backup valves to ensure that no single failure will prevent the system as a whole from performing its isolation function; e.g., a check valve inside the containment and motor-operated valve outside the containment or two motor-operated valves in series, each powered from a separate ESF bus.

E. Lines which penetrate the containment and which either are part of the RCPB, connect directly to the containment atmosphere, or do not meet the requirements for a closed system, except instrument sensing lines, are provided with one of the following valve arrangements conforming to the requirements of 10 CFR 50, Appendix A, General Design Criteria 55 and 56, as follows:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment.
2. One automatic isolation valve inside and one locked closed isolation valve outside containment.
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment.)
4. One automatic isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment.)

Isolation valves outside containment are located as close to the containment as practical, and upon loss of actuating power, air-operated automatic isolation valves fail closed.

F. Each line which penetrates the containment and is neither part of the RCPB nor connected directly to the containment atmosphere and satisfies the requirements of a closed system has at least one containment isolation valve which is either automatic, locked closed, or capable of remote-manual operation. The valve is outside the containment and located as close to the containment as practical. A

simple check valve is not used as the automatic isolation valve. This design is in compliance with 10 CFR 50, Appendix A, General Design Criterion 57.

G. Instrument lines penetrating the containment and the containment pressure instrument lines are designed in accordance with NRC Regulatory Guide 1.141.

H. The containment isolation system is designed in accordance with Seismic Category 1 requirements as specified in section 3.2. The components (and supporting structures) of any system, equipment, or structure which is non-Seismic Category 1 and whose collapse could result in loss of a required function of the containment isolation system through either impact or flooding are analytically checked to determine that they will not collapse when subjected to seismic loading resulting from an SSE.

Air-operated isolation valves fail in the shut position upon loss of air if they are not required to operate after a design basis accident. Containment isolation system valves required to be operated after a design basis accident are powered by the Class 1E electric power system.

6.2.4.4 Tests and Inspections

Preoperational testing is described in chapter 14. The containment isolation system is testable through the operational sequence that is postulated to take place following an accident, including operation of applicable portions of the protection system and the transfer between normal and standby power sources.

The piping and valves associated with the containment penetration are designed and located to permit preservice and inservice inspection in accordance with ASME Section XI, as discussed in section 6.6.

Each line penetrating the containment is provided with testing features to allow containment leak rate tests in accordance with 10 CFR 50, Appendix J, as discussed in subsection 6.2.6.

6.2.4.5 Instrumentation Application

The generation of CIA or CVI signals which automatically isolate the appropriate containment isolation valves is described in section 7.3.

For those valves for which automatic closure is not desired, based on the system safety function, remote-manual operation is available from the control room.

Containment isolation valves which are equipped with power operators and which are automatically actuated may also be controlled individually by positioning hand switches in the control room. Also, in the case of certain valves with actuators, a manual override of an automatic isolation signal is installed to permit manual control of the associated valve. The override control function can be performed only subsequent to resetting of the actuation signal; that is, deliberate manual action is required to change the position of containment isolation valves in addition to resetting the original actuation signal. The design does not allow ganged reopening of the containment isolation valves. Reopening of the isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis. Safety injection signals take precedence over manual overrides of other isolation signals, for example, a safety injection signal causes isolation valve closure even though the high radiation signal is being overridden by the operator. Overrides are input to the system status monitoring panel, described in subsection 7.5.5. Containment isolation valves with power operators are provided with

open/closed indication, which is displayed in the control room. The valve mechanism also provides a local, mechanical indication of valve position.

In mode 6 during core alterations and movement of irradiated fuel assemblies inside containment, automatic or system-level manual initiation CVI capability no longer applies. During these refueling operations, manual containment ventilation isolation is permitted, along with open personnel and emergency air lock doors. Manual CVI capability, using individual valve hand switches, is performed during these refueling operations. The following conditions apply:

- One personnel air lock door and one emergency air lock door must be operable, and
- At least 23 feet of water is maintained above the reactor vessel flange, and
- A designated individual is available to close the doors. The emergency air lock will not normally be open during core alterations or fuel movement inside containment. Therefore, in the event the emergency air lock is open at the same time the personnel air lock is open, a separate individual shall be responsible for closing the emergency air lock (within 15 minutes) in addition to the individual designated to close the personnel air lock.

All power supplies and control functions necessary for containment isolation are Class 1E, as described in chapters 7 and 8.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Technical Specifications (TS) Amendment Number 134/113 eliminated the requirements regarding containment hydrogen recombiners and relaxed the requirements for hydrogen monitors. The hydrogen recombiners and monitors have been deleted from the TS. The hydrogen monitors are included in the post accident monitoring instrument program.

Following a loss-of-coolant accident (LOCA), hydrogen may be produced inside the reactor containment by radiolysis of the core and sump solutions, by corrosion of aluminum and zinc, by reaction of the Zircaloy fuel cladding with water, and by release of the hydrogen dissolved in the reactor coolant and contained in the pressurizer vapor space. To ensure that the containment hydrogen concentration is maintained at a level low enough to preclude endangering containment integrity, a combustible gas control system is provided. This subsection describes the systems that are provided in accordance with General Design Criterion 41 to control the buildup of hydrogen within the containment.

Five mechanisms for monitoring and controlling hydrogen inside the containment are considered in the VEGP design:

- Hydrogen recombiners.
- Post-LOCA containment hydrogen purge.
- Post-LOCA cavity hydrogen purge.
- Containment hydrogen monitoring.
- Containment hydrogen mixing.

6.2.5.1 Design Bases

6.2.5.1.1 Electric Hydrogen Recombiners

The following design bases apply to the electric hydrogen recombiners:

A. The recombiners are designed to sustain all normal and accident loads including safe shutdown earthquake and pressure transients from a design basis LOCA.

B. The recombiners are designed for a lifetime of 40 years, consistent with that of the plant.[a]

C. All materials used in the recombiners are selected to be compatible with the environmental conditions inside the reactor containment during normal operation or during accident conditions.

D. Process capacity is such that the containment hydrogen concentration will not exceed 4 volume% based on the Nuclear Regulatory Commission (NRC) TID release model as indicated in Regulatory Guide 1.7.

E. Two redundant electric hydrogen recombiners are provided to meet the single-failure criterion.

6.2.5.1.2 Post-LOCA Containment Hydrogen Purge System

A. The containment post-LOCA purge exhaust system constitutes a 100% single train backup to the hydrogen recombiner. It is designed as Seismic Category 1 except for ductwork inside the containment upstream of the containment isolation valve and ductwork at the filter inlet.

B. The post-LOCA containment pressure provides the motive force to purge the containment.

C. This system may be used post-LOCA to maintain hydrogen levels in the containment below 4 volume% in conjunction with a portable air compressor through the Seismic Category 1 portion of the service air piping.

D. The redundant motor-operated isolation valves located inside the containment are in parallel and powered from the 480-V, Class 1E buses. All other system components are powered from the normal ac bus. The outside isolation valve is locked closed and manually controlled by the operator. The isolation valves inside the containment have position indication in the control room and are normally closed.

E. The system, including the isolation valves and the filtration unit, is designed to be manually started for operation.

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[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

6.2.5.1.3 Post-LOCA Cavity Hydrogen Purge System

A. The post-LOCA cavity purge system prevents hydrogen pocketing in the reactor cavity after a LOCA by supplying air to the reactor cavity for dilution.

B. The post-LOCA cavity purge system is capable of accomplishing its function with the single failure of an active component.

C. The post-LOCA cavity purge system is capable of performing its function with Class 1E power, each redundant train being connected to separate Class 1E safety buses.

D. The system is designed to automatically start upon receipt of a safety injection signal.

E. The system is designed to meet Seismic Category 1 requirements.

6.2.5.1.4 Containment Hydrogen Monitoring System

A. The hydrogen monitoring system is designed as a Class 1E, Seismic Category 1 system. It is designed to retain its integrity and operability following a design basis accident (DBA).

B. All materials and equipment required by this system are selected to be compatible with the environmental conditions anticipated during accident operation and are suitable for a lifetime consistent with that of the plant[a].

C. The system samples containment air, providing the means to measure the containment hydrogen concentration and to alert the operator in the event that a high hydrogen concentration is detected, in accordance with the requirements of Regulatory Guide 1.7.

D. The hydrogen monitoring system consists of two identical units that are completely independent of each other and are powered from independent Class 1E power sources. Assuming a single failure and compensatory operator actions from the control room, capability is available to monitor the hydrogen concentration in the containment.

E. Proper shielding and other provisions are incorporated into the design to ensure that personnel exposure does not exceed the limits of General Design Criterion 19 and that the required radiological analysis can be performed on the containment air sample.

6.2.5.1.5 Containment Hydrogen Mixing

The following design bases apply to mechanisms or systems for mixing of hydrogen-bearing gases inside the reactor containment:

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[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

A. Local hydrogen concentrations inside the reactor containment shall be maintained at less than 4 volume%.

B. Any active systems required for mixing containment air should meet the same redundancy, environmental, seismic, and quality requirements as the hydrogen recombiner system as described in paragraph 6.2.5.1.1.

6.2.5.2 System Design

6.2.5.2.1 Electric Hydrogen Recombiners

The applicable codes and standards used in the design of the electric hydrogen recombiner are listed in table 6.2.5-1, and a typical electrical recombiner is shown in figure 6.2.5-1.

Each recombiner system consists of a control panel located in the control building, a power supply cabinet located on level B of the control building, and a recombiner located above the operating deck at el 261 ft in the containment. There are no moving parts or controls inside the containment. Heating air within the unit causes airflow by natural convection. The recombiner is a completely passive device.

To regulate the power supply to the recombiner, the power supply cabinet located in the control building contains an isolation transformer and a controller. This equipment will not be exposed to the post-LOCA environment. The controls for the power supply are located in the control building beside the power supply panel and are manually actuated.

Each hydrogen recombiner consists of the following components:

A. A preheater section, consisting of a shroud placed around the central heaters to take advantage of heat conduction through the central walls, for preheating incoming air.

B. An orifice plate to regulate the rate of airflow through the unit.

C. A heater section, consisting of four banks of metal-sheathed electric resistance heaters, to heat the air flowing through it to hydrogen-oxygen recombination temperatures.

D. An exhaust chamber which mixes and dilutes the hot effluent with containment air to lower the temperature of the discharge stream.

E. An outer enclosure to protect the unit from impingement by containment spray.

The hydrogen recombiner has no need for external services except electrical power.

The containment atmosphere is heated within the recombiner in a vertical duct, causing it to rise by natural convection. As it rises, replacement air is drawn through intake louvers downward through a preheater section which will temper the air and lower its relative humidity. The preheated air then flows through an orifice plate, sized to maintain a 100-sf^3/min flowrate, to the heater section. The airflow is heated to a temperature above 1150°F, the reaction temperature for the hydrogen-oxygen reaction. Any free hydrogen present reacts with atmospheric oxygen to form water vapor. After passing through the heater section, the flow enters a mixing section, which is a louvered chamber where the hot gases are mixed and cooled with containment atmosphere before the gases are discharged directly into the containment. The air discharge louvers are located on three sides of the recombiner. To avoid short-circuiting previously processed air, no discharge louvers are located on the intake side of the recombiner.

Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters (paragraph 6.2.5.4) but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit cannot occur.

Two recombiners are provided to meet the requirements for redundancy and independence. Each recombiner is powered from a separate safeguard bus and is provided with a separate power panel and control panel. This system and the other safety-related subsystems are not interdependent.

The unit is manufactured of corrosion-resistant, high-temperature material. The electric hydrogen recombiner uses commercial-type electric resistance heaters sheathed with Incoloy-800, which is an excellent corrosion-resistant material for this service. The recombiner heaters operate at significantly lower power densities than similar heaters used in commercial practice.

The recombiner is operated manually from a control panel located in the control building. The recombiner, power supply panel, and control panel are shown schematically in figure 6.2.5-2. The power panel for the recombiner contains an isolation transformer and a controller to regulate power into the recombiner. This equipment is not exposed to the post-LOCA containment environment.

To control the recombination process, the correct power input to bring the recombiner above the threshold temperature for recombination is set on the controller. The correct power required for recombination depends upon containment atmosphere conditions and is determined when recombiner operation is required. A thermocouple readout instrument is also provided in the control panel to monitor temperatures in the recombiner.

6.2.5.3 System Design

6.2.5.3.1 Containment Hydrogen Purge System.

The post-LOCA containment hydrogen purge system is provided as a backup means of controlling hydrogen inside the containment. It provides a means of purging the hydrogen from the containment and is intended as a backup to the hydrogen recombiner system. The exhaust filters and exhaust duct to plant vent are designed as Seismic Category 1.

The system consists of an exhaust penetration line and a filtered exhaust system; it is shown in drawings 1X4DB213-1 and 1X4DB213-2. Design data for principal system components are presented in table 6.2.5-3. The containment isolation valves and interconnecting piping are Seismic Category 1; all other portions of the system are Seismic Category 2.

The purge exhaust filter unit includes in the direction of airflow:

- A demister.
- Electrical heating coil.
- High-efficiency particulate air (HEPA) prefilter.
- A 4-in. charcoal adsorber.
- A HEPA afterfilter.

The hydrogen purge exhaust filter unit is located in the equipment building, and the hydrogen purge intake point is located in the containment dome. The ductwork is fastened to the inside of the containment and routed through a containment penetration. The flowrate through the filter unit is 500 ft^3/min.

The system is actuated manually. The operator unlocks and opens the manual isolation valve located outside the containment. From the control room, the operator opens the remote-manual isolation valves located inside the containment. The outward purge flow is due to the pressure differential existing between the containment atmosphere and the environment. Periodically, the operator will dilute the containment atmosphere by charging air into the containment via the instrument and service air system.

6.2.5.3.2 Post-LOCA Cavity Purge System

The post-LOCA cavity purge system is designed to prevent hydrogen pocketing in the reactor cavity following a LOCA by supplying air to the reactor cavity for dilution of the hydrogen released in the cavity area. The system fans take a suction on the atmosphere within the generator compartments and discharge the air into the cavity above (el 193 ft 2 in.) and below (169 ft 9 in.) the reactor vessel nozzles. The air flows out of the cavity along the reactor vessel nozzles or through the ventilation openings surrounding the seal ring.

The system consists of two 100%-capacity fans. Each fan is powered from an independent Class 1E power supply. The fans automatically start on a safety injection signal. There is an independent discharge pipe which routes the dilution air from the fans to the cavity. A failure modes and effects analysis is provided in table 6.2.5-2. Distribution of the air in the cavity area is accomplished via common discharge headers. All portions of the system are designed to Seismic Category 1 requirements.

Design data for principal components are provided in table 6.2.5-4. The system is schematically shown in drawing 1X4DB214-2.

6.2.5.3.3 Containment Hydrogen Monitoring System

Each redundant hydrogen monitoring train in the hydrogen monitoring system consists of a hydrogen analyzer and two associated sample lines with solenoid-operated isolation valves inside and outside the containment. These sampling lines are designed to be free of water traps (runs where liquid could accumulate) and are equipped with sufficient heat tracing to prevent condensation from the sample being supplied to the analyzers.

After the sample has been analyzed, it is returned to the containment. The analyzers are located in accessible areas outside the containment. The hydrogen monitoring subsystem piping is in accordance with the criteria of Regulatory Guide 1.26, Quality Group B. Solenoid-operated isolation valves are arranged to obtain samples from two locations within the containment for each train. The operator may select either of these sampling points from the main control room.

The operation of the hydrogen gas analyzer is based on the measurement of thermal conductivity of the gaseous containment atmosphere sample. The thermal conductivity of the gas mixture changes in proportion to the changes in the concentration of the individual gas constituents of the mixture. The thermal conductivity of hydrogen is far greater (approximately seven times the thermal conductivity of air) than any other gases or vapors expected to be present. This operation of the hydrogen monitoring system is not limited due to radiation,

moisture, or temperature expected at the equipment location. The monitors are designed to function under design pressure conditions of -2 to 60 psig.

The containment hydrogen monitors are aligned for operation within 60 minutes after initiating safety injection following a LOCA. Accurate indication of hydrogen concentration is available within 30 min of initiating flow through the monitors. This is accomplished by operating the monitors in standby during normal plant operation. Therefore, indication of containment hydrogen concentration is available to the operators within 90 minutes of initiating safety injection following a LOCA.

The range of the monitors is 0 to 10 volume% with an accuracy of ±5.0% of scale.

The output signal of the hydrogen monitors is indicated and alarmed locally as well as indicated, recorded, and alarmed in the control room. In addition to the high hydrogen alarm, a common malfunction alarm is located in the control room to indicate loss of power, low gas pressure, low analyzer chamber temperature, analyzer cell failure, or high hydrogen concentration.

Design data for principal system components are presented in table 6.2.5-5. The system is schematically shown in drawings 1X4DB213-1 and 1X4DB213-2.

The hydrogen monitoring system meets the requirements of TMI Action Plan Task II.F.1 with the clarification that accurate indication of containment hydrogen concentration is available to the operators within 90 minutes of initiating safety injection following a LOCA.

6.2.5.3.4 Containment Hydrogen Mixing

Hydrogen mixing is facilitated by the containment fan coolers, which take suction from above the operating deck and discharge to the lower levels of the containment. Functional descriptions of the containment coolers are provided in subsections 6.2.2 and 9.4.6. A flow diagram for the containment coolers is provided in drawings 1X4DB251-1, 1X4DB252, and 1X4DB253-1.

In addition, the post-LOCA cavity purge system described in paragraph 6.2.5.2.3 is available for hydrogen mixing.

6.2.5.4 Design Evaluation

6.2.5.4.1 Hydrogen Production and Accumulation

6.2.5.4.1.1 Zirconium-Water Reaction. A major source of hydrogen immediately following a LOCA is caused by the reaction of the Zircaloy fuel cladding with water. The extent of the zirconium-water reaction depends upon the effectiveness of the emergency core cooling systems (ECCS). An evaluation of the VEGP ECCS shows the zirconium-water reaction to be less than 0.3%.

Zirconium reacts with steam according to the following equation:

$$Zr + 2\ H_2O \rightarrow Zr\ O_2 + 2\ H_2 + \text{Heat}$$

The hydrogen produced is calculated as follows:

$$\frac{2\ \text{lb}-\text{mole H}_2/\text{lb}-\text{mole Zr}}{91.22\ \text{lb Zr/lb}-\text{mole Zr}} = \frac{0.022\ \text{lb}-\text{mole H}_2}{\text{lb Zr}}$$

The NRC model suggested in Regulatory Guide 1.7 and Standard Review Plan 6.2.5 conservatively assumes a 1.5% zirc-water reaction (five times the maximum amount calculated in the ECCS evaluations).(1) There are approximately 45,914 lb of zirconium metal in the reactor core. The hydrogen produced by the reaction of 689 lb of zirconium is 15.15 lb-moles. This hydrogen is assumed to be immediately released to the containment atmosphere.

6.2.5.4.1.2 Radiolysis Core and Sump Solutions. Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the equation:

$$H_2O \leftrightarrow H_2 + \frac{1}{2} O_2$$

An extensive program was conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the DBA. During the investigation it became apparent that post-accident conditions in the containment create two distinct radiolytic environments. One environment exists inside the reactor vessel, where radiolysis can occur when energy emitted by decaying fission products in the fuel is absorbed by the solution pumped through the reactor to cool the core. The other environment exists outside the reactor vessel, in the containment sump solution, where radiolysis can also occur when decay energy emitted by dissolved fission products is absorbed by the sump solution. The two basic differences between the core environment and the sump environment that affect the rate of hydrogen production are the rate of energy absorption and the type of flow regime. The results of these investigations are discussed in reference 1.

The rate of hydrogen production by radiolysis depends upon the rate of energy absorption by the solution. A detailed analysis of energy deposition in the reactor core where decaying fission products are retained in the fuel shows that beta radiation (which represents roughly 50% of the total decay energy) is emitted at an energy level too low to permit its penetration of the fuel and cladding. As a result, roughly 50% of the total decay energy emitted by fission products in the fuel is absorbed by the fuel and cladding and therefore does not contribute significantly to the rate of energy absorption by the water. Furthermore, approximately 7% of the gamma energy is absorbed by the core solution; the rest is absorbed by the fuel, cladding, or other core components.

In the containment sump, where fission products are assumed to be dissolved in the sump solution, energy is emitted directly to the solution. Since the depth of the sump is relatively large compared to the penetrating capability of even gamma energy, effectively 100% of the decay energy of the fission products dissolved in solution is absorbed by the solution. The other significant difference between the core and sump environment is the type of flow regime to which the products of radiolysis are exposed.

Radiolytic decomposition of water is a reversible reaction. In the core, where the products of radiolysis are continuously flushed away by the circulation of cooling solutions, there is little chance for hydrogen and oxygen to accumulate. Consequently, recombination of hydrogen and oxygen is assumed not to occur because significant quantities of the two reactants are not available. The sump, however, is a relatively deep and static environment, where the products of radiolysis are removed by molecular diffusion. Experimental tests simulating sump conditions demonstrate that there is significant reverse reaction in the sump. Hence, there is an apparent reduction in the quantity of hydrogen produced per unit energy absorbed.

The results of Westinghouse and Oak Ridge National Laboratory studies indicate maximum hydrogen yields of 0.44 molecules per 100 eV for core radiolysis and 0.3 molecules per 100 eV for sump radiolysis. The results of these studies are published in references 2, 3, and 4. This analysis, based on the conservative recommendations of Regulatory Guide 1.7 and Standard Review Plan 6.2.5, assumes a hydrogen yield of 0.5 molecules per 100 eV of energy absorbed for both core and sump radiolysis.

The rate of hydrogen gas production from radiolysis has been evaluated using the methodology presented in Appendix A to Standard Review Plan 6.2.5, and the calculational assumptions detailed in Regulatory Guide 1.7. Table 6.2.5-6 provides a summary of the assumptions made in the analysis.

6.2.5.4.1.3 Corrosion of Metals and Paints in Containment. Following a LOCA, hydrogen may be produced inside the containment by corrosion of aluminum and zinc.

Extensive corrosion testing has been conducted to determine the behavior and compatibility of various materials with alkaline borate solution.[7][8][9] Metals tested included Zircaloy, Inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel, and copper. The tests showed that only aluminum and zinc will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

Aluminum is found in the containment as aluminum metal components. Zinc will be in the form of either galvanized steel, zinc metal, or zinc-based paint.

Aluminum corrosion may be described by the overall reaction:

$$2\,Al + 3\,H_2O \rightarrow Al_2O_3 + 3\,H_2$$

Three moles of hydrogen gas are produced for every two moles of aluminum that is oxidized. Approximately 0.0556 lb-moles of hydrogen gas are produced for each pound of aluminum corroded.

The corrosion of zinc may be described by the overall reaction:

$$Zn + H_2O \rightarrow Zn(OH)_2 + H_2$$

One mole of hydrogen gas is produced for each mole of zinc that is oxidized. Approximately 0.0153 lb-moles of hydrogen gas are produced for each pound of zinc corroded.

The time-temperature cycle (table 6.2.5-9) considered in the calculation of aluminum and zinc corrosion is a representation of the postulated post-accident containment temperature transient. The corrosion rates at the various temperatures are shown in table 6.2.5-10. With these corrosion rates and the baseline aluminum and zinc inventory given in table 6.2.5-6, the contribution of aluminum and zinc corrosion to the hydrogen accumulation in the containment following the DBA was calculated. No credit was taken for the protective shielding effects of insulation or enclosures; i.e., complete and continuous immersion in spray was assumed.

Calculations based on Regulatory Guide 1.7 are performed by increasing the aluminum corrosion rate during the final interval of the post-accident containment temperature transient (table 6.2.5-9) to 200 mils/year. The aluminum and zinc corrosion rates earlier in the accident sequence are shown in table 6.2.5-10.

In order to determine an allowable margin of increase for the zinc and aluminum inventories, the hydrogen accumulation in the containment following a LOCA was calculated for different cases of varying zinc and aluminum inventories. Three cases were evaluated. They included the case where no recombiners were in operation, the case where one recombiner is in operation starting at day 2, and the case where one recombiner is in operation at 3.5 v/o of hydrogen.

Based on the zinc and aluminum inventories provided by table 6.2.5-6, it was determined that the bounding case is 3 times the baseline zinc inventory or 3 times the baseline aluminum inventory or any combination thereof which satisfies the following equation:

$$\frac{N_{al}}{O_{al}} + \frac{N_{zn} + (N_{zbp} - O_{zbp})}{O_{zn}} \leq 3.8$$

where:

N_{al} = new total surface area of aluminum (ft^2)
O_{al} = baseline surface area of aluminum
O_{al} = 3607 ft^2 (table 6.2.5-6)

N_{zn} = new total surface area of zinc (ft^2)
O_{zn} = baseline surface area of zinc
O_{zn} = 182,053 ft^2 (table 6.2.5-6)

N_{zbp} = new total surface area of zinc-based paint (ft^2)
O_{zbp} = baseline surface area of zinc-based paint
O_{zbp} = 680,722 ft^2 (table 6.2.5-6)

Future additions of aluminum and zinc beyond those quantities identified in table 6.2.5-6 are documented and used in this equation to ensure that the limits of hydrogen generation are not exceeded.

The above equation is valid if at least one recombiner is turned on at a hydrogen concentration of 3.03 v/o (2 days), or less, following a LOCA. It is dependent on surface area only and not dependent on the total mass of zinc and aluminum in containment.

Although 4.0 v/o is the acceptance limit, the maximum H_2 v/o should not exceed 3.3 v/o to account for instrument inaccuracies.

6.2.5.4.1.4 Hydrogen in the Primary Coolant. During normal operation of the plant, hydrogen is dissolved in the reactor coolant and is also contained in the pressurizer vapor space. Following a LOCA, this hydrogen is assumed to be immediately released to the containment atmosphere. The maximum equilibrium quantity of hydrogen from this source is 1600 sf^3.

The pressurizer vapor space hydrogen is based on the following:

A. Reactor coolant hydrogen concentration of 50 cm^3 (STP)/kg of coolant.

B. Normal pressurizer heaters turned on 50% of the time and all of the heat going to the boiling water.

C. Bypass spray rate of 1 gal/min.

D. Normal liquid level in pressurizer (60%).

E. Pressurizer relief valves closed.

6.2.5.4.1.5 Hydrogen Mixing. Experiments (references 10 through 16) demonstrate that for the period of high hydrogen evolution during and following blowdown, bulk turbulence and natural convective transport will be available to distribute and diffuse hydrogen throughout the containment.

Following this period, long-term mixing within and between the lower volumes of the containment and the region above the operating deck will be provided by the containment sprays (if operating) and the emergency containment coolers. The reactor vessel head vent system will provide for the release of any concentrated hydrogen from the primary loop at a controlled rate and in a location so as to allow complete dispersion due to the natural diffusion tendencies of hydrogen and by augmented means such as the containment cooler discharge, which is ducted so as to maintain hydrogen concentration equilibrium between the upper and lower containment regions through forced convection.

6.2.5.4.1.6 Conclusions. Figure 6.2.5-5 shows bulk containment volume% hydrogen versus time following a LOCA. The results from the post-LOCA hydrogen generation calculation have determined that for the three cases of varying aluminum/zinc inventory with a single 95% efficient, 100-sf^3/min recombiner starting operation on the second day following a LOCA or when the containment hydrogen concentration reaches 3.03 v/o, the hydrogen concentration peaks below 3.5 v/o and is maintained well below 4.0 v/o, thus showing ample margin in the hydrogen control system.

The integrated hydrogen production for each source is shown in figure 6.2.5-7.

6.2.5.5 Tests and Inspections

6.2.5.5.1 Electric Hydrogen Recombiners

The electric hydrogen recombiners underwent extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principal tests, and full-scale prototype testing. The full-scale prototype tests included the effects of:

- Varying hydrogen concentrations.
- Alkaline spray atmosphere.
- Steam effects.
- Convection currents.
- Seismic effects.

A detailed discussion of these tests is provided in references 17 through 24.

6.2.5.5.2 Post-LOCA Containment Hydrogen Purge System

Safety-related equipment is qualified by the vendor to meet the codes and standards required by the system classification. Functional testing is performed after installation but prior to plant startup to verify the system performance capability. Periodic testing of the system components will be performed in accordance with plant procedures.

6.2.5.5.3 Post-LOCA Cavity Hydrogen Purge System

Safety-related equipment is qualified by the vendor to meet the codes and standards required by the system classification. Functional testing is performed after installation but prior to plant startup to verify the system performance capability.

6.2.5.5.4 Post-Accident Hydrogen Monitoring System

Equipment for this system is vendor qualified to meet the codes and standards required by the system classification. Functional and preoperational testing is performed after installation and prior to plant startup to verify the system performance capability. Periodic testing of the system and the isolation and sample selector valves will be performed in accordance with plant procedures.

6.2.5.6 Instrumentation Requirements

6.2.5.6.1 Electric Hydrogen Recombiner

The recombiners do not require any instrumentation inside the containment for proper operation after a LOCA. The recombiners are started manually after a LOCA. The sampling system is used to obtain containment atmosphere samples that indicate when the recombiners or the venting system should be actuated. Control measures can be initiated when the hydrogen concentration reaches 3 volume%.

6.2.5.6.2 Post-LOCA Containment Hydrogen Purge System

Instrumentation and controls for this system are located outside of the containment in the equipment building or in the control room. Control switches and status indication for the fans, isolation valves, and control valves are provided in the control room. Indication monitoring the operation of the filter exhaust unit is also provided in the control room.

6.2.5.6.3 Post-LOCA Cavity Hydrogen Purge System

Control switches for manual operation of the fans and low-flow alarms for each discharge line are provided on the HVAC panel in the control room.

6.2.5.6.4 Containment Hydrogen Monitoring System

The control switches for the sample selector valves and containment isolation valves are located on the process control panel in the control room. Operation of the hydrogen analyzers is controlled remotely from the main control board. Hydrogen concentration is both indicated and recorded on the main control board.

6.2.5.7 Materials

The materials of construction for the hydrogen control systems are selected for their compatibility with the post-LOCA environment.

The major structural components of the hydrogen recombiners are manufactured from 300-Series stainless steel. Incoloy-800 is used for the heater sheaths and for other parts such as the heat duct, which operates at high temperature.

There are no radiolytic or pyrolytic composition products from these materials.

6.2.5.8 References

1. NRC Regulations:

   Regulatory Guide 1.7, Rev. 2, November 1978.

   Standard Review Plan 6.2.5, "Combustible Gas Control in Containment," July 1981.

   Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident."

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13. "Nuclear Safety Quarterly Report-August, September, and October 1968," BNWL-926, December 1968.

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23. Wilson, J. F., "Electric Hydrogen Recombiner IEEE 323-1974 Qualification," WCAP-7709-L, Supplement 6 (Proprietary), and WCAP-7820, Supplement 6 (Nonproprietary), October 1976.

24. Wilson, J. F., "Electric Hydrogen Recombiner LWR Containments Supplemental Test Number 2," WCAP-7709-L, Supplement 7 (Proprietary), and WCAP-7820, Supplement 7 (Nonproprietary), August 1977.

6.2.6 CONTAINMENT LEAKAGE TESTING[a]

The reactor containment, containment penetrations, and containment isolation barriers are designed to permit periodic leakage rate testing as required by 10 CFR 50, Appendix A, General Design Criteria (GDC) 52, 53, and 54. The containment leak test requirements are outlined and the acceptance criteria for such tests are established in 10 CFR 50, Appendix J, Option B. The objective of the leakage rate testing is to ensure that the leakage from the containment is within the limits set by the Technical Specifications.

Compliance with 10 CFR 50, Appendix J, Option B, Types A, B, and C, testing is discussed in paragraphs 6.2.6.1, 6.2.6.2, 6.2.6.3, and 6.2.6.4.

6.2.6.1 Containment Integrated Leakage Rate Test (Type A Test)

The design leakage rate (L_d) for the containment is 0.2% free volume per day for the first 24 h. The actual leakage rate will be determined by using the methods and requirements of 10 CFR 50, Appendix J, Option B, for Type A tests.

[a] The 10 CFR 50 Appendix J Program is credited as a license renewal aging management program (see subsection 19.2.29).

The acceptance criteria specified in Appendix J for the integrated leakage rate test (ILRT) includes a margin for possible deterioration of the containment leakage integrity during the service intervals between tests. The measured leakage rate (L_{am}) will be less than 0.75 of the maximum allowable leakage rate value L_a.

6.2.6.1.1 ILRT Pretest Requirements

Several pretest requirements are to be met before the ILRT is performed. A general inspection of the accessible nonconcrete interior and exterior surfaces of the containment structures and components for any evidence of structural deterioration which may affect either the containment structural integrity or leaktightness will be made. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage. Any evidence of structural deterioration will be evaluated and corrected if necessary before the Type A test is performed.

Systems that are required to maintain the plant in a safe condition during the test, such as the nuclear service cooling water lines to the containment air coolers, are operable in their normal mode and need not be vented.

Systems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system, need not be vented or drained. Systems which are not vented to containment or drained during Type A tests are identified in table 6.2.6-1.

The steam generator tubes and shell and the associated piping systems passing through the containment liner are considered an extension of the containment. Therefore, the secondary side of the steam generator and connecting systems are not vented to the containment atmosphere. However, since the secondary side of the steam generator is a valid leak path, the main steam lines are vented outside of containment. The penetrations associated with the secondary side of the steam generator are identified in figure 6.2.4-1.

The reactor coolant drain tank, pressurizer relief tank, and accumulator tanks are vented to the containment atmosphere. This is done to protect the tanks from the external pressure of the test and to preclude leakage to or from the tanks which could affect the accuracy of the test results.

An equipment protection list is provided in the ILRT procedure to identify any components that cannot withstand the ILRT test pressure.

During preoperational testing, a structural integrity test (SIT) is performed. The SIT is a pressure test conducted to verify that the containment structural response due to the induced load is consistent with the predicted behavior. Paragraph 3.8.1.7 describes the SIT deflection measurements and concrete crack inspections.

Following the SIT, the preoperational ILRT is performed.

6.2.6.1.2 ILRT Test Method

The ILRT will be conducted in accordance with 10 CFR 50 Appendix J, Option B. A short duration ILRT may be performed in accordance with Bechtel's Topical Report BN-TOP-1. The

test procedure used during the preoperational Type A test is described in chapter 14. Drawing 1X4DB132 shows the test arrangement for a Type A test. Two permanent flowthrough tubes have been added in a spare electrical penetration (electrical penetration number 31) to be utilized for the flow verification and pressure sensing lines. After completion of the ILRT, the flowthrough tubes are sealed and the spare electrical penetration is restored to its original configuration and Type B tested. For penetrations which are exempt from Type B or C tests the leakage testing is accomplished by the Type A test, as noted in table 6.2.4-1.

Containment dry bulb temperature, pressure, and dewpoint temperature are periodically monitored during the test. These data are analyzed as they are taken so that the leakage rate and its statistical significance are known as the test progresses. Once the leakage rate has been found with sufficient accuracy, a known additional leak is imposed and the measurements are continued, giving additional verification of the leakage rate.

6.2.6.2 Containment Penetration Leakage Rate Tests (Type B Tests)

Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds; airlocks and lock-door seals; equipment and access hatch seals; and electrical canister and modular type penetrations receive a preoperational and periodic Type B leakage rate test in accordance with 10 CFR 50, Appendix J, Option B.

Electrical penetrations are of modular- or canister-type design (66 modular and 6 canister) and their leakage testing provisions were designed and initially tested to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) 317. Each of the 66 modular-type electrical penetrations consists of a single header plate sealed to a nozzle on the containment exterior by double O-rings with interspace connection. Feedthrough modules carrying conductors are sealed into the header plate by a metal compression fitting assembly with interspace connection. The feedthrough conductors are sealed to the feedthrough module by double, high-temperature, thermoplastic seals with interspace connection. The seal interspaces (header plate O-rings, feedthrough module compression fittings, and thermoplastic seals) are pressurized with nitrogen; the pressure is periodically monitored to detect leakage. Each of the canister-type electrical penetrations is similar in design to the modular type, except that the canister type extends through the length of the nozzle with header plates at each end, through which the conductor modules are sealed and terminate in electrical ceramic bushings. Each of the 72 electrical penetrations is provided with a local pressure gauge.

Expansion bellows utilized on the fuel transfer tube penetration accommodate relative movement between the refueling pool liner and the containment penetration and do not form part of the containment pressure boundary. Expansion bellows utilized on the containment emergency sump suction line guard pipes accommodate relative movement between the auxiliary and fuel handling buildings with respect to the containment and do not form part of the containment pressure boundary.

The equipment hatch, personnel lock, and escape hatch doors are fitted with double seals with an interspace test connection.

The test connection for the equipment hatch is located inside the containment. Test connections for the personnel lock and escape hatch doors are located such that testing is accomplished without entering the containment. Two pressure gauges are provided, one inside the lock which penetrates the bulkhead at the inner airlock door to measure containment pressure and one located outside the airlock to penetrate the bulkhead at the outer airlock door to read lock pressure. The handwheel shafts are provided with double seals and test connections where the shafts penetrate the airlock bulkheads.

The containment leak rate penetrations (table 6.2.4-1, penetrations 64A, 64B, 68, and 87) are provided with blind flanges that have testable double O-ring seals and are Type B tested. The eddy current/sludge lancing penetrations (table 6.2.4-1, penetrations 5, 55, and 90) are provided with blind flanges that have O-ring type joints per ANSI B16.5 with valve test connections and are Type B tested.

Type B tests are conducted at calculated peak post-LOCA containment internal pressure (P_a). The acceptance criteria and leakage rate limits are given in the Technical Specifications. Test methods and equipment are described in paragraph 6.2.6.3.

6.2.6.3 <u>Containment Isolation Valve Leakage Rate Tests</u>

Containment isolation valves are Type C tested in accordance with 10 CFR 50, Appendix J, Option B.

The process piping, instrumentation tubing, and personnel access penetrations are listed in table 6.2.4-1. Figure 6.2.4-1 shows the location of all test vent and drain connections and the normal direction of flow.

The CIVs for each piping penetration and process isolation valve for those piping penetrations where no GDC is applicable are tabulated in table 6.2.4-1, together with the test type.

Type B and C tests are performed by local pressurization utilizing either the pressure decay or flowmeter method. For the pressure decay method, the test volume is pressurized with air or nitrogen to at least P_a. The rate of decay of pressure of the known free air test volume is monitored to calculate the leakage rate. In the flowmeter method, pressure is maintained in the test volume by makeup air, nitrogen, or water (if applicable) through a calibrated flowmeter. The flowmeter fluid flowrate is the isolation valve leakage rate.

The leakage will be measured in the same direction as would occur in an accident, unless it can be determined that leakage measured in a different direction will provide an equivalent or more conservative result. The test medium used for pressurization is determined by the valve's post-accident condition. Valves which could be exposed to the containment atmosphere subsequent to an accident will be tested with air or nitrogen. Valves which are in lines designated to be filled with a liquid for at least 30 days subsequent to an accident may be leakage-rate tested with water.

Type C testing of the safety injection lines, residual heat removal lines, high head safety injection lines of the chemical and volume control system, RCP seal injection lines, the containment emergency sump lines to the residual heat removal and containment spray pumps, and nuclear service cooling water lines to and from the containment fan coolers is not performed. The justification for this is that these valves are either normally open at the time of a LOCA or are opened at some time after the accident to effect immediate and long term core cooling. These systems are closed systems outside containment except for the NSCW system which is a closed system inside containment, designed and constructed to ASME III, Class 2 and Seismic Category I requirements, and as such they do not constitute a potential containment atmosphere leak path during or following a loss-of-coolant accident with a single active failure of a system component. Should the valves leak slightly when closed, the fluid seal within the pipe or the closed piping system outside/inside containment would preclude release of containment atmosphere to the environs. Furthermore, inservice testing and inspection of these isolation valves and the associated piping system outside the containment is performed periodically under the inservice inspection requirements of ASME XI as described in subsection 3.9.6 and section 6.6. During normal operation, the systems are water filled, and

degradation of valves or piping is readily detected. Containment penetrations not vented to containment or drained during Type A testing are identified in table 6.2.6-1.

The steam generator and associated secondary system piping that form the primary barrier to the outside, much the same as the containment liner plate, are subjected to Type A test as shown in table 6.2.4-1. The barriers against fission product release to the environment are the steam generator tubes and piping associated with the steam generators.

Containment pressure monitoring lines are considered an extension of the containment boundary, and therefore the isolation valves are not Type C tested.

Isolation valves will be positioned to their post accident position by the normal method with no accompanying adjustments. Exercising valves for the purpose of improving leakage performance shall not be permitted.

For larger test volumes, a pressure decay method may be utilized to determine the leakage rate. The makeup flow rate or the pressure decay methods or other proven techniques may be used to determine the leakage.

The total leakage rate for Type B and C tests must be less than 0.6 L_a.

The criteria for determining the direction in which the test pressure is applied to the isolation valves are as follows:

A. Gate Valves

1. Parallel Disc

a. Test in the design basis accident (DBA) direction.

b. Testing can be performed between the discs if a test connection or drain is provided in the valve design.

2. Flexible Wedge

a. Test in the DBA direction.

b. Testing can be performed between the wedge sections if a test connection or drain is provided in the valve design.

3. Solid Wedge

a. Test in the DBA direction.

B. Globe Valves

If the DBA flow direction is over the disc (flow to close), the valve may be tested in the reverse direction. However, if the DBA flow direction is under the disc (flow to open), then the valve must be tested in this direction.

C. Butterfly Valves

Test in the DBA direction for Type C tests.

D. Flanges

Test in either direction.

The leakage rate test acceptance criteria for penetrations and isolation valves subject to Type B and C tests are given in the Technical Specifications.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Type A, B, and C tests are conducted at the intervals specified in the containment leakage testing program as specified in the Technical Specifications. These intervals are in accordance with 10 CFR 50, Appendix J, Option B, with approved exceptions.

A post outage report will be prepared, presenting the results of the previous cycle's Type B and Type C tests; and Type A, Type B, and Type C tests if performed during that outage. Sufficient documentation will be collected and retained so that the effectiveness of the implementation of the containment leakage testing program can be reviewed and determined. The technical contents of the report will be available onsite for NRC review.

The preoperational test report contains a schematic of the leakage measuring system, instrumentation used, supplemental test method, test program, and analysis and interpretation of the leakage test data for the Type A test.

6.2.6.5 Special Testing Requirements

VEGP does not have a subatmospheric containment or a secondary containment, hence there are no special testing requirements beyond those delineated in paragraphs 6.2.6.1 through 6.2.6.4.

6.2.7 FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

In accordance with General Design Criterion (GDC) 51, the reactor containment pressure boundary is designed with sufficient margin to ensure that under operating, maintenance, testing, and design basis accident conditions, its ferritic materials behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

For VEGP, the reactor containment pressure boundary components with ferritic materials are:

- Containment liner plate.
- Containment penetration sleeve assemblies.
- Equipment hatch.
- Personnel lock.
- Escape lock.
- Flued heads.
- Containment isolation boundary piping.
- Containment isolation boundary valves.

6.2.7.1 Design Bases

A. In accordance with GDC 1, the containment pressure boundary is designed to quality standards commensurate with the safety function performed.

B. In accordance with GDC 16, the containment pressure boundary is designed to provide a barrier against the uncontrolled release of radioactivity to the environment.

6.2.7.2 Specifications for Ferritic Materials

A. Containment Liner Plate

Since the containment liner plate is only 1/4 in. thick, it is exempt from fracture toughness testing. However, liner plate greater than 1/4 in. thick (referred to as thickened liner plate) satisfies the fracture toughness test requirements presented in NE-2320 of Subsection NE of the following code editions and addenda:

1. All thickened liner plate up to the basemat liner plate at el 169 ft 0 in., 1971 edition through summer 1973 addenda.
2. All thickened liner plate above el 169 ft 0 in., 1974 edition through summer 1975 addenda.

B. Penetration Sleeve Assemblies

The penetration sleeve assemblies are composed of the sleeve, thickened liner plate (as reinforcing), anchor rings and stiffeners, gusset plates, and cap plates for spare penetrations. The penetration sleeve assemblies for the electrical penetrations contain a weld neck flange with captive nuts attached to receive the electrical header plates. Sleeves are fabricated from SA-516 Grade 70 plates and weld neck flanges are steel forgings of SA-105 material. The impact requirements for fracture toughness for penetration sleeve assemblies in the containment shell are in accordance with American Society of Mechanical Engineers (ASME) Section III, Division 1, Subsection NE, 1974 edition through summer 1975 addenda. Penetration sleeves in the containment basemat are in accordance with ASME Section III, Division 1, Subsection NE, 1971 edition through summer 1973 addenda.

C. Equipment Hatch, Personnel Lock, and Escape Lock

These items are fracture toughness tested in accordance with ASME Section III, Division 1, Subsection NE, 1974 edition through summer 1975 addenda.

D. Flued Heads

The flued heads are fracture toughness tested in accordance with ASME Section III, Division 1, Subsection NC, 1977 through summer 1978 addenda.

E. Containment Pressure Boundary Piping

Those portions of the containment pressure boundary piping comprising the main steam and feedwater system are designed and fracture toughness tested in accordance with ASME Section III, Division 1, Subsection NC, 1974 edition through summer 1975 addenda. The remaining containment pressure boundary piping is designed in accordance with ASME III, 1974 edition through summer

1975 addenda, without fracture toughness testing. The fracture toughness of these items is further addressed in paragraph 6.2.7.4.

The configuration of containment pressure boundary piping is shown in figure 6.2.4-1.

F. Containment Pressure Boundary Valves

Pressure boundary materials of the main steam isolation valves and main feedwater isolation valves are fracture toughness tested in accordance with Subarticle NC-2300 of ASME Section III, Division 1, Subsection NC, 1977 edition. For other containment pressure boundary valves in the balance of plant scope, the specific code edition and addenda to which the valves were procured did not require fracture toughness testing. These valves were procured in accordance with the 1974 edition up to and including the summer 1975 addenda of the ASME Code. The fracture toughness of these items is further addressed in paragraph 6.2.7.4.

Containment isolation valves within the nuclear steam supply system scope are fabricated of austenitic stainless steel or carbon steel and therefore need not be fracture toughness tested.

6.2.7.3 Documentation

A. ASME Code Data Reports

ASME Code Data Reports are prepared by the cognizant organization which applies the appropriate "N" type symbol to certify that the design, fabrication, installation, inspection, testing, and stamping are in accordance with ASME III for containment pressure boundary systems.

B. Certified Material Test Reports

Certified material test reports, when applicable, are provided to document the actual results of required chemical analyses, tests, examinations, and weld repairs of materials comprising the containment pressure boundary.

C. Other Documents

Drawings and related supplemental information are available as applicable to convey information necessary to fabricate, install, test, etc., the containment pressure boundary piping system.

6.2.7.4 Evaluation

The containment isolation boundary piping and balance of plant containment isolation boundary valves were procured to meet the minimum requirements of the 1974 edition through summer 1975 addenda of the ASME Code, which did not require impact testing. The specific requirements of NX-2300 of this code effective date are that the Design Specification shall state whether or not impact testing is required. Due to the exclusion criteria of subparagraph NX-2300 of the code, the piping and valves discussed in Sections 6.2.7.2.E and F did not require impact testing.

The containment pressure boundary piping and valves meet or exceed the minimum ASME Code requirements for design, materials, fabrication, examination, testing, and code stamping.

TABLE 6.2.1-1

CONTAINMENT DESIGN LIMITS AND CALCULATED
CONTAINMENT PEAK PRESSURE AND TEMPERATURE

Break	Peak Pressure (psig)	Available Margin (psi)	Peak Temperature (°F)
Primary Side Ruptures			
Double-ended pump suction, maximum safety injection	35.9[b]	16.1[b]	247[c]
Double-ended pump suction, minimum safety injection	35.9[b]	16.1[b]	247[c]
Double-ended hot leg	36.5	15.5	250

Containment Design Pressure

52 psig
-3 psig

Containment Atmosphere Design Temperature

381°F[a]

a. A peak containment atmosphere temperature of 381°F was used in calculating the thermal gradients across the containment wall.

b. Per LDCR 2005051, the pressure is being increased by 1.3 psig from 34.6 psig, and the margin is correspondingly reduced temporarily until re-evaluated during a future analysis requiring re-evaluation of the containment mass and energy releases.

c. Per LDCR 2005051, there will be an insignificant increase in peak temperature. However, the peak temperature remains well below the design temperature of 381 °F.

TABLE 6.2.1-2

ASSUMPTIONS FOR CONTAINMENT ANALYSIS - PART 1

Service water temperature (°F)	95
Refueling water temperature (°F)	130
Refueling water storage tank volume (gal) (deliverable volume)	710,700
Initial Containment	
Temperature (°F)	120
Initial pressure (psia)	17.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.75×10^6

TABLE 6.2.1-3

ASSUMPTIONS FOR CONTAINMENT ANALYSIS - PART 2

	Accident Analysis	Design
Fan coolers		
Number operating	4	8
Flowrate per fan cooler (ft^3/min)	43,500 (slow speed)	97,000 (full speed)
Heat removal capacity per fan cooler (Btu/h)	8.15×10^5	2.605×10^6
Spray pumps		
Number available	1	2
Flowrate per pump (gal/min)	2597	3200

TABLE 6.2.1-4 (SHEET 1 of 2)

CONTAINMENT STRUCTURAL HEAT SINKS

Passive Heat Sinks

Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
1. Dome	29260	Epoxy	0.00025
		Inorganic zinc paint	0.0002083
		Carbon steel	0.02083
		Concrete	3.30
2. Shell	65181	Inorganic zinc paint	0.0002083
		Carbon steel	0.02083
		Concrete	3.75
3. Miscellaneous interior concrete	15971	Epoxy	0.00154
		Concrete	2.82
4. Primary shields	2106	Inorganic zinc paint	0.0002083
		Carbon steel	0.05795
		Epoxy	0.001125
		Concrete	1.5
5. Mechanical equipment	209	Stainless steel	0.0158
6 Refueling canal wall	6455	Stainless steel	0.02083
		Epoxy	0.001125
		Concrete	4.133
7. Miscellaneous interior concrete	4693	Epoxy	0.000042
		Concrete	2.893
8. Miscellaneous interior concrete	16147	Epoxy	0.01125
		Concrete	2.371
9. Miscellaneous interior concrete	58883	Epoxy	0.001125
		Concrete	1.4705

TABLE 6.2.1-4 (SHEET 2 of 2)

Wall Description		Heat Transfer Area (ft^2)	Material	Thickness (ft)
10.	Refueling canal slab	1255	Stainless steel Concrete	0.02083 4.0
11.	Structural steel	192088	Galvanizing Carbon steel	0.00033 0.0141
12.	Structural steel	183286	Inorganic zinc paint Carbon steel	0.000208 0.0263
13.	Piping	72630	Inorganic zinc paint Carbon steel	0.000208 0.03676 0.03676
14.	Piping	34158	Stainless steel	0.0163
15.	Piping	1395	Stainless steel	0.0340
16.	Mechanical equipment	98158	Inorganic zinc paint Carbon steel	0.000208 0.142
17.	Mechanical equipment	24750	Inorganic zinc paint Carbon steel	0.000208 0.0208
18.	Miscellaneous steel	40599	Inorganic zinc paint Carbon steel	0.000208 0.01288
19.	Basement	13702	Inorganic zinc paint Carbon steel Concrete	0.000208 0.02083 10.5
20.	Reactor cavity	4263	Inorganic zinc paint Carbon steel Concrete	0.000208 0.02083 8.0

TABLE 6.2.1-5

THERMOPHYSICAL PROPERTIES OF CONTAINMENT HEAT SINKS

Material	Thermal Conductivity (Btu/h-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Paint	0.9	20.0
Carbon steel	28.0	52.5
Stainless steel	9.4	60.1
Concrete	0.65	28.8
Galvanizing	64.8	40.9
Epoxy	0.97	20.0

TABLE 6.2.1-6

ACCIDENT SEQUENCE FOR DOUBLE-ENDED PUMP SUCTION BREAK
MINIMUM SAFETY INJECTION

	Event	Time of Occurrence (s)
1.	Accumulators begin injecting	14.5
2.	Containment peak pressure	17.6
3.	End of blowdown	22.0
4.	Safety injection begins	43.3
5.	Accumulator injection stopped	54.3
6.	Fan coolers start	100.8
7.	Spray start	108.0
8.	End of reflood	225.8
9.	Recirculation, injection	3952.0
10.	Recirculation, spray	5000.0

TABLE 6.2.1-7

DELETED

TABLE 6.2.1-8

POSTULATED SUBCOMPARTMENT PIPE BREAKS

Subcompartment	High-Energy Line	Break Area ($in.^2$)	Maximum ΔP (psid)
Reactor cavity	1201-009-27.5 in.	144	192.90
Steam generator (pipe break below el 220 ft)	1201-008-31 in.	436	23.50
Steam generator (pipe break above el 220 ft)	1305-064-16 in.	144.5	6.77
Pressurizer	1201-053-14 in.	308	20.7

TABLE 6.2.1-9 (SHEET 1 OF 6)

REACTOR CAVITY FLOW MODEL: NODE CHARACTERISTICS
(INITIAL CONDITIONS REMAIN THE SAME)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psi)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Approximately 1/8 of the upper inspection annulus.	3.625	28.4	120	13.2	25	192.89	102.95
2	Approximately 1/8 of the upper inspection annulus.	3.625	28.22	120	13.2	25	83.17	102.286
3	Approximately 1/8 of the upper inspection annulus.	3.625	28.034	120	13.2	25	55.42	101.624
4	Approximately 1/8 of the upper inspection annulus.	3.625	28.22	120	13.2	25	50.29	102.286
5	Approximately 1/8 of the upper inspection annulus.	3.625	28.4	120	13.2	25	50.46	102.95
6	Approximately 1/8 of the upper inspection annulus.	3.625	28.22	120	13.2	25	192.63	102.286
7	Approximately 1/8 of the upper inspection annulus.	3.625	28.034	120	13.2	25	100.85	101.624
8	Approximately 1/8 of the upper inspection annulus.	3.625	28.22	120	13.2	25	63.71	102.286
9	Approximately 1/8 of the lower inspection annulus.	2.083	34.20	120	13.2	25	193.0	71.245
10	Approximately 1/8 of the lower inspection annulus.	2.083	33.88	120	13.2	25	83.04	70.57

TABLE 6.2.1-9 (SHEET 2 OF 6)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psi)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Approximately 1/8 of the lower inspection annulus.	2.083	33.56	120	13.2	25	55.64	69.91
12	Approximately 1/8 of the lower inspection annulus.	2.083	33.88	120	13.2	25	50.56	70.57
13	Approximately 1/8 of the lower inspection annulus.	2.083	34.20	120	13.2	25	50.6	71.245
14	Approximately 1/8 of the lower inspection annulus.	2.083	33.88	120	13.2	25	192.45	70.57
15	Approximately 1/8 of the lower inspection annulus.	2.083	33.56	120	13.2	25	100.91	69.91
16	Approximately 1/8 of the lower inspection annulus.	2.083	33.88	120	13.2	25	64.26	70.57
17	Approximately 1/8 of the upper inner cavity.	7.125	1.287	120	13.2	25	159.86	9.169
18	Approximately 1/8 of the upper inner cavity.	7.125	1.33	120	13.2	25	135.98	9.447
19	Approximately 1/8 of the upper inner cavity.	7.125	1.365	120	13.2	25	127.72	9.7265
20	Approximately 1/8 of the upper inner cavity.	7.125	1.326	120	13.2	25	124.98	9.447
21	Approximately 1/8 of the upper inner cavity.	7.125	1.287	120	13.2	25	124.99	9.169
22	Approximately 1/8 of the upper inner cavity.	7.125	1.33	120	13.2	25	159.75	9.447

TABLE 6.2.1-9 (SHEET 3 OF 6)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psi)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Approximately 1/8 of the upper inner cavity.	7.125	1.365	120	13.2	25	139.1	9.7265
24	Approximately 1/8 of the upper inner cavity.	7.125	1.326	120	13.2	25	128.38	9.447
25	Approximately 1/8 of the lower inner cavity.	4.72	0.99	120	13.2	25	177.58	4.678
26	Approximately 1/8 of the lower inner cavity.	4.72	1.05	120	13.2	25	98.84	4.96
27	Approximately 1/8 of the lower inner cavity.	4.72	1.11	120	13.2	25	56.69	5.24
28	Approximately 1/8 of the lower inner cavity.	4.72	1.051	120	13.2	25	52.15	4.96
29	Approximately 1/8 of the lower inner cavity.	4.72	0.99	120	13.2	25	52.41	4.678
30	Approximately 1/8 of the lower inner cavity.	4.72	1.051	120	13.2	25	177.48	4.96
31	Approximately 1/8 of the lower inner cavity.	4.72	1.11	120	13.2	25	102.44	5.24
32	Approximately 1/8 of the lower inner cavity.	4.72	1.051	120	13.2	25	65.39	4.96
33	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	117.24	20.005
34	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	94.04	20.005

TABLE 6.2.1-9 (SHEET 4 OF 6)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psi)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	54.65	20.005
36	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	51.09	20.005
37	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	50.69	20.005
38	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	117.29	20.005
39	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	98.57	20.005
40	Approximately 1/8 of the upper cavity annulus.	6.45	3.101	120	13.2	25	64.37	20.005
41	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	76.9	20.005
42	Approximately 1/8 of the lower cavity annulus. annulus.	6.45	3.101	120	13.2	25	74.16	20.005
43	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	48.21	20.005
44	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	46.97	20.005
45	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	47.11	20.005
46	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	76.94	20.005

TABLE 6.2.1-9 (SHEET 5 OF 6)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psi)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
47	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	74.93	20.005
48	Approximately 1/8 of the lower cavity annulus.	6.45	3.101	120	13.2	25	51.3	20.005
49	Lower RX Cavity region el 160 ft 10 3/4 in. to 169 ft 4 1/2 in.	8.48	121.81	120	13.2	25	12.4	1032.93
50	Lower RX cavity region el 156 ft to 160 ft 10 3/4 in.	4.90	274.66	120	13.2	25	11.91	1345.85
51	Lower RX cavity; near guide tube; supports region 156 ft.	4.75	301.30	120	13.2	25	11.25	1431.16
52	Located between first and second guide tube supports in lower RX cavity.	7.75	303.9	120	13.2	25	10.59	2355.23
53	Incore instrumentation tunnel.	10.0	70.60	120	13.2	25	10.2	705.96
54	Instrumentation tunnel horizontal passageway.	10.0	100.85	120	13.2	25	9.71	1008.52
55	Instrumentation tunnel access shaft	15.52	52.96	120	13.2	25	9.69	821.894
56	Instrumentation tunnel access shaft	23.48	22.58	120	13.2	25	9.68	530.23
57	Instrumentation tunnel	11.0	83.42	120	13.2	25	8.43	917.62
58	Instrumentation tunnel	8.0	94.44	120	13.2	25	6.74	755.48

TABLE 6.2.1-9 (SHEET 6 OF 6)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psi)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
59	Instrumentation tunnel.		94.44	120	13.2	25	5.88	755.48
60	Instrumentation tunnel.	8.0	117.30	120	13.2	25	5.01	938.39
61	Instrumentation tunnel.	9.0	112.39	120	13.2	25	4.34	1011.51
62	Instrumentation tunnel.	8.0	88.09	120	13.2	25	3.68	704.69
63	South steam generator (SG) compartment.	NA	NA	120	13.2	25	2.25	79,242.0
64	North SG compartment.	NA	NA	120	13.2	25	2.147	79,242.0
65	Free containment.	NA	NA	120	13.2	25	0.97	2.75E+06

TABLE 6.2.1-10 (SHEET 1 OF 3)

REACTOR CAVITY FLOW MODEL: FLOW CHARACTERISTICS

Vent Path	Node Number (From-To)	Description of Flow Choked	Description of Flow Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	1-2	x		6.5	0	0	1.0	0.231	1.231	0.7725
2	1-6		*	6.5	0	0	1.0	0.231	1.231	0.7725
3	1-17	x		0.743	0	0	1.0	0.48	1.48	0.957
4	1-9		*	18.524	0	0	1.0	0.188	1.188	0.124
5	1-63	x		3.31	0	0	1.0	0.45	1.45	2.77
6	2-3	x		6.25	0	0	1.0	0.24	1.24	0.778
7	2-10		*	18.28	0	0	1.0	0.19	1.19	0.125
8	2-63	x		6.45	0	0	1.0	0.40	1.40	1.26
9	3-4		*	6.25	0	0	1.0	0.24	1.24	0.778
10	3-11		*	18.0	0	0	1.0	0.197	1.197	0.125
11	3-63	x		6.24	0	0	1.0	0.40	1.40	0.715
12	4-5		*	6.5	0	0	1.0	0.231	1.231	0.7725
13	4-12		*	18.28	0	0	1.0	0.19	1.19	0.125
14	4-63	x		6.45	0	0	1.0	0.40	1.40	1.26
15	5-8		*	6.5	0	0	1.0	0.231	1.231	0.7725
16	5-13		*	18.28	0	0	1.0	0.188	1.188	0.125
17	5-63	x		3.31	0	0	1.0	0.45	1.45	2.77
18	5-64	x		3.31	0	0	1.0	0.45	1.45	2.77
19	6-7	x		6.25	0	0	1.0	0.24	1.24	0.778
20	6-14		*	18.28	0	0	1.0	0.19	1.19	0.125
21	6-22	x		0.743	0	0	1.0	0.48	1.48	0.957
22	6-64	x		3.12	0	0	1.0	0.45	1.45	1.43
23	7-8	x		6.25	0	0	1.0	0.24	1.24	0.778
24	7-15		*	18.0	0	0	1.0	0.197	1.197	0.125
25	7-64	x		6.24	0	0	1.0	0.40	1.40	0.715
26	8-16		*	18.28	0	0	1.0	0.19	1.19	0.125
27	8-64	x		6.45	0	0	1.0	0.40	1.40	1.26
28	9-10	x		4.16	0	0	1.0	0.286	1.286	1.12
29	9-14		*	4.16	0	0	1.0	0.286	1.286	1.12
30	9-25		*	0.743	0	0	1.0	0.48	1.48	0.983
31	9-25		*	2.71	0	1.12	1.0	0.5	2.62	3.00
32	9-33	x		0.347	0	0	1.0	0.5	1.5	6.32
33	9-63	x		2.34	0	0	1.0	0.45	1.45	3.93
34	10-11	x		6.25	0	0	1.0	0.18	1.18	1.13
35	10-26		*	2.71	0	1.12	1.0	0.5	2.62	3.00
36	10-34		*	0.347	0	0	1.0	0.5	1.5	6.32
37	10-63	x		4.48	0	0	1.0	0.4	1.4	2.08
38	11-12		*	6.25	0	0	1.0	0.18	1.18	1.13
39	11-27		*	2.71	0	1.12	1.0	0.5	2.62	3.00
40	11-35		*	0.347	0	0	1.0	0.5	1.5	6.32
41	11-63	x		4.288	0	0	1.0	0.4	1.4	1.04
42	12-13		*	4.16	0	0	1.0	0.286	1.286	1.12
43	12-28		*	2.71	0	1.12	1.0	0.5	2.62	3.00
44	12-36		*	0.347	0	0	1.0	0.5	1.5	6.32
45	12-63	x		4.48	0	0	1.0	0.4	1.40	1.36
46	13-16		*	4.16	0	0	1.0	0.286	1.286	1.12
47	13-29		*	2.71	0	1.12	1.0	0.5	2.62	3.00
48	13-37		*	0.347	0	0	1.0	0.5	1.50	6.32

TABLE 6.2.1-10 (SHEET 2 OF 3)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
49	13-63	x		2.144	0	0	1.0	0.45	1.45	2.08
50	13-64	x		2.34	0	0	1.0	0.45	1.45	3.93
51	14-15	x		6.25	0	0	1.0	0.18	1.18	1.13
52	14-30		*	0.743	0	0	1.0	0.48	1.48	0.983
53	14-30		*	2.71	0	1.12	1.0	0.5	2.62	3.00
54	14-38	x		0.347	0	0	1.0	0.5	1.50	6.32
55	14-64	x		2.144	0	0	1.0	0.45	1.45	2.08
56	15-16	x		6.25	0	0	1.0	0.18	1.18	1.13
57	15-31		*	2.71	0	1.12	1.0	0.5	2.62	3.00
58	15-39		*	0.347	0	0	1.0	0.5	1.50	6.32
59	15-64	x		4.288	0	0	1.0	0.4	1.40	1.04
60	16-32		*	2.71	0	1.12	1.0	0.5	2.62	3.00
61	16-40		*	0.347	0	0	1.0	0.5	1.50	6.32
62	16-64	x		4.48	0	0	1.0	0.4	1.40	1.36
63	17-18	x		0.91	0	0	1.0	0.18	1.18	4.8
64	17-22		*	0.91	0	0	1.0	0.18	1.18	4.8
65	17-25	x		0.38	0	0	1.0	0.22	1.22	5.15
66	18-19		*	0.975	0	0	1.0	0.16	1.16	4.662
67	18-26		*	0.316	0	0	1.0	0.26	1.26	4.93
68	19-20		*	0.975	0	0	1.0	0.16	1.16	4.662
69	19-27	x		0.25	0	0	1.0	0.31	1.31	4.74
70	20-21		*	0.91	0	0	1.0	0.18	1.18	4.8
71	20-28	x		0.316	0	0	1.0	0.26	1.26	4.93
72	21-24		*	0.91	0	0	1.0	0.18	1.18	4.8
73	21-29	x		0.38	0	0	1.0	0.22	1.22	5.15
74	22-23	x		0.975	0	0	1.0	0.16	1.16	4.662
75	22-30	x		0.316	0	0	1.0	0.26	1.26	4.93
76	23-24		*	0.975	0	0	1.0	0.16	1.16	4.662
77	23-31		*	0.25	0	0	1.0	0.31	1.31	4.74
78	24-32	x		0.316	0	0	1.0	0.26	1.26	4.93
79	25-26	x		0.427	0	0	1.0	0.274	1.274	9.3
80	25-30		*	0.427	0	0	1.0	0.274	1.274	9.3
81	25-33	x		1.254	0.178	0	1.0	0.032	1.21	3.43
82	26-27	x		0.494	0	0	1.0	0.24	1.24	9.03
83	26-34		*	1.254	0.178	0	1.0	0.032	1.21	3.43
84	27-28		*	0.494	0	0	1.0	0.24	1.24	9.03
85	27-35		*	1.254	0.178	0	1.0	0.032	1.21	3.43
86	28-29		*	0.427	0	0	1.0	0.274	1.274	9.3
87	28-36		*	1.254	0.178	0	1.0	0.032	1.21	3.43
88	29-32		*	0.427	0	0	1.0	0.274	1.274	9.3
89	29-37		*	1.254	0.178	0	1.0	0.032	1.21	3.43
90	30-31	x		0.494	0	0	1.0	0.24	1.24	9.03
91	30-38	x		1.254	0.178	0	1.0	0.032	1.21	3.43
92	31-32	x		0.494	0	0	1.0	0.24	1.24	9.03
93	31-39		*	1.254	0.178	0	1.0	0.032	1.21	3.43
94	32-40		*	1.254	0.178	0	1.0	0.032	1.21	3.43
95	33-34		*	1.226	0.2085	0	- 1.0	0	1.21	5.38
96	33-38		*	1.226	0.2085	0	1.0	0	1.21	5.38
97	33-41	x		1.25	0.21	0	1.0	0	1.21	5.154
98	34-35	x		0.613	0.21	0	1.0	0.25	1.46	5.38

TABLE 6.2.1-10 (SHEET 3 OF 3)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
99	34-42		*	1.25	0.21	0	1.0	0	1.21	5.154
100	35-36		*	1.226	0.21	0	1.0	0	1.21	5.38
101	35-43		*	1.25	0.21	0	1.0	0	1.21	5.154
102	36-37		*	1.226	0.21	0	1.0	0	1.21	5.38
103	36-44		*	1.25	0.21	0	1.0	0	1.21	5.154
104	37-40		*	1.226	0.21	0	1.0	0	1.21	5.38
105	37-45		*	1.25	0.21	0	1.0	0	1.21	5.154
106	38-39		*	1.226	0.21	0	1.0	0	1.21	5.38
107	38-46	x		1.25	0.21	0	1.0	0	1.21	5.154
108	39-47	x		1.25	0.21	0	1.0	0	1.21	5.154
109	40-39	x		0.613	0.21	0	1.0	0.25	1.46	5.38
110	40-48		*	1.25	0.21	0	1.0	0	1.21	5.154
111	41-42		*	1.226	0.21	0	1.0	0	1.21	5.384
112	41-46		*	1.226	0.21	0	1.0	0	1.21	5.384
113	41-49	x		1.254	0.12	0	1.0	0	1.12	2.63
114	42-43	x		0.613	0.21	0	1.0	0.25	1.46	5.38
115	42-49	x		1.254	0.12	0	1.0	0	1.12	2.63
116	43-44		*	1.226	0.21	0	1.0	0	1.21	5.384
117	43-49	x		1.254	0.12	0	1.0	0	1.12	2.63
118	44-45		*	1.226	0.21	0	1.0	0	1.21	5.384
119	44-49	x		1.254	0.12	0	1.0	0	1.12	2.63
120	45-48		*	1.226	0.21	0	1.0	0	1.21	5.384
121	45'49	x		1.254	0.12	0	1.0	0	1.12	2.63
122	46-47		*	1.226	0.21	0	1.0	0	1.21	5.384
123	46-49	x		1.254	0.12	0	1.0	0	1.12	2.63
124	47-49	x		1.254	0.12	0	1.0	0	1.12	2.63
125	48-47		*	0.613	0.21	0	1.0	0.25	1.46	5.384
126	48-49	x		1.254	0.12	0	1.0	0.0	1.12	2.63
127	49-50		*	93.49	0	0	1.0	0.083	1.083	0.069
128	50-51		*	84.92	0	0	1.0	0.225	1.225	0.045
129	51-52		*	101.86	0	0.5	1.0	0.25	1.75	0.033
130	52-53		*	103.56	0	0	1.0	0.223	1.223	0.1
131	53-54		*	94.34	0	0	1.0	0.13	1.13	0.063
132	54-55		*	47.66	0	1.12	1.0	0.29	2.41	0.177
133	54-57		*	59.66	0	0	1.0	0.266	1.266	0.106
134	55-56		*	22.58	0.525	0	1.0	0.05	1.575	0.777
135	57-58		*	71.73	0	1.12	1.0	0.14	2.26	0.082
136	58-59		*	71.73	0	0	1.0	0.141	1.141	0.08
137	59-60		*	71.73	0	0	1.0	0.141	1.141	0.08
138	60-61		*	82.08	0	0	1.0	0.155	1.155	0.076
139	61-62		*	82.46	0	0	1.0	0.153	1.153	0.064
140	62-65		*	47.25	0	0	1.0	0.301	1.301	0.0264
141	63-64		*	248.0	0	0	1.0	0.5	1.5	0.11
142	63-65		*	452.0	0	0	1.0	0.45	1.45	0.071
143	64-65		*	452.0	0	0	1.0	0.45	1.45	0.071

TABLE 6.2.1-11 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): NODE CHARACTERISTICS LOOP CLOSURE WELD (336-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	7.11	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	13.58	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	15.89	1661.50
4	By SG 4; from el 171 ft 9 in. to 183 ft.	11.25	83.6	120	13.2	25	13.65	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	13.27	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	13.24	3124.98
7	Between hot, cold and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	289.03	120	13.2	25	14.96	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	12.76	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	12.89	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	10.93	837.79

TABLE 6.2.1-11 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	7.62	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	11.11	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	7.60	2815.04
14	Interface of two halves of SG compartment; between quadrants 1 and 2.	20.25	131.71	120	13.2	25	5.85	2400.44
15	Over node 4; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	13.52	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	13.28	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	12.88	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.44	120	13.2	25	10.96	796.98
19	Over node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	13.56	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	15.9	814.29
21	Over-node 7; from el 187 to 195 ft.	8.0	229.18	120	13.2	25	14.93	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	10.26	1631.41

TABLE 6.2.1-11 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	8.56	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	8.61	1572.59
25	Around RCP 4; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	13.01	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	13.11	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	13.2	25	13.2	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120	13.2	25	12.77	145.4
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	12.06	324.23
30	Between SG 4 and concrete beam; from el 195 to 200 ft.	5.0	176.1	120	13.2	25	12.06	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	12.73	460.28
32	Opposite node 27; from el 195 to 200 ft.	5.0	197.63	120	13.2	25	12.98	788.23
33	Between SG 1 and concrete beam; from el 195 to 290 ft.	5.0	176.84	120	13.2	25	11.55	671.4
34	Between SG 1 and wall; from el 195 to 200 ft.	5.0	81.91	120	13.2	25	11.14	315.67

TABLE 6.2.1-11 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	10.68	146.0
36	Between SG 1 and RCP 1; from el 195 to 200 ft.	5.0	99.23	120	13.2	25	8.59	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	8.60	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	8.61	513.74
39	Opposite node 36; from el 195 to 200 ft.	5.0	225.29	120	13.2	25	9.12	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	10.38	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	11.38	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	11.42	635.37
43	Between HVAC shaft, RCP 4, and SG 4; from el 200 to 220 ft.	20.0	144.38	120	13.2	25	11.53	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	11.50	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	11.46	697.96

TABLE 6.2.1-11 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	11.27	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	11.49	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	11.58	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	124.07	120	13.2	25	11.53	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120	13.2	25	10.89	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	11.07	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	10.96	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	10.8	402.25
54	Between HVAC shaft, RCP 1, and SG 1; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	8.55	2640.6

TABLE 6.2.1-11 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	8.33	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	8.36	1986.14
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	8.66	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	9.40	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	3.75	918.0
60	SW node of left half of SG 4 dog-house; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	4.41	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	3.28	404.2
62	SE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	3.88	1356.2
63	NE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	3.87	1234.0

TABLE 6.2.1-11 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	3.75	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	3.76	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	4.12	596.4
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	3.20	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	3.60	918.0
69	HVAC shaft 4	-	-	120	13.2	25	11.41	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	7.91	838.6
71	HVAC duct 4.	-	-	120	13.2	25	8.64	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	6.06	1231.6
73	Quadrants 2 and 3 of SG compartment.	-	-	120	13.2	25	3.06	8.36E+04
74	Containment atmosphere.	-	-	120	13.2	25	2.74	2.75E+06

TABLE 6.2.1-11A

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK ABOVE EL. 220 ft):
NODE CHARACTERISTICS

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Quadrant 1 from el 216 ft to 238 ft	12	149.7	120	14.7	20	8.06	1796
2	Quadrant 2 from el 229 ft to 238 ft	9	45.4	120	14.7	20	5.31	409
3	Quadrant 3 from el 229 ft to 238 ft	9	48.2	120	14.7	20	5.23	434
4	Quadrant 4 from el 229 ft to 238 ft	9	73.0	120	14.7	20	6.22	657
5	Quadrant 2 from el 216 ft to 229 ft	13	54.4	120	14.7	20	6.65	707
6	Quadrant 3 from el 216 ft to 229 ft	13	60.4	120	14.7	20	6.56	785
7	Quadrant 4 from el 216 ft to 229	13	79.9	120	14.7	20	6.90	1039
8	Quadrant 2 from el 200 ft to 216 ft	16	76.3	120	14.7	20	6.58	1221
9	Quadrant 3 from el 200 ft to 216 ft	16	166.56	120	14.7	20	6.49	2665
10	Quadrant 4 from el 200 ft to 216 ft	16	91.3	120	14.7	20	6.58	1461
11	Quadrant 1 from el 200 ft to 216 ft	16	186.13	120	14.7	20	6.60	2978
12	Containment Atmosphere	-	-	120	14.7	20	3.29	2.75×10^6

TABLE 6.2.1-12 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL 220 ft):
NODE CHARACTERISTICS STEAM GENERATOR INLET NOZZLE (306-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	3.08	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	7.43	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	7.91	1661.50
4	By SG 4; from el 171 ft 9 in. to 183 ft.	11.25	83.6	120	13.2	25	8.21	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	8.48	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	9.0	3124.98
7	Between hot, cold, and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	189.03	120	13.2	25	9.0	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	8.24	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	8.38	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	6.35	837.79

TABLE 6.2.1-12 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	3.90	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	6.57	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	3.95	2815.04
14	Interface of two halves of SG compartment; between quadrants 1 and 2.	20.25	131.71	120	13.2	25	2.71	2400.44
15	Over node 4; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	8.19	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	8.46	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	8.82	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.44	120	13.2	25	6.25	796.98
19	Over node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	7.43	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	7.98	814.29
21	Over node 7; from el 187 to 195 ft.	8.0	229.18	120	13.2	25	8.92	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	5.82	1631.41

TABLE 6.2.1-12 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	4.66	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	4.54	1572.59
25	Around RCP 4; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	7.19	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	7.16	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	13.2	25	7.18	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120	13.2	25	7.11	145.4
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	6.90	324.23
30	Between SC 4 and concrete beam; from el 195 to 200 ft.	5.0	176.1	120	13.2	25	6.94	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	7.13	460.28
32	Opposite node 27; from el 195 to 200 ft.	5.0	197.63	120	13.2	25	7.18	788.23
33	Between SG 1 and concrete beam; from el 195 to 200 ft.	5.0	176.84	120	13.2	25	6.70	671.4
34	Between SG 1 and wall; from el 195 to 200 ft.	5.0	81.91	120	13.2	25	6.43	315.67

TABLE 6.2.1-12 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	6.0	146.0
36	Between SG 1 and RCP 1; from el 195 to 200 ft.	5.0	99.23	120	13.2	25	4.67	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	4.54	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	4.53	513.74
39	Opposite node 36; from el 195 to 200 ft.	5.0	225.29	120	13.2	25	4.87	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	5.85	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	6.12	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	6.10	635.37
43	Between HVAC shaft, RCP 4, and SG 4. from el 200 to 220 ft.	20.0	144.38	120	13.2	25	6.22	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	6.23	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	6.39	697.96

TABLE 6.2.1-12 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	6.19	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	6.30	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	6.30	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	124.07	120	13.2	25	6.22	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120	13.2	25	6.03	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	6.23	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	6.07	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	5.97	402.25
54	Between HVAC shaft, RCP 1, and SG 1; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	4.38	2640.6

TABLE 6.2.1-12 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	4.23	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	4.27	1986.14
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	4.47	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	4.99	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	2.10	918.0
60	SW node of left half of SG 4 doghouse; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	2.33	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	1.94	404.2
62	SE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	2.17	1356.2
63	NE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	2.15	1234.0

TABLE 6.2.1-12 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	2.11	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	2.11	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	2.24	596.4
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	1.92	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	2.05	918.0
69	HVAC shaft 4	-	-	120	13.2	25	6.06	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	3.98	838.6
71	HVAC duct 4.	-	-	120	13.2	25	3.99	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	3.0	1231.6
73	Quadrants 2 and 3 of SG compartment.	-	-	120	13.2	25	1.85	8.36E+04
74	Containment atmosphere.	-	-	120	13.2	25	1.75	2.75E+06

TABLE 6.2.1-13 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL 220 ft):
NODE CHARACTERISTICS STEAM GENERATOR OUTLET NOZZLE (436-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	8.71	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	16.79	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	21.33	1661.50
4	By SG 4; from el 171 ft 9 in. to183 ft.	11.25	83.6	120	13.2	25	17.26	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	16.41	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	16.27	3124.98
7	Between hot, cold, and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	189.03	120	13.2	25	17.45	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	15.72	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	15.85	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	15.98	837.79

TABLE 6.2.1-13 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	9.83	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	13.90	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	9.81	2815.04
14	Interface of two halves of SG compartment; between quadrants 1 and 2.	20.25	131.71	120	13.2	25	7.53	2400.44
15	Over node 4; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	17.17	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	16.39	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	15.90	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.44	120	13.2	25	15.94	796.98
19	Over node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	16.83	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	24.79	814.29
21	Over-node 7; from el 187 to 195 ft.	8.0	229.18	120	13.2	25	17.57	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	12.96	1631.41

TABLE 6.2.1-13 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	11.15	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	11.08	1572.59
25	Around RCP 4; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	16.26	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	16.54	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	13.2	25	17.76	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120.	13.2	25	16.72	145.4
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	15.26	324.23
30	Between SG 4 and concrete beam; from el 195 to200 ft.	5.0	176.1	120	13.2	25	15.22	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	15.87	460.28
32	Opposite node 27; from el 195 to200 ft.	5.0	197.63	120	13.2	25	16.20	788.23
33	Between SG 1 and concrete beam; from el 195 to 200 ft.	5.0	176.84	120	13.2	25	14.66	671.4
34	Between SG 1 and wall; from el195 to 200 ft.	5.0	81.91	120	13.2	25	14.71	315.67

TABLE 6.2.1-13 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	14.36	146.0
36	Between SG 1 and RCP 1; from el195 to 200 ft.	5.0	99.23	120	13.2	25	11.14	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	11.18	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	11.10	513.74
39	Opposite node 36; from el 195 to200 ft.	5.0	225.29	120	13.2	25	11.55	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	13.20	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	14.58	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	14.54	635.37
43	Between HVAC shaft, RCP 4, and SG 4. from el 200 to 220 ft.	20.0	144.38	120	13.2	25	14.71	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	14.87	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	14.69	697.96

TABLE 6.2.1-13 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	14.48	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	14.63	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	14.74	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	124.07	120	13.2	25	14.7	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120 °	13.2	25	13.96	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	14.37	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	14.19	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	14.13	402.25
54	Between HVAC shaft, RCP 1, and SG 1,; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	11.02	2640.6

TABLE 6.2.1-13 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	10.75	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	10.77	1986.14
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	11.12	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	12.04	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	4.45	918.0
60	SW node of left half of SG 4 dog house; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	5.28	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	3.85	404.2
62	SE node of left half of SG 4 dog house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	4.62	1356.2
63	NE node of-left half of SG 4 dog house; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	4.60	1234.0

TABLE 6.2.1-13 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	4.47	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	4.48	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	4.96	596.4
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	3.76	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	4.28	918.0
69	HVAC shaft 4	-	-	120	13.2	25	14.63	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	10.20	838.6
71	HVAC duct 4.	-	-	120	13.2	25	11.34	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	7.88	1231.6
73	Quadrants 2 and 3 of SG compartment.	-	-	120	13.2	25	3.58	8.36E+04
74	Containment atmosphere.	-	-	120	13.2	25	3.15	2.75E+06

TABLE 6.2.1-14 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): NODE CHARACTERISTICS REACTOR COOLANT PUMP OUTLET NOZZLE (236-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	7.62	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	14.32	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	13.42	1661.50
4	By SG 4; from el 171 ft 9 in. to 183 ft.	11.25	83.6	120	13.2	25	12.51	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	12.36	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	12.34	3124.98
7	Between hot, cold, and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	189.03	120	13.2	25	14.33	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	11.92	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	12.0	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	10.56	837.79

TABLE 6.2.1-14 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	7.35	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	10.44	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	7.34	2815.04
14	Interface of two halves of SG compartment; between quadrants 1 and 2.	20.25	131.71	120	13.2	25	5.79	2400.44
15	Over node 14; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	12.41	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	12.32	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	12.0	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.44	120	13.2	25	10.55	796.98
19	0ver node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	14.37	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	13.37	814.29
21	Over node 7; from el 187 to 1'95 ft.	8.0	229.18	120	13.2	25	14.25	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	9.69	1631.41

TABLE 6.2.1-14 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	8.25	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	8.25	1572.59
25	Around RCP 14; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	13.08	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	13.04	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	13.2	25	12.47	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120	13.2	25	12.0	145.14
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	11.41	324.23
30	Between SG 4 and concrete beam; from el 195 to 200 ft.	5.0	176.1	120	13.2	25	11.43	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	12.24	460.28
32	Opposite node 27; from el 195 to 200 ft.	5.0	197.63	120	13.2	25	12.63	788.23
33	Between SG 1 and concrete beam; from el 195 to 200 ft.	5.0	176.84	120	13.2	25	10.87	671.14
34	Between SG 1 and wall; from el 195 to 200 ft.	5.0	81.91	120	13.2	25	10.69	315.67

TABLE 6.2.1-14 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	10.22	146.0
36	Between SG 1 and RCP 1; from el 195 to 200 ft.	5.0	99.23	120	13.2	25	8.25	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	8.23	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	8.23	513.74
39	Opposite node 36; from el 195 to 200 ft.	5.0	225.29	120	13.2	25	8.60	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	9.83	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	10.94	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	10.9	635.37
43	Between HVAC shaft, RCP 4, and SG 4. from el 200 to 220 ft.	20.0	144.38	120	13.2	25	10.99	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	10.95	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	10.93	697.96

TABLE 6.2.1-14 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	10.74	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	10.95	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	11.07	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	214.07	120	13.2	25	11.0	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120	13.2	25	10.36	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	10.58	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	10.46	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	10.32	402.25
54	Between HVAC shaft, RCP 1, and SG 1; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	8.21	2640.6

TABLE 6.2.1-14 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	7.98	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	8.0	1986.14
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	8.26	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	8.95	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	4.01	918.0
60	SW node of left half of SG 4 dog-house; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	4.60	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	3.59	404.2
62	SE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	4.13	1356.2
63	NE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	4.12	1234.0

TABLE 6.2.1-14 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	4.0	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	4.01	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	4.33	596.4
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	3.51	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	3.86	918.0
69	HVAC shaft 4		-	120	13.2	25	10.82	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	7.63	838.6
71	HVAC duct 4.	-	-	120	13.2	25	8.27	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	5.96	1231.6
73	Quadrants 2 and 3 of SG compartment.	-	-	120	13.2	25	3.41	8.36E+04
74	Containment atmosphere.	-	-	120	13.2	25	3.10	2.75E+06

TABLE 6.2.1-15 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft):
NODE CHARACTERISTICS REACTOR COOLANT PUMP INLET NOZZLE (336-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	7.13	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	13.64	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	15.95	1661.50
4	By SG 4; from el 171 ft 9 in. to 183 ft.	11.25	83.6	120	13.2	25	13.71	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	13.33	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	13.33	3124.98
7	Between hot, cold, and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	189.03	120	13.2	25	15.02	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	12.83	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	12.96	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	11.06	837.79

TABLE 6.2.1-15 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	7.76	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	11.19	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	7.75	2815.04
14	Interface of two halves of SG compartment between quadrants 1 and 2.	20.25	131.71	120	13.2	25	5.96	2400.44
15	Over node 14; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	13.58	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	13.32	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	12.94	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.144	120	13.2	25	11.06	796.98
19	Over node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	13.63	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	16.06	814.29
21	Over node 7; from el 187 to 195 ft.	8.0	229.18	120	13.2	25	14.98	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	10.33	1631.41

TABLE 6.2.1-15 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	8.73	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	8.75	1572.59
25	Around RCP 4; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	13.10	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	13.15	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	13.2	25	13.31	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120	13.2	25	12.84	145.14
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	12.11	324.23
30	Between SG 4 and concrete beam; from el 195 to 200 ft.	5.0	176.1	120	13.2	25	12.14	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	12.82	460.28
32	Opposite node 27; from el 195 to 200 ft.	5.0	197.63	120	13.2	25	13.06	788.23
33	Between SG 1 and concrete beam; from el 195 to 200 ft.	5.0	176.84	120	13.2	25	11.63	671.14
34	Between SG 1 and wall; from el 195 to 200 ft.	5.0	81.91	120	13.2	25	11.24	315.67

TABLE 6.2.1-15 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	10.75	146.0
36	Between SG 1 and RCP 1; from el 195 to 200 ft.	5.0	99.23	120	13.2	25	8.75	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	8.83	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	8.76	513.74
39	Opposite node 36; from el 195 to 200 ft.	5.0	225.29	120	13.2	25	9.13	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	10.46	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	11.48	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	11.42	635.37
43	Between HVAC shaft, RCP 4, and SG 4. from el 200 to 22O ft.	20.0	144.38	120	13.2	25	11.56	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	11.67	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	11.57	697.96

TABLE 6.2.1-15 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft²)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft³)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	11.37	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	11.56	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	11.67	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	124.07	120	13.2	25	11.58	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120	13.2	25	10.97	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	11.14	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	11.03	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	10.88	402.25
54	Between HVAC shaft, RCP 1, and SG 1; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	8.69	2640.6

TABLE 6.2.1-15 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	8.46	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	8.48	1986.14
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	8.76	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	9.49	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	3.76	918.0
60	SW node of left half of SG 14 doghouse; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	4.42	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	3.29	1404.2
62	SE node of left half of SG 14 dog-house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	3.89	1356.2
63	NE node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	3.88	1234.0

TABLE 6.2.1-15 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	3.76	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	3.77	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	4.13	596.14
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	3.20	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	3.60	918.0
69	HVAC shaft 4		-	120	13.2	25	11.45	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	8.04	838.6
71	HVAC duct 4.	-	-	120	13.2	25	8.72	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	6.20	1231.6
73	Quadrants 2 and 3 of SG compartment.	-	-	120	13.2	25	3.06	8.36E+O14
74	Containment atmosphere.	-	-	120	13.2	25	2.73	2.75E+O6

TABLE 6.2.1-16 (SHEET 1 OF 7)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL 220 ft):
NODE CHARACTERISTICS STEAM GENERATOR INLET ELBOW (763-in.^{2} BREAK)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
1	Interface of two halves of SG compartment.	6.56	222.97	120	13.2	25	10.54	1316.42
2	Between cold leg and wall; from el 171 ft 9 in. to 187 ft.	15.25	176.0	120	13.2	25	19.63	2283.14
3	Between RCP 4, SG 4, and wall; from el 171 ft 9 in. to 187 ft.	15.25	132.24	120	13.2	25	20.52	1661.50
4	By SG 4; from el 171 ft 9 in. to 183 ft.	11.25	83.6	120	13.2	25	20.97	809.05
5	Adjacent to SG 4; from el 171 ft 9 in. to 183 ft.	11.25	95.78	120	13.2	25	21.34	895.04
6	Around SG 4; from el 171 ft 9 in. to 195 ft.	23.25	165.73	120	13.2	25	22.13	3124.98
7	Between hot, cold, and suction legs; from el 171 ft 9 in. to 187 ft.	15.25	189.03	120	13.2	25	22.16	3721.19
8	Around SG 1; from el 171 ft 9 in. to 195 ft.	23.25	163.51	120	13.2	25	20.93	3078.62
9	Adjacent to SG 1; from el 171 ft 9 in. to 183 ft.	11.25	95.31	120	13.2	25	21.12	890.21
10	By SG 1; from el 171 ft 9 in. to 183 ft.	11.25	86.44	120	13.2	25	17.50	837.79

TABLE 6.2.1-16 (SHEET 2 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
11	Between RCP 1, SG 1, and wall; from el 171 ft 9 in. to 187 ft.	15.25	137.8	120	13.2	25	12.69	1737.79
12	Between hot, cold, and suction legs 1; from el 171 ft 9 in. to 187 ft.	15.25	289.64	120	13.2	25	18.17	3729.58
13	Between hot leg 1 and wall; from el 171 ft 9 in. to 187 ft.	15.25	219.7	120	13.2	25	12.66	2815.04
14	Interface of two halves of SG compartment; between quadrants 1 and 2.	20.25	131.71	120	13.2	25	9.83	2400.44
15	Over node 4; from el 183 to 195 ft.	12.0	83.6	120	13.2	25	20.90	766.38
16	Over node 5; from el 183 to 195 ft.	12.0	95.78	120	13.2	25	21.22	892.25
17	Over node 9 in quadrant 1; from el 183 to 195 ft.	12.0	95.31	120	13.2	25	21.03	887.09
18	Over node 10 in quadrant 1; from el 183 to 195 ft.	12.0	86.44	120	13.2	25	17.52	796.98
19	Over node 2; from el 187 to 195 ft.	8.0	179.22	120	13.2	25	19.66	1251.37
20	Over node 3; from el 187 to 195 ft.	8.0	113.1	120	13.2	25	20.54	814.29
21	Over node 7; from el 187 to 195 ft.	8.0	229.18	120	13.2	25	22.03	1581.52
22	Over node 12; from el 187 to 195 ft.	8.0	236.11	120	13.2	25	16.74	1631.41

TABLE 6.2.1-16 (SHEET 3 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
23	Over node 11; from el 187 to 195 ft.	8.0	110.51	120	13.2	25	14.13	795.67
24	Over node 13; from el 187 to 195 ft.	8.0	223.84	120	13.2	25	14.21	1572.59
25	Around RCP 4; from el 195 to 200 ft.	5.0	89.33	120	13.2	25	19.22	402.0
26	Between RCP 4 and wall; from el 195 to 200 ft.	5.0	31.62	120	13.2	25	19.26	142.3
27	Between SG 4 and RCP 4; from el 195 to 200 ft.	5.0	89.14	120	19.28	25	19.28	348.52
28	Between SG 4 and HVAC shaft; from el 195 to 200 ft.	5.0	32.94	120	13.2	25	19.24	145.14
29	Between SG 4 and wall; from el 195 to 200 ft.	5.0	83.81	120	13.2	25	18.83	324.23
30	Between SG 4 and concrete beam; from el 195 to 200 ft.	5.0	176.1	120	13.2	25	18.93	668.07
31	Adjacent to node 32; from el 195 to 200 ft.	5.0	102.29	120	13.2	25	19.31	460.28
32	Opposite node 27; from el 195 to 200 ft. 200 ft.	5.0	197.63	120	13.2	25	19.34	788.23
33	Between SG 1 and concrete beam; from el 195 to 200 ft.	5.0	176.84	120	13.2	25	18.44	671.4
34	Between SG 1 and wall; from el 195 to 200 ft.	5.0	81.91	120	13.2	25	17.86	315.67

TABLE 6.2.1-16 (SHEET 4 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
35	Between SG 1 and HVAC shaft; from el 195 to 200 ft.	5.0	33.08	120	13.2	25	17.03	146.0
36	Between SG 1 and RCP 1; from el 195 to 200 ft.	5.0	99.23	120	13.2	25	14.21	393.9
37	Between RCP 1 and wall; from el 195 to 200 ft.	5.0	37.59	120	13.2	25	14.41	169.15
38	Around RCP 1; from el 195 to 200 ft.	5.0	114.17	120	13.2	25	14.22	513.74
39	Opposite node 36; from el 195 to 200 ft.	5.0	225.29	120	13.2	25	14.93	912.67
40	Adjacent to node 39; from el 195 to 200 ft.	5.0	103.1	120	13.2	25	16.82	463.95
41	Between RCP 4 and northern wall of quadrant; from el 200 to 220 ft.	20.0	100.1	120	13.2	25	17.55	1796.2
42	Between RCP 4 and SW wall of quadrant; from el 200 to 220 ft.	20.0	35.3	120	13.2	25	17.52	635.37
43	Between HVAC shaft, RCP 4, and SG 4; from el 200 to 220 ft.	20.0	144.38	120	13.2	25	17.72	2577.68
44	Between SG 4 and southern wall of quadrant; from el 200 to 215 ft.	15.0	31.7	120	13.2	25	17.72	419.36
45	Between node 44 and quadrant 1; from el 200 to 207 ft.	7.0	111.91	102	13.2	25	17.94	697.96

TABLE 6.2.1-16 (SHEET 5 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
46	Over node 45; from el 207 to 215 ft.	8.0	110.59	120	13.2	25	17.63	796.22
47	Between node 48 and quadrant 1; from el 200 to 215 ft.	15.0	225.12	120	13.2	25	17.82	3015.61
48	Between SG 4 and northern wall, adjacent to node 49; from el 200 to 215 ft.	15.0	93.84	120	13.2	25	17.82	1258.89
49	Adjacent to node 43; between RCP 4 and northern wall; from el 200 to 220 ft.	20.0	124.07	120	13.2	25	17.73	2214.86
50	Between node 58 and quadrant 4; from el 200 to 215 ft.	15.0	224.85	120	13.2	25	17.28	3012.0
51	Between node 53 and quadrant 4; from el 200 to 207 ft.	7.0	111.55	120	13.2	25	17.71	695.7
52	Over node 51; from el 207 to 215 ft.	8.0	110.91	120	13.2	25	17.38	798.54
53	Between SG 1 and southern wall of quadrant 1; from el 200 to 215 ft.	15.0	30.43	120	13.2	25	17.20	402.25
54	Between HVAC shaft, RCP 1, and SG 1; from el 200 to 220 ft.	20.0	147.88	120	13.2	25	14.08	2640.6

TABLE 6.2.1-16 (SHEET 6 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
55	Between RCP 1 and SE wall of quadrant 1; from el 200 to 220 ft.	20.0	35.91	120	13.2	25	13.78	646.32
56	Between RCP 1 and northern wall of quadrant 1; from el 200 to 220 ft.	20.0	110.65	120	13.2	25	13.82	1986.114
57	Adjacent to node 54; between RCP 1 and northern wall; from el 200 to 220 ft.	20.0	139.29	120	13.2	25	14.23	2488.86
58	Between SG 1 and northern wall, adjacent to node 57; from el 200 to 215 ft.	15.0	100.43	120	13.2	25	15.26	1347.85
59	NW node of left half of SG 4 doghouse; from el 215 to 238 ft.	23.0	50.73	120	13.2	25	5.65	918.0
60	SW node of left half of SG 4 dog-house; from el 215 to 229 ft.	14.0	52.98	120	13.2	25	6.62	596.4
61	Over node 60; from el 229 to 238 ft.	9.0	52.84	120	13.2	25	4.92	404.2
62	SE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.51	120	13.2	25	5.85	1356.2
63	NE node of left half of SG 4 dog-house; from el 215 to 238 ft.	23.0	79.87	120	13.2	25	5.83	1234.0

TABLE 6.2.1-16 (SHEET 7 OF 7)

Volume No.	Description	Height (ft)	Cross-Sectional Area (ft^2)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psig)	Net Free Volume (ft^3)
64	Symmetric of node 63 in quadrant 1.	23.0	79.87	120	13.2	25	5.69	1234.0
65	Symmetric of node 62 in quadrant 1.	23.0	79.51	120	13.2	25	5.71	1356.2
66	Symmetric of node 60 in quadrant 1.	14.0	52.98	120	13.2	25	6.30	596.4
67	Symmetric of node 61 in quadrant 1.	9.0	52.84	120	13.2	25	14.83	404.2
68	Symmetric of node 59 in quadrant 1.	23.0	50.73	120	13.2	25	5.47	918.0
69	HVAC shaft 14	-	-	120	13.2	25	17.45	838.6
70	HVAC shaft 1.	-	-	120	13.2	25	13.11	838.6
71	HVAC duct 14.	-	-	120	13.2	25	13.69	1231.6
72	HVAC duct 1.	-	-	120	13.2	25	10.24	1231.6
73	Quadrants 2 and 3 of SG compartment.	-		120	13.2	25	14.63	8.36E+04
74	Containment atmosphere.	-	-	120	13.2	25	14.07	2.75E+06

TABLE 6.2.1-16A

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK ABOVE EL 220 ft):
FLOW CHARACTERISTICS

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	2-1		X	22.8	0	0	1	0.41	1.41	0.329
2	1-4		X	39.1	0	0	1	0.41	1.41	0.234
3	1-5		X	52.4	0	0	1	0.41	1.41	0.097
4	1-7		X	75.8	0	0	1	0.41	1.41	0.070
5	1-11		X	23.2	0	0	1	0.35	1.35	0.286
6	1-12		X	73.0	2.04	0	1	0.20	3.24	0.139
7	2-3		X	30.5	0	0	1	0.41	1.41	0.777
8	2-5		X	32.6	0	0	1	0.16	1.16	0.211
9	2-12		X	45.4	0	0	1	0.02	1.02	0.099
10	3-4		X	22.8	0	0	1	0.41	1.41	0.637
11	3-6		X	34.3	0	0	1	0.16	1.16	0.199
12	3-12		X	48.26	0	0	1	0.02	1.02	0.089
13	4-7		X	49.1	0	0	1	0.18	1.18	0.131
14	4-12		X	73.0	2.04	0	1	0.20	3.24	0.059
15	5-6		X	63.5	0	0	1	0.41	1.41	0.186
16	5-8		X	13.44	0	0	1	0.38	1.38	0.302
17	6-7		X	52.4	0	0	1	0.41	1.41	0.155
18	6-9		X	19.06	0	0	1	0.35	1.35	0.256
19	7-10		X	19.12	0	0	1	0.38	1.38	0.219
20	8-9		X	86.0	0	0	1	0.29	1.29	0.139
21	8-11		X	178.6	0	0	1	0.29	1.29	0.054
22	9-10		X	52.1	0	0	1	0.29	1.29	0.168
23	10-11		X	93.7	0	0	1	0.41	1.41	0.126
24	9-12		X	29.5	0	0	1	0.45	1.45	0.198

TABLE 6.2.1-17 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
STEAM GENERATOR INLET ELBOW (763-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19		X	29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21		X	58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15		X	36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	9-8		X	113.9	0	0	1.0	0.25	1.25	0.08965
30	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185

TABLE 6.2.1-17 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
59	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-23		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55

TABLE 6.2.1-17 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	4	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74	X		43.29	0	0	1.0	0.26	1.26	0.123
118	56-74	X		43.29	0	0	1.0	0.26	1.26	0.123
119	42-74	X		30.16	0	0	1.0	0.04	1.04	0.338
120	55-74	X		30.16	0	0	1.0	0.04	1.04	0.338
121	43-74	X		9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74	X		9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74	X		15.82	0	0	1.0	0.43	1.43	0.095
124	57-74	X		15.82	0	0	1.0	0.43	1.43	0.095
125	50-64	X		15.1	0	0	1.0	0.46	1.46	0.276
126	47-63	X		15.1	0	0	1.0	0.46	1.46	0.276
127	52-65	X		21.55	0	0	1.0	0.39	1.39	0.259
128	46-62	X		21.55	0	0	1.0	0.39	1.39	0.259
129	53-66	X		8.63	0	0	1.0	0.35	1.35	0.475
130	44-60	X		8.63	0	0	1.0	0.35	1.35	0.475

TABLE 6.2.1-17 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60	X		10.05	0	0	1.0	0.46	1.46	0.465
133	58-68	X		9.95	0	0	1.0	0.44	1.44	0.374
134	48-59	X		9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.2	1.2
172	62-63		X	0.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55

TABLE 6.2.1-17 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.90	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69		X	5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74	X		7.2	0	0	1.0	0.44	1.44	0.776
193	70-74	X		7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	38.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74	X		39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-18 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
STEAM GENERATOR INLET NOZZLE (306-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19		X	29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21		X	58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15		X	36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	9-8		X	113.9	0	0	1.0	0.25	1.25	0.08965
30	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389

TABLE 6.2.1-18 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
59	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-28		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	0	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119

TABLE 6.2.1-18 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74		X	43.29	0	0	1.0	0.26	1.26	0.123
118	56-74		X	43.29	0	0	1.0	0.26	1.26	0.123
119	42-74		X	30.16	0	0	1.0	0.04	1.04	0.338
120	55-74		X	30.16	0	0	1.0	0.04	1.04	0.338
121	43-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74		X	15.82	0	0	1.0	0.43	1.43	0.095
124	57-74		X	15.82	0	0	1.0	0.43	1.43	0.095
125	50-64		X	15.1	0	0	1.0	0.46	1.46	0.276
126	47-63		X	15.1	0	0	1.0	0.46	1.46	0.276
127	52-65		X	21.55	0	0	1.0	0.39	1.39	0.259
128	46-62		X	21.55	0	0	1.0	0.39	1.39	0.259
129	53-66		X	8.63	0	0	1.0	0.35	1.35	0.475
130	44-60		X	8.63	0	0	1.0	0.35	1.35	0.475
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60		X	10.05	0	0	1.0	0.46	1.46	0.465
133	58-68		X	9.95	0	0	1.0	0.44	1.44	0.374
134	48-59		X	9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127

TABLE 6.2.1-18 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.0	1.2
172	62-63		X	30.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.09	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69		X	5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74		X	7.2	0	0	1.0	0.44	1.44	0.776
193	70-74		X	7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	33.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795

TABLE 6.2.1-18 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-19 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
STEAM GENERATOR OUTLET NOZZLE (436-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19	X		29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21	X		58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15	X		36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	9-8		X	113.9	0	0	1.0	0.25	1.25	0.08965
30	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389

TABLE 6.2.1-19 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
59	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-23		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	0	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119

TABLE 6.2.1-19 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow Choked	Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74	X		43.29	0	0	1.0	0.26	1.26	0.123
118	56-74		X	43.29	0	0	1.0	0.26	1.26	0.123
119	42-74	X		30.16	0	0	1.0	0.04	1.04	0.338
120	55-74		X	30.16	0	0	1.0	0.04	1.04	0.338
121	43-74	X		9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74	X		15.82	0	0	1.0	0.43	1.43	0.095
124	57-74		X	15.82	0	0	1.0	0.43	1.43	0.095
125	50-64	X		15.1	0	0	1.0	0.46	1.46	0.276
126	47-63	X		15.1	0	0	1.0	0.46	1.46	0.276
127	52-65	X		21.55	0	0	1.0	0.39	1.39	0.259
128	46-62	X		21.55	0	0	1.0	0.39	1.39	0.259
129	53-66		X	8.63	0	0	1.0	0.35	1.35	0.475
130	44-60		X	8.63	0	0	1.0	0.35	1.35	0.475
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60		X	10.05	0	0	1.0	0.46	1.46	0.465
133	58-68		X	9.95	0	0	1.0	0.44	1.44	0.374
134	48-59	X		9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127

TABLE 6.2.1-19 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.0	1.2
172	62-63		X	30.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.90	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69	X		5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74	X		7.2	0	0	1.0	0.44	1.44	0.776
193	70-74		X	7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	33.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795

TABLE 6.2.1-19 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-20 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
REACTOR COOLANT PUMP OUTLET NOZZLE (236-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19		X	29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21		X	58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15		X	36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	9-8		X	113.9	0	0	1.0	0.25	1.25	0.08965
30	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185

TABLE 6.2.1-20 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
59	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-23		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55

TABLE 6.2.1-20 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	0	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74		X	43.29	0	0	1.0	0.26	1.26	0.123
118	56-74		X	43.29	0	0	1.0	0.26	1.26	0.123
119	42-74		X	30.16	0	0	1.0	0.04	1.04	0.338
120	55-74		X	30.16	0	0	1.0	0.04	1.04	0.338
121	43-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74		X	15.82	0	0	1.0	0.43	1.43	0.095
124	57-74		X	15.82	0	0	1.0	0.43	1.43	0.095
125	50-64		X	15.1	0	0	1.0	0.46	1.46	0.276
126	47-63		X	15.1	0	0	1.0	0.46	1.46	0.276
127	52-65		X	21.55	0	0	1.0	0.39	1.39	0.259
128	46-62		X	21.55	0	0	1.0	0.39	1.39	0.259
129	53-66		X	8.63	0	0	1.0	0.35	1.35	0.475
130	44-60		X	8.63	0	0	1.0	0.35	1.35	0.475

TABLE 6.2.1-20 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow Choked	Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60		X	10.05	0	0	1.0	0.46	1.46	0.465
133	58-68		X	9.95	0	0	1.0	0.44	1.44	0.374
134	48-59		X	9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.0	1.2
172	62-63		X	30.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55

TABLE 6.2.1-20 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow		Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
		Choked	Unchoked							
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.90	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69		X	5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74		X	7.2	0	0	1.0	0.44	1.44	0.776
193	70-74		X	7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	33.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-21 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
REACTOR COOLANT PUMP INLET NOZZLE (336-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow		Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
		Choked	Unchoked							
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19		X	29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21		X	58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15		X	36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	9-8		X	113.9	0	0	1.0	0.25	1.25	0.08965
30	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389

TABLE 6.2.1-21 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
59	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-23		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	0	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119

TABLE 6.2.1-21 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74	X		43.29	0	0	1.0	0.26	1.26	0.123
118	56-74		X	43.29	0	0	1.0	0.26	1.26	0.123
119	42-74	X		30.16	0	0	1.0	0.04	1.04	0.338
120	55-74		X	30.16	0	0	1.0	0.04	1.04	0.338
121	43-74	X		9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74	X		15.82	0	0	1.0	0.43	1.43	0.095
124	57-74		X	15.82	0	0	1.0	0.43	1.43	0.095
125	50-64		X	15.1	0	0	1.0	0.46	1.46	0.276
126	47-63		X	15.1	0	0	1.0	0.46	1.46	0.276
127	52-65		X	21.55	0	0	1.0	0.39	1.39	0.259
128	46-62		X	21.55	0	0	1.0	0.39	1.39	0.259
129	53-66		X	8.63	0	0	1.0	0.35	1.35	0.475
130	44-60		X	8.63	0	0	1.0	0.35	1.35	0.475
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60		X	10.05	0	0	1.0	0.46	1.46	0.465
133	58-68		X	9.95	0	0	1.0	0.44	1.44	0.374
134	48-59		X	9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127

TABLE 6.2.1-21 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.0	1.2
172	62-63		X	30.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.90	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69		X	5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74	X		7.2	0	0	1.0	0.44	1.44	0.776
193	70-74		X	7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	33.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795

TABLE 6.2.1-21 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow Choked	Description of Flow Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-22 (SHEET 1 OF 5)

STEAM GENERATOR COMPARTMENT MODEL (PIPE BREAK BELOW EL. 220 ft): FLOW CHARACTERISTICS
LOOP CLOSURE WELD (336-in.^{2} BREAK)

Vent Path	Node Number (From-To)	Description of Flow		Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
		Choked	Unchoked							
1	1-74		X	20.25	0	0	1.0	0.38	1.38	0.096
2	1-73		X	97.08	0	0	1.0	0	1.0	0.079
3	20-19		X	29.58	0	0	1.0	0.27	1.27	0.295
4	23-24		X	29.58	0	0	1.0	0.27	1.27	0.295
5	2-1		X	92.1	0	0	1.0	0.25	1.25	0.122
6	3-2		X	63.12	0	0	1.0	0.24	1.24	0.151
7	11-13		X	63.12	0	0	1.0	0.24	1.24	0.151
8	7-2		X	94.0	0	0	1.0	0.27	1.27	0.0623
9	13-12		X	125.8	0	0	1.0	0.27	1.27	0.0623
10	21-19		X	64.65	0	0	1.0	0.20	1.20	0.136
11	22-24		X	80.0	0	0	1.0	0.20	1.20	0.134
12	3-7		X	47.57	0	0	1.0	0.41	1.41	0.0698
13	11-12		X	47.57	0	0	1.0	0.41	1.41	0.0698
14	20-21		X	58.72	0	0	1.0	0.29	1.29	0.297
15	23-22		X	58.72	0	0	1.0	0.29	1.29	0.297
16	3-4		X	63.65	0	0	1.0	0.28	1.28	0.103
17	11-10		X	63.65	0	0	1.0	0.28	1.28	0.103
18	20-15		X	36.93	0	0	1.0	0.25	1.25	0.192
19	23-18		X	36.93	0	0	1.0	0.25	1.25	0.192
20	7-6		X	129.19	0	0	1.0	0.27	1.27	0.0509
21	12-8		X	129.19	0	0	1.0	0.27	1.27	0.0509
22	21-6		X	70.58	0	0	1.0	0.26	1.26	0.1303
23	22-8		X	70.58	0	0	1.0	0.26	1.26	0.1303
24	4-5		X	99.53	0	0	1.0	0.07	1.07	0.1265
25	10-9		X	99.53	0	0	1.0	0.07	1.07	0.1265
26	15-16		X	81.3	0	0	1.0	0.17	1.17	0.1093
27	18-17		X	81.3	0	0	1.0	0.17	1.17	0.1093
28	5-6		X	113.9	0	0	1.0	0.25	1.25	0.08965
29	16-6		X	94.2	0	0	1.0	0.29	1.29	0.153
31	17-8		X	94.2	0	0	1.0	0.29	1.29	0.153
32	6-8		X	247.93	0	0	1.0	0.16	1.16	0.0310
33	5-9		X	76.37	0	0	1.0	0.17	1.17	0.1071
34	16-17		X	68.31	0	0	1.0	0.22	1.22	0.115
35	13-14		X	92.65	0	0	1.0	0.30	1.30	0.086
36	24-14		X	13.7	0	0	1.0	0.38	1.38	0.163
37	14-73		X	44.86	0	0	1.0	0.40	1.40	0.1063
38	14-74		X	50.6	0	0	1.0	0.35	1.35	0.42
39	19-25		X	80.58	0	0	1.0	0.05	1.05	0.0798
40	24-38		X	80.58	0	0	1.0	0.05	1.05	0.0798
41	19-26		X	28.52	0	0	1.0	0.18	1.18	0.185
42	24-37		X	28.52	0	0	1.0	0.18	1.18	0.185
43	19-32		X	53.45	0	0	1.0	0	1.0	0.112
44	24-39		X	53.45	0	0	1.0	0	1.0	0.112
45	20-27		X	62.58	0	0	1.0	0.15	1.15	0.0794
46	23-36		X	62.58	0	0	1.0	0.15	1.15	0.0794
47	21-32		X	81.5	0	0	1.0	0.27	1.27	0.0389

TABLE 6.2.1-22 (SHEET 2 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
48	22-39		X	81.5	0	0	1.0	0.27	1.27	0.0389
49	21-31		X	57.19	0	0	1.0	0.22	1.22	0.0778
50	22-40		X	57.19	0	0	1.0	0.22	1.22	0.0778
51	15-28		X	12.58	0	0	1.0	0.30	1.30	0.2903
52	18-35		X	12.58	0	0	1.0	0.30	1.30	0.2903
53	15-29		X	21.43	0	0	1.0	0.16	1.16	0.2396
54	18-34		X	21.43	0	0	1.0	0.16	1.16	0.2396
55	4-15		X	66.88	0	0	1.0	0.10	1.10	0.161
56	10-18		X	66.88	0	0	1.0	0.10	1.10	0.161
57	5-16		X	76.62	0	0	1.0	0.10	1.10	0.1378
58	9-17		X	76.62	0	0	1.0	0.10	1.10	0.1378
5	6-30		X	80.58	0	0	1.0	0.26	1.26	0.113
60	8-33		X	80.58	0	0	1.0	0.26	1.26	0.113
61	16-29		X	11.0	0	0	1.0	0.40	1.40	0.195
62	17-34		X	11.0	0	0	1.0	0.40	1.40	0.195
63	16-30		X	6.89	0	0	1.0	0.37	1.37	0.321
64	17-33		X	6.89	0	0	1.0	0.37	1.37	0.321
65	2-19		X	107.33	0	0	1.0	0.20	1.20	0.0711
66	13-24		X	107.33	0	0	1.0	0.20	1.20	0.0711
67	3-20		X	85.83	0	0	1.0	0.18	1.18	0.102
68	11-23		X	85.83	0	0	1.0	0.18	1.18	0.102
69	7-21		X	174.0	0	0	1.0	0.20	1.20	0.0482
70	12-22		X	174.0	0	0	1.0	0.20	1.20	0.0482
71	25-26		X	18.42	0	0	1.0	0.74	1.74	0.43
72	38-37		X	18.42	0	0	1.0	0.74	1.74	0.43
73	25-32		X	34.77	0	0	1.0	0.39	1.39	0.30
74	38-39		X	34.77	0	0	1.0	0.39	1.39	0.30
75	26-27		X	17.66	0	0	1.0	0.74	1.74	0.44
76	37-36		X	17.66	0	0	1.0	0.74	1.74	0.44
77	27-32		X	29.7	0	0	1.0	0.28	1.28	0.36
78	36-39		X	29.7	0	0	1.0	0.28	1.28	0.36
79	27-28		X	26.72	0	0	1.0	0.20	1.20	0.31
80	36-35		X	26.72	0	0	1.0	0.20	1.20	0.31
81	32-31		X	68.93	0	0	1.0	0.10	1.10	0.13
82	39-40		X	68.93	0	0	1.0	0.10	1.10	0.13
83	31-30		X	58.55	0	0	1.0	0.34	1.34	0.15
84	40-33		X	58.55	0	0	1.0	0.34	1.34	0.15
85	28-29		X	13.75	0	0	1.0	0.56	1.56	0.55
86	35-34		X	13.75	0	0	1.0	0.56	1.56	0.55
87	29-30		X	26.7	0	0	1.0	0.23	1.23	0.35
88	34-33		X	26.7	0	0	1.0	0.23	1.23	0.35
89	29-34		X	4.56	0	0	1.0	0.38	1.38	1.04
90	30-33		X	34.72	0	0	1.0	0.35	1.35	0.23
91	25-41		X	51.3	0	0	1.0	0.21	1.21	0.154
92	38-56		X	51.3	0	0	1.0	0.21	1.21	0.154
93	26-42		X	18.36	0	0	1.0	0.21	1.21	0.434
94	37-55		X	18.36	0	0	1.0	0.21	1.21	0.434
95	27-43		X	54.96	0	0	1.0	0.22	1.22	0.138
96	36-54		X	54.96	0	0	1.0	0.22	1.22	0.138
97	32-49		X	58.97	0	0	1.0	0.25	1.25	0.119

TABLE 6.2.1-22 (SHEET 3 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
98	39-57		X	58.97	0	0	1.0	0.25	1.25	0.119
99	32-43		X	9.55	0	0	1.0	0.29	1.29	0.615
100	39-54		X	9.55	0	0	1.0	0.29	1.29	0.615
101	32-48		X	46.3	0	0	1.0	0.13	1.13	0.17
102	39-58		X	46.3	0	0	1.0	0.13	1.13	0.17
103	31-48		X	15.89	0	0	1.0	0.10	1.10	0.56
104	40-58		X	15.89	0	0	1.0	0.10	1.10	0.56
105	31-47		X	50.26	0	0	1.0	0.13	1.13	0.16
106	40-50		X	50.26	0	0	1.0	0.13	1.13	0.16
107	28-43		X	7.65	0	0	1.0	0.27	1.27	0.814
108	35-54		X	7.65	0	0	1.0	0.27	1.27	0.814
109	28-44		X	9.16	0	0	1.0	0.18	1.18	0.76
110	35-53		X	9.16	0	0	1.0	0.18	1.18	0.76
111	30-45		X	26.8	0	0	1.0	0.09	1.09	0.201
112	33-51		X	26.8	0	0	1.0	0.09	1.09	0.201
113	30-47		X	116.2	0	0	1.0	0.10	1.10	0.076
114	33-50		X	116.2	0	0	1.0	0.10	1.10	0.076
115	29-45		X	56.71	0	0	1.0	0.10	1.10	0.16
116	34-51		X	56.71	0	0	1.0	0.10	1.10	0.16
117	41-74	X		43.29	0	0	1.0	0.26	1.26	0.123
118	56-74		X	43.29	0	0	1.0	0.26	1.26	0.123
119	42-74	X		30.16	0	0	1.0	0.04	1.04	0.338
120	55-74		X	30.16	0	0	1.0	0.04	1.04	0.338
121	43-74	X		9.63	0	0	1.0	0.47	1.47	0.0794
122	54-74		X	9.63	0	0	1.0	0.47	1.47	0.0794
123	49-74	X		15.82	0	0	1.0	0.43	1.43	0.095
124	57-74		X	15.82	0	0	1.0	0.43	1.43	0.095
125	50-64		X	15.1	0	0	1.0	0.46	1.46	0.276
126	47-63		X	15.1	0	0	1.0	0.46	1.46	0.276
127	52-65		X	21.55	0	0	1.0	0.39	1.39	0.259
128	46-62		X	21.55	0	0	1.0	0.39	1.39	0.259
129	53-66		X	8.63	0	0	1.0	0.35	1.35	0.475
130	44-60		X	8.63	0	0	1.0	0.35	1.35	0.475
131	54-66		X	10.05	0	0	1.0	0.46	1.46	0.465
132	43-60		X	10.05	0	0	1.0	0.46	1.46	0.465
133	58-68		X	9.95	0	0	1.0	0.44	1.44	0.374
134	48-59		X	9.95	0	0	1.0	0.44	1.44	0.374
135	45-46		X	91.35	0	0	1.0	0.04	1.04	0.0828
136	51-52		X	91.35	0	0	1.0	0.04	1.04	0.0828
137	41-42		X	81.18	0	0	1.0	0.25	1.25	0.094
138	56-55		X	81.18	0	0	1.0	0.25	1.25	0.094
139	41-49		X	128.07	0	0	1.0	0.23	1.23	0.056
140	56-57		X	128.07	0	0	1.0	0.23	1.23	0.056
141	42-43		X	78.89	0	0	1.0	0.08	1.08	0.128
142	55-54		X	78.89	0	0	1.0	0.08	1.08	0.128
143	43-49		X	107.19	0	0	1.0	0.24	1.24	0.0893
144	54-57		X	107.19	0	0	1.0	0.24	1.24	0.0893
145	43-48		X	85.73	0	0	1.0	0.33	1.33	0.0482
146	54-58		X	85.73	0	0	1.0	0.33	1.33	0.0482
147	43-44		X	95.48	0	0	1.0	0.31	1.31	0.127

TABLE 6.2.1-22 (SHEET 4 OF 5)

Vent Path	Node Number (From-To)	Description of Flow Choked	Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
148	54-53		X	95.48	0	0	1.0	0.31	1.31	0.127
149	49-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
150	57-58		X	109.22	0	0	1.0	0.27	1.27	0.0406
151	48-47		X	178.11	0	0	1.0	0.53	1.53	0.053
152	58-50		X	178.11	0	0	1.0	0.53	1.53	0.053
153	44-45		X	22.12	0	0	1.0	0.56	1.56	0.308
154	53-51		X	22.12	0	0	1.0	0.56	1.56	0.308
155	44-46		X	29.81	0	0	1.0	0.50	1.50	0.276
156	53-52		X	29.81	0	0	1.0	0.50	1.50	0.276
157	45-47		X	27.88	0	0	1.0	0.61	1.61	0.186
158	51-50		X	27.88	0	0	1.0	0.61	1.61	0.186
159	46-47		X	53.57	0	0	1.0	0.55	1.55	0.173
160	52-50		X	53.57	0	0	1.0	0.55	1.55	0.173
161	47-50		X	175.22	0	0	1.0	0.16	1.16	0.065
162	45-51		X	35.04	0	0	1.0	0.25	1.25	0.26
163	46-52		X	58.39	0	0	1.0	0.14	1.14	0.229
164	59-60		X	21.0	0	0	1.0	0.27	1.27	0.60
165	68-66		X	21.0	0	0	1.0	0.27	1.27	0.60
166	59-61		X	13.5	0	0	1.0	0.35	1.35	0.937
167	68-67		X	13.5	0	0	1.0	0.35	1.35	0.937
168	60-62		X	16.38	0	0	1.0	0.18	1.18	0.77
169	66-65		X	16.38	0	0	1.0	0.18	1.18	0.77
170	61-62		X	10.53	0	0	1.0	0	1.0	1.2
171	67-65		X	10.53	0	0	1.0	0	1.0	1.2
172	62-63		X	30.59	0	0	1.0	0.13	1.13	0.41
173	65-64		X	30.59	0	0	1.0	0.13	1.13	0.41
174	63-59		X	23.0	0	0	1.0	0.16	1.16	0.55
175	64-68		X	23.0	0	0	1.0	0.16	1.16	0.55
176	59-74		X	38.1	0	0	1.0	0.08	1.08	0.276
177	68-74		X	38.1	0	0	1.0	0.08	1.08	0.276
178	61-74		X	42.0	0	0	1.0	0.03	1.03	0.11
179	67-74		X	42.0	0	0	1.0	0.03	1.03	0.11
180	62-74		X	47.25	0	0	1.0	0.10	1.10	0.215
181	65-74		X	47.25	0	0	1.0	0.10	1.10	0.215
182	63-74		X	29.4	0	0	1.0	0.23	1.23	0.236
183	64-74		X	29.4	0	0	1.0	0.23	1.23	0.236
184	60-61		X	15.54	0	0	1.0	0.32	1.32	0.181
185	66-67		X	15.54	0	0	1.0	0.32	1.32	0.181
186	3-69		X	21.6	0	0.90	1.0	0.50	2.40	1.256
187	11-70		X	21.6	0	0.90	1.0	0.50	2.40	1.256
188	20-69		X	5.4	0	0.90	1.0	0.50	2.40	1.273
189	23-70		X	5.4	0	0.90	1.0	0.50	2.40	1.273
190	69-43		X	75.6	0	0	1.0	0.35	1.35	0.0869
191	70-54		X	75.6	0	0	1.0	0.35	1.35	0.0869
192	69-74	X		7.2	0	0	1.0	0.44	1.44	0.776
193	70-74		X	7.2	0	0	1.0	0.44	1.44	0.776
194	69-71		X	33.88	0.012	0	1.0	0.44	1.45	0.98
195	70-72		X	33.88	0.012	0	1.0	0.44	1.45	0.98
196	71-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795
197	72-74		X	39.26	0.045	1.38	1.0	0.45	2.88	0.795

TABLE 6.2.1-22 (SHEET 5 OF 5)

Vent Path	Node Number (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (K)	Turning and Obstruction (K)	Expansion (K)	Contraction (K)	Total (K_T)	L/A (ft^{-1})
198	73-74		X	391.01	0	0	1.0	0.29	1.29	0.008

TABLE 6.2.1-23 (SHEET 1 OF 2)

PRESSURIZER COMPARTMENT MODEL: FLOW CHARACTERISTICS (SPRAY LINE)
LOSS COEFFICIENT (K)

Vent Parameters	Node 5 (From-To)	Description Flow of Flow Choked	Unchoked	Area (ft^2)	Turning Friction (ft^2-h)	and Obstruction	Expansion	Contraction	Total	L/A (ft^{-1})
1	1-28		X	172.0	0	0	1.0	0.82	1.82	0.027
2	2-1		X	27.4	0	0	1.0	0.23	1.23	0.190
3	2-3		X	18.1	0	0	1.0	0.32	1.32	0.157
4	3-1		X	23.6	0	0	1.0	0.29	1.29	0.201
5	3-4		X	25.0	0	0	1.0	0.29	1.29	0.114
6	4-1		X	26.8	0	0	1.0	0.26	1.26	0.153
7	4-5		X	26.2	0	0	1.0	0.32	1.32	0.119
8	5-1		X	33.4	0	0	1.0	0.23	1.23	0.143
9	5-2		X	27.5	0	0	1.0	0.38	1.38	0.105
10	6-2		X	16.4	0	0	1.0	0.29	1.29	0.319
11	6-7		X	20.5	0	0	1.0	0.38	1.38	0.180
12	7-3		X	12.7	0	0	1.0	0.35	1.35	0.352
13	7-8		X	30.0	0	0	1.0	0.32	1.32	0.126
14	8-4		X	19.2	0	0	1.0	0.32	1.32	0.253
15	8-9		X	33.0	0	0	1.0	0.32	1.32	0.129
16	9-5		X	19.9	0	0	1.0	0.35	1.35	0.231
17	9-6		X	35.0	0	0	1.0	0.29	1.29	0.117
18	10-6		X	16.1	0	0	1.0	0.35	1.35	0.303
19	10-11		X	14.0	0	0	1.0	0.38	1.38	0.213
20	10-28		X	12.7	0	0	1.06	0.26	2.32	0.517
21	11-7		X	13.5	0	0	1.0	0.35	1.35	0.335
22	11-12		X	25.2	0	0	1.0	0.32	1.32	0.150
23	12-8		X	19.2	0	0	1.0	0.32	1.32	0.241
24	12-13		X	27.8	0	0	1.0	0.32	1.32	0.154
25	13-9		X	22.9	0	0	1.0	0.32	1.32	0.219
26	13-10		X	29.4	0	0	1.0	0.32	1.32	0.140
27	14-10		X	27.9	0	0	1.0	0.21	1.21	0.225
28	14-15		X	9.5	0	0	1.0	0.35	1.35	0.350
29	15-11		X	24.9	0	0	1.0	0.26	1.26	0.248
30	15-16		X	15.3	0	0	1.0	0.21	1.21	0.449
31	16-12		X	32.5	0	0	1.0	0.21	1.21	0.178
32	16-17		X	16.5	0	0	1.0	0.32	1.32	0.255
33	17-13		X	37.3	0	0	1.0	0.21	1.21	0.163
34	17-14		X	17.8	0	0	1.0	0.29	1.29	0.410
35	18-14		X	14.5	0	0	1.0	0.35	1.35	0.161
36	18-19		X	51.0	0	0	1.0	0.26	1.26	0.100
37	19-15		X	10.5	0	0	1.0	0.35	1.35	0.178
38	19-20		X	23.0	0	0	1.0	0.32	1.32	0.330
39	20-16		X	24.2	0	0	1.0	0.32	1.32	0.186
40	20-21		X	29.3	0	0	1.0	0.32	1.32	0.143
41	21-17		X	26.0	0	0	1.0	0.32	1.32	0.170
42	21-18		X	31.5	0	0	1.0	0.29	1.29	0.152
43	22-18		X	33.0	0	0	1.0	0.32	1.32	0.154

TABLE 6.2.1-23 (SHEET 2 OF 2)

Vent Parameters	Node 5 (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (ft^2-h)	Turning and Obstruction	Expansion	Contraction	Total	L/A (ft^{-1})
44	22-23		X	51.8	0	0	1.0	0.26	1.26	0.099
45	22-28		X	0.20	0	0	1.0	0.49	1.49	12.5
46	23-19		X	29.2	0	0	1.0	0.32	1.32	0.167
47	23-24		X	27.5	0	0	1.0	0.32	1.32	0.162
48	24-20		X	19.2	0	0	1.0	0.32	1.32	0.238
49	24-25		X	29.7	0	0	1.0	0.32	1.32	0.141
50	25-21		X	24.9	0	0	1.0	0.32	1.32	0.218
51	25-22		X	32.0	0	0	1.0	0.29	1.29	0.149
52	25-28		X	1.23	0	0	1.0	0.49	1.49	2.08
53	26-22		X	26.1	0	0	1.0	0.35	1.35	0.099
54	26-23		X	27.0	0	0	1.0	0.32	1.32	0.104
55	26-24		X	13.4	0	0	1.0	0.35	1.35	0.140
56	26-25		X	16.4	0	0	1.0	0.35	1.35	0.130
57	26-27		X	161.5	0	0	1.0	0.26	1.26	0.043
58	26-28		X	5.62	0	0	1.0	0.49	1.49	0.532
59	27-28		X	304.3	0	0	1.0	0.23	1.23	0.024

TABLE 6.2.1-24 (SHEET 1 OF 2)

PRESSURIZER COMPARTMENT FLOW CHARACTERISTIC MODE (SURGE LINE)
LOSS COEFFICIENT (K)

Vent Parameters	Node 5 (From-To)	Description of Flow: Choked	Description of Flow: Unchoked	Flow Area (ft^2)	Friction (ft^2-h)	Turning and Obstruction	Expansion	Contraction	Total	L/A (ft^{-1})
1	1-28		X	172.0	0	0	1.0	0.82	1.82	0.027
2	2-1		X	27.4	0	0	1.0	0.23	1.23	0.190
3	2-3		X	18.1	0	0	1.0	0.32	1.32	0.157
4	3-1		X	23.6	0	0	1.0	0.29	1.29	0.201
5	3-4		X	25.0	0	0	1.0	0.29	1.29	0.114
6	4-1		X	26.8	0	0	1.0	0.26	1.26	0.153
7	4-5		X	26.2	0	0	1.0	0.32	1.32	0.119
8	5-1		X	33.4	0	0	1.0	0.23	1.23	0.143
9	5-2		X	27.5	0	0	1.0	0.38	1.38	0.105
10	6-2		X	16.4	0	0	1.0	0.29	1.29	0.319
11	6-7		X	20.5	0	0	1.0	0.38	1.38	0.180
12	7-3		X	12.7	0	0	1.0	0.35	1.35	0.352
13	7-8		X	30.0	0	0	1.0	0.32	1.32	0.126
14	8-4		X	19.2	0	0	1.0	0.32	1.32	0.253
15	8-9		X	33.0	0	0	1.0	0.32	1.32	0.129
16	9-5		X	19.9	0	0	1.0	0.35	1.35	0.231
17	9-6		X	35.0	0	0	1.0	0.29	1.29	0.117
18	10-6		X	16.1	0	0	1.0	0.35	1.35	0.303
19	10-11		X	14.0	0	0	1.0	0.38	1.38	0.213
20	10-24	X		12.7	0	1.06	1.0	0.26	2.32	0.517
21	11-7		X	13.5	0	0	1.0	0.35	1.35	0.335
22	11-12		X	25.2	0	0	1.0	0.32	1.32	0.150
23	12-8		X	19.2	0	0	1.0	0.32	1.32	0.241
24	12-13		X	27.8	0	0	1.0	0.32	1.32	0.154
25	13-9		X	22.9	0	0	1.0	0.32	1.32	0.219
26	13-10		X	29.4	0	0	1.0	0.32	1.32	0.140
27	14-10		X	27.9	0	0	1.0	0.21	1.21	0.225
28	14-15		X	9.5	0	0	1.0	0.35	1.35	0.350
29	15-11		X	24.9	0	0	1.0	0.26	1.26	0.248
30	15-16		X	15.3	0	0	1.0	0.21	1.21	0.449
31	16-12		X	32.5	0	0	1.0	0.21	2.21	0.178
32	16-17		X	16.5	0	0	1.0	0.32	1.32	0.255
33	17-13		X	37.3	0	0	1.0	0.21	1.21	0.163
34	17-14		X	17.8	0	0	1.0	0.29	1.29	0.410
35	18-14		X	14.5	0	0	1.0	0.35	1.35	0.161
36	18-19		X	51.0	0	0	1.0	0.26	1.26	0.100
37	19-15		X	10.5	0	0	1.0	0.35	1.35	0.178
38	19-20		X	23.0	0	0	1.0	0.32	1.32	0.330
39	20-16		X	24.2	0	0	1.0	0.32	1.32	0.186
40	20-21		X	29.3	0	0	1.0	0.32	1.32	0.143
41	21-17		X	26.0	0	0	1.0	0.32	1.32	0.170
42	21-18		X	31.5	0	0	1.0	0.29	1.29	0.152

TABLE 6.2.1-24 (SHEET 2 OF 2)

Vent Parameters	Node 5 (From-To)	Description of Flow Choked	Description of Flow Unchoked	Flow Area (ft^2)	Friction (ft^2-h)	Turning and Obstruction	Expansion	Contraction	Total	L/A (ft^{-1})
43	22-18		X	33.0	0	0	1.0	0.32	1.32	0.154
44	22-23		X	51.8	0	0	1.0	0.26	1.26	0.099
45	22-28	X		0.2	0	0	1.0	0.49	1.49	12.5
46	23-19		X	29.2	0	0	1.0	0.32	1.32	0.167
47	23-24		X	27.5	0	0	1.0	0.32	1.32	0.162
48	24-20		X	19.2	0	0	1.0	0.32	1.32	0.238
49	24-25		X	29.7	0	0	1.0	0.32	1.32	0.141
50	25-21		X	24.9	0	0	1.0	0.32	1.32	0.218
51	25-22		X	32.0	0	0	1.0	0.29	1.29	0.149
52	25-28	X		1.23	0	0	1.0	0.49	1.49	2.08
53	26-22		X	26.1	0	0	1.0	0.35	1.35	0.099
54	26-23		X	27.0	0	0	1.0	0.32	1.32	0.104
55	26-24		X	13.4	0	0	1.0	0.35	1.35	0.140
56	26-25		X	16.4	0	0	1.0	0.35	1.35	0.130
57	26-27	X		161.5	0	0	1.0	0.26	1.26	0.043
58	26-28	X		5.62	0	0	1.0	0.49	1.49	0.532
59	27-28		X	304.3	0	0	1.0	0.23	1.23	0.024

TABLE 6.2.1-25 (SHEET 1 OF 4)

PRESSURIZER COMPARTMENT MODEL: NODE CHARACTERISTICS

Node	Description	Height (ft)	Cross-Sectional Area (ft)	Net Volume (ft^3)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psi)
1	In pressurizer compartment between el 252 ft 4 in. and 265 ft.	12.67	215.7	2596	120	13.2	50.0	3.15
2	Northwest quadrant in pressurizer compartment between el 242 ft 0 in. and 252 ft 4 in.	10.33	49.3	400	120	13.2	50.0	5.71
3	Northeast quadrant in pressurizer compartment between el 242 ft 0 in. and 252 ft 4 in.	10.33	46.3	367	120	13.2	50.0	5.84
4	Southeast quadrant in pressurizer compartment between el 242 ft 0 in. and 252 ft 4 in.	10.33	58.1	485	120	13.2	50.0	5.70
5	Southwest quadrant in pressurizer compartment between el 242 ft 0 in. and 252 ft 4 in.	10.33	62.0	524	120	13.2	50.0	5.64
6	Northwest quadrant in pressurizer compartment between el 231 ft 0 in. and 242 ft 0 in.	11.0	49.3	350	120	13.2	50.0	7.41
7	Northeast quadrant in pressurizer compartment between el 231 ft 0 in. and 242 ft 0 in.	11.0	46.3	317	120	13.2	50.0	7.46

TABLE 6.2.1-25 (SHEET 2 OF 4)

Node	Description	Height (ft)	Cross-Sectional Area (ft)	Net Volume (ft^3)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psi)
8	Southeast quadrant in pressurizer compartment between el 231 ft 0 in. and 242 ft 0 in.	11.0	58.1	441	120	13.2	50.0	7.36
9	Southwest quadrant in pressurizer compartment between el 231 ft 0 in. and 242 ft 0 in.	11.0	62.0	480	120	13.2	50.0	7.44
10	Northwest quadrant of pressurizer compartment between el 221 ft 7 7/8 in. and 231 ft 0 in.	9.34	49.3	296	120	13.2	50.0	10.5
11	Northeast quadrant of pressurizer compartment between el 221 ft 7 7/8 in. and 231 ft 0 in.	9.34	46.3	269	120	13.2	50.0	10.7
12	Southeast quadrant of pressurizer compartment between el 221 ft 7 7/8 in. and 231 ft 0 in.	9.34	58.1	374	120	13.2	50.0	10.7
13	Southwest quadrant of pressurizer compartment between el 221 ft 7 7/8 in. and 231 ft 0 in.	9.34	62.0	409	120	13.2	50.0	10.6
14	Northwest quadrant of pressurizer compartment between el 216 ft 0 in. and 221 ft 7 7/8 in.	5.66	49.3	180	120	13.2	50.0	11.8
15	Northeast quadrant of pressurizer compartment between el 216 ft 0 in. and 221 ft 7 7/8 in.	5.66	46.3	163	120	13.2	50.0	11.7

TABLE 6.2.1-25 (SHEET 3 OF 4)

Node	Description	Height (ft)	Cross-Sectional Area (ft)	Net Volume (ft^3)	Initial Conditions Temperature (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	Calc Peak Pressure Differential (psi)
16	Southeast quadrant of pressurizer compartment between el 216 ft 0 in. and 221 ft 7 7/8 in.	5.66	58.1	227	120	13.2	50.0	11.9
17	Southwest quadrant of pressurizer compartment between el 216 ft 0 in. and 221 ft 7 7/8 in.	5.66	62.0	244	120	13.2	50.0	11.9
18	Northwest quadrant in pressurizer compartment between el 206 ft 0 in. and 216 ft 0 in.	10.0	81.3	622	120	13.2	50.0	15.8
19	Northeast quadrant in pressurizer compartment between el 206 ft 0 in. and 216 ft 0 in.	10.0	76.2	573	120	13.2	50.0	15.8
20	Southeast quadrant in pressurizer compartment between el 206 ft 0 in. and 216 ft 0 in.	10.0	58.1	401	120	13.2	50.0	15.4
21	Southwest quadrant in pressurizer compartment between el 206 ft 0 in. and 216 ft 0 in.	10.0	62.0	438	120	13.2	50.0	15.4
22	Northwest quadrant in pressurizer compartment between el 195 ft 9 3/4 in. and 206 ft 0 in.	10.19	81.3	633	120	13.2	50.0	17.3
23	Northeast quadrant in pressurizer compartment between el 195 ft 9 3/4 in. and 206 ft 0 in.	10.19	76.2	584	120	13.2	50.0	17.2

TABLE 6.2.1-25 (SHEET 4 OF 4)

Node	Description	Height (ft)	Cross-Sectional Area (ft)	Net Volume (ft^3)	Initial Conditions Temperature (°F)	Pressure (psia)	Humidity (%)	Calc Peak Pressure Differential (psi)
24	Southeast quadrant in pressurizer compartment between el 195 ft 9 3/4 in. and 206 ft 0 in.	10.19	58.1	409	120	13.2	50.0	17.1
25	Southwest quadrant in pressurizer compartment between el 195 ft 9 3/4 in. and 306 ft 0 in.	10.19	62.0	445	120	13.2	50.0	17.2
26	Inside pressurizer compartment between el 185 ft 0 in. and 195 ft 9 3/4 in.	10.81	277.7	2419	120	13.2	50.0	20.7
27	Inside pressurizer compartment between el 171 ft 9 in. and 185 ft 0 in.	13.25	277.7	3496	120	13.2	50.0	6.51
28	Containment building free volume.	--	--	2.75E+06	120	13.2	50.0	--

a. Note that this value includes allowances for volumes occupied by large equipment as well as a 5 percent reduction to account for smaller objects.

b. Results are for a 308-in. surge line break except nodes 1 through 5, which are for a spray line break at the pressurizer nozzle.

TABLE 6.2.1-26 (SHEET 1 OF 7)

REACTOR CAVITY TIME DEPENDENT
FLOW CONDITIONS

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.00000,	0.0,	0.,
0.00250,	1.3586147E+4,	7.6261951E+6,
0.00500,	1.7940995E+4,	1.0071084E+7,
0.00752,	2.0648819E+4,	1.1591079E+7,
0.01001,	2.2913458E+4,	1.2857206E+7,
0.01251,	2.4062034E+4,	1.3493659E+7,
0.01501,	2.3856512E+4,	1.3362610E+7,
0.01751,	2.6598904E+4,	1.4911117E+7,
0.02003,	2.7096058E+4,	1.5177837E+7,
0.02253,	2.6427985E+4,	1.4785699E+7,
0.02507,	2.6171937E+4,	1.4631801E+7,
0.02756,	2.6566208E+4,	1.4848680E+7,
0.03005,	2.6676953E+4,	1.4905002E+7,
0.03252,	2.7098266E+4,	1.5139974E+7,
0.03510,	2.7742642E+4,	1.5502190E+7,
0.03751,	2.8169787E+4,	1.5741567E+7,
0.04005,	2.8575125E+4,	1.5968826E+7,
0.04257,	2.8940040E+4,	1.6173308E+7,
0.04508,	2.9015063E+4,	1.6212095E+7,
0.04753,	2.8845152E+4,	1.6111615E+7,
0.05000,	2.8677447E+4,	1.6013188E+7,
0.05251,	2.8523145E+4,	1.5923076E+7,
0.05510,	2.8416210E+4,	1.5860480E+7,
0.05752,	2.8345423E+4,	1.5818843E+7,
0.06003,	2.8188576E+4,	1.5728214E+7,
0.06257,	2.7840963E+4,	1.5529144E+7,
0.06505,	2.7390292E+4,	1.5271875E+7,
0.06759,	2.7024295E+4,	1.5063533E+7,
0.07005,	2.6934059E+4,	1.5012643E+7,
0.07261,	2.7117701E+4,	1.5117598E+7,
0.07501,	2.7372228E+4,	1.5262759E+7,
0.07759,	2.7520339E+4,	1.5346859E+7,
0.08005,	2.7439861E+4,	1.5300528E+7,
0.08255,	2.7153041E+4,	1.5136868E+7,
0.08502,	2.6724728E+4,	1.4893012E+7,
0.08761,	2.6197717E+4,	1.4593675E+7,
0.09002,	2.5681851E+4,	1.4301168E+7,
0.09253,	2.5203708E+4,	1.4030333E+7,
0.09502,	2.4835535E+4,	1.3822267E+7,
0.09754,	2.4578990E+4,	1.3677537E+7,
0.10013,	2.4419910E+4,	1.3588081E+7,
0.10253,	2.4355729E+4,	1.3552374E+7,

TABLE 6.2.1-26 (SHEET 2 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.10500,	2.4365128E+4,	1.3558416E+7,
0.10751,	2.4455309E+4,	1.3610180E+7,
0.11010,	2.4625670E+4,	1.3707282E+7,
0.11255,	2.4845949E+4,	1.3832514E+7,
0.11503,	2.5086307E+4,	1.3969057E+7,
0.12010,	2.5466453E+4,	1.4184861E+7,
0.12260,	2.5551384E+4,	1.4232951E+7,
0.12513,	2.5567152E+4,	1.4241650E+7,
0.12761,	2.5523282E+4,	1.4216391E+7,
0.13014,	2.5427294E+4,	1.4161608E+7,
0.13265,	2.5276425E+4,	1.4075754E+7,
0.13507,	2.5071667E+4,	1.3959456E+7,
0.13754,	2.4813584E+4,	1.3813099E+7,
0.14005,	2.4539029E+4,	1.3657649E+7,
0.14258,	2.4313941E+4,	1.3530409E+7,
0.14509,	2.4165958E+4,	1.3446888E+7,
0.14765,	2.4092923E+4,	1.3405844E+7,
0.15005,	2.4058579E+4,	1.3386724E+7,
0.15256,	2.4033247E+4,	1.3372693E+7,
0.15502,	2.3993516E+4,	1.3350446E+7,
0.15762,	2.3909192E+4,	1.3302862E+7,
0.16014,	2.3778645E+4,	1.3229134E+7,
0.16258,	2.3624748E+4,	1.3142275E+7,
0.16515,	2.3480926E+4,	1.3061202E+7,
0.16764,	2.3386045E+4,	1.3007825E+7,
0.17012,	2.3341080E+4,	1.2982715E+7,
0.17261,	2.3337016E+4,	1.2980780E+7,
0.17506,	2.3360386E+4,	1.2994312E+7,
0.17760,	2.3404110E+4,	1.3019393E+7,
0.18014,	2.3465021E+4,	1.3054108E+7,
0.18255,	2.3537716E+4,	1.3095460E+7,
0.18503,	2.3632725E+4,	1.3149368E+7,
0.18760,	2.3744212E+4,	1.3212592E+7,
0.19013,	2.3871936E+4,	1.3284937E+7,
0.19261,	2.4004296E+4,	1.3359858E+7,
0.19513,	2.4135817E+4,	1.3434247E+7,
0.19765,	2.4247269E+4,	1.3497285E+7,
0.20012,	2.4332207E+4,	1.3545267E+7,
0.20255,	2.4395464E+4,	1.3580953E+7,
0.20503,	2.4443214E+4,	1.3607826E+7,
0.20755,	2.4475122E+4,	1.3625711E+7,
0.21005,	2.4480973E+4,	1.3628774E+7,
0.21252,	2.4448240E+4,	1.3609927E+7,
0.21513,	2.4370781E+4,	1.3565733E+7,
0.21757,	2.4275725E+4,	1.3511689E+7,

TABLE 6.2.1-26 (SHEET 3 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.22002,	2.4182117E+4,	1.3458565E+7,
0.22262,	2.4107319E+4,	1.3416180E+7,
0.22513,	2.4069692E+4,	1.3394923E+7,
0.22754,	2.4059667E+4,	1.3389325E+7,
0.23021,	2.4062445E+4,	1.3390990E+7,
0.23263,	2.4067014E+4,	1.3393646E+7,
0.23504,	2.4070423E+4,	1.3395637E+7,
0.23752,	2.4078331E+4,	1.3400188E+7,
0.24011,	2.4100562E+4,	1.3412891E+7,
0.24259,	2.4140469E+4,	1.3435579E+7,
0.24509,	2.4200933E+4,	1.3469938E+7,
0.24750,	2.4265283E+4,	1.3506417E+7,
0.25003,	2.4321768E+4,	1.3538422E+7,
0.25261,	2.4353615E+4,	1.3556377E+7,
0.25505,	2.4344223E+4,	1.3550864E+7,
0.25757,	2.4290066E+4,	1.3519957E+7,
0.26001,	2.4201115E+4,	1.3459386E+7,
0.26258,	2.4078015E+4,	1.3399541E+7,
0.26505,	2.3945320E+4,	1.3324386E+7,
0.26754,	2.3810797E+4,	1.3248271E+7,
0.27004,	2.3687905E+4,	1.3178772E+7,
0.27253,	2.3586351E+4,	1.3121419E+7,
0.27511,	2.3502079E+4,	1.3073899E+7,
0.27752,	2.3455387E+4,	1.3047656E+7,
0.28006,	2.3438152E+4,	1.3038153E+7,
0.28252,	2.3453951E+4,	1.3047366E+7,
0.28511,	2.3503649E+4,	1.3075784E+7,
0.28760,	2.3577850E+4,	1.3117988E+7,
0.29004,	2.3669172E+4,	1.3169845E+7,
0.29253,	2.3774953E+4,	1.3229813E+7,
0.29506,	2.3886183E+4,	1.3292821E+7,
0.29756,	2.3996260E+4,	1.3355146E+7,
0.30005,	2.4097381E+4,	1.3412378E+7,
0.30261,	2.4184345E+4,	1.3461533E+7,
0.30501,	2.4247804E+4,	1.3497363E+7,
0.30766,	2.4289171E+4,	1.3520597E+7,
0.31011,	2.4303098E+4,	1.3528275E+7,
0.31262,	2.4292119E+4,	1.3521774E+7,
0.31508,	2.4261540E+4,	1.3504208E+7,
0.31755,	2.4219339E+4,	1.3480093E+7,
0.32003,	2.4174469E+4,	1.3454522E+7,
0.32255,	2.4138798E+4,	1.3434171E+7,
0.32507,	2.4117832E+4,	1.3422282E+7,
0.32755,	2.4111291E+4,	1.3418580E+7,
0.33000,	2.4108026E+4,	1.3416724E+7,

TABLE 6.2.1-26 (SHEET 4 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.33255,	2.4097899E+4,	1.3410959E+7,
0.33503,	2.4079157E+4,	1.3400306E+7,
0.33753,	2.4059258E+4,	1.3389024E+7,
0.34000,	2.4048690E+4,	1.3383058E+7,
0.34262,	2.4055783E+4,	1.3387132E+7,
0.34511,	2.4075206E+4,	1.3398197E+7,
0.34758,	2.4094905E+4,	1.3409386E+7,
0.35002,	2.4103796E+4,	1.3414398E+7,
0.35256,	2.4096385E+4,	1.3410127E+7,
0.35504,	2.4075163E+4,	1.3398034E+7,
0.35762,	2.4045807E+4,	1.3381330E+7,
0.36008,	2.4017017E+4,	1.3364990E+7,
0.36254,	2.3991591E+4,	1.3350596E+7,
0.36508,	2.3972836E+4,	1.3340000E+7,
0.36762,	2.3966208E+4,	1.3336299E+7,
0.37000,	2.3971003E+4,	1.3339075E+7,
0.37253,	2.3979807E+4,	1.3344103E+7,
0.37510,	2.3980967E+4,	1.3344761E+7,
0.37760,	2.3966909E+4,	1.3336760E+7,
0.38008,	2.3941814E+4,	1.3322495E+7,
0.38251,	2.3916742E+4,	1.3308297E+7,
0.38502,	2.3903255E+4,	1.3300691E+7,
0.38579,	2.3908591E+4,	1.3303782E+7,
0.39003,	2.3926500E+4,	1.3313992E+7,
0.39251,	2.3950989E+4,	1.3327913E+7,
0.39509,	2.3969493E+4,	1.3338405E+7,
0.39751,	2.3976395E+4,	1.3342291E+7,
0.40006,	2.3979176E+4,	1.3343952E+7,
0.40261,	2.3986081E+4,	1.3347733E+7,
0.40508,	2.3999851E+4,	1.3355527E+7,
0.40770,	2.4020182E+4,	1.3367041E+7,
0.41005,	2.4041523E+4,	1.3379124E+7,
0.41280,	2.4061453E+4,	1.3390382E+7,
0.41514,	2.4079853E+4,	1.3400786E+7,
0.41763,	2.4100740E+4,	1.3412580E+7,
0.42010,	2.4144685E+4,	1.3426111E+7,
0.42256,	2.4148532E+4,	1.3439591E+7,
0.42511,	2.4166027E+4,	1.3449429E+7,
0.42760,	2.4173080E+4,	1.3453348E+7,
0.43009,	2.4172381E+4,	1.3452864E+7,
0.43256,	2.4169703E+4,	1.3451262E+7,
0.43504,	2.4168789E+4,	1.3450676E+7,
0.43778,	2.4169276E+4,	1.3450904E+7,
0.44017,	2.4165751E+4,	1.3448854E+7,
0.44263,	2.4154330E+4,	1.3442313E+7,

TABLE 6.2.1-26 (SHEET 5 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.44509,	2.4135122E+4,	1.3431355E+7,
0.44753,	2.4113758E+4,	1.3419208E+7,
0.45012,	2.4094979E+4,	1.3408512E+7,
0.45263,	2.4084151E+4,	1.3402379E+7,
0.45510,	2.4078610E+4,	1.3399245E+7,
0.45766,	2.4072112E+4,	1.3395561E+7,
0.46015,	2.4061009E+4,	1.3389283E+7,
0.46261,	2.4044888E+4,	1.3380159E+7,
0.46507,	2.4028418E+4,	1.3370837E+7,
0.46754,	2.4014861E+4,	1.3363172E+7,
0.47015,	2.4007389E+4,	1.3358971E+7,
0.47259,	2.4007662E+4,	1.3359161E+7,
0.47505,	2.4013697E+4,	1.3362618E+7,
0.47760,	2.4023721E+4,	1.3368333E+7,
0.48009,	2.4036380E+4,	1.3375569E+7,
0.48262,	2.4052248E+4,	1.3384600E+7,
0.48509,	2.4074545E+4,	1.3397290E+7,
0.48760,	2.4107246E+4,	1.3415863E+7,
0.49013,	2.4148401E+4,	1.3439192E+7,
0.49260,	2.4192806E+4,	1.3464343E+7,
0.49504,	2.4232365E+4,	1.3486726E+7,
0.49750,	2.4262844E+4,	1.3503931E+7,
0.50009,	2.4280483E+4,	1.3513829E+7,
0.51004,	2.4229105E+4,	1.3484245E+7,
0.52019,	2.4120069E+4,	1.3422231E+7,
0.53012,	2.3944743E+4,	1.3322982E+7,
0.54004,	2.3980354E+4,	1.3343661E+7,
0.55002,	2.4145342E+4,	1.3437383E+7,
0.56008,	2.4221418E+4,	1.3480274E+7,
0.57014,	2.4209329E+4,	1.3473144E+7,
0.58011,	2.4144979E+4,	1.3436505E+7,
0.59000,	2.4107243E+4,	1.3415227E+7,
0.60006,	2.4159685E+4,	1.3445105E+7,
0.61010,	2.4217996E+4,	1.3478169E+7,
0.62013,	2.4261479E+4,	1.3502731E+7,
0.63013,	2.4256432E+4,	1.3499721E+7,
0.64003,	2.4203746E+4,	1.3469774E+7,
0.65008,	2.4162745E+4,	1.3446618E+7,
0.66009,	2.4159154E+4,	1.3444706E+7,
0.67000,	2.4187283E+4,	1.3460781E+7,
0.68000,	2.4231721E+4,	1.3486025E+7,
0.69011,	2.4263450E+4,	1.3503981E+7,
0.70012,	2.4273260E+4,	1.3509453E+7,
0.71012,	2.4252390E+4,	1.3503212E+7,
0.72015,	2.4246713E+4,	1.3494349E+7,

TABLE 6.2.1-26 (SHEET 6 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.73014,	2.4255830E+4,	1.3499597E+7,
0.74019,	2.4284243E+4,	1.3515760E+7,
0.75002,	2.4295893E+4,	1.3522331E+7,
0.76012,	2.4285562E+4,	1.3516424E+7,
0.77013,	2.4270343E+4,	1.3507794E+7,
0.78001,	2.4266732E+4,	1.3505797E+7,
0.79003,	2.4279302E+4,	1.3513005E+7,
0.80011,	2.4295891E+4,	1.3522447E+7,
0.81004,	2.4310716E+4,	1.3530822E+7,
0.82001,	2.4318677E+4,	1.3535287E+7,
0.83012,	2.4318126E+4,	1.3534984E+7,
0.84019,	2.4323296E+4,	1.3537931E+7,
0.85007,	2.4333920E+4,	1.3543954E+7,
0.86001,	2.4343134E+4,	1.3549166E+7,
0.87001,	2.4345149E+4,	1.3550301E+7,
0.88004,	2.4338762E+4,	1.3546668E+7,
0.89006,	2.4335464E+4,	1.3544826E+7,
0.90005,	2.4335806E+4,	1.3545058E+7,
0.91005,	2.4340583E+4,	1.3547769E+7,
0.92020,	2.4348287E+4,	1.3552160E+7,
0.93008,	2.4355085E+4,	1.3556021E+7,
0.94000,	2.4360745E+4,	1.3559249E+7,
0.95007,	2.4367196E+4,	1.3562914E+7,
0.96009,	2.4374109E+4,	1.3566815E+7,
0.97015,	2.4378996E+4,	1.3569608E+7,
0.98005,	2.4379731E+4,	1.3570001E+7,
0.99007,	2.4378418E+4,	1.3569255E+7,
1.00010,	2.4375674E+4,	1.3567753E+7,
1.05019,	2.4380088E+4,	1.3570353E+7,
1.10005,	2.4382999E+4,	1.3572130E+7,
1.15002,	2.4383322E+4,	1.3572555E+7,
1.20009,	2.4377813E+4,	1.3569670E+7,
1.25007,	2.4374905E+4,	1.3568424E+7,
1.30007,	2.4378444E+4,	1.3570840E+7,
1.35018,	2.4377156E+4,	1.3570652E+7,
1.40010,	2.4370716E+4,	1.3567578E+7,
1.45004,	2.4382101E+4,	1.3574759E+7,
1.50024,	2.4361620E+4,	1.3563940E+7,
1.55003,	2.4349789E+4,	1.3558156E+7,
1.60012,	2.4323449E+4,	1.3544249E+7,
1.65001,	2.4297197E+4,	1.3530488E+7,
1.70009,	2.4265055E+4,	1.3513472E+7,
1.75003,	2.4230524E+4,	1.3495182E+7,
1.80007,	2.4195149E+4,	1.3476460E+7,
1.85006,	2.4154185E+4,	1.3454644E+7,

TABLE 6.2.1-26 (SHEET 7 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.90005,	2.4110518E+4,	1.3431345E+7,
1.95001,	2.4061980E+4,	1.3405353E+7,
2.00002,	2.4006998E+4,	1.3375804E+7,
10.0000,	2.4006998E+4,	1.3375804E+7,

TABLE 6.2.1-26A

STEAM GENERATOR COMPARTMENT TIME DEPENDENT FLOW CONDITIONS
(16 IN. FEEDWATER LINE CONDITION)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
0.0	1.3371E4	425.11
5.0	1.3371E4	425.11

TABLE 6.2.1-27 (SHEET 1 OF 36)

STEAM GENERATOR COMPARTMENT TIME
DEPENDENT FLOW CONDITIONS
(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	1.0904410E+04,	6.0480220E+06,
.00101,	2.3103719E+04,	1.2714696E+07,
.00202,	2.2381056E+04,	1.2345418E+07,
.00301,	2.5906342E+04,	1.4299363E+07,
.00402,	2.5436400E+04,	1.4035341E+07,
.00501,	2.5656111E+04,	1.4159121E+07,
.00602,	2.5357243E+04,	1.3989857E+07,
.00701,	2.4931699E+04,	1.3752451E+07,
.00800,	2.4571808E+04,	1.3551184E+07,
.00904,	2.4187011E+04,	1.3336488E+07,
.01002,	2.3880219E+04,	1.3165588E+07,
.01103,	2.3651012E+04,	1.3037901E+07,
.01202,	2.3496965E+04,	1.2952142E+07,
.01303,	2.3396491E+04,	1.2896245E+07,
.01401,	2.3336370E+04,	1.2862813E+07,
.01500,	2.3332478E+04,	1.2843994E+07,
.01602,	2.3286653E+04,	1.2835217E+07,
.01702,	2.3282699E+04,	1.2833050E+07,
.01801,	2.3290222E+04,	1.2837292E+07,
.01901,	2.3318349E+04,	1.2853066E+07,
.02001,	2.3382726E+04,	1.2889117E+07,
.02100,	2.3507313E+04,	1.2958857E+07,
.02200,	2.3716174E+04,	1.3075720E+07,
.02301,	2.4020877E+04,	1.3246185E+07,
.02402,	2.4407558E+04,	1.3462450E+07,
.02502,	2.4805974E+04,	1.3685261E+07,
.02603,	2.5179215E+04,	1.3893997E+07,
.02702,	2.5476048E+04,	1.4059991E+07,
.02801,	2.5688755E+04,	1.4178854E+07,
.02900,	2.5815616E+04,	1.4249685E+07,
.03001,	2.5872569E+04,	1.4281352E+07,
.03100,	2.5871120E+04,	1.4280349E+07,
.03202,	2.5831980E+04,	1.4258255E+07,
.03300,	2.5797867E+04,	1.4238755E+07,
.03403,	2.5729553E+04,	1.4200369E+07,
.03502,	2.5666245E+04,	1.4164838E+07,
.03601,	2.5615037E+04,	1.4136130E+07,
.03701,	2.5584268E+04,	1.4118888E+07,
.03802,	2.5576045E+04,	1.4114292E+07,
.03900,	2.5593825E+04,	1.4124256E+07,

TABLE 6.2.1-27 (SHEET 2 OF 36)

(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.04000,	2.5636989E+04,	1.4148441E+07,
.04101,	2.5709142E+04,	1.4189612E+07,
.04201,	2.5803415E+04,	1.4242315E+07,
.04300,	3.9073270E+04,	2.1622832E+07,
.04400,	3.6123909E+04,	1.9959244E+07,
.04500,	3.7514640E+04,	2.0734863E+07,
.04600,	3.7123722E+04,	2.0519158E+07,
.04701,	3.7595199E+04,	2.0781432E+07,
.04801,	3.7990734E+04,	2.1004792E+07,
.04901,	3.8569457E+04,	2.1328762E+07,
.05000,	3.9090779E+04,	2.1621351E+07,
.05101,	3.9522644E+04,	2.1863598E+07,
.05203,	3.9924193E+04,	2.2089344E+07,
.05301,	4.0328238E+04,	2.2316935E+07,
.05402,	4.0780930E+04,	2.2571622E+07,
.05502,	4.1058245E+04,	2.2726524E+07,
.05602,	4.1028097E+04,	2.2707867E+07,
.05701,	4.0827711E+04,	2.2594210E+07,
.05801,	4.0614117E+04,	2.2473446E+07,
.05902,	4.0306748E+04,	2.2298999E+07,
.06005,	3.9705141E+04,	2.1958902E+07,
.06101,	3.9019798E+04,	2.1573731E+07,
.06203,	3.8524030E+04,	2.1296608E+07,
.06306,	3.8264535E+04,	2.1151257E+07,
.06402,	3.8043324E+04,	2.1027233E+07,
.06506,	3.7528857E+04,	2.0963996E+07,
.06609,	3.8047881E+04,	2.1031529E+07,
.06710,	3.8248523E+04,	2.1144413E+07,
.06810,	3.8431685E+04,	2.1247369E+07,
.06904,	3.8577465E+04,	2.1329199E+07,
.07010,	3.8681749E+04,	2.1387566E+07,
.07103,	3.8716321E+04,	2.1406737E+07,
.07203,	3.8694016E+04,	2.1393940E+07,
.07308,	3.8610314E+04,	2.1346583E+07,
.07404,	3.8477661E+04,	2.1271803E+07,
.07508,	3.8286475E+04,	2.1164259E+07,
.07601,	3.8094753E+04,	2.1056554E+07,
.07705,	3.7869887E+04,	2.0930259E+07,
.07808,	3.7625546E+04,	2.0793040E+07,
.07903,	3.7377536E+04,	2.0653886E+07,
.08003,	3.7108876E+04,	2.0503174E+07,
.08109,	3.6815615E+04,	2.0338789E+07,

TABLE 6.2.1-27 (SHEET 3 OF 36)

(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.08203,	3.6570906E+04,	2.0201650E+07,
.08307,	3.6322997E+04,	2.0062868E+07,
.08403,	3.6146874E+04,	1.9964492E+07,
.08504,	3.6011558E+04,	1.9888990E+07,
.08601,	3.5921555E+04,	1.9838855E+07,
.08704,	3.5854749E+04,	1.9801639E+07,
.08815,	3.5799896E+04,	1.9771041E+07,
.08902,	3.5757645E+04,	1.9747418E+07,
.09014,	3.5703170E+04,	1.9716932E+07,
.09105,	3.5652585E+04,	1.9688585E+07,
.09211,	3.5585871E+04,	1.9651180E+07,
.09305,	3.5518181E+04,	1.9613209E+07,
.09403,	3.5440371E+04,	1.9569576E+07,
.09505,	3.5365526E+04,	1.9527655E+07,
.09610,	3.5293384E+04,	1.9487297E+07,
.09704,	3.5246775E+04,	1.9461321E+07,
.09814,	3.5215885E+04,	1.9444199E+07,
.09909,	3.5206027E+04,	1.9438841E+07,
.10014,	3.5208081E+04,	1.9440142E+07,
.10502,	3.5235098E+04,	1.9455575E+07,
.11015,	3.5216755E+04,	1.9445382E+07,
.11517,	3.5019113E+04,	1.9334344E+07,
.12014,	3.4879507E+04,	1.9256199E+07,
.12506,	3.4768055E+04,	1.9193846E+07,
.13009,	3.4649578E+04,	1.9127645E+07,
.13501,	3.4580271E+04,	1.9389133E+07,
.14013,	3.4693845E+04,	1.9153331E+07,
.14508,	3.4881703E+04,	1.9258839E+07,
.15008,	3.4994292E+04,	1.9322028E+07,
.15511,	3.4990962E+04,	1.9320158E+07,
.16007,	3.4968263E+04,	1.9307534E+07,
.16503,	3.4964778E+04,	1.9305667E+07,
.17010,	3.4993582E+04,	1.9321916E+07,
.17505,	3.5092374E+04,	1.9377433E+07,
.18005,	3.5274019E+04,	1.9479423E+07,
.18511,	3.5445015E+04,	1.9575282E+07,
.19012,	3.5513330E+04,	1.9613385E+07,
.19511,	3.5476643E+04,	1.9592519E+07,
.20000,	3.5389047E+04,	1.9543169E+07,
.21012,	3.5255198E+04,	1.9468035E+07,
.22007,	3.5256683E+04,	1.9468911E+07,

TABLE 6.2.1-27 (SHEET 4 OF 36)

(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.23004,	3.5231527E+04,	1.9454787E+07,
.24009,	3.5109052E+04,	1.9386133E+07,
.25004,	3.5068150E+04,	1.9363427E+07,
.26004,	3.5121658E+04,	1.9393549E+07,
.27000,	3.5159026E+04,	1.9414539E+07,
.28002,	3.5246849E+04,	1.9463869E+07,
.29009,	3.5606682E+04,	1.9665651E+07,
.30013,	3.6110831E+04,	1.9947367E+07,
.31010,	3.5883817E+04,	1.9819724E+07,
.32015,	3.5731357E+04,	1.9733879E+07,
.33008,	3.5406465E+04,	1.9552039E+07,
.34008,	3.5403076E+04,	1.9550425E+07,
.35014,	3.5428013E+04,	1.9564578E+07,
.36003,	3.5367341E+04,	1.9530647E+07,
.37004,	3.5316627E+04,	1.9502441E+07,
.38005,	3.5402063E+04,	1.9550400E+07,
.39001,	3.5506771E+04,	1.9609110E+07,
.40001,	3.5628132E+04,	1.9677021E+07,
.41008,	3.5647173E+04,	1.9687547E+07,
.42013,	3.5587259E+04,	1.9653881E+07,
.43005,	3.5518685E+04,	1.9615449E+07,
.44011,	3.5458889E+04,	1.9582016E+07,
.45002,	3.5423570E+04,	1.9562332E+07,
.46007,	3.5389368E+04,	1.9543272E+07,
.47014,	3.5360551E+04,	1.9527243E+07,
.48003,	3.5344742E+04,	1.9518465E+07,
.49004,	3.5331713E+04,	1.9511278E+07,
.50014,	3.5464944E+04,	1.9586157E+07,
.51004,	3.5455419E+04,	1.9580555E+07,
.52016,	3.5171002E+04,	1.9421101E+07,
.53013,	3.4956311E+04,	1.9301100E+07,
.54020,	3.4967429E+04,	1.9307806E+07,
.55003,	3.5173577E+04,	1.9423487E+07,
.56006,	3.5249947E+04,	1.9466290E+07,
.57007,	3.5274526E+04,	1.9480020E+07,
.58008,	3.5192533E+04,	1.9434100E+07,
.59009,	3.5173286E+04,	1.9423484E+07,
.60004,	3.5207910E+04,	1.9443066E+07,
.61004,	3.5263128E+04,	1.9474124E+07,
.62003,	3.5289369E+04,	1.9488901E+07,
.63008,	3.5280693E+04,	1.9484094E+07,

TABLE 6.2.1-27 (SHEET 5 OF 36)

(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.64011,	3.5205381E+04,	1.9441981E+07,
.65018,	3.5185255E+04,	1.9430942E+07,
.66010,	3.5247094E+04,	1.9465823E+07,
.67011,	3.5294402E+04,	1.9492475E+07,
.68006,	3.5270057E+04,	1.9478927E+07,
.69010,	3.5224067E+04,	1.9453332E+07,
.70027,	3.5205546E+04,	1.9443190E+07,
.71002,	3.5205036E+04,	1.9443150E+07,
.72011,	3.5204349E+04,	1.9443007E+07,
.73008,	3.5216969E+04,	1.9450347E+07,
.74015,	3.5248956E+04,	1.9468535E+07,
.75007,	3.5278490E+04,	1.9485325E+07,
.76011,	3.5294514E+04,	1.9494545E+07,
.77010,	3.5324933E+04,	1.9511865E+07,
.78003,	3.5323636E+04,	1.9511366E+07,
.80009,	3.5349418E+04,	1.9526430E+07,
.81007,	3.5364223E+04,	1.9535023E+07,
.82009,	3.5337390E+04,	1.9520270E+07,
.83007,	3.5315749E+04,	1.9508490E+07,
.84002,	3.5304547E+04,	1.9502577E+07,
.85010,	3.5315673E+04,	1.9509200E+07,
.86007,	3.5398275E+04,	1.9555915E+07,
.87000,	3.5422401E+04,	1.9569725E+07,
.88005,	3.5471641E+04,	1.9597672E+07,
.89000,	3.5514959E+04,	1.9622268E+07,
.90011,	3.5544815E+04,	1.9639309E+07,
.91008,	3.5574072E+04,	1.9656051E+07,
.92015,	3.5609336E+04,	1.9676169E+07,
.93004,	3.5644462E+04,	1.9696218E+07,
.94007,	3.5681054E+04,	1.9717092E+07,
.95006,	3.5708010E+04,	1.9732566E+07,
.96004,	3.5732460E+04,	1.9746650E+07,
.97008,	3.5767399E+04,	1.9766649E+07,
.98003,	3.5808071E+04,	1.9789841E+07,
.99019,	3.5852729E+04,	1.9815310E+07,
1.00012,	3.5895623E+04,	1.9839774E+07,
1.05008,	3.6112647E+04,	1.9963381E+07,
1.10007,	3.6338443E+04,	2.0092128E+07,
1.15002,	3.6608642E+04,	2.0245910E+07,
1.20007,	3.6888439E+04,	2.0405162E+07,
1.25001,	3.7127241E+04,	2.0541579E+07,

TABLE 6.2.1-27 (SHEET 6 OF 36)

(236-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.30009,	3.7127043E+04,	2.0656265E+07,
1.35002,	3.7484280E+04,	2.0747071E+07,
1.40001,	3.7596272E+04,	2.0812441E+07,
1.45005,	3.7694652E+04,	2.0870059E+07,
1.50008,	3.7762278E+04,	2.0910243E+07,
1.55002,	3.7934417E+04,	2.1009055E+07,
1.60009,	3.7995308E+04,	2.1045152E+07,
1.65015,	3.8086014E+04,	2.1098323E+07,
1.70001,	3.8172222E+04,	2.1148978E+07,
1.73004,	3.8252964E+04,	2.1196588E+07,
1.80009,	3.8321222E+04,	2.1237084E+07,
1.85004,	3.8383561E+04,	2.1274274E+07,
1.90007,	3.8435015E+04,	2.1305277E+07,
1.95004,	3.8437624E+04,	2.1308836E+07,
2.00011,	3.8421768E+04,	2.1302004E+07,
2.05008,	3.8369901E+04,	2.1274851E+07,
2.10006,	3.8189733E+04,	2.1175694E+07,
2.15010,	3.8119992E+04,	2.1139027E+07,
2.20004,	3.7912908E+04,	2.1024926E+07,
2.25002,	3.7738727E+04,	2.0929322E+07,
2.30000,	3.7497730E+04,	2.0796234E+07,
2.35011,	3.7281923E+04,	2.0677130E+07,
2.40001,	3.7071816E+04,	2.0561347E+07,
2.45002,	3.7082174E+04,	2.0570251E+07,
2.50011,	3.6948336E+04,	2.0497962E+07,
2.55008,	3.6903695E+04,	2.0476484E+07,
2.60014,	3.6908560E+04,	2.0483188E+07,
2.65001,	3.6817126E+04,	2.0435988E+07,
2.70001,	3.6667266E+04,	2.0355691E+07,
2.75018,	3.6307184E+04,	2.0157719E+07,
2.80003,	3.5923918E+04,	1.9946870E+07,
2.85001,	3.5715709E+04,	1.9835248E+07,
2.90018,	3.5330716E+04,	1.9623353E+07,
2.95009,	3.4981705E+04,	1.9432201E+07,
3.00005,	3.4718072E+04,	1.9289429E+07,

TABLE 6.2.1-27 (SHEET 7 OF 36)

(306-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	1.0904418E+04,	6.9546146E+06,
.00101,	2.6783296E+04,	1.7817432E+07,
.00201,	2.6059905E+04,	1.6564051E+07,
.00301,	2.6630723E+04,	1.6923731E+07,
.00401,	2.6299962E+04,	1.6714564E+07,
.00500,	2.6431184E+04,	1.6797431E+07,
.00602,	2.6349622E+04,	1.6749019E+07,
.00700,	2.6305987E+04,	1.6768774E+07,
.00802,	2.6365680E+04,	1.6755933E+07,
.00903,	2.6380583E+04,	1.6765424E+07,
.01003,	2.6309531E+04,	1.6771220E+07,
.01101,	2.6428212E+04,	1.6748873E+07,
.01203,	2.6455432E+04,	1.6813421E+07,
.01302,	2.6431622E+04,	1.6830145E+07,
.01403,	2.6494736E+04,	1.6830535E+07,
.01502,	2.6507040E+04,	1.6846432E+07,
.01605,	2.6532537E+04,	1.6862820E+07,
.01703,	2.6614296E+04,	1.6919424E+07,
.01804,	2.6873875E+04,	1.7032259E+07,
.01901,	2.7467003E+04,	1.7463110E+07,
.02004,	2.8525431E+04,	1.8142452E+07,
.02103,	2.9703593E+04,	1.8893017E+07,
.02204,	3.0491958E+04,	1.9401975E+07,
.02304,	3.0380016E+04,	1.9461919E+07,
.02403,	3.0268393E+04,	1.9254922E+07,
.02502,	2.9002195E+04,	1.9006799E+07,
.02607,	2.9394309E+04,	1.8822112E+07,
.02707,	2.9400495E+04,	1.8702756E+07,
.02807,	2.9276976E+04,	1.8618401E+07,
.02906,	2.9210971E+04,	1.9581326E+07,
.03005,	2.9233801E+04,	1.8590523E+07,
.03107,	2.9300991E+04,	1.8689301E+07,
.03207,	2.9433417E+04,	1.8719351E+07,
.03305,	2.9508009E+04,	1.8818687E+07,
.03400,	2.9757005E+04,	1.8927234E+07,
.03507,	2.9977287E+04,	1.9068795E+07,
.03603,	3.0194624E+04,	1.9208524E+07,
.03700,	3.0444230E+04,	1.9369022E+07,
.03803,	3.0748400E+04,	1.9364717E+07,
.03906,	3.1197470E+04,	1.9795836E+07,
.04000,	3.1497312E+04,	2.0047030E+07,

TABLE 6.2.1-27 (SHEET 8 OF 36)

(306-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.04100,	3.1024106E+04,	2.0256915E+07,
.04205,	3.2108183E+04,	2.0435911E+07,
.04305,	3.2203201E+04,	2.0499450E+07,
.04406,	3.2156609E+04,	2.0468593E+07,
.04507,	3.2011837E+04,	2.8374677E+07,
.04602,	3.1845034E+04,	2.0266726E+07,
.04700,	3.1651549E+04,	2.0141733E+07,
.04808,	3.1498506E+04,	2.0042877E+07,
.04909,	3.1359642E+04,	1.9953230E+07,
.05001,	3.1253206E+04,	1.9004520E+07,
.05109,	3.1149257E+04,	1.9817440E+07,
.05206,	3.1072705E+04,	1.9760037E+07,
.05300,	3.1004020E+04,	1.9723740E+07,
.05401,	3.0954130E+04,	1.9691557E+07,
.05501,	3.0912015E+04,	1.9664420E+07,
.05611,	3.0660596E+04,	1.9644190E+07,
.05711,	3.0067966E+04,	1.9626102E+07,
.05811,	3.0871682E+04,	1.9630555E+07,
.05914,	3.0091361E+04,	1.9651626E+07,
.06007,	3.0920740E+04,	1.9670236E+07,
.06130,	3.0951744E+04,	1.9690274E+07,
.06200,	3.0975342E+04,	1.9705433E+07,
.06308,	3.0977493E+04,	1.9706693E+07,
.06401,	3.0958439E+04,	1.9694300E+07,
.06504,	3.0916964E+04,	1.9667593E+07,
.06605,	3.0876672E+04,	1.9641563E+07,
.06700,	3.0837718E+04,	1.9616490E+07,
.06803,	3.0817279E+04,	1.9603306E+07,
.06903,	3.0811027E+04,	1.9599425E+07,
.07008,	3.0821414E+04,	1.9606100E+07,
.07106,	3.0043773E+04,	1.9620637E+07,
.07201,	3.0874702E+04,	1.9640660E+07,
.07206,	3.0915265E+04,	1.9666707E+07,
.07409,	3.0958322E+04,	1.9694572E+07,
.07507,	3.1001246E+04,	1.9722270E+07,
.07607,	3.1044705E+04,	1.9750317E+07,
.07715,	3.1000957E+04,	1.9770337E+07,
.07814,	3.1123120E+04,	1.9882146E+07,
.07900,	3.1157700E+04,	1.9823092E+07,
.08009,	3.1183245E+04,	1.9839430E+07,
.08107,	3.1200280E+04,	1.9850340E+07,

TABLE 6.2.1-27 (SHEET 9 OF 36)

(306-in.^2 BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.08206,	3.1203954E+04,	1.9852600E+07,
.08308,	3.1193945E+04,	1.9846042E+07,
.08404,	3.1170566E+04,	1.9838001E+07,
.08501,	3.1134922E+04,	1.9807821E+07,
.08601,	3.1087116E+04,	1.9776942E+07,
.08709,	3.1029503E+04,	1.9739000E+07,
.08804,	3.0974094E+04,	1.9704012E+07,
.08901,	3.0917695E+04,	1.9667644E+07,
.09089,	3.0853531E+04,	1.9626200E+07,
.09110,	3.8796140E+04,	1.9569278E+07,
.09201,	3.0743409E+04,	1.9555347E+07,
.09301,	3.0685573E+04,	1.9518020E+07,
.09409,	3.0624348E+04,	1.9478562E+07,
.09501,	3.0573333E+04,	1.9445674E+07,
.09600,	3.0516623E+04,	1.9489168E+07,
.09713,	3.0452875E+04,	1.9368057E+07,
.09814,	3.0400141E+04,	1.9334105E+07,
.09912,	3.0350250E+04,	1.9301972E+07,
.10012,	3.0300946E+04,	1.9270230E+07,
.10516,	3.0004490E+04,	1.9131006E+07,
.11013,	2.9947775E+04,	1.9043465E+07,
.11505,	2.9896911E+04,	1.9019056E+07,
.12002,	2.9827896E+04,	1.8967496E+07,
.12505,	2.9681357E+04,	1.8873476E+07,
.13003,	2.9477222E+04,	1.8742507E+07,
.13504,	2.9200877E+04,	1.8616757E+07,
.14007,	2.9125707E+04,	1.8517710E+07,
.14509,	2.8981100E+04,	1.8425514E+07,
.15006,	2.8811361E+04,	1.8217077E+07,
.15504,	2.8594534E+04,	1.8178459E+07,
.16002,	2.8344040E+04,	1.8010209E+07,
.16506,	2.8067722E+04,	1.7041673E+07,
.17004,	2.7764467E+04,	1.7647852E+07,
.17501,	2.7423570E+04,	1.7429901E+07,
.18001,	2.7069411E+04,	1.7283536E+07,
.18504,	2.6715030E+04,	1.6977152E+07,
.19002,	2.6493250E+04,	1.6778063E+07,
.19500,	2.6124672E+04,	1.6682797E+07,
.20018,	2.5841963E+04,	1.6419880E+07,
.21000,	2.5138580E+04,	1.5970348E+07,
.22013,	2.4375517E+04,	1.5482003E+07,

TABLE 6.2.1-27 (SHEET 10 OF 36)

(306-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.23016,	2.3823925E+04,	1.5130751E+07,
.24014,	2.3369722E+04,	1.4049362E+07,
.25007,	2.2929182E+04,	1.4559762E+07,
.26013,	2.2691429E+04,	1.4408545E+07,
.27011,	2.2497616E+04,	1.4205460E+07,
.28001,	2.2356313E+04,	1.4196081E+07,
.29006,	2.2180437E+04,	1.4004375E+07,
.30001,	2.2073196E+04,	1.4016719E+07,
.31009,	2.1944465E+04,	1.3935290E+07,
.32009,	2.1820878E+04,	1.3857087E+07,
.33014,	2.1706070E+04,	1.3734560E+07,
.34090,	2.1612880E+04,	1.3725889E+07,
.35014,	2.1490487E+04,	1.3648525E+07,
.36013,	2.1391424E+04,	1.3586161E+07,
.37011,	2.1260616E+04,	1.3583657E+07,
.38014,	2.1181301E+04,	1.3454095E+07,
.39882,	2.1112816E+04,	1.3411754E+07,
.40019,	2.1013401E+04,	1.3349771E+07,
.41010,	2.0924470E+04,	1.3294601E+07,
.42003,	2.0869091E+04,	1.3261205E+07,
.43008,	2.0889126E+04,	1.3224993E+07,
.44008,	2.0723134E+04,	1.3172146E+07,
.45083,	2.0620172E+04,	1.3113971E+07,
.46002,	2.0515965E+04,	1.3044064E+07,
.47010,	2.0475523E+04,	1.3920890E+07,
.48024,	2.0433069E+04,	1.2995489E+07,
.49003,	2.0389661E+04,	1.2971487E+07,
.50003,	2.0340102E+04,	1.2940015E+07,
.51002,	2.0306981E+04,	1.2924773E+07,
.52010,	2.0267614E+04,	1.2902050E+07,
.53003,	2.0235873E+04,	1.2885929E+07,
.54001,	2.0205732E+04,	1.2869978E+07,
.55014,	2.0176703E+04,	1.2854603E+07,
.56001,	2.0151311E+04,	1.2841861E+07,
.57001,	2.0126861E+04,	1.2020991E+07,
.58013,	2.0095633E+04,	1.2012343E+07,
.59003,	2.0062537E+04,	1.2794139E+07,
.60003,	2.0030307E+04,	1.2776336E+07,
.61809,	2.0002602E+04,	1.2761137E+07,
.62004,	1.9977344E+04,	1.2747300E+07,
.63007,	1.9955914E+04,	1.2735706E+07,

TABLE 6.2.1-27 (SHEET 11 OF 36)

(306-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.64002,	1.9937743E+04,	1.2723896E+07,
.65013,	1.9919953E+04,	1.2716322E+07,
.66007,	1.9892660E+04,	1.2649427E+07,
.67006,	1.9861153E+04,	1.2679012E+07,
.68007,	1.9846173E+04,	1.2678675E+07,
.69009,	1.9830751E+04,	1.2660895E+07,
.70009,	1.9816368E+04,	1.2651831E+07,
.71012,	1.9805505E+04,	1.2644140E+07,
.72004,	1.9795400E+04,	1.2637627E+07,
.73002,	1.9786282E+04,	1.2635125E+07,
.74012,	1.9778482E+04,	1.2623013E+07,
.75003,	1.9772434E+04,	1.2610364E+07,
.76010,	1.9767230E+04,	1.2613063E+07,
.77003,	1.9762966E+04,	1.2608096E+07,
.78023,	1.9759570E+04,	1.2603391E+07,
.79001,	1.9757060E+04,	1.2598978E+07,
.80001,	1.9753590E+04,	1.2594963E+07,
.81010,	1.9755388E+04,	1.2591514E+07,
.82005,	1.9755886E+04,	1.2588207E+07,
.83011,	1.9756346E+04,	1.2584745E+07,
.84102,	1.9756783E+04,	1.2583036E+07,
.85005,	1.9758443E+04,	1.2577762E+07,
.86809,	1.9763788E+04,	1.2576771E+07,
.87017,	1.9772159E+04,	1.2577829E+07,
.88007,	1.9780558E+04,	1.2578600E+07,
.89004,	1.9780392E+04,	1.2578731E+07,
.90001,	1.9797611E+04,	1.2580141E+07,
.91004,	1.9837670E+04,	1.2581783E+07,
.92005,	1.9817255E+04,	1.2583110E+07,
.93003,	1.9828011E+04,	1.2565229E+07,
.94097,	1.9841004E+04,	1.2563790E+07,
.95039,	1.9854040E+04,	1.2592694E+07,
.96003,	1.9869443E+04,	1.2597593E+07,
.97010,	1.9884994E+04,	1.2602041E+07,
.98004,	1.9899603E+04,	1.2607500E+07,
.99014,	1.9913539E+04,	1.2611643E+07,
1.00014,	1.9927001E+04,	1.2616035E+07,
1.05808,	2.0003035E+04,	1.2648624E+07,
1.10006,	2.0001506E+04,	1.2666790E+07,
1.15083,	2.0159717E+04,	1.2683963E+07,

TABLE 6.2.1-27 (SHEET 12 OF 36)

(306-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.23001,	2.0229722E+04,	1.2703628E+07,
1.25006,	2.0271963E+04,	1.2714947E+07,
1.30002,	2.0329809E+04,	1.2733996E+07,
1.35006,	2.0445870E+04,	1.2730469E+07,
1.40006,	2.0595390E+04,	1.2069199E+07,
1.45001,	2.0797697E+04,	1.2973763E+07,
1.50001,	2.2597603E+04,	1.4130098E+07,
1.55006,	2.1727914E+04,	1.3531673E+07,
1.60003,	2.2033142E+04,	1.3603610E+07,
1.65008,	2.2569311E+04,	1.3902740E+07,
1.70001,	2.7069624E+04,	1.6559945E+07,
1.75004,	2.9992324E+04,	1.8156965E+07,
1.83007,	3.1111373E+04,	1.8901704E+07,
1.85009,	3.8141272E+04,	1.8262443E+07,
1.90010,	3.8279306E+04,	1.8347891E+07,
1.95302,	3.8503591E+04,	1.8471000E+07,
2.00011,	3.8724291E+04,	1.8593102E+07,
2.05007,	3.1050715E+04,	1.8793001E+07,
2.10001,	3.1246237E+04,	1.8937338E+07,
2.15006,	3.1427902E+04,	1.9017557E+07,
2.20014,	3.1528101E+04,	1.9070014E+07,
2.25007,	3.1406946E+04,	1.8997013E+07,
2.30001,	3.1327452E+04,	1.9951002E+07,
2.35006,	3.1352832E+04,	1.8979205E+07,
2.40002,	3.1247389E+04,	1.8921466E+07,
2.45090,	3.1046631E+04,	1.8000368E+07,
2.50003,	3.0660505E+04,	1.8564644E+07,
2.55022,	3.0143320E+04,	1.8247507E+07,
2.60003,	2.9549638E+04,	1.7070145E+07,
2.65004,	2.8990574E+04,	1.7535716E+07,
2.70011,	2.8403195E+04,	1.7166309E+07,
2.75007,	2.8107991E+04,	1.6999214E+07,
2.80009,	2.7602489E+04,	1.6695309E+07,
2.85005,	2.7334948E+04,	1.6590445E+07,
2.90004,	2.7149613E+04,	1.6473829E+07,
2.95002,	2.6945061E+04,	1.6376299E+07,
3.00010,	2.6740150E+04,	1.6273687E+07,

TABLE 6.2.1-27 (SHEET 13 OF 36)

(336-in.^{2} CLOSURE WELD BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	1.0904410E+04,	6.0400220E+06,
.00102,	3.1315332E+04,	1.7242679E+07,
.00200,	2.7940955E+04,	1.5372641E+07,
.00300,	2.9058594E+04,	1.6005069E+07,
.00401,	2.9324630E+04,	1.6141002E+07,
.00501,	2.0732274E+04,	1.5817640E+07,
.00601,	2.9150326E+04,	1.6049104E+07,
.00703,	2.9082265E+04,	1.6010471E+07,
.00801,	2.9371131E+04,	1.6173599E+07,
.00901,	3.0350271E+04,	1.6720200E+07,
.01002,	3.2337123E+04,	1.7030637E+07,
.01102,	3.4808617E+04,	1.9207799E+07,
.01200,	3.5977677E+04,	1.9858237E+07,
.01306,	3.5252689E+04,	1.9437759E+07,
.01402,	3.4082369E+04,	1.8706010E+07,
.01503,	3.3472383E+04,	1.8449149E+07,
.01606,	3.3290402E+04,	1.8340323E+07,
.01703,	3.3200217E+04,	1.8298546E+07,
.01801,	3.3225546E+04,	1.8313396E+07,
.01900,	3.3391040E+04,	1.8406700E+07,
.02004,	3.3702763E+04,	1.8580301E+07,
.02101,	3.4041621E+04,	1.8769631E+07,
.02203,	3.4247080E+04,	1.8882856E+07,
.02386,	3.4120032E+04,	1.8810176E+07,
.02406,	3.3755622E+04,	1.8606049E+07,
.02503,	3.3380219E+04,	1.8397173E+07,
.02602,	3.3167359E+04,	1.8279444E+07,
.02705,	3.3119492E+04,	1.8253315E+07,
.02805,	3.3145091E+04,	1.8268110E+07,
.02901,	3.3190096E+04,	1.8292848E+07,
.03007,	3.3253045E+04,	1.8327994E+07,
.03101,	3.3313912E+04,	1.8361057E+07,
.03200,	3.3353834E+04,	1.8303881E+07,
.03302,	3.3343114E+04,	1.8377561E+07,
.03408,	3.3272446E+04,	1.8337869E+07,
.03507,	3.3183214E+04,	1.8200103E+07,
.03602,	3.3123953E+04,	1.8255267E+07,
.03707,	3.3116108E+04,	1.8251151E+07,
.03805,	3.3150502E+04,	1.8270439E+07,

TABLE 6.2.1-27 (SHEET 14 OF 36)

(336-in.^2 CLOSURE WELD BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.03901,	3.3197302E+04,	1.8296529E+07,
.04005,	3.3242461E+04,	1.8321653E+07,
.04103,	3.3275084E+04,	1.8340223E+07,
.04206,	3.3299100E+04,	1.8353138E+07,
.04304,	3.3307218E+04,	1.8357510E+07,
.04401,	3.3505517E+04,	1.8470440E+07,
.04502,	3.4449731E+04,	1.8992921E+07,
.04601,	3.5262496E+04,	1.9440836E+07,
.04702,	3.5879893E+04,	1.9779923E+07,
.04806,	3.6236755E+04,	1.9972897E+07,
.04900,	3.6269583E+04,	1.9985931E+07,
.05004,	3.6060590E+04,	1.9867704E+07,
.05102,	3.5924401E+04,	1.9795340E+07,
.05201,	3.6126195E+04,	1.9912190E+07,
.05306,	3.6575781E+04,	2.0161982E+07,
.05406,	3.6849965E+04,	2.0310129E+07,
.05506,	3.6897666E+04,	2.0333794E+07,
.05602,	3.6882889E+04,	2.0331701E+07,
.05705,	3.6999489E+04,	2.0392556E+07,
.05805,	3.7219065E+04,	2.0515462E+07,
.05900,	3.7476960E+04,	2.0658312E+07,
.06006,	3.7707201E+04,	2.0794494E+07,
.06103,	3.7832928E+04,	2.0852950E+07,
.06203,	3.7925042E+04,	2.0904186E+07,
.06364,	3.8068676E+04,	2.0985074E+07,
.06406,	3.8264791E+04,	2.1094051E+07,
.06503,	3.8403280E+04,	2.1169090E+07,
.06604,	3.8394878E+04,	2.1161822E+07,
.06701,	3.8260995E+04,	2.1085868E+07,
.06800,	3.8071294E+04,	2.0980414E+07,
.06906,	3.7910358E+04,	2.0892004E+07,
.07002,	3.7831074E+04,	2.0848952E+07,
.07103,	3.7803983E+04,	2.0834629E+07,
.07206,	3.7811991E+04,	2.0839661E+07,
.07308,	3.7856688E+04,	2.0865381E+07,
.07402,	3.7957543E+04,	2.0922723E+07,
.07507,	3.8165873E+04,	2.1040090E+07,
.07608,	3.8451658E+04,	2.1200016E+07,
.07702,	3.8764481E+04,	2.1374287E+07,
.07808,	3.9132485E+04,	2.1579074E+07,

TABLE 6.2.1-27 (SHEET 15 OF 36)

(336-in.^{2} CLOSURE WELD BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.07906,	3.9492063E+04,	2.1779484E+07,
.08001,	3.9879446E+04,	2.1995774E+07,
.08103,	4.0353849E+04,	2.2260672E+07,
.08211,	4.0908980E+04,	2.2570087E+07,
.08308,	4.1412318E+04,	2.2850120E+07,
.26011,	4.0316515E+04,	2.2239430E+07,
.27014,	4.0058097E+04,	2.2097363E+07,
.28016,	3.9818692E+04,	2.1967568E+07,
.29008,	3.9879564E+04,	2.2005338E+07,
.30006,	4.0093220E+04,	2.2127686E+07,
.31003,	4.0207611E+04,	2.2194375E+07,
.32005,	4.0201445E+04,	2.2194242E+07,
.33030,	4.0216076E+04,	2.2206028E+07,
.34000,	4.0219714E+04,	2.2211666E+07,
.35014,	4.0145695E+04,	2.2174008E+07,
.36008,	3.9994375E+04,	2.2093549E+07,
.37012,	3.9893347E+04,	2.2041597E+07,
.38021,	3.9907754E+04,	2.2054342E+07,
.39002,	3.9952948E+04,	2.2083866E+07,
.40002,	3.9935496E+04,	2.2078383E+07,
.41003,	3.9923464E+04,	2.2076291E+07,
.42007,	3.9978870E+04,	2.2111915E+07,
.43000,	4.0047123E+04,	2.2154536E+07,
.44009,	4.0054216E+04,	2.2163031E+07,
.45011,	4.0115169E+04,	2.2202630E+07,
.46005,	4.0211520E+04,	2.2260985E+07,
.47009,	4.0184188E+04,	2.2250703E+07,
.48008,	4.0117447E+04,	2.2218402E+07,
.49011,	4.0013181E+04,	2.2165700E+07,
.50003,	4.0008453E+04,	2.2168665E+07,
.51004,	4.0050068E+04,	2.2197559E+07,
.52010,	4.0110299E+04,	2.2236690E+07,
.53007,	4.0097367E+04,	2.2234944E+07,
.54009,	4.0068467E+04,	2.2224382E+07,
.55008,	4.0018860E+04,	2.2202150E+07,
.56003,	3.9933661E+04,	2.2160059E+07,
.57006,	3.9836338E+04,	2.2111598E+07,
.58006,	3.9806384E+04,	2.2100931E+07,
.59003,	3.9786229E+04,	2.2095666E+07,
.60003,	3.9758272E+04,	2.2086180E+07,

TABLE 6.2.1-27 (SHEET 16 OF 36)

(336-in.^{2} CLOSURE WELD BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.61011,	3.9733403E+04,	2.2078365E+07,
.62019,	3.9687180E+04,	2.2058478E+07,
.63005,	3.9633316E+04,	2.2034454E+07,
.64013,	3.9595001E+04,	2.2019153E+07,
.65008,	3.9544578E+04,	2.1996974E+07,
.66007,	3.9478791E+04,	2.1966303E+07,
.67001,	3.9415717E+04,	2.1937149E+07,
.68024,	3.9345246E+04,	2.1903947E+07,
.69004,	3.9272848E+04,	2.1869592E+07,
.70003,	3.9209415E+04,	2.1860313E+07,
.71001,	3.9153229E+04,	2.1815160E+07,
.72008,	3.9101670E+04,	2.1792552E+07,
.73012,	3.9048780E+04,	2.1769073E+07,
.74000,	3.8988046E+04,	2.1741096E+07,
.75012,	3.8920540E+04,	2.1709414E+07,
.76006,	3.8849061E+04,	2.1675378E+07,
.77002,	3.8768716E+04,	2.1636330E+07,
.78001,	3.8683025E+04,	2.1594241E+07,
.79014,	3.8595052E+04,	2.1550884E+07,
.80009,	3.8507172E+04,	2.1507415E+07,
.81009,	3.8420234E+04,	2.1464492E+07,
.82009,	3.8379224E+04,	2.1447517E+07,
.83007,	3.8367035E+04,	2.1446716E+07,
.84007,	3.8347948E+04,	2.1441847E+07,
.85001,	3.8302612E+04,	2.1422006E+07,
.86003,	3.8221354E+04,	2.1381845E+07,
.87009,	3.8138299E+04,	2.1340767E+07,
.88011,	3.8079445E+04,	2.1313324E+07,
.89010,	3.8030601E+04,	2.1291442E+07,
.90001,	3.7979574E+04,	2.1268167E+07,
.91006,	3.7917809E+04,	2.1238829E+07,
.92001,	3.7850827E+04,	2.1206416E+07,
.93010,	3.7789108E+04,	2.1176988E+07,
.94008,	3.7734890E+04,	2.1151639E+07,
.95003,	3.7684520E+04,	2.1128412E+07,
.96004,	3.7635988E+04,	2.1106175E+07,
.97008,	3.7584676E+04,	2.1082300E+07,
.98008,	3.7531491E+04,	2.1057291E+07,
.99001,	3.7479837E+04,	2.1033066E+07,

TABLE 6.2.1-27 (SHEET 17 OF 36)

(336-in.^{2} CLOSURE WELD BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.00009,	3.7430349E+04,	2.1010098E+07,
1.05001,	3.7616906E+04,	2.0996744E+07,
1.10010,	3.7766230E+04,	2.0901135E+07,
1.15010,	3.8045396E+04,	2.1157066E+07,
1.20002,	3.8355584E+04,	2.1341372E+07,
1.25003,	3.8657749E+04,	2.1520398E+07,
1.30009,	3.8968707E+04,	2.1704303E+07,
1.35004,	3.9164171E+04,	2.1023411E+07,
1.40001,	3.9307973E+04,	2.1912914E+07,
1.45005,	3.9326743E+04,	2.1932111E+07,
1.50008,	3.9363693E+04,	2.1961100E+07,
1.55004,	3.9457534E+04,	2.2021794E+07,
1.60008,	3.9468121E+04,	2.2031990E+07,
1.65008,	3.9500104E+04,	2.2061126E+07,
1.70019,	3.9585397E+04,	2.2072423E+07,
1.75005,	3.9431749E+04,	2.2039800E+07,
1.80008,	3.9341064E+04,	2.1997493E+07,
1.85010,	3.9261865E+04,	2.1961693E+07,
1.90003,	3.9296701E+04,	2.1991160E+07,
1.95006,	3.9235303E+04,	2.1994017E+07,
2.00088,	3.9153504E+04,	2.1931664E+07,
2.05010,	3.8974005E+04,	2.1042375E+07,
2.10010,	3.8742276E+04,	2.1723844E+07,
2.15027,	3.8304543E+04,	2.1490275E+07,
2.20002,	3.7877215E+04,	2.1261769E+07,
2.25003,	3.7703755E+04,	2.1177745E+07,
2.30003,	3.7377102E+04,	2.1004598E+07,
2.35003,	3.7050456E+04,	2.0831472E+07,
2.40012,	3.6771210E+04,	2.0686139E+07,
2.45007,	3.6533469E+04,	2.0565091E+07,
2.50003,	3.6394909E+04,	2.0503197E+07,
2.55003,	3.5981609E+04,	2.0286802E+07,
2.60006,	3.5565120E+04,	2.0060275E+07,
2.65008,	3.5170092E+04,	1.9360301E+07,
2.70007,	3.4761140E+04,	1.9641536E+07,
2.75001,	3.4506170E+04,	1.9511926E+07,
2.80004,	3.4050260E+04,	1.9272390E+07,
2.85006,	3.3531241E+04,	1.8903699E+07,
2.90035,	3.2924740E+04,	1.8661662E+07,
2.95008,	3.2356095E+04,	1.8353048E+07,
3.00006,	3.1975899E+04,	1.8153529E+07,

TABLE 6.2.1-27 (SHEET 18 OF 36)

(336-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	1.0904410E+04,	6.0480220E+06,
.00102,	3.1319824E+04,	1.7245649E+07,
.00201,	2.7946215E+04,	1.5375841E+07,
.00302,	2.9121894E+04,	1.6040584E+07,
.00400,	2.9399851E+04,	1.6183312E+07,
.00503,	2.8826792E+04,	1.5870642E+07,
.00603,	2.9234675E+04,	1.6096228E+07,
.00703,	3.0946665E+04,	1.7044444E+07,
.00804,	3.0758539E+04,	1.6933443E+07,
.00902,	2.9383448E+04,	1.6176691E+07,
.01003,	2.9135851E+04,	1.6040374E+07,
.01102,	2.9233906E+04,	1.6095455E+07,
.01200,	2.9381316E+04,	1.6175652E+07,
.01302,	2.9281523E+04,	1.6121855E+07,
.01400,	2.9575649E+04,	1.6206387E+07,
.01500,	3.0328683E+04,	1.6706998E+07,
.01600,	3.1745495E+04,	1.7497768E+07,
.01701,	3.3610875E+04,	1.8537603E+07,
.01802,	3.5095205E+04,	1.9361516E+07,
.01905,	3.5434097E+04,	1.9544472E+07,
.02005,	3.4878420E+04,	1.9230944E+07,
.02103,	3.4073856E+04,	1.8782276E+07,
.02205,	3.3541485E+04,	1.8487222E+07,
.02301,	3.3323030E+04,	1.8366196E+07,
.02402,	3.3191577E+04,	1.8278652E+07,
.02504,	3.3082371E+04,	1.8232257E+07,
.02601,	3.3002613E+04,	1.8187853E+07,
.02706,	3.2932539E+04,	1.8148856E+07,
.02806,	3.2888128E+04,	1.8124209E+07,
.02902,	3.2876176E+04,	1.8117719E+07,
.03001,	3.2912300E+04,	1.8138147E+07,
.03103,	3.3023916E+04,	1.8200783E+07,
.03205,	3.3225583E+04,	1.8313675E+07,
.03302,	3.3480035E+04,	1.8455733E+07,
.03407,	3.3738721E+04,	1.8599652E+07,
.03503,	3.3867245E+04,	1.8670587E+07,
.03601,	3.3840843E+04,	1.8654958E+07,
.03708,	3.3661255E+04,	1.8554098E+07,
.03803,	3.3438771E+04,	1.8429893E+07,
.03902,	3.3247515E+04,	1.8323650E+07,

TABLE 6.2.1-27 (SHEET 19 OF 36)

(336-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.04001,	3.3149606E+04,	1.8269451E+07,
.04102,	3.4858290E+04,	1.9234289E+07,
.04204,	3.6596807E+04,	2.0194634E+07,
.04300,	3.7811404E+04,	2.0861141E+07,
.04403,	3.8537083E+04,	2.1255744E+07,
.04505,	3.8789200E+04,	2.1389356E+07,
.04602,	3.8713281E+04,	2.1342235E+07,
.04705,	3.8376772E+04,	2.1151524E+07,
.04804,	3.7912998E+04,	2.0892494E+07,
.04904,	3.7450575E+04,	2.0636400E+07,
.05005,	3.7085790E+04,	2.0435341E+07,
.05106,	3.6835150E+04,	2.0297233E+07,
.05203,	3.6663090E+04,	2.0201827E+07,
.05305,	3.6510859E+04,	2.0117200E+07,
.05408,	3.6418075E+04,	2.0066715E+07,
.05503,	3.6484002E+04,	2.0105915E+07,
.05604,	3.6795543E+04,	2.0282122E+07,
.05702,	3.7285754E+04,	2.0555863E+07,
.05801,	3.7915465E+04,	2.0848865E+07,
.05903,	3.8213922E+04,	2.1066913E+07,
.06005,	3.8376449E+04,	2.1153361E+07,
.06105,	3.8307867E+04,	2.1111641E+07,
.06205,	3.8019257E+04,	2.0947908E+07,
.06300,	3.7570319E+04,	2.0696225E+07,
.06402,	3.6975766E+04,	2.0365317E+07,
.06508,	3.6374761E+04,	2.0032690E+07,
.06608,	3.5905768E+04,	1.9774418E+07,
.06702,	3.5615671E+04,	1.9615745E+07,
.06808,	3.5480589E+04,	1.9543488E+07,
.06905,	3.5539272E+04,	1.9578348E+07,
.07000,	3.5753754E+04,	1.9699418E+07,
.07103,	3.6140128E+04,	1.9915977E+07,
.07202,	3.6666822E+04,	2.0210557E+07,
.07308,	3.7386655E+04,	2.0612760E+07,
.07404,	3.8159480E+04,	2.1043976E+07,
.07506,	3.9061995E+04,	2.1546568E+07,
.07607,	3.9923806E+04,	2.2025178E+07,
.07705,	4.0654471E+04,	2.2430038E+07,
.07800,	4.1238952E+04,	2.2753206E+07,
.07905,	4.1713347E+04,	2.3014682E+07,

TABLE 6.2.1-27 (SHEET 20 OF 36)

(336-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.08005,	4.2016658E+04,	2.3181200E+07,
.08105,	4.2198966E+04,	2.3280508E+07,
.08210,	4.2279767E+04,	2.3323496E+07,
.08310,	4.2281321E+04,	2.3322819E+07,
.08404,	4.2239756E+04,	2.3298599E+07,
.08501,	4.2177672E+04,	2.3263276E+07,
.08611,	4.2104329E+04,	2.3221916E+07,
.08704,	4.2052709E+04,	2.3193003E+07,
.08807,	4.2020811E+04,	2.3175363E+07,
.08909,	4.2025770E+04,	2.3178449E+07,
.09009,	4.2065480E+04,	2.3200843E+07,
.09110,	4.2124772E+04,	2.3233807E+07,
.09211,	4.2172432E+04,	2.3259759E+07,
.09310,	4.2175533E+04,	2.3260295E+07,
.09411,	4.2105578E+04,	2.3219849E+07,
.09511,	4.1961821E+04,	2.3138422E+07,
.09603,	4.1778332E+04,	2.3035338E+07,
.09706,	4.1547717E+04,	2.2906509E+07,
.09803,	4.1339159E+04,	2.2790489E+07,
.09907,	4.1143922E+04,	2.2682407E+07,
.10005,	4.1030160E+04,	2.2619962E+07,
.10504,	4.1277515E+04,	2.2760967E+07,
.11006,	4.1684568E+04,	2.2984817E+07,
.11501,	4.0861846E+04,	2.2522864E+07,
.12002,	4.0110925E+04,	2.2107457E+07,
.12508,	4.0271513E+04,	2.2200062E+07,
.13011,	4.0672513E+04,	2.2423415E+07,
.13509,	4.0462782E+04,	2.2304174E+07,
.14009,	4.0129215E+04,	2.2119836E+07,
.14508,	4.0128710E+04,	2.2120539E+07,
.15011,	4.0036140E+04,	2.2068352E+07,
.15500,	3.9694514E+04,	2.1878076E+07,
.16006,	3.9472130E+04,	2.1755828E+07,
.16505,	3.9703270E+04,	2.1886290E+07,
.17002,	3.9968584E+04,	2.2033203E+07,
.17509,	3.9622431E+04,	2.1838486E+07,
.18011,	3.9104870E+04,	2.1550930E+07,
.18516,	3.9013925E+04,	2.1502278E+07,
.19010,	3.9202980E+04,	2.1608665E+07,
.19500,	3.9485145E+04,	2.1766258E+07,

TABLE 6.2.1-27 (SHEET 21 OF 36)

(336-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.20002,	3.9794566E+04,	2.1938705E+07,
.21006,	4.0042339E+04,	2.2075738E+07,
.22010,	4.0034978E+04,	2.2071672E+07,
.23003,	4.0069032E+04,	2.2090788E+07,
.24000,	3.9921154E+04,	2.2008323E+07,
.25003,	3.9753266E+04,	2.1915694E+07,
.26002,	3.9661005E+04,	2.1865181E+07,
.27003,	3.9609295E+04,	2.1837809E+07,
.28007,	3.9784416E+04,	2.1936634E+07,
.29010,	3.9855442E+04,	2.1977128E+07,
.30007,	3.9896288E+04,	2.2001107E+07,
.31006,	3.9954873E+04,	2.2035214E+07,
.32012,	3.9985827E+04,	2.2053889E+07,
.33004,	3.9951025E+04,	2.2036286E+07,
.34005,	3.9962792E+04,	2.2044922E+07,
.35017,	3.9932590E+04,	2.2030225E+07,
.36001,	3.9882825E+04,	2.2004858E+07,
.37009,	3.9819271E+04,	2.1971978E+07,
.38001,	3.9741062E+04,	2.1931179E+07,
.39010,	3.9729809E+04,	2.1928009E+07,
.40014,	3.9790731E+04,	2.1965091E+07,
.41008,	3.9848676E+04,	2.2000469E+07,
.42008,	3.9901479E+04,	2.2033191E+07,
.43006,	3.9943806E+04,	2.2059947E+07,
.44011,	3.9928071E+04,	2.2054514E+07,
.45004,	3.9890825E+04,	2.2037290E+07,
.46002,	3.9879296E+04,	2.2034773E+07,
.47001,	4.0099196E+04,	2.2161891E+07,
.48010,	4.0052539E+04,	2.2139260E+07,
.49004,	4.0009473E+04,	2.2119273E+07,
.50003,	3.9967072E+04,	2.2100057E+07,
.51013,	4.0033119E+04,	2.2141239E+07,
.52000,	3.9986071E+04,	2.2119181E+07,
.53008,	3.9995917E+04,	2.2129457E+07,
.54009,	4.0062363E+04,	2.2171146E+07,
.55003,	4.0939414E+04,	2.2162845E+07,
.56000,	3.9982026E+04,	2.2135457E+07,
.57007,	3.9916363E+04,	2.2183738E+07,
.58002,	3.9855869E+04,	2.2074846E+07,
.59007,	3.9797104E+04,	2.2047191E+07,

TABLE 6.2.1-27 (SHEET 22 OF 36)

(336-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.60011,	3.9801350E+04,	2.2054765E+07,
.61007,	3.9813400E+04,	2.2066753E+07,
.62001,	3.9806850E+04,	2.2068233E+07,
.63011,	3.9785117E+04,	2.2061235E+07,
.64010,	3.9745377E+04,	2.2044375E+07,
.65010,	3.9683897E+04,	2.2015240E+07,
.66002,	3.9627585E+04,	2.1989164E+07,
.67003,	3.9582612E+04,	2.1969491E+07,
.68012,	3.9535443E+04,	2.1948660E+07,
.69009,	3.9490536E+04,	2.1929156E+07,
.70008,	3.9446800E+04,	2.1909657E+07,
.71006,	3.9396414E+04,	2.1887528E+07,
.72007,	3.9348911E+04,	2.1866617E+07,
.73009,	3.9308340E+04,	2.1849517E+07,
.74006,	3.9266575E+04,	2.1831721E+07,
.75003,	3.9221422E+04,	2.1812018E+07,
.76008,	3.9167478E+04,	2.1787392E+07,
.77027,	3.9101610E+04,	2.1756044E+07,
.78014,	3.9029067E+04,	2.1720941E+07,
.79004,	3.8954031E+04,	2.1684527E+07,
.80006,	3.8880815E+04,	2.1649063E+07,
.81003,	3.8810827E+04,	2.1615375E+07,
.82008,	3.8743384E+04,	2.1583084E+07,
.83008,	3.8703083E+04,	2.1566116E+07,
.84013,	3.8698525E+04,	2.1569190E+07,
.85007,	3.8686489E+04,	2.1567963E+07,
.86008,	3.8646193E+04,	2.1550798E+07,
.87003,	3.8596282E+04,	2.1522539E+07,
.88036,	3.8509193E+04,	2.1484667E+07,
.89009,	3.8441697E+04,	2.1452192E+07,
.90005,	3.8394362E+04,	2.1431046E+07,
.91010,	3.8355403E+04,	2.1414617E+07,
.92009,	3.8312887E+04,	2.1396098E+07,
.93010,	3.8262021E+04,	2.1372841E+07,
.94004,	3.8201289E+04,	2.1343908E+07,
.95009,	3.8138982E+04,	2.1314158E+07,
.96010,	3.8085254E+04,	2.1288159E+07,
.97009,	3.8037122E+04,	2.1267278E+07,
.98005,	3.7991015E+04,	2.1246553E+07,
.99008,	3.7942798E+04,	2.1224593E+07,

TABLE 6.2.1-27 (SHEET 23 OF 36)

(336-in.^2 BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.00007,	3.7889110E+04,	2.1199669E+07,
1.05003,	3.7810879E+04,	2.1179703E+07,
1.10019,	3.7798934E+04,	2.1196647E+07,
1.15003,	3.8081679E+04,	2.1163255E+07,
1.20005,	3.8384633E+04,	2.1332920E+07,
1.25003,	3.8697696E+04,	2.1518565E+07,
1.30003,	3.8991710E+04,	2.1693715E+07,
1.35000,	3.9204347E+04,	2.1823086E+07,
1.40002,	3.9341273E+04,	2.1909702E+07,
1.45010,	3.9359873E+04,	2.1929688E+07,
1.50011,	3.9377289E+04,	2.1948093E+07,
1.55009,	3.9503057E+04,	2.2027510E+07,
1.60007,	3.9505578E+04,	2.2037144E+07,
1.65004,	3.9515997E+04,	2.2051235E+07,
1.70006,	3.9488035E+04,	2.2044140E+07,
1.75012,	3.9404968E+04,	2.2006330E+07,
1.80012,	3.9322910E+04,	2.1968669E+07,
1.85000,	3.9242225E+04,	2.1931719E+07,
1.90002,	3.9276580E+04,	2.1988992E+07,
1.95007,	3.9252680E+04,	2.1956123E+07,
2.00004,	3.9113078E+04,	2.1887882E+07,
2.05024,	3.8941004E+04,	2.1801685E+07,
2.10008,	3.8699189E+04,	2.1676477E+07,
2.15006,	3.8304963E+04,	2.1465963E+07,
2.20010,	3.7870398E+04,	2.1232248E+07,
2.25001,	3.7731360E+04,	2.1165724E+07,
2.30003,	3.7417022E+04,	2.0998164E+07,
2.35015,	3.7105293E+04,	2.0831994E+07,
2.40009,	3.6817342E+04,	2.0680490E+07,
2.45005,	3.6586350E+04,	2.0562003E+07,
2.50012,	3.6424999E+04,	2.0485987E+07,
2.55001,	3.6005489E+04,	2.0265259E+07,
2.60002,	3.5606846E+04,	2.0055890E+07,
2.65010,	3.5240474E+04,	1.9862977E+07,
2.70000,	3.4855162E+04,	1.9657410E+07,
2.75000,	3.4606376E+04,	1.9530965E+07,
2.80001,	3.4147934E+04,	1.9285313E+07,
2.85006,	3.3611093E+04,	1.8996154E+07,
2.90003,	3.2983430E+04,	1.8657303E+07,
2.95005,	3.2431145E+04,	1.8159907E+07,
3.00012,	3.2121901E+04,	1.8199162E+07,

TABLE 6.2.1-27 (SHEET 24 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	.	.
.00101,	1.0912112E+04,	6.0279802E+06,
.00202,	1.5876922E+04,	8.7501640E+06,
.00301,	1.9107228E+04,	1.0528134E+07,
.00401,	2.2394932E+04,	1.2352347E+07,
.00501,	2.6556709E+04,	1.4651538E+07,
.00601,	2.9317134E+04,	1.6155227E+07,
.00701,	3.0770794E+04,	1.6953810E+07,
.00804,	3.2095162E+04,	1.7683326E+07,
.00902,	3.3063797E+04,	1.8214192E+07,
.01003,	3.3630048E+04,	1.8519761E+07,
.01101,	3.3690832E+04,	1.8547935E+07,
.01202,	3.3572936E+04,	1.8481671E+07,
.01301,	3.3414537E+04,	1.8392567E+07,
.01402,	3.3201386E+04,	1.8274230E+07,
.01500,	3.3292380E+04,	1.8216214E+07,
.01600,	3.3335582E+04,	1.8356246E+07,
.01704,	3.4004110E+04,	1.8727901E+07,
.01804,	3.4615891E+04,	1.9061515E+07,
.01904,	3.4722628E+04,	1.9109429E+07,
.02006,	3.3412646E+04,	1.8379770E+07,
.02106,	3.2920608E+04,	1.8115117E+07,
.02201,	3.2594040E+04,	1.7928432E+07,
.02301,	3.2123172E+04,	1.7672891E+07,
.02402,	3.1960616E+04,	1.7582280E+07,
.02500,	3.1616771E+04,	1.7391027E+07,
.02604,	3.1391551E+04,	1.7269437E+07,
.02703,	3.1250702E+04,	1.7190452E+07,
.02801,	3.1060788E+04,	1.7086540E+07,
.02903,	3.1177394E+04,	1.7155856E+07,
.03005,	3.1416908E+04,	1.7284245E+07,
.03102,	3.1233809E+04,	1.7178519E+07,
.03202,	3.0877682E+04,	1.6981724E+07,
.03304,	3.0644644E+04,	1.6856355E+07,
.03401,	3.0690733E+04,	1.6883908E+07,
.03504,	3.0785240E+04,	1.6935342E+07,
.03605,	3.0786569E+04,	1.6934025E+07,
.03703,	3.0631717E+04,	1.6847147E+07,
.03800,	3.0466928E+04,	1.6756951E+07,
.03901,	3.0358633E+04,	1.6698084E+07,

TABLE 6.2.1-27 (SHEET 25 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.04000,	3.0438252E+04,	1.6745592E+07,
.04102,	3.0846188E+04,	1.6973687E+07,
.04201,	3.1253101E+04,	1.7193824E+07,
.04301,	3.1106960E+04,	1.7106070E+07,
.04400,	3.0710999E+04,	1.6888803E+07,
.04502,	3.0570958E+04,	1.6815748E+07,
.04600,	3.0688257E+04,	1.6881790E+07,
.04700,	3.0701911E+04,	1.6885873E+07,
.04801,	3.0574827E+04,	1.6818120E+07,
.04902,	3.0832885E+04,	1.6964301E+07,
.05000,	3.1189824E+04,	1.7162053E+07,
.05101,	3.1912778E+04,	1.7569750E+07,
.05200,	3.3570213E+04,	1.8500191E+07,
.05301,	3.6325123E+04,	2.0032669E+07,
.05402,	3.8686112E+04,	2.1316538E+07,
.05502,	3.9264176E+04,	2.1613079E+07,
.05602,	3.9164552E+04,	2.1560199E+07,
.05703,	3.9631396E+04,	2.1830694E+07,
.05804,	4.0664118E+04,	2.2406067E+07,
.05902,	4.1526297E+04,	2.2876487E+07,
.06004,	4.2026121E+04,	2.3149620E+07,
.06103,	4.2469172E+04,	2.3395779E+07,
.06203,	4.2856047E+04,	2.3606824E+07,
.06301,	4.2978961E+04,	2.3670597E+07,
.06401,	4.3026975E+04,	2.3700508E+07,
.06502,	4.3549054E+04,	2.4001981E+07,
.06600,	4.4721753E+04,	2.4661505E+07,
.06702,	4.6168579E+04,	2.5462312E+07,
.06807,	4.7325363E+04,	2.0096691E+07,
.06906,	4.8123725E+04,	2.6536014E+07,
.07004,	4.8775747E+04,	2.6895365E+07,
.07103,	4.9224183E+04,	2.7137644E+07,
.07203,	4.9270169E+04,	2.7153251E+07,
.07305,	4.8890924E+04,	2.6936090E+07,
.07400,	4.8385139E+04,	2.6654382E+07,
.07500,	4.7968558E+04,	2.6391960E+07,
.07606,	4.7563277E+04,	2.6202327E+07,
.07704,	4.7314610E+04,	2.6064344E+07,
.07806,	4.7036794E+04,	2.5908764E+07,

TABLE 6.2.1-27 (SHEET 26 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.07904,	4.6723180E+04,	2.5733981E+07,
.08000,	4.6426196E+04,	2.5570341E+07,
.08106,	4.6243806E+04,	2.5472215E+07,
.08202,	4.6246547E+04,	2.5476682E+07,
.08303,	4.6390383E+04,	2.5558327E+07,
.08402,	4.6584489E+04,	2.5665636E+07,
.08504,	4.6699180E+04,	2.5726088E+07,
.08606,	4.6613903E+04,	2.5674637E+07,
.08701,	4.6324981E+04,	2.5511111E+07,
.08803,	4.5908154E+04,	2.5279153E+07,
.08901,	4.5514602E+04,	2.5062319E+07,
.09005,	4.5170201E+04,	2.4872751E+07,
.09101,	4.4878381E+04,	2.4710995E+07,
.09203,	4.4550294E+04,	2.4529427E+07,
.09300,	4.4258533E+04,	2.4369254E+07,
.09402,	4.4041145E+04,	2.4250486E+07,
.09504,	4.3883324E+04,	2.4163816E+07,
.09602,	4.3748654E+04,	2.4089902E+07,
.09704,	4.3731915E+04,	2.4085325E+07,
.09803,	4.4055327E+04,	2.4270897E+07,
.09906,	4.4593085E+04,	2.4568299E+07,
.10004,	4.4893727E+04,	2.4729042E+07,
.10506,	4.4915203E+04,	2.4738296E+07,
.11009,	4.5843578E+04,	2.5265301E+07,
.11508,	4.7961126E+04,	2.6441304E+07,
.12006,	4.8641610E+04,	2.6811261E+07,
.12511,	4.8600261E+04,	2.6791935E+07,
.13003,	4.8897292E+04,	2.6954703E+07,
.13507,	4.7822446E+04,	2.6353890E+07,
.14007,	4.6760026E+04,	2.5769811E+07,
.14510,	4.6613100E+04,	2.5690052E+07,
.15004,	4.5690755E+04,	2.5177113E+07,
.15504,	4.5251619E+04,	2.4940772E+07,
.16009,	4.5889095E+04,	2.5299141E+07,
.16521,	4.6090261E+04,	2.5411637E+07,
.17003,	4.6472870E+04,	2.5629872E+07,
.17512,	4.7251129E+04,	2.6064692E+07,
.18005,	4.7225150E+04,	2.6048738E+07,
.18515,	4.6856876E+04,	2.5847037E+07,
.19008,	4.6705900E+04,	2.5764932E+07,
.19500,	4.6273001E+04,	2.5525670E+07,

TABLE 6.2.1-27 (SHEET 27 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.20008,	4.6134180E+04,	2.5453781E+07,
.21005,	4.6386621E+04,	2.5599180E+07,
.22011,	4.7146235E+04,	2.6028554E+07,
.23005,	4.7015697E+04,	2.5959264E+07,
.24013,	4.6347097E+04,	2.5591888E+07,
.25001,	4.6162333E+04,	2.5496921E+07,
.26009,	4.6307811E+04,	2.5584812E+07,
.27006,	4.6570319E+04,	2.5736626E+07,
.28006,	4.6444756E+04,	2.5672954E+07,
.29009,	4.6083901E+04,	2.5479119E+07,
.30002,	4.6141090E+04,	2.5517881E+07,
.31009,	4.6525927E+04,	2.5739030E+07,
.32012,	4.6713708E+04,	2.5849861E+07,
.33009,	4.6460700E+04,	2.5714637E+07,
.34001,	4.6141225E+04,	2.5543500E+07,
.35007,	4.6109222E+04,	2.5533279E+07,
.36007,	4.6227538E+04,	2.5606394E+07,
.37007,	4.6178311E+04,	2.5586123E+07,
.38005,	4.5918254E+04,	2.5448585E+07,
.39016,	4.5767288E+04,	2.5372179E+07,
.40008,	4.8571189E+04,	2.5438048E+07,
.41016,	4.6015706E+04,	2.5526360E+07,
.42012,	4.6063981E+04,	2.5527136E+07,
.43008,	4.5902058E+04,	2.5477926E+07,
.44004,	4.5873316E+04,	2.5469601E+07,
.45003,	4.5921183E+04,	2.5504168E+07,
.46003,	4.5928820E+04,	2.5516775E+07,
.47005,	4.5863777E+04,	2.5489062E+07,
.48007,	4.5822618E+04,	2.5474726E+07,
.49020,	4.5868611E+04,	2.5509101E+07,
.50011,	4.5941758E+04,	2.5558760E+07,
.51002,	4.5968058E+04,	2.5582434E+07,
.52003,	4.6069171E+04,	2.5648956E+07,
.53004,	4.6064521E+04,	2.5654282E+07,
.54004,	4.6024512E+04,	2.5640936E+07,
.55004,	4.6017318E+04,	2.5645755E+07,
.56007,	4.5951087E+04,	2.5618073E+07,
.57003,	4.5875425E+04,	2.5585545E+07,
.58003,	4.5836519E+04,	2.5573007E+07,
.59001,	4.5786252E+04,	2.5554223E+07,
.60015,	4.5761354E+04,	2.5550060E+07,

TABLE 6.2.1-27 (SHEET 28 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.61002,	4.5679475E+04,	2.5513841E+07,
.62006,	4.5614735E+04,	2.5487332E+07,
.63001,	4.5580814E+04,	2.5477627E+07,
.64010,	4.5537959E+04,	2.5463058E+07,
.65006,	4.5472967E+04,	2.5436110E+07,
.66004,	4.5368041E+04,	2.5386841E+07,
.67008,	4.5276146E+04,	2.5344907E+07,
.68008,	4.5212833E+04,	2.5318834E+07,
.69010,	4.5146649E+04,	2.5291273E+07,
.70013,	4.5063020E+04,	2.5253774E+07,
.71008,	4.4970006E+04,	2.5211126E+07,
.72015,	4.4899570E+04,	2.5180997E+07,
.73015,	4.4842223E+04,	2.5157915E+07,
.74007,	4.4768907E+04,	2.5125831E+07,
.75000,	4.4675265E+04,	2.5082213E+07,
.76013,	4.4575080E+04,	2.5034875E+07,
.77005,	4.4486279E+04,	2.4993566E+07,
.78003,	4.4393089E+04,	2.4949759E+07,
.79011,	4.4280814E+04,	2.4895178E+07,
.80002,	4.4159534E+04,	2.4835323E+07,
.81009,	4.4047896E+04,	2.4780782E+07,
.82019,	4.4002782E+04,	2.4763851E+07,
.83017,	4.3971608E+04,	2.4754636E+07,
.84015,	4.3897477E+04,	2.4720952E+07,
.85012,	4.3807403E+04,	2.4677902E+07,
.86011,	4.3719832E+04,	2.4636057E+07,
.87001,	4.3641735E+04,	2.4599452E+07,
.88002,	4.3538419E+04,	2.4548494E+07,
.89016,	4.3425841E+04,	2.4492077E+07,
.90029,	4.3323151E+04,	2.4441194E+07,
.91012,	4.3237202E+04,	2.4399583E+07,
.92001,	4.3197013E+04,	2.4383525E+07,
.93007,	4.3135609E+04,	2.4354941E+07,
.94009,	4.3075249E+04,	2.4326592E+07,
.95014,	4.3008400E+04,	2.4294520E+07,
.96017,	4.2928460E+04,	2.4254898E+07,
.97012,	4.2855367E+04,	2.4218973E+07,
.98005,	4.2795246E+04,	2.4190366E+07,
.99007,	4.2780821E+04,	2.4187831E+07,

TABLE 6.2.1-27 (SHEET 29 OF 36)

(436-in.^{2} BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.00003,	4.2752246E+04,	2.4177108E+07,
1.05004,	4.2539695E+04,	2.4082305E+07,
1.10007,	4.2334300E+04,	2.3988517E+07,
1.15004,	4.2209823E+04,	2.3939257E+07,
1.20025,	4.2149730E+04,	2.3926104E+07,
1.25006,	4.2092436E+04,	2.3914261E+07,
1.30021,	4.2110966E+04,	2.3946258E+07,
1.35002,	4.2078652E+04,	2.3949385E+07,
1.40027,	4.1966932E+04,	2.3905238E+07,
1.45004,	4.1784741E+04,	2.3818295E+07,
1.50000,	4.1727473E+04,	2.3801830E+07,
1.55013,	4.1699789E+04,	2.3801814E+07,
1.60003,	4.1558067E+04,	2.3735546E+07,
1.65008,	4.1435150E+04,	2.3679377E+07,
1.70000,	4.1483172E+04,	2.3722962E+07,
1.75006,	4.1504535E+04,	2.3754789E+07,
1.80008,	4.1483365E+04,	2.3718907E+07,
1.85002,	4.1227437E+04,	2.3642725E+07,
1.90002,	4.1016295E+04,	2.3550974E+07,
1.95014,	4.0870107E+04,	2.3501156E+07,
2.00003,	4.0455307E+04,	2.3295283E+07,
2.05006,	3.9934443E+04,	2.3025753E+07,
2.10007,	3.9753225E+04,	2.2954260E+07,
2.15010,	3.9290013E+04,	2.2718398E+07,
2.20016,	3.8805140E+04,	2.2466812E+07,
2.25009,	3.8502236E+04,	2.2320490E+07,
2.30029,	3.8306991E+04,	2.2241263E+07,
2.35003,	3.7919102E+04,	2.2052489E+07,
2.40001,	3.7768094E+04,	2.2012574E+07,
2.45017,	3.7325121E+04,	2.1803745E+07,
2.50005,	3.6821854E+04,	2.1557137E+07,
2.55005,	3.6306581E+04,	2.1301116E+07,
2.60014,	3.5600723E+04,	2.0928827E+07,
2.65026,	3.4826145E+04,	2.0513480E+07,
2.70004,	3.4058488E+04,	2.0100456E+07,
2.75012,	3.3204439E+04,	1.9641103E+07,
2.80013,	3.2227744E+04,	1.9115998E+07,
2.85005,	3.1113386E+04,	1.8509041E+07,
2.90000,	3.0258151E+04,	1.8052239E+07,
2.95002,	3.0030674E+04,	1.7966679E+07,
3.00007,	2.9782520E+04,	1.7874253E+07,

TABLE 6.2.1-27 (SHEET 30 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.00000,	0. ,	0. ,
.00102,	7.7823700E+03,	4.9504728E+06,
.00201,	1.1086818E+04,	7.0489849E+06,
.00303,	2.2100842E+04,	1.4051387E+07,
.00403,	3.4238857E+04,	2.1774422E+07,
.00500,	4.1741684E+04,	2.6547063E+07,
.00602,	4.6481946E+04,	2.9556339E+07,
.00702,	4.9110254E+04,	3.1218089E+07,
.00800,	5.0377887E+04,	3.2012324E+07,
.00901,	5.0849559E+04,	3.2301064E+07,
.01003,	5.0878346E+04,	3.2311861E+07,
.01104,	5.0761880E+04,	3.2234808E+07,
.01202,	5.0695144E+04,	3.2192830E+07,
.01301,	5.0769941E+04,	3.2242396E+07,
.01400,	5.1006142E+04,	3.2394472E+07,
.01503,	5.1391505E+04,	3.2640344E+07,
.01601,	5.1837302E+04,	3.2923221E+07,
.01701,	5.2309698E+04,	3.3221666E+07,
.01801,	5.2748453E+04,	3.3497754E+07,
.01903,	5.3127817E+04,	3.3735439E+07,
.02003,	5.3411771E+04,	3.3911858E+07,
.02102,	5.3587249E+04,	3.4019292E+07,
.02202,	5.3662466E+04,	3.4063267E+07,
.02300,	5.3650130E+04,	3.4052182E+07,
.02403,	5.3565754E+04,	3.3995967E+07,
.02504,	5.3436329E+04,	3.3912156E+07,
.02601,	5.3290526E+04,	3.3819075E+07,
.02702,	5.1347931E+04,	3.3729261E+07,
.02802,	5.3036867E+04,	3.3660527E+07,
.02904,	5.2971948E+04,	3.3622112E+07,
.03000,	5.2968743E+04,	3.3623478E+07,
.03100,	5.3029110E+04,	3.3665533E+07,
.03201,	5.3153722E+04,	3.3748506E+07,
.03302,	5.3331104E+04,	3.3864763E+07,
.03402,	5.3549493E+04,	3.4006787E+07,
.03501,	5.3794178E+04,	3.4165108E+07,
.03601,	5.4062063E+04,	3.4337820E+07,
.03702,	5.4337548E+04,	3.4515029E+07,

TABLE 6.2.1-27 (SHEET 31 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.03801,	5.4603859E+04,	3.4686046E+07,
.03903,	5.4871375E+04,	3.4857557E+07,
.04002,	5.5115012E+04,	3.5013532E+07,
.04104,	5.5345808E+04,	3.5161050E+07,
.04203,	5.5556895E+04,	3.5295796E+07,
.04302,	5.5742463E+04,	3.5414074E+07,
.04402,	5.5914907E+04,	3.5523738E+07,
.04500,	5.6061661E+04,	3.5616908E+07,
.04602,	5.6195171E+04,	3.5701406E+07,
.04701,	5.6304293E+04,	3.5770270E+07,
.04800,	5.6400251E+04,	3.5830544E+07,
.04906,	5.6479238E+04,	3.5879874E+07,
.05003,	5.6538079E+04,	3.5916299E+07,
.05101,	5.6580392E+04,	3.5942104E+07,
.05202,	5.6607447E+04,	3.5958098E+07,
.05301,	5.6618964E+04,	3.5964131E+07,
.05403,	5.6614906E+04,	3.5960166E+07,
.05507,	5.6595249E+04,	3.5948240E+07,
.05604,	5.6563240E+04,	3.5924545E+07,
.05706,	5.6517173E+04,	3.5893887E+07,
.05808,	5.6459413E+04,	3.5855845E+07,
.05903,	5.6395002E+04,	3.5813728E+07,
.06000,	5.6323849E+04,	3.5767426E+07,
.06105,	5.6238056E+04,	3.5711826E+07,
.06204,	5.6152107E+04,	3.5656354E+07,
.06303,	5.6061848E+04,	3.5598461E+07,
.06406,	5.5969187E+04,	3.5539262E+07,
.06503,	5.5882016E+04,	3.5483752E+07,
.06600,	5.5797599E+04,	3.5430181E+07,
.06705,	5.5711644E+04,	3.5375809E+07,
.06803,	5.5631219E+04,	3.5325235E+07,
.06907,	5.5559086E+04,	3.5280130E+07,
.07002,	5.5492411E+04,	3.5238611E+07,
.07103,	5.5432132E+04,	3.5201303E+07,
.07201,	5.5378454E+04,	3.5168222E+07,
.07301,	5.5330161E+04,	3.5138690E+07,
.07402,	5.5287428E+04,	3.5112743E+07,
.07501,	5.5249761E+04,	3.5090042E+07,
.07601,	5.5215621E+04,	3.5069674E+07,
.07701,	5.5185947E+04,	3.5052144E+07,

TABLE 6.2.1-27 (SHEET 32 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.07800,	5.5515956E+04,	3.5036721E+07,
.07900,	5.5135988E+04,	3.5023107E+07,
.08000,	5.5114523E+04,	3.5010852E+07,
.08100,	5.5095190E+04,	3.4999922E+07,
.08201,	5.5077046E+04,	3.4989734E+07,
.08301,	5.5059845E+04,	3.4980098E+07,
.08402,	5.5042363E+04,	3.4970280E+07,
.08501,	5.5024404E+04,	3.4960147E+07,
.08600,	5.5005332E+04,	3.4949397E+07,
.08701,	5.4984427E+04,	3.4937654E+07,
.08803,	5.4961479E+04,	3.4924813E+07,
.08902,	5.4937813E+04,	3.4911638E+07,
.09001,	5.4912757E+04,	3.4897802E+07,
.09100,	5.4887504E+04,	3.4884017E+07,
.09201,	5.4862924E+04,	3.4870828E+07,
.09303,	5.4839475E+04,	3.4859563E+07,
.09402,	5.4818668E+04,	3.4848096E+07,
.09503,	5.4801182E+04,	3.4839804E+07,
.09601,	5.4787478E+04,	3.4833887E+07,
.09700,	5.4776706E+04,	3.4829916E+07,
.09802,	5.4768772E+04,	3.4827716E+07,
.09902,	5.4762804E+04,	3.4826644E+07,
.10002,	5.4757413E+04,	3.4825796E+07,
.10505,	5.4634156E+04,	3.4756575E+07,
.11004,	5.3966335E+04,	3.4334414E+07,
.11506,	5.2981994E+04,	3.3711323E+07,
.12000,	5.2125082E+04,	3.3169628E+07,
.12507,	5.1385350E+04,	3.2700539E+07,
.13006,	5.0623676E+04,	3.2217691E+07,
.13502,	4.9952734E+04,	3.1794294E+07,
.14002,	4.9624244E+04,	3.1590996E+07,
.14507,	4.9659208E+04,	3.1619886E+07,
.15002,	4.9826967E+04,	3.1730674E+07,
.15504,	4.9958804E+04,	3.1816492E+07,
.16014,	4.9974844E+04,	3.1827172E+07,
.16512,	4.9873952E+04,	3.1762947E+07,
.17007,	4.9676572E+04,	3.1637817E+07,
.17500,	4.9424401E+04,	3.1478882E+07,
.18002,	4.9169537E+04,	3.1319320E+07,
.18507,	4.8977379E+04,	3.1200412E+07,
.19008,	4.8891947E+04,	3.1149864E+07,

TABLE 6.2.1-27 (SHEET 33 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.19512,	4.8905230E+04,	3.1162286E+07,
.20017,	4.8971499E+04,	3.1208073E+07,
.21011,	4.9132117E+04,	3.1316312E+07,
.22004,	4.9118318E+04,	3.1311984E+07,
.23006,	4.8895594E+04,	3.1176020E+07,
.24001,	4.8547872E+04,	3.0963726E+07,
.25009,	4.8202471E+04,	3.0756576E+07,
.26004,	4.7929553E+04,	3.0596316E+07,
.27003,	4.7702439E+04,	3.0462393E+07,
.28009,	4.7508889E+04,	3.0345842E+07,
.29002,	4.7371087E+04,	3.0261785E+07,
.30001,	4.7306930E+04,	3.0223088E+07,
.31008,	4.7280472E+04,	3.0207649E+07,
.32012,	4.7230590E+04,	3.0177807E+07,
.33006,	4.7094913E+04,	3.0094690E+07,
.34005,	4.6887690E+04,	2.9968310E+07,
.35010,	4.6650874E+04,	2.9824836E+07,
.36013,	4.6424525E+04,	2.9688533E+07,
.37004,	4.6213540E+04,	2.9562593E+07,
.38012,	4.6028987E+04,	2.9452992E+07,
.39005,	4.5905610E+04,	2.9381484E+07,
.40008,	4.5854401E+04,	2.9355651E+07,
.41013,	4.5863015E+04,	2.9367338E+07,
.42003,	4.5897394E+04,	2.9394808E+07,
.43010,	4.5915956E+04,	2.9411982E+07,
.44013,	4.5895521E+04,	2.9403662E+07,
.45009,	4.5822802E+04,	2.9361067E+07,
.46003,	4.5702115E+04,	2.9287550E+07,
.47007,	4.5560837E+04,	2.9201025E+07,
.48012,	4.5440327E+04,	2.9128036E+07,
.49005,	4.5370502E+04,	2.9087580E+07,
.50001,	4.5356383E+04,	2.9082777E+07,
.51009,	4.5385682E+04,	2.9105652E+07,
.52009,	4.5432669E+04,	2.9139564E+07,
.53005,	4.5468619E+04,	2.9166079E+07,
.54006,	4.5468395E+04,	2.9169117E+07,
.55003,	4.5416172E+04,	2.9138892E+07,
.56000,	4.5321254E+04,	2.9081899E+07,
.57006,	4.5210656E+04,	2.9015687E+07,
.58001,	4.5115650E+04,	2.8960172E+07,

TABLE 6.2.1-27 (SHEET 34 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
.59004,	4.5044293E+04,	2.8920623E+07,
.60012,	4.5010443E+04,	2.8906032E+07,
.61010,	4.5013950E+04,	2.8915961E+07,
.62013,	4.5038708E+04,	2.8939886E+07,
.63009,	4.5059370E+04,	2.8961219E+07,
.64014,	4.5060179E+04,	2.8969927E+07,
.65011,	4.5035184E+04,	2.8962020E+07,
.66013,	4.4986898E+04,	2.8939247E+07,
.67009,	4.4925663E+04,	2.8908217E+07,
.68003,	4.4865426E+04,	2.8877950E+07,
.69012,	4.4818243E+04,	2.8855888E+07,
.70008,	4.4791555E+04,	2.8846866E+07,
.71002,	4.4780332E+04,	2.8847766E+07,
.72000,	4.4772711E+04,	2.8850961E+07,
.73002,	4.4760797E+04,	2.8851406E+07,
.74007,	4.4741266E+04,	2.8847037E+07,
.75002,	4.4712307E+04,	2.8836656E+07,
.76001,	4.4673720E+04,	2.8820326E+07,
.77004,	4.4627231E+04,	2.8799190E+07,
.78001,	4.4576132E+04,	2.8775296E+07,
.79005,	4.4521680E+04,	2.8749623E+07,
.80009,	4.4469922E+04,	2.8726022E+07,
.81002,	4.4425386E+04,	2.8707297E+07,
.82009,	4.4385236E+04,	2.8691873E+07,
.83004,	4.4346941E+04,	2.8677848E+07,
.84013,	4.4306106E+04,	2.8662665E+07,
.85007,	4.4261121E+04,	2.8644991E+07,
.86005,	4.4201427E+04,	2.8618225E+07,
.87006,	4.4147923E+04,	2.8595735E+07,
.88001,	4.4100598E+04,	2.8577514E+07,
.89009,	4.4061089E+04,	2.8564640E+07,
.90003,	4.4041189E+04,	2.8564429E+07,
.91005,	4.4031122E+04,	2.8570789E+07,
.92010,	4.4021949E+04,	2.8577993E+07,
.93001,	4.4004878E+04,	2.8580150E+07,
.94003,	4.3975310E+04,	2.8574674E+07,
.95000,	4.3932837E+04,	2.8561112E+07,
.96007,	4.3877626E+04,	2.8539750E+07,
.97011,	4.3813505E+04,	2.8512862E+07,
.98001,	4.3746579E+04,	2.8434177E+07,
.99004,	4.3679310E+04,	2.8455659E+07,

TABLE 6.2.1-27 (SHEET 35 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
1.00009,	4.3616582E+04,	2.8430269E+07,
1.05004,	4.3362851E+04,	2.8343817E+07,
1.10009,	4.3106996E+04,	2.8203634E+07,
1.15007,	4.2656312E+04,	2.8059859E+07,
1.20002,	4.2276635E+04,	2.7914949E+07,
1.25002,	4.1996846E+04,	2.7844326E+07,
1.30006,	4.1778160E+04,	2.7808761E+07,
1.35006,	4.1502658E+04,	2.7729192E+07,
1.40003,	4.1145142E+04,	2.7590021E+07,
1.45007,	4.0719791E+04,	2.7403645E+07,
1.50011,	4.0278464E+04,	2.7196215E+07,
1.55009,	3.9836748E+04,	2.6978928E+07,
1.60001,	3.9415183E+04,	2.6769679E+07,
1.65012,	3.8998375E+04,	2.6559661E+07,
1.70012,	3.8602806E+04,	2.6359988E+07,
1.75011,	3.8221231E+04,	2.6165738E+07,
1.80006,	3.7854297E+04,	2.5973809E+07,
1.85007,	3.7502521E+04,	2.5782014E+07,
1.90000,	3.7155071E+04,	2.5583127E+07,
1.95010,	3.6808801E+04,	2.5377270E+07,
2.00009,	3.6453550E+04,	2.5162290E+07,
2.05014,	3.6085701E+04,	2.4941049E+07,
2.10011,	3.5697676E+04,	2.4713441E+07,
2.15005,	3.5290806E+04,	2.4482273E+07,
2.20000,	3.4873172E+04,	2.4253393E+07,
2.25004,	3.4442934E+04,	2.4025041E+07,
2.30012,	3.4020636E+04,	2.3801494E+07,
2.35007,	3.3630302E+04,	2.3589986E+07,
2.40002,	3.3267885E+04,	2.3389694E+07,
2.45004,	3.2900597E+04,	2.3195920E+07,
2.50007,	3.2674711E+04,	2.3007341E+07,
2.55013,	3.2398991E+04,	2.2818065E+07,
2.60007,	3.2112088E+04,	2.2622375E+07,
2.65003,	3.1797713E+04,	2.2417317E+07,
2.70013,	3.1441037E+04,	2.2198094E+07,
2.75000,	3.1057741E+04,	2.1971109E+07,
2.80009,	3.0669535E+04,	2.1741814E+07,
2.85004,	3.0303664E+04,	2.1519357E+07,

TABLE 6.2.1-27 (SHEET 36 OF 36)

(763-in.^{2} HOT LEG BREAK)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)
2.90009,	2.9952884E+04,	2.1304776E+07,
2.95003,	2.9588947E+04,	2.1034229E+07,
3.00005,	2.9263868E+04,	2.0871117E+07,

TABLE 6.2.1-28 (SHEET 1 OF 7)

PRESSURIZER COMPARTMENT MASS AND ENERGY RELEASE
PRESSURIZER SURGE LINE BREAK

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.00000	0.00	0.00	0.00
0.00251	5.6681148E+04	1.1296008E+07	677.17
0.00501	1.6556361E+04	1.1212058E+07	677.21
0.00752	1.6699069E+04	1.1302997E+07	676.86
0.01002	1.9033506E+04	1.2830006E+07	674.07
0.01250	2.2089365E+04	1.4828262E+07	671.29
0.01501	2.1648161E+04	1.4533929E+07	671.37
0.01754	2.1247911E+04	1.4268193E+07	671.51
0.02002	2.0465838E+04	1.3752132E+07	671.96
0.02250	2.0393611E+04	1.3704347E+07	671.99
0.02505	2.0706044E+04	1.3907231E+07	671.65
0.02752	2.0931729E+04	1.4053966E+07	671.42
0.03001	2.0998600E+04	1.4096217E+07	671.29
0.03258	2.0967919E+04	1.4074876E+07	671.26
0.03503	2.0990414E+04	1.4088700E+07	671.20
0.03750	2.1019840E+04	1.4107187E+07	671.14
0.04009	2.1062241E+04	1.4134287E+07	671.07
0.04259	2.1156624E+04	1.4195514E+07	670.97
0.04512	2.1160405E+04	1.4197306E+07	670.94
0.04761	2.1098863E+04	1.4156324E+07	670.95
0.05009	2.1066994E+04	1.4134990E+07	670.95
0.05264	2.1095761E+04	1.4153509E+07	670.92
0.05505	2.1085426E+04	1.4146331E+07	670.91
0.05751	2.1000054E+04	1.4089897E+07	670.95
0.06008	2.0863697E+04	1.4000108E+07	671.03
0.06255	2.0694171E+04	1.3888722E+07	671.14
0.06512	2.0509657E+04	1.3767663E+07	671.28
0.06759	2.0407265E+04	1.3700589E+07	671.36
0.07002	2.0418448E+04	1.3708034E+07	671.36
0.07250	2.0481072E+04	1.3749121E+07	671.31
0.07507	2.0519777E+04	1.3774417E+07	671.28
0.07754	2.0521037E+04	1.3775092E+07	671.27
0.08008	2.0488129E+04	1.3753315E+07	671.28
0.08255	2.0410939E+04	1.3702503E+07	671.33
0.08504	2.0305825E+04	1.3633481E+07	671.41
0.08755	2.0203846E+04	1.3566573E+07	671.48
0.09006	2.0124810E+04	1.3514710E+07	671.54
0.09261	2.0067276E+04	1.3476990E+07	671.59
0.09501	2.0049927E+04	1.3465634E+07	671.61
0.09751	2.0091047E+04	1.3492584E+07	671.57
0.10011	2.0190095E+04	1.3557427E+07	671.49

TABLE 6.2.1-28 (SHEET 2 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.10259	2.0322401E+04	1.3644022E+07	671.38
0.10513	2.0470869E+04	1.3741182E+07	671.26
0.10762	2.0613302E+04	1.3834228E+07	671.13
0.11001	2.0698652E+04	1.3889843E+07	671.05
0.11251	2.0710177E+04	1.3896965E+07	671.02
0.11504	2.0648143E+04	1.3855823E+07	671.04
0.11753	2.0539220E+04	1.3783979E+07	671.11
0.12009	2.0392131E+04	1.3687301E+07	671.21
0.12255	2.0235027E+04	1.3584155E+07	671.32
0.12505	2.0076107E+04	1.3479844E+07	671.44
0.12762	1.9907462E+04	1.3369294E+07	671.57
0.13001	1.9721330E+04	1.3247406E+07	671.73
0.13253	1.9515669E+04	1.3112767E+07	671.91
0.13513	1.9325524E+04	1.2988402E+07	672.09
0.13760	1.9143891E+04	1.2869713E+07	672.26
0.14000	1.8989530E+04	1.2768843E+07	672.41
0.14263	1.8858030E+04	1.2682997E+07	672.55
0.14517	1.8757129E+04	1.2617164E+07	672.66
0.14757	1.8681814E+04	1.2568006E+07	672.74
0.15014	1.8626059E+04	1.2531659E+07	672.80
0.15265	1.8590127E+04	1.2508250E+07	672.84
0.15504	1.8569023E+04	1.2494497E+07	672.87
0.15764	1.8551204E+04	1.2482868E+07	672.89
0.16005	1.8525873E+04	1.2466307E+07	672.91
0.16250	1.8477290E+04	1.2434527E+07	672.96
0.16508	1.8394683E+04	1.2380511E+07	673.05
0.16765	1.8282083E+04	1.2306962E+07	673.17
0.17002	1.8146081E+04	1.2218190E+07	673.32
0.17258	1.7994507E+04	1.2119261E+07	673.50
0.17510	1.7851733E+04	1.2026175E+07	673.67
0.17754	1.7718757E+04	1.1939508E+07	673.83
0.18004	1.7607714E+04	1.1867137E+07	673.97
0.18257	1.7516255E+04	1.1807583E+07	674.09
0.18510	1.7440413E+04	1.1758207E+07	674.19
0.18753	1.7384799E+04	1.1721993E+07	674.27
0.19006	1.7331275E+04	1.1687159E+07	674.34
0.19258	1.7283259E+04	1.1655900E+07	674.40
0.19513	1.7239302E+04	1.1627267E+07	674.46
0.19757	1.7196585E+04	1.1599461E+07	674.52
0.20000	1.7153255E+04	1.1571232E+07	674.58
0.20257	1.7114304E+04	1.1545858E+07	674.63
0.20515	1.7075604E+04	1.1520650E+07	674.68
0.20764	1.7047204E+04	1.1502146E+07	674.72
0.21013	1.7030892E+04	1.1491517E+07	674.75

TABLE 6.2.1-28 (SHEET 3 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.21262	1.7026041E+04	1.1488355E+07	674.75
0.21509	1.7031337E+04	1.1491802E+07	674.74
0.21755	1.7044630E+04	1.1500441E+07	674.73
0.22014	1.7064281E+04	1.1513204E+07	674.70
0.22261	1.7085144E+04	1.1526752E+07	674.67
0.22518	1.7104103E+04	1.1539031E+07	674.64
0.22757	1.7120939E+04	1.1549913E+07	674.61
0.23002	1.7133999E+04	1.1558321E+07	674.58
0.23261	1.7139862E+04	1.1562022E+07	674.57
0.23511	1.7135943E+04	1.1559323E+07	674.57
0.23756	1.7121402E+04	1.1549700E+07	674.58
0.24007	1.7091493E+04	1.1530055E+07	674.61
0.24253	1.7044464E+04	1.1499260E+07	674.66
0.24509	1.6981075E+04	1.1457784E+07	674.74
0.24752	1.6909137E+04	1.1410815E+07	674.83
0.25009	1.6816993E+04	1.1350678E+07	674.95
0.25256	1.6725891E+04	1.1291225E+07	675.07
0.25505	1.6633374E+04	1.1230921E+07	675.20
0.25751	1.6542717E+04	1.1171824E+07	675.33
0.26005	1.6466594E+04	1.1122207E+07	675.44
0.26251	1.6403107E+04	1.1080847E+07	675.53
0.26504	1.6352876E+04	1.1048104E+07	675.61
0.26751	1.6321725E+04	1.1027787E+07	675.65
0.27009	1.6296620E+04	1.1011400E+07	675.69
0.27265	1.6276846E+04	1.0998473E+07	675.71
0.27509	1.6255893E+04	1.0984763E+07	675.74
0.27751	1.6228524E+04	1.0966874E+07	675.78
0.28003	1.6185426E+04	1.0938740E+07	675.84
0.28253	1.6127011E+04	1.0900616E+07	675.92
0.28512	1.6053276E+04	1.0852567E+07	676.03
0.28755	1.5970294E+04	1.0798521E+07	676.16
0.29001	1.5898962E+04	1.0752073E+07	676.28
0.29256	1.5832910E+04	1.0709102E+07	676.38
0.29509	1.5829868E+04	1.0707076E+07	676.38
0.29751	1.5825312E+04	1.0704075E+07	676.39
0.30009	1.5808551E+04	1.0693108E+07	676.41
0.30265	1.5808165E+04	1.0692807E+07	676.41
0.30517	1.5803688E+04	1.0689818E+07	676.41
0.30753	1.5753769D+04	1.0657243E+07	676.49
0.31006	1.5744724E+04	1.0651197E+07	676.49
0.31256	1.5753012E+04	1.0656385E+07	676.47
0.31501	1.5760820E+04	1.0661304E+07	676.44
0.31767	1.5737615E+04	1.0646068E+07	676.47
0.32014	1.5735895E+04	1.0644859E+07	676.47

TABLE 6.2.1-28 (SHEET 4 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.32273	1.5734566E+04	1.0643956E+07	676.47
0.32509	1.5732595E+04	1.0642618E+07	676.47
0.32753	1.5729905E+04	1.0640794E+07	676.47
0.33024	1.5726344E+04	1.0638378E+07	676.47
0.33251	1.5723018E+04	1.0636121E+07	676.47
0.33526	1.5718891E+04	1.0633317E+07	676.47
0.33752	1.5715349E+04	1.0630912E+07	676.47
0.34007	1.5711537E+04	1.0628319E+07	676.47
0.34274	1.5707870E+04	1.0625820E+07	676.46
0.34506	1.5704632E+04	1.0623609E+07	676.46
0.34770	1.5701190E+04	1.0621254E+07	676.46
0.35025	1.5698041E+04	1.0619096E+07	676.46
0.35261	1.5695247E+04	1.0617177E+07	676.46
0.35516	1.5692136E+04	1.0615038E+07	676.46
0.35762	1.5688914E+04	1.0612827E+07	676.45
0.36011	1.5685466E+04	1.0610463E+07	676.45
0.36259	1.5681810E+04	1.0607961E+07	676.45
0.36518	1.5677911E+04	1.0605294E+07	676.45
0.36758	1.5674219E+04	1.0602767E+07	676.45
0.37034	1.5670037E+04	1.0599906E+07	676.44
0.37267	1.5666639E+04	1.0597581E+07	676.44
0.37504	1.5663437E+04	1.0595386E+07	676.44
0.37757	1.5660250E+04	1.0593197E+07	676.44
0.38027	1.5657086E+04	1.0591016E+07	676.44
0.38277	1.5654322E+04	1.0589105E+07	676.43
0.38516	1.5651744E+04	1.0587321E+07	676.43
0.38754	1.5649144E+04	1.0585526E+07	676.43
0.39027	1.5646820E+04	1.0583914E+07	676.43
0.39262	1.5644725E+04	1.0582456E+07	676.42
0.39522	1.5642789E+04	1.0581100E+07	676.42
0.39760	1.5641070E+04	1.0579890E+07	676.42
0.40004	1.5639449E+04	1.0578741E+07	676.41
0.40257	1.5637925E+04	1.0577652E+07	676.41
0.40505	1.5636591E+04	1.0576689E+07	676.41
0.40753	1.5635413E+04	1.0575828E+07	676.40
0.41013	1.5634337E+04	1.0575027E+07	676.40
0.41251	1.5633562E+04	1.0574432E+07	676.39
0.41504	1.5632975E+04	1.0573960E+07	676.39
0.41775	1.4532551E+04	1.0573584E+07	676.38
0.42001	1.5632366E+04	1.0573381E+07	676.38
0.42268	1.5632315E+04	1.0573254E+07	676.37
0.42508	1.5632378E+04	1.0573212E+07	676.37
0.42781	1.5632537E+04	1.0573215E+07	676.36
0.43027	1.5632708E+04	1.0573240E+07	676.35

TABLE 6.2.1-28 (SHEET 5 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.43267	1.5632877E+04	1.0573264E+07	676.35
0.43529	1.5633025E+04	1.0573264E+07	676.34
0.43752	1.4533096E+04	1.0573227E+07	676.34
0.44015	1.5633088E+04	1.0573125E+07	676.33
0.44255	1.5632964E+04	1.0572955E+07	676.32
0.44523	1.5632658E+04	1.0572657E+07	676.32
0.44756	1.5632172E+04	1.0572247E+07	676.31
0.45019	1.5631366E+04	1.0571613E+07	676.31
0.45237	1.5630249E+04	1.0570770E+07	676.30
0.45517	1.5629049E+04	1.0569885E+07	676.30
0.45757	1.5627623E+04	1.0568848E+07	676.29
0.46022	1.5625881E+04	1.0567590E+07	676.29
0.46289	1.5623874E+04	1.0566152E+07	676.28
0.46514	1.5622119E+04	1.0564900E+07	676.28
0.46777	1.5619955E+04	1.0563361E+07	676.27
0.47036	1.5617741E+04	1.0561792E+07	676.27
0.47288	1.5615520E+04	1.0560223E+07	676.26
0.47503	1.5613565E+04	1.0558844E+07	676.26
0.47757	1.5611133E+04	1.0557133E+07	676.26
0.48011	1.5608643E+04	1.0555384E+07	676.25
0.48252	1.5606080E+04	1.0553588E+07	676.25
0.48512	1.5603492E+04	1.0551775E+07	676.24
0.48758	1.5600916E+04	1.0549973E+07	676.24
0.49009	1.5598319E+04	1.0548154E+07	676.24
0.49273	1.5595641E+04	1.0546276E+07	676.23
0.49505	1.5593323E+04	1.0544647E+07	676.23
0.49786	1.5590573E+04	1.0542711E+07	676.22
0.50029	1.5588315E+04	1.0541119E+07	676.22
0.51010	1.5579889E+04	1.0535146E+07	676.20
0.52041	1.5571914E+04	1.0529460E+07	676.18
0.53013	1.5564820E+04	1.0524389E+07	676.17
0.54025	1.5557270E+04	1.0518995E+07	676.15
0.55044	1.5550075E+04	1.0513839E+07	676.13
0.56029	1.5544285E+04	1.0509628E+07	676.11
0.57048	1.5539811E+04	1.0506279E+07	676.09
0.58035	1.5537036E+04	1.0504053E+07	676.07
0.59012	1.5535838E+04	1.0502863E+07	676.04
0.60013	1.5535793E+04	1.0502401E+07	676.01
0.61039	1.5536152E+04	1.0502172E+07	675.98
0.62027	1.5535777E+04	1.0501463E+07	675.95
0.63022	1.5533739E+04	1.0499646E+07	675.93
0.64028	1.5529494E+04	1.0496323E+07	675.90
0.65009	1.5523642E+04	1.0491943E+07	675.87
0.66010	1.5516674E+04	1.0486820E+07	675.84

TABLE 6.2.1-28 (SHEET 6 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.67006	1.5509527E+04	1.0481657E+07	675.82
0.68002	1.5503246E+04	1.0476939E+07	675.79
0.69043	1.5497613E+04	1.0472703E+07	675.76
0.70003	1.5493333E+04	1.0469413E+07	675.74
0.71008	1.5489729E+04	1.0466564E+07	675.71
0.72062	1.5486462E+04	1.0463919E+07	675.68
0.73049	1.5483628E+04	1.0461579E+07	675.65
0.74006	1.5481145E+04	1.0459476E+07	675.63
0.75052	1.5478710E+04	1.0457353E+07	675.60
0.76039	1.5476578E+04	1.0455447E+07	675.57
0.77046	1.5474438E+04	1.0453521E+07	675.53
0.78005	1.5472213E+04	1.0451554E+07	675.50
0.79034	1.5469339E+04	1.0449119E+07	675.47
0.80022	1.5465740E+04	1.0446217E+07	675.44
0.81004	1.5461051E+04	1.0442584E+07	675.41
0.82013	1.5455070E+04	1.0438084E+07	675.38
0.83054	1.5447988E+04	1.0432852E+07	675.35
0.84013	1.5441170E+04	1.0427849E+07	675.33
0.85057	1.5433935E+04	1.0422535E+07	675.30
0.86010	1.5427803E+04	1.0418005E+07	675.27
0.87064	1.5421686E+04	1.0413445E+07	675.25
0.88049	1.5416519E+04	1.0409555E+07	675.22
0.89029	1.5411736E+04	1.0405923E+07	675.19
0.90062	1.5406860E+04	1.0402204E+07	675.17
0.91043	1.5402219E+04	1.0398661E+07	675.14
0.92014	1.5397541E+04	1.0395092E+07	675.11
0.93032	1.5392543E+04	1.0391286E+07	675.09
0.94013	1.5387606E+04	1.0387534E+07	675.06
0.95020	1.5382356E+04	1.0383562E+07	675.03
0.96044	1.5376745E+04	1.0379341E+07	675.00
0.97022	1.5371025E+04	1.0375073E+07	674.98
0.98023	1.5364719E+04	1.0370408E+07	674.95
0.99000	1.5358118E+04	1.0365561E+07	674.92
1.00024	1.5350854E+04	1.0360257E+07	674.90
1.05009	1.5319038E+04	1.0336848E+07	674.77
1.10020	1.5295400E+04	1.0318842E+07	674.64
1.15027	1.5270563E+04	1.0299964E+07	674.50
1.20065	1.5244981E+04	1.0280607E+07	674.36
1.25022	1.5225611E+04	1.0265425E+07	674.22
1.30001	1.5204826E+04	1.0249226E+07	674.08
1.35008	1.5181682E+04	1.0231482E+07	673.94
1.40047	1.5158958E+04	1.0214058E+07	673.80
1.45008	1.5135654E+04	1.0196263E+07	673.66
1.50052	1.5108628E+04	1.0175997E+07	673.52

TABLE 6.2.1-28 (SHEET 7 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
1.55020	1.5082453E+04	1.0156373E+07	673.39
1.60001	1.5056220E+04	1.0136713E+07	673.26
1.65041	1.5029192E+04	1.0116506E+07	673.12
1.70057	1.5003251E+04	1.0097044E+07	672.99
1.75042	1.4978790E+04	1.0078568E+07	672.86
1.80008	1.4954574E+04	1.0060259E+07	672.72
1.85001	1.4910178E+04	1.0027829E+07	672.55
1.90030	1.4937484E+04	1.0045448E+07	672.50
1.95350	1.4846071E+04	9.9803806E+06	672.26
2.00022	1.4821801E+04	9.9621501E+06	672.13

TABLE 6.2.1-28A (SHEET 1 OF 7)

PRESSURIZER COMPARTMENT
MASS AND ENERGY RELEASE
PRESSURIZER SPRAY LINE BREAK

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.00000	0.00	0.00	0.00
0.00251	5.5520269E+03	3.4074591E+06	613.73
0.00502	5.7566695E+03	3.5214197E+06	611.71
0.00751	5.6923083E+03	3.4814317E+06	611.60
0.01002	5.6156477E+03	3.4348228E+06	611.65
0.01251	5.5820416E+03	3.4132128E+06	611.46
0.01502	5.6059056E+03	3.4246446E+06	610.90
0.01755	5.9216506E+03	3.6028096E+06	608.41
0.02003	6.0469291E+03	3.6722760E+06	607.30
0.02255	6.0807799E+03	3.6897530E+06	606.79
0.02505	6.0942687E+03	3.6961327E+06	606.49
0.02754	6.2322326E+03	3.7737113E+06	605.52
0.03004	6.3772227E+03	3.8554807E+06	604.57
0.03259	6.4620881E+03	3.9026378E+06	603.93
0.03507	6.4752834E+03	3.9088990E+06	603.66
0.03753	6.5151813E+03	3.9305146E+06	603.29
0.04002	6.5143568E+03	3.9287741E+06	603.09
0.04251	6.4438043E+03	3.8870628E+06	603.22
0.04511	6.3427181E+03	3.8281043E+06	603.54
0.04763	6.2773775E+03	3.7899141E+06	603.74
0.05005	6.2551735E+03	3.7764886E+06	603.74
0.05263	6.2524046E+03	3.7741758E+06	603.64
0.05518	6.2668754E+03	3.7818297E+06	603.46
0.05769	6.2974621E+03	3.7987461E+06	603.22
0.06003	6.3249883E+03	3.8139264E+06	602.99
0.06259	6.3350379E+03	3.8191169E+06	602.86
0.06510	6.3288577E+03	3.8150521E+06	602.80
0.06750	6.3276385E+03	3.8139641E+06	602.75
0.07005	6.3455081E+03	3.8238003E+06	602.60
0.07250	6.3655132E+03	3.8354407E+06	602.44
0.07503	6.3688606E+03	3.8363851E+06	602.37
0.07765	6.3404989E+03	3.8197353E+06	602.43
0.08003	6.2872107E+03	3.7888759E+06	602.63
0.08254	6.2175863E+03	3.7487047E+06	602.92
0.08512	6.1352700E+03	3.7014579E+06	603.31
0.08757	6.0708749E+03	3.6645461E+06	603.63
0.09001	6.0409120E+03	3.6473208E+06	603.77
0.09257	6.0539679E+03	3.6546606E+06	603.68
0.09509	6.1075851E+03	3.6851210E+06	603.33
0.09759	6.1876085E+03	3.7306914E+06	602.93
0.10002	6.2735718E+03	3.7796317E+06	602.47
0.10257	6.3664980E+03	3.8325765E+06	601.99

TABLE 6.2.1-28A (SHEET 2 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.10507	6.4429047E+03	3.8761465E+06	601.61
0.10758	6.4965895E+03	3.9067176E+06	601.35
0.11001	6.5270618E+03	3.9240262E+06	601.19
0.11253	6.5344748E+03	3.9281326E+06	601.14
0.11510	6.5204882E+03	3.9199857E+06	601.18
0.11761	6.4843044E+03	3.8991168E+06	601.32
0.12009	6.4325725E+03	3.8693579E+06	601.53
0.12256	6.3726344E+03	3.8349813E+06	601.79
0.12518	6.3034597E+03	3.7953523E+06	602.11
0.12761	6.2383094E+03	3.7580645E+06	602.42
0.13012	6.1820566E+03	3.7258948E+06	602.70
0.13264	6.1395327E+03	3.7015958E+06	602.91
0.13504	6.1127110E+03	3.6862746E+06	603.05
0.13756	6.0998879E+03	3.6789327E+06	603.11
0.14010	6.0988357E+03	3.6783086E+06	603.12
0.14267	6.1080626E+03	3.6835439E+06	603.06
0.14510	6.1240390E+03	3.6926378E+06	602.97
0.14758	6.1442581E+03	3.7041585E+06	602.87
0.15011	6.1667410E+03	3.7169743E+06	602.75
0.15260	6.1867163E+03	3.7283579E+06	602.64
0.15502	6.1992384E+03	3.7354879E+06	602.57
0.15755	6.2031636E+03	3.7377038E+06	602.55
0.16011	6.1939722E+03	3.7324198E+06	602.59
0.16265	6.1745723E+03	3.7213319E+06	602.69
0.16510	6.1423746E+03	3.7029436E+06	602.85
0.16751	6.1097626E+03	3.6843379E+06	603.02
0.17014	6.0750772E+03	3.6645786E+06	603.22
0.17253	6.0482471E+03	3.6492888E+06	603.36
0.17512	6.0289841E+03	3.6383136E+06	603.47
0.17760	6.0193184E+03	3.6328088E+06	603.52
0.18008	6.0197962E+03	3.6330808E+06	603.52
0.18258	6.0276833E+03	3.6375701E+06	603.48
0.18501	6.0400469E+03	3.6446001E+06	603.41
0.18766	6.0528858E+03	3.6518945E+06	603.33
0.19013	6.0646210E+03	3.6585557E+06	603.26
0.19254	6.0708106E+03	3.6620600E+06	603.22
0.19500	6.0717703E+03	3.6625777E+06	603.21
0.19754	6.0668335E+03	3.6597328E+06	603.24
0.20010	6.0570812E+03	3.6541472E+06	603.29
0.20274	6.0452208E+03	3.6473601E+06	603.35
0.20509	6.0330499E+03	3.6404113E+06	603.41
0.20760	6.0241617E+03	3.6353420E+06	603.46
0.21005	6.0191035E+03	3.6324522E+06	603.49
0.21354	6.0197127E+03	3.6327939E+06	603.48

TABLE 6.2.1-28A (SHEET 3 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.21510	6.0257956E+03	3.6362513E+06	603.45
0.21754	6.0357699E+03	3.6419285E+06	603.39
0.22016	6.0480859E+03	3.6489364E+06	603.32
0.22256	6.0625560E+03	3.6571731E+06	603.24
0.22504	6.0749929E+03	3.6642500E+06	603.17
0.22753	6.0866417E+03	3.6708838E+06	603.10
0.23007	6.0951080E+03	3.6757012E+06	603.06
0.23259	6.1001899E+03	3.6785898E+06	603.05
0.23508	6.1011008E+03	3.6791040E+06	603.02
0.23753	6.0973306E+03	3.6769487E+06	603.04
0.24023	6.0876426E+03	3.6714129E+06	603.09
0.24255	6.0753055E+03	3.6643848E+06	603.16
0.24503	6.0564102E+03	3.6536132E+06	603.26
0.24761	6.0364519E+03	3.6422456E+06	603.38
0.25009	6.0184816E+03	3.6320047E+06	603.48
0.25280	6.0028077E+03	3.6230838E+06	603.56
0.25504	5.9931687E+03	3.6175960E+06	603.62
0.25759	5.9900445E+03	3.6158252E+06	603.64
0.26004	5.9938317E+03	3.6179873E+06	603.62
0.26255	6.0039391E+03	3.6237539E+06	603.56
0.26501	6.0188653E+03	3.6322491E+06	603.48
0.26759	6.0373297E+03	3.6427624E+06	603.37
0.27011	6.0551031E+03	3.6528848E+06	603.27
0.27258	6.0711548E+03	3.6620230E+06	603.18
0.27506	6.0821145E+03	3.6682550E+06	603.12
0.27752	6.0879002E+03	3.6715416E+06	603.09
0.28003	6.0867031E+03	3.6708478E+06	603.09
0.28252	6.0784821E+03	3.6661507E+06	603.14
0.28504	6.0651856E+03	3.6585621E+06	603.21
0.28754	6.0434821E+03	3.6461959E+06	603.33
0.29006	6.0196102E+03	3.6325934E+06	603.46
0.29255	5.9947341E+03	3.6184236E+06	603.60
0.29520	5.9687892E+03	3.6036635E+06	603.75
0.29758	5.9450244E+03	3.5901412E+06	603.89
0.30004	5.9252478E+03	3.5788953E+06	604.01
0.30284	5.9052411E+03	3.5675125E+06	604.13
0.30504	5.8913813E+03	3.5596397E+06	604.21
0.30760	5.8782215E+03	3.5521680E+06	604.29
0.31001	5.8700234E+03	3.5475147E+06	604.34
0.31256	5.8639105E+03	3.5440436E+06	604.38
0.31501	5.8607978E+03	3.5422734E+06	604.40
0.31753	5.8599636E+03	3.5417959E+06	604.41
0.32004	5.8608011E+03	3.5422668E+06	604.40
0.32254	5.8629988E+03	3.5435071E+06	604.38
0.32505	5.8656482E+03	3.5450037E+06	604.37

TABLE 6.2.1-28A (SHEET 4 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.32784	5.8686407E+03	3.5466905E+06	604.35
0.33009	5.8714918E+03	3.5482967E+06	604.33
0.33273	5.8741682E+03	3.5498034E+06	604.31
0.33508	5.8771774E+03	3.5514986E+06	604.29
0.33756	5.8802735E+03	3.5532429E+06	604.26
0.34009	5.8841700E+03	3.5554426E+06	604.24
0.34260	5.8884789E+03	3.5578797E+06	604.21
0.34507	5.8930780E+03	3.5604779E+06	604.18
0.34755	5.8975952E+03	3.5630314E+06	604.15
0.35027	5.9014291E+03	3.5651946E+06	604.12
0.35257	5.9039809E+03	3.5666279E+06	604.11
0.35506	5.9044769E+03	3.5668906E+06	604.10
0.35763	5.9024245E+03	3.5657013E+06	604.11
0.36011	5.8983668E+03	3.5633740E+06	604.13
0.36265	5.8905550E+03	3.5589069E+06	604.17
0.36502	5.8818991E+03	3.5539585E+06	604.22
0.36763	5.8702901E+03	3.5473402E+06	604.29
0.37011	5.8585785E+03	3.5406607E+06	604.35
0.37259	5.8478945E+03	3.5345723E+06	604.42
0.37512	5.8384907E+03	3.5292171E+06	604.47
0.37757	5.8315699E+03	3.5252690E+06	604.51
0.38001	5.8283142E+03	3.5234085E+06	604.53
0.38255	5.8284433E+03	3.5234795E+06	604.53
0.38504	5.8325211E+03	3.5257837E+06	604.50
0.38764	5.8405465E+03	3.5303363E+06	604.45
0.39003	5.8503172E+03	3.5358907E+06	604.39
0.39255	5.8632583E+03	3.5432350E+06	604.31
0.39507	5.8769944E+03	3.5510359E+06	604.23
0.39752	5.8901754E+03	3.5585146E+06	604.14
0.40002	5.9026209E+03	3.5655867E+06	604.07
0.40266	5.9140894E+03	3.5720883E+06	604.00
0.40506	5.9222442E+03	3.5767163E+06	603.95
0.40761	5.9272907E+03	3.5795690E+06	603.91
0.41010	5.9287654E+03	3.5803886E+06	603.90
0.41261	5.9266016E+03	3.5791393E+06	603.91
0.41515	5.9207163E+03	3.5757700E+06	603.94
0.41756	5.9120146E+03	3.5708002E+06	603.99
0.42010	5.9007105E+03	3.5643532E+06	604.05
0.42264	5.8873278E+03	3.5567264E+06	604.13
0.42510	5.8733889E+03	3.5487826E+06	604.21
0.42776	5.8596323E+03	3.5409490E+06	604.30
0.43002	5.8458384E+03	3.5330901E+06	604.38
0.43273	5.8331063E+03	3.5258467E+06	604.45
0.43505	5.8221311E+03	3.5196022E+06	604.52

TABLE 6.2.1-28A (SHEET 5 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.43751	5.8130794E+03	3.5144558E+06	604.58
0.44011	5.8051154E+03	3.5099211E+06	604.63
0.44254	5.7994338E+03	3.5066831E+06	604.66
0.44512	5.7943623E+03	3.5037894E+06	604.69
0.44756	5.7908671E+03	3.5017915E+06	604.71
0.45006	5.7884098E+03	3.5003828E+06	604.72
0.45255	5.7868114E+03	3.4994613E+06	604.73
0.45501	5.7860305E+03	3.4990035E+06	604.73
0.45755	5.7859404E+03	3.4989362E+06	604.73
0.46008	5.7865023E+03	3.4992415E+06	604.72
0.46255	5.7876727E+03	3.4998863E+06	604.71
0.46505	5.7891053E+03	3.5006850E+06	604.70
0.46756	5.7908624E+03	3.5016617E+06	604.69
0.47006	5.7926250E+03	3.5026427E+06	604.67
0.47253	5.7940101E+03	3.5034098E+06	604.66
0.47514	5.7949974E+03	3.5039478E+06	604.65
0.47751	5.7952080E+03	3.5040456E+06	604.65
0.48017	5.7945401E+03	3.5036413E+06	604.65
0.48257	5.7927826E+03	3.5026177E+06	604.65
0.48509	5.7901457E+03	3.5010969E+06	604.66
0.48757	5.7864539E+03	3.4989748E+06	604.68
0.49011	5.7818731E+03	3.4963467E+06	604.71
0.49257	5.7768716E+03	3.4934807E+06	604.74
0.49505	5.7713451E+03	3.4903136E+06	604.77
0.49760	5.7657954E+03	3.4871343E+06	604.80
0.50011	5.7603602E+03	3.4840234E+06	604.83
0.51009	5.7491923E+03	3.4775926E+06	604.88
0.52009	5.7609480E+03	3.4841862E+06	604.79
0.53006	5.7844690E+03	3.4974703E+06	604.63
0.54005	5.7987721E+03	3.5055003E+06	604.52
0.55000	5.7970116E+03	3.5044053E+06	604.52
0.56006	5.7863331E+03	3.4982404E+06	604.57
0.57002	5.7741241E+03	3.4912139E+06	604.63
0.58007	5.7631540E+03	3.4848951E+06	604.69
0.59036	5.7548799E+03	3.4801119E+06	604.72
0.60004	5.7492097E+03	3.4768060E+06	604.75
0.61010	5.7426034E+03	3.4729615E+06	604.77
0.62004	5.7325199E+03	3.4671373E+06	604.82
0.63014	5.7211947E+03	3.4605989E+06	604.87
0.64006	5.7162332E+03	3.4576773E+06	604.89
0.65003	5.7176191E+03	3.4583568E+06	604.86
0.66012	5.7204578E+03	3.4598599E+06	604.82
0.67003	5.7223865E+03	3.4608454E+06	604.79
0.68006	5.7257890E+03	3.4626698E+06	604.75

TABLE 6.2.1-28A (SHEET 6 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
0.69002	5.7317272E+03	3.4659330E+06	604.69
0.70010	5.7366181E+03	3.4686004E+06	604.64
0.71004	5.7366207E+03	3.4684965E+06	604.62
0.72003	5.7320245E+03	3.4657852E+06	604.64
0.73012	5.7251012E+03	3.4617556E+06	604.66
0.74006	5.7172129E+03	3.4571775E+06	604.70
0.75003	5.7094890E+03	3.4526916E+06	604.73
0.76011	5.7035213E+03	3.4491984E+06	604.75
0.77015	5.6997968E+03	3.4469781E+06	604.75
0.78006	5.6960407E+03	3.4447350E+06	604.76
0.79006	5.6906934E+03	3.4415867E+06	604.77
0.80005	5.6856760E+03	3.4386253E+06	604.79
0.81021	5.6845992E+03	3.4379041E+06	604.78
0.82005	5.6883128E+03	3.4399046E+06	604.73
0.83010	5.6942403E+03	3.4431581E+06	604.67
0.84013	5.6985107E+03	3.4454743E+06	604.63
0.85000	5.6994327E+03	3.4458899E+06	604.60
0.86014	5.6970829E+03	3.4444488E+06	604.60
0.87001	5.6929918E+03	3.4420216E+06	604.61
0.88010	5.6890670E+03	3.4396914E+06	604.61
0.89006	5.6860123E+03	3.4378527E+06	604.62
0.90008	5.6830007E+03	3.4360407E+06	604.62
0.91009	5.6783432E+03	3.4332928E+06	604.63
0.92011	5.6715216E+03	3.4293170E+06	604.66
0.93018	5.6647085E+03	3.4253454E+06	604.68
0.94003	5.6629152E+03	3.4242285E+06	604.68
0.95003	5.6673076E+03	3.4266195E+06	604.63
0.96010	5.6722227E+03	3.4293052E+06	604.58
0.97011	5.6735480E+03	3.4299524E+06	604.55
0.98001	5.6715705E+03	3.4287262E+06	604.55
0.99004	5.6708125E+03	3.4281936E+06	604.53
1.00005	5.6717795E+03	3.4286424E+06	604.51
1.05002	5.6542380E+03	3.4181930E+06	604.54
1.10009	5.6578617E+03	3.4197733E+06	604.43
1.15011	5.6427925E+03	3.4107498E+06	604.44
1.20005	5.6426635E+03	3.4102125E+06	604.36
1.25004	5.6325971E+03	3.4040406E+06	604.35
1.30006	5.6245751E+03	3.3990528E+06	604.32
1.35009	5.6163063E+03	3.3939444E+06	604.30
1.40023	5.6056208E+03	3.3874824E+06	604.30
1.45004	5.5990362E+03	3.3833512E+06	604.27
1.50006	5.5888967E+03	3.3772132E+06	604.27
1.55004	5.5798754E+03	3.3717233E+06	604.26
1.60010	5.5703163E+03	3.3659471E+06	604.26

TABLE 6.2.1-28A (SHEET 7 OF 7)

Time (s)	Mass Flow (lb/s)	Energy Flow (Btu/s)	Average Enthalpy (Btu/lb)
1.65004	5.5670531E+03	3.3637635E+06	604.23
1.70008	5.5583218E+03	3.3584925E+06	604.23
1.75012	5.5601180E+03	3.3591994E+06	604.16
1.80010	5.5588433E+03	3.3581573E+06	604.11
1.85006	5.5580249E+03	3.3573797E+06	604.06
1.90001	5.5513019E+03	3.3532687E+06	604.05
1.95012	5.5450010E+03	3.3494269E+06	604.04
2.00000	5.5358334E+03	3.3439853E+06	604.06

TABLE 6.2.1-29

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (s)	(lbm/s)	Thousand (Btu/s)	(lbm/s)	Thousand (Btu/s)
0.0000	0.0	0.0	0.0	0.0
0.0503	42452.7	23717.3	23884.1	13269.7
0.100	42477.6	23781.0	22007.5	12268.7
0.250	43738.3	24766.2	24698.8	13790.4
0.450	45532.2	26371.7	23441.7	13118.1
0.700	44738.7	26657.0	20526.6	11504.7
1.30	37135.2	23390.3	18674.9	10501.9
2.50	26961.5	18469.6	18221.0	10254.8
2.80	22569.4	15708.3	17541.1	9880.5
3.40	19872.4	13967.1	16413.6	9263.4
4.30	18259.2	12735.7	14784.9	8363.0
5.25	17440.3	11908.7	13610.2	7705.3
5.50	17631.0	11941.9	14324.4	8111.6
5.75	17013.2	11406.1	14341.5	8120.9
6.25	13138.8	9669.6	13750.2	7783.1
7.00	17339.3	12063.4	13198.4	7473.1
7.75	19110.4	12856.5	12481.5	7069.1
8.25	25395.3	16695.2	11754.4	6654.1
9.00	25795.2	16541.7	10565.4	5976.8
9.50	25151.6	15997.0	9857.1	5568.7
10.0	23378.7	14868.0	9265.5	5221.3
10.3	11791.1	7528.6	9398.9	5305.8
10.5	8184.8	5542.7	9303.3	5279.1
10.8	7267.3	5102.7	9492.4	5392.7
11.3	8748.9	5693.9	9678.6	5491.4
11.5	8904.4	6167.3	9314.0	5277.8
11.8	6683.6	5609.3	9465.3	5376.3
13.0	6218.1	5121.7	8558.9	4905.3
15.0	4748.8	4248.4	6899.7	3906.8
16.0	3841.2	3967.5	6259.4	3268.9
17.8	1792.5	2239.4	3863.0	1754.3
18.5	1391.3	1750.4	2980.7	1162.3
19.3	884.7	1118.0	3387.0	1209.2
20.3	286.0	363.9	722.3	259.2
22.0	0.0	0.0	0.0	0.0

TABLE 6.2.1-30

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED HOT LEG GUILLOTINE

Time	Break Path No. 1 Flow		Break Path No. 2 Flow	
		Thousand		Thousand
(s)	(lbm/s)	(Btu/s)	(lbm/s)	(Btu/s)
0.0000	0.0	0.0	0.0	0.0
0.0507	38706.4	25536.9	28739.9	18622.6
0.100	42320.7	27947.5	28199.4	18281.1
0.200	36861.1	24360.0	23814.9	15352.6
0.350	35128.6	23213.6	20763.4	13095.8
0.750	33808.0	22364.6	18400.5	10939.9
1.30	31743.0	21475.4	17941.3	10257.0
2.10	28464.4	19660.5	18631.7	10405.6
3.20	24775.4	17144.8	18081.1	10086.5
4.20	23104.0	15666.3	16889.2	9499.1
4.70	23182.3	15399.6	16063.0	9078.4
5.00	23877.7	15568.4	15239.8	8640.2
5.25	17131.5	12528.8	14889.5	8469.0
6.00	18769.3	12994.2	13227.6	7574.8
6.50	20547.3	13761.0	12435.3	7146.3
6.75	27819.8	18329.9	12036.4	6926.3
7.00	29507.8	19265.7	11669.9	6724.5
7.25	29030.5	18672.1	11298.0	6518.5
7.75	29436.5	18610.1	10299.7	5966.2
8.75	28906.2	18096.5	8432.4	4957.7
9.25	27483.4	17015.2	7583.8	4509.8
9.50	18993.4	11798.8	7184.9	4303.1
9.75	16879.4	10322.2	6812.6	4112.8
10.0	9611.9	7793.2	6495.8	3954.9
10.3	10818.6	7971.5	6269.8	3844.4
10.8	10674.0	7894.7	5971.6	3685.0
12.5	10937.3	7822.4	4928.9	3150.3
13.0	8893.1	6668.0	4565.7	2976.4
14.0	7453.4	6124.6	3710.3	2579.7
14.8	5067.7	4829.4	2957.8	2212.7
17.5	1500.8	1780.3	1361.6	1486.4
18.5	817.9	1017.8	1091.9	1296.4
21.0	76.3	99.2	96.8	122.3
24.3	630.3	776.0	252.8	318.8
25.0	0.0	0.0	0.0	0.0

TABLE 6.2.1-31

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MINIMUM SAFETY INJECTION

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (s)	(lbm/s)	Thousand (Btu/s)	(lbm/s)	Thousand (Btu/s)
22.0	0.0	0.0	0.0	0.0
23.0	0.0	0.0	0.0	0.0
23.1	89.1	105.2	0.0	0.0
23.4	22.5	26.6	0.0	0.0
24.5	78.5	92.6	0.0	0.0
27.1	147.3	173.9	0.0	0.0
28.1	195.5	230.9	1135.8	166.3
28.8	456.0	540.8	4373.5	687.1
29.1	471.9	559.8	4495.8	721.1
30.1	470.7	558.5	4480.6	727.1
31.1	462.4	548.5	4404.8	718.4
35.1	427.9	507.3	4081.8	678.3
39.1	397.7	471.3	3787.7	640.7
43.1	372.0	440.6	3529.2	607.4
44.1	397.3	470.8	3810.3	638.0
46.1	386.4	457.7	3701.2	623.8
47.1	381.2	451.5	3648.9	617.0
49.1	371.3	439.8	3548.6	603.8
53.1	353.3	418.3	3363.3	579.5
54.1	297.6	352.1	2675.8	502.8
54.3	292.6	346.1	2674.8	495.6
55.1	511.4	607.1	347.3	288.1
56.1	516.2	612.9	349.2	291.2
60.1	456.5	541.5	322.0	254.7
64.1	406.8	482.1	299.3	224.9
68.1	364.0	431.1	280.0	199.7
69.1	354.3	419.5	275.7	194.0
77.1	288.6	341.4	246.5	156.3
85.1	241.0	284.8	225.8	129.9
99.1	189.8	224.2	204.0	102.6
111.1	167.9	198.2	194.9	91.3
125.1	156.7	185.0	190.3	85.6
141.1	153.4	181.0	188.8	83.7
167.1	155.3	183.4	189.2	84.2
225.8	165.4	195.3	192.5	88.2

TABLE 6.2.1-32

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SAFETY INJECTION

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (s)	(lbm/s)	Thousand (Btu/s)	(lbm/s)	Thousand (Btu/s)
22.0	0.0	0.0	0.0	0.0
23.0	0.0	0.0	0.0	0.0
23.1	89.1	105.2	0.0	0.0
23.4	22.5	26.6	0.0	0.0
24.5	78.5	92.6	0.0	0.0
27.1	147.3	173.9	0.0	0.0
28.1	195.5	230.9	1135.8	166.3
28.8	456.0	540.8	4373.5	687.1
29.1	471.9	559.8	4495.8	721.1
30.1	470.7	558.5	4480.6	727.1
31.1	462.4	548.5	4404.8	718.4
35.1	427.9	507.3	4081.8	678.3
39.1	397.7	471.3	3787.7	640.7
43.1	372.0	440.6	3529.2	607.4
44.1	435.3	516.2	4209.5	683.2
46.1	424.4	503.1	4104.6	669.4
50.1	404.7	479.5	3911.0	643.8
54.1	387.2	458.7	3736.3	620.6
55.2	367.4	435.1	474.6	196.0
56.2	370.6	438.9	467.4	196.4
58.2	368.9	436.9	471.6	196.2
66.2	361.9	428.6	488.5	195.5
81.2	346.5	410.2	524.2	194.9
87.2	338.7	400.9	541.0	195.1
101.2	315.6	373.4	586.5	196.4
109.2	298.7	353.4	617.3	198.0
117.2	278.8	329.7	652.2	200.6
125.2	255.1	301.6	691.9	204.4
127.2	248.5	293.8	702.6	205.6
135.2	218.8	258.6	750.0	211.4
143.2	181.4	214.3	807.3	219.8
145.2	170.3	201.1	823.9	222.6
147.2	165.9	195.8	831.1	222.9
157.2	161.5	190.7	840.8	220.6
189.2	148.1	174.8	870.9	213.9

TABLE 6.2.1-33

PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MINIMUM SAFETY INJECTION

Time (s)	Flooding Temp (°F)	Flooding Rate (in./s)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection Total (lbm/s)	Injection Accumulator (lbm/s)	Injection Spill (lbm/s)	Enthalpy (Btu/lbm)
22.0	240.8	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
22.8	236.2	21.570	0.000	0.62	1.50	0.000	7488.3	7488.3	0.0	99.54
23.0	233.1	23.702	0.000	1.10	1.37	0.000	7406.8	7406.8	0.0	99.54
23.4	231.8	2.948	0.130	1.34	2.00	0.217	7305.1	7305.1	0.0	99.54
23.5	231.7	2.968	0.151	1.36	2.30	0.243	7279.5	7279.5	0.0	99.54
24.3	231.2	2.596	0.312	1.50	4.76	0.328	7065.0	7065.0	0.0	99.54
25.1	230.8	2.520	0.414	1.61	7.06	0.347	6888.4	6888.4	0.0	99.54
28.8	229.3	4.517	0.643	2.00	16.05	0.587	5728.4	5728.4	0.0	99.54
29.1	229.1	4.542	0.654	2.04	16.06	0.586	5632.9	5632.9	0.0	99.54
30.1	228.5	4.349	0.681	2.16	16.07	0.584	5468.5	5468.5	0.0	99.54
33.5	227.0	3.903	0.722	2.50	16.07	0.577	5052.1	5052.1	0.0	99.54
39.7	226.1	3.476	0.745	3.00	16.07	0.561	4467.3	4467.3	0.0	99.54
43.1	228.2	3.315	0.749	3.25	16.07	0.552	4206.3	4206.3	0.0	99.54
44.1	226.3	3.450	0.751	3.32	16.07	0.566	4523.8	4042.5	0.0	99.37
46.7	226.5	3.349	0.753	3.50	16.07	0.561	4356.7	3872.3	0.0	99.36
54.3	228.0	2.807	0.754	4.00	16.07	0.510	3214.7	2712.7	0.0	99.29
55.1	228.3	4.027	0.759	4.06	15.85	0.623	453.4	0.0	0.0	97.96
61.1	230.6	3.488	0.759	4.52	13.91	0.617	468.2	0.0	0.0	97.96
68.7	235.2	2.874	0.756	5.00	12.19	0.607	485.5	0.0	0.0	97.96
78.1	241.9	2.331	0.753	5.50	10.87	0.592	498.6	0.0	0.0	97.96
89.6	249.9	1.901	0.749	6.00	10.09	0.572	507.2	0.0	0.0	97.96
105.1	258.2	1.594	0.746	6.57	9.87	0.549	511.9	0.0	0.0	97.96
118.5	263.9	1.476	0.746	7.00	10.07	0.538	513.5	0.0	0.0	97.96
135.1	269.8	1.421	0.750	7.50	10.53	0.533	514.1	0.0	0.0	97.96
152.2	275.0	1.408	0.756	8.00	11.07	0.533	514.2	0.0	0.0	97.96
157.1	276.4	1.407	0.758	8.14	11.24	0.534	514.2	0.0	0.0	97.96
171.1	280.0	1.409	0.763	8.53	11.70	0.536	514.1	0.0	0.0	97.96
188.0	283.8	1.416	0.769	9.00	12.26	0.539	513.9	0.0	0.0	97.96
207.1	287.5	1.423	0.777	9.51	12.87	0.542	513.6	0.0	0.0	97.96
225.8	290.6	1.431	0.784	10.00	13.46	0.545	513.4	0.0	0.0	97.96

TABLE 6.2.1-34

PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SAFETY INJECTION

Time (s)	Flooding Temp (°F)	Flooding Rate (in./s)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection Total (lbm/s)	Injection Accumulator (lbm/s)	Injection Spill (lbm/s)	Enthalpy (Btu/lbm)
22.0	240.8	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
22.8	236.2	21.570	0.000	0.62	1.50	0.000	7488.3	7488.3	0.0	99.54
23.0	233.1	23.702	0.000	1.10	1.37	0.000	7406.8	7406.8	0.0	99.54
23.4	231.8	2.948	0.130	1.34	2.00	0.217	7305.1	7305.1	0.0	99.54
23.5	231.7	2.968	0.151	1.36	2.30	0.243	7279.5	7279.5	0.0	99.54
24.3	231.2	2.596	0.312	1.50	4.76	0.328	7065.0	7065.0	0.0	99.54
25.1	230.8	2.520	0.414	1.61	7.06	0.347	6888.4	6888.4	0.0	99.54
28.8	229.3	4.517	0.643	2.00	16.05	0.587	5728.4	5728.4	0.0	99.54
29.1	229.1	4.542	0.654	2.04	16.06	0.586	5632.9	5632.9	0.0	99.54
30.1	228.5	4.349	0.681	2.16	16.07	0.584	5468.5	5468.5	0.0	99.54
33.5	227.0	3.903	0.722	2.50	16.07	0.577	5052.1	5052.1	0.0	99.54
39.7	226.1	3.476	0.745	3.00	16.07	0.561	4467.3	4467.3	0.0	99.54
43.1	226.2	3.315	0.749	3.25	16.07	0.552	4206.3	4206.3	0.0	99.54
44.1	226.2	3.666	0.752	3.32	16.07	0.583	4981.3	3925.1	0.0	99.20
46.5	226.5	3.571	0.754	3.50	16.07	0.579	4831.0	3770.9	0.0	99.19
53.6	227.7	3.340	0.757	4.00	16.07	0.568	4444.2	3374.7	0.0	99.16
55.2	228.1	3.232	0.758	4.10	16.07	0.557	1076.5	0.0	0.0	97.96
56.2	228.3	3.238	0.758	4.17	16.07	0.557	1074.8	0.0	0.0	97.96
62.2	231.0	3.182	0.760	4.55	16.07	0.558	1076.1	0.0	0.0	97.96
69.4	235.8	3.114	0.763	5.00	16.07	0.558	1077.7	0.0	0.0	97.96
78.2	243.2	3.023	0.766	5.54	16.07	0.558	1079.9	0.0	0.0	97.96
86.3	250.4	2.929	0.769	6.00	16.07	0.557	1082.3	0.0	0.0	97.96
97.2	258.7	2.785	0.773	6.60	16.07	0.554	1086.2	0.0	0.0	97.96
105.1	263.7	2.668	0.775	7.00	16.07	0.550	1089.6	0.0	0.0	97.96
117.2	270.1	2.462	0.778	7.58	16.07	0.537	1095.9	0.0	0.0	97.96
126.8	274.3	2.273	0.780	8.00	16.07	0.520	1101.7	0.0	0.0	97.96
141.2	279.4	1.924	0.781	8.56	16.07	0.471	1111.6	0.0	0.0	97.96
154.8	283.2	1.737	0.783	9.00	16.07	0.441	1115.6	0.0	0.0	97.96
173.2	287.4	1.643	0.789	9.56	16.07	0.441	1115.9	0.0	0.0	97.96
189.2	290.4	1.564	0.794	10.00	16.07	0.441	1116.2	0.0	0.0	97.96

TABLE 6.2.1-35

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MINIMUM SAFETY INJECTION

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (s)	(lbm/s)	Thousand (Btu/s)	(lbm/s)	Thousand (Btu/s)
225.9	238.4	298.4	281.8	134.5
260.9	237.1	296.8	283.0	133.4
295.9	234.4	293.4	285.8	132.6
300.9	234.8	293.9	285.4	132.3
310.9	233.6	292.4	286.6	132.2
315.9	233.8	292.7	286.4	131.9
325.9	233.2	291.9	287.0	131.6
330.9	232.4	290.9	287.8	131.7
340.9	232.3	290.8	287.9	131.2
360.9	230.3	288.2	289.9	130.9
365.9	230.6	288.6	289.6	130.6
375.9	230.0	287.9	290.2	130.3
290.9	228.4	285.9	291.8	130.1
395.9	228.6	286.1	291.6	129.8
425.9	226.0	282.9	294.2	129.2
430.9	226.0	282.9	294.2	129.0
465.9	223.6	279.9	296.5	128.1
615.9	223.4	279.6	296.8	128.0
616.0	98.7	122.6	421.5	160.9
645.9	97.9	121.6	422.3	159.6
795.9	93.7	116.3	426.5	157.6
800.9	93.5	116.2	426.6	157.3
945.9	90.3	112.1	429.9	153.5
950.9	90.2	112.0	430.0	153.1
160.9	86.5	107.4	433.7	152.6
1280.9	84.6	104.9	435.6	148.6
1305.9	84.2	104.4	436.0	148.0
1425.9	82.2	102.0	438.0	148.8
1570.9	82.1	101.9	438.1	147.0
1571.0	79.0	90.9	441.2	58.4
3600.0	64.0	73.7	456.2	61.1
3600.1	50.7	58.4	469.5	46.0
3952.2	50.7	58.4	469.5	46.0
3952.3	52.3	60.3	462.3	61.6
9999.9	52.3	60.3	462.3	61.6
10000.0	38.2	43.9	476.5	63.5
100000.0	20.4	23.5	494.3	65.9
1000000.0	8.7	10.1	505.9	67.4

TABLE 6.2.1-36

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SAFETY INJECTION

Time (s)	Break Path No. 1 Flow (lbm/s)	Break Path No. 1 Flow Thousand (Btu/s)	Break Path No. 2 Flow (lbm/s)	Break Path No. 2 Flow Thousand (Btu/s)
189.2	162.3	202.1	957.3	218.8
199.2	161.7	201.3	957.9	218.5
224.2	161.9	201.5	957.7	217.2
259.2	161.0	200.4	958.7	215.9
274.2	161.1	200.5	958.5	215.1
289.2	160.3	199.6	959.3	214.6
299.2	160.9	200.3	958.7	214.0
334.2	159.4	198.4	960.3	212.7
344.2	159.8	198.9	959.8	212.1
359.2	159.6	198.6	960.1	211.4
374.2	158.4	197.2	961.2	211.0
389.2	158.8	197.7	960.8	210.1
409.2	157.9	196.5	961.8	209.4
419.2	158.3	197.0	961.4	208.8
439.2	157.3	195.9	962.3	213.2
459.2	157.7	196.3	961.9	212.1
509.2	156.4	194.6	963.3	209.7
514.2	156.8	195.2	962.8	209.3
559.2	155.6	193.7	964.0	207.1
569.2	155.9	194.1	963.7	206.5
639.2	154.6	192.5	965.0	207.6
919.2	154.6	192.5	965.0	207.6
919.3	91.9	113.5	1027.7	217.7
924.2	91.8	113.3	1027.8	217.3
999.2	90.1	111.3	1029.5	214.3
1064.2	89.1	109.9	1030.6	214.5
1219.2	86.6	106.8	1033.1	214.5
1299.2	85.3	105.2	1034.3	212.9
1430.6	85.3	105.2	1034.3	212.9
1430.7	81.9	94.2	1037.7	115.2
2397.3	81.9	94.2	1037.7	115.2
2397.4	84.8	97.6	1016.2	170.8
3599.0	84.8	97.6	1016.2	170.8
3600.0	65.0	74.8	1036.0	174.1
3600.1	53.4	61.4	1047.7	157.5
10000.0	38.8	44.7	1062.2	159.6
100000.0	20.7	23.9	1080.3	162.4
1000000.0	8.9	10.2	1092.1	164.1

TABLE 6.2.1-37

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MINIMUM SAFETY INJECTION

Time (s)	0.00	22.00	22.00	225.84	620.90	1570.94	3600.00
				Mass (Thousand lbm)			
Initial Mass (In RCS and ACC)	735.26	735.26	735.26	735.26	735.26	735.26	735.26
Added Mass (Pumped Injection)	0.00	0.00	0.00	92.43	297.90	791.36	1835.65
Total Available	735.26	735.26	735.26	827.68	1033.16	1526.62	2570.91
Distribution							
• Reactor Coolant	513.22	48.53	67.98	125.36	125.36	125.36	125.36
• Accumulator	222.04	176.45	156.99	0.00	0.00	0.00	0.00
Effluent							
• Break Flow	0.00	510.28	510.28	691.11	896.58	1390.04	2434.33
• ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Total Accountable	735.26	735.25	735.25	816.46	1021.94	1515.40	2559.69

TABLE 6.2.1-38

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SAFETY INJECTION

	Time (s)	0.00	22.00	22.00	189.19	924.20	1430.60	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	735.26	735.26	735.26	735.26	735.26	735.26	735.26
Added Mass	Pumped Injection	0.00	0.00	0.00	159.76	982.69	1547.31	3935.89
	Total Added	0.00	0.00	0.00	159.76	982.69	1547.31	3935.89
Total Available		735.26	735.26	735.26	895.02	1717.95	2282.57	4671.15
Distribution	Reactor Coolant	513.22	48.53	67.98	130.18	130.18	130.18	130.18
	Accumulator	222.04	176.45	156.99	0.00	0.00	0.00	0.00
	Total Contents	735.26	224.97	224.97	130.18	130.18	130.18	130.18
Effluent	Break Flow	0.00	510.28	510.28	753.61	1576.54	2141.16	4529.74
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	510.28	510.28	753.61	1576.54	2141.16	4529.74
Total Accountable		735.26	735.25	735.25	883.80	1706.72	2271.35	4659.93

TABLE 6.2.1-39

MASS BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	Time (s)	0.00	25.00	25.00
			Mass (Thousand lbm)	
Initial	In RCS and ACC	735.26	735.26	735.26
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
Total Available		735.26	735.26	735.26
Distribution	Reactor Coolant	513.22	93.74	113.19
	Accumulator	222.04	144.38	124.93
	Total Contents	735.26	238.12	238.12
Effluent	Break Flow	0.00	497.13	497.13
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	497.13	497.13
Total Accountable		735.26	735.25	735.25

TABLE 6.2.1-40

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MINIMUM SAFETY INJECTION

	Time (s)	0.00	22.00	22.00	225.84	620.90	1570.94	3600.00
				Energy (Million Btu)				
Initial Energy	In RCS, ACC, S Gen	898.43	898.43	898.43	898.43	898.43	898.43	898.43
Added Energy	Pumped Injection	0.00	0.00	0.00	9.05	29.18	79.98	219.17
	Decay Heat	0.00	8.68	8.68	33.02	70.83	142.86	263.30
	Heat from Secondary	0.00	-1.78	-1.78	-1.78	0.92	6.51	6.51
	Total Added	0.00	6.90	6.90	40.89	100.94	229.35	488.98
Total Available		898.43	905.33	905.33	939.32	999.37	1127.78	1387.41
Distribution	Reactor Coolant	309.49	13.70	15.63	34.74	34.74	34.74	34.74
	Accumulator	22.10	17.56	15.63	0.00	0.00	0.00	0.00
	Core Stored	25.04	13.26	13.26	4.95	4.75	4.29	3.33
	Primary Metal	154.32	146.70	146.70	120.18	88.47	64.63	49.37
	Secondary Metal	114.85	114.42	114.42	105.35	85.36	57.36	43.51
	Steam Generator	272.63	277.93	277.93	251.64	200.58	135.68	103.99
	Total Contents	898.43	583.57	583.57	516.86	413.89	296.70	234.93
Effluent	Break Flow	0.00	321.76	321.76	410.78	573.79	819.39	1140.79
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	321.76	321.76	410.78	573.79	819.39	1140.79
Total Accountable		898.43	905.33	905.33	927.63	987.68	1116.09	1375.72

TABLE 6.2.1-41

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MAXIMUM SAFETY INJECTION

	Time (s)	0.00	22.00	22.00	189.19	924.20	1430.60	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, S Gen	898.43	898.43	898.43	898.43	898.43	898.43	898.43
Added Energy	Pumped Injection	0.00	0.00	0.00	15.65	96.26	158.86	517.84
	Decay Heat	0.00	8.68	8.68	29.71	95.70	133.17	263.31
	Heat From Secondary	0.00	-1.78	-1.78	-1.78	3.25	5.86	5.86
	Total Added	0.00	6.90	6.90	43.58	195.22	297.88	787.01
Total Available		898.43	905.33	905.33	942.00	1093.65	1196.31	1685.44
Distribution	Reactor Coolant	309.49	13.70	15.63	36.09	36.09	36.09	36.09
	Accumulator	22.10	17.56	15.63	0.00	0.00	0.00	0.00
	Core Stored	25.04	13.26	13.26	4.95	4.75	4.48	3.33
	Primary Metal	154.32	146.70	146.70	119.02	79.46	65.95	49.38
	Secondary Metal	114.85	114.42	114.42	103.30	73.72	57.94	43.50
	Steam Generator	272.63	277.93	277.93	245.74	171.66	136.25	103.29
	Total Contents	898.43	583.57	583.57	509.09	365.67	300.71	235.59
Effluent	Break Flow	0.00	321.76	321.76	421.23	716.30	883.92	1438.17
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	321.76	321.76	421.23	716.30	883.92	1438.17
Total Accountable		898.43	905.33	905.33	930.32	1081.97	1184.63	1673.76

TABLE 6.2.1-42

ENERGY BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	Time (s)	0.00	25.00	25.00
		Energy (Million Btu)		
Initial Energy	In RCS, ACC, S Gen	898.43	898.43	898.43
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	9.63	9.63
	Heat From Secondary	0.00	-4.06	-4.06
	Total Added	0.00	5.57	5.57
Total Available		898.43	904.00	904.00
Distribution	Reactor Coolant	309.49	22.68	24.62
	Accumulator	22.10	14.37	12.44
	Core Stored	25.04	10.07	10.07
	Primary Metal	154.32	144.41	144.41
	Secondary Metal	114.85	112.08	112.08
	Steam Generator	272.63	270.80	270.80
	Total Contents	898.43	574.41	574.41
Effluent	Break Flow	0.00	329.59	329.59
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	329.59	329.59
Total Accountable		898.43	904.00	904.00

TABLE 6.2.1-43

SYSTEM PARAMETERS

PARAMETER	VALUE
Core inlet temperature (includes +6.0°F allowance for instrument error and deadband)	562.8°F
Core outlet temperature (includes +6.0°F allowance for instrument error and deadband)	631.04°F
Initial steam generator steam pressure	969.6 psia
Assumed maximum containment back pressure	66.7 psia
Pumped injection, assumed for FROTH:	
Minimum (lb/s)	514.67
Maximum (lb/s)	1101.03

TABLE 6.2.1-44

THROUGH

TABLE 6.2.1-59

DELETED

TABLE 6.2.1-60

SPECTRUM OF SECONDARY SYSTEM PIPE RUPTURES ANALYZED[a]

Case	Size	Type Of Rupture	% Power[b]
1	Full	Double-ended	102
2	Full	Double-ended	70
3	Full	Double-ended	30
4	Full	Double-ended	0
5	0.60 ft^2	Double-ended with entrainment	102
6	0.53 ft^2	Double-ended with entrainment	70
7	0.36 ft^2	Double-ended with entrainment	30
8	0.20 ft^2	Double-ended with entrainment	0
9	0.33 ft^2	Double-ended without entrainment	102
10	0.32 ft^2	Double-ended without entrainment	70
11	0.22 ft^2	Double-ended without entrainment	30
12	0.10 ft^2	Double-ended without entrainment	0
13	0.86 ft^2	Split rupture	102
14	0.908 ft^2	Split rupture	70
15	0.944 ft^2	Split rupture	30
16	0.40 ft^2	Split rupture	0

a. Case 1 through case 16 have assumed the loss of a diesel.
b. % power of 3579 MWt.

TABLE 6.2.1-61 (SHEET 1 OF 9)

MASS AND ENERGY RELEASE DATA FOR CASE 16 - PEAK CALCULATED CONTAINMENT PRESSURE FOR MSLB

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
0.00	0.00	0.000	9.40	843.43	1.005	18.60	799.79	0.954
0.20	906.37	1.077	9.60	842.21	1.003	18.80	798.43	0.953
0.40	904.86	1.075	9.80	841.85	1.003	19.00	797.53	0.952
0.60	903.06	1.073	10.00	841.43	1.002	19.20	797.05	0.951
0.80	901.31	1.071	10.20	839.56	1.000	19.40	795.86	0.950
1.00	899.95	1.070	10.40	838.11	0.999	19.60	795.70	0.950
1.20	898.72	1.068	10.60	836.90	0.997	19.80	794.60	0.948
1.40	897.06	1.066	10.80	836.71	0.997	20.00	793.65	0.947
1.60	894.80	1.064	11.00	836.34	0.997	20.20	792.95	0.946
1.80	892.96	1.062	11.20	834.53	0.994	20.40	792.33	0.946
2.00	892.40	1.061	11.40	833.05	0.993	20.60	791.61	0.945
2.20	890.84	1.059	11.60	831.83	0.991	20.80	791.36	0.945
2.40	888.62	1.057	11.80	831.34	0.991	21.00	791.16	0.944
2.60	886.71	1.054	12.00	829.79	0.989	21.20	790.40	0.943
2.80	885.95	1.054	12.20	828.86	0.988	21.40	789.33	0.942
3.00	884.39	1.052	12.40	827.84	0.987	21.60	788.93	0.942
3.20	882.89	1.050	13.00	824.87	0.983	21.80	788.56	0.941
3.40	880.80	1.048	13.20	823.95	0.982	22.00	787.66	0.940
3.60	879.46	1.046	13.40	822.96	0.981	22.20	787.82	0.940
3.80	878.70	1.045	13.60	822.04	0.980	22.40	786.59	0.939
4.00	877.29	1.044	13.80	821.04	0.979	22.60	786.22	0.939
4.20	875.24	1.041	14.00	820.14	0.978	22.80	786.48	0.939
4.40	873.61	1.039	14.20	819.17	0.977	23.00	785.80	0.938
4.80	870.88	1.036	14.40	818.88	0.976	23.20	785.75	0.938
5.00	869.46	1.035	14.60	817.85	0.975	23.40	784.42	0.936
5.20	869.10	1.034	14.80	816.40	0.973	23.60	783.92	0.936
5.40	867.83	1.033	15.00	815.61	0.973	24.00	783.12	0.935
5.60	865.86	1.030	15.20	815.16	0.972	24.20	782.74	0.934
5.80	864.34	1.029	15.40	814.13	0.971	24.80	781.65	0.933
6.20	861.82	1.026	15.60	812.69	0.969	25.40	780.51	0.932
6.40	860.80	1.025	15.80	811.88	0.968	26.00	779.34	0.930
6.60	860.07	1.024	16.00	811.45	0.968	26.40	778.53	0.930
6.80	858.87	1.022	16.20	810.44	0.967	27.00	777.29	0.928
7.00	857.05	1.020	16.40	809.04	0.965	27.60	776.01	0.927
7.20	855.98	1.019	16.60	808.20	0.964	28.20	774.70	0.925
7.40	855.36	1.018	16.80	807.86	0.964	28.80	773.36	0.924
7.60	854.30	1.017	17.00	806.87	0.962	29.80	771.07	0.921
7.80	853.01	1.016	17.20	805.45	0.961	30.40	769.67	0.919
8.00	851.60	1.014	17.40	804.57	0.960	31.40	767.32	0.917
8.20	850.38	1.013	17.60	804.24	0.959	32.60	764.48	0.913

TABLE 6.2.1-61 (SHEET 2 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
8.40	848.89	1.011	17.80	803.29	0.958	34.80	759.23	0.907
8.60	847.83	1.010	18.00	801.91	0.957	39.00	749.16	0.895
8.80	847.27	1.009	18.20	801.00	0.956	41.20	743.90	0.889
9.00	845.58	1.007	18.40	800.77	0.955	43.40	738.67	0.883
45.60	733.49	0.877	116.20	578.27	0.695	128.40	506.00	0.609
47.80	728.34	0.871	116.40	576.70	0.693	128.80	504.21	0.607
50.00	723.23	0.865	116.60	575.15	0.691	129.20	502.44	0.604
52.00	718.64	0.860	116.80	573.61	0.689	129.60	500.70	0.602
54.20	713.63	0.854	117.00	572.09	0.687	129.80	499.84	0.601
56.40	708.68	0.848	117.20	570.59	0.685	130.20	498.13	0.599
58.60	703.79	0.842	117.40	569.10	0.684	130.60	496.45	0.597
60.80	698.94	0.837	117.60	567.63	0.682	130.80	495.62	0.596
62.60	695.01	0.832	117.80	566.17	0.680	131.20	493.98	0.594
62.80	694.74	0.832	118.00	564.73	0.679	131.60	492.37	0.592
65.00	689.99	0.826	118.20	563.31	0.677	131.80	491.57	0.591
67.20	685.30	0.821	118.40	561.90	0.675	132.20	489.99	0.590
69.40	680.65	0.815	118.60	560.50	0.674	132.60	488.43	0.588
71.60	676.05	0.810	118.80	559.12	0.672	132.80	487.66	0.587
73.80	671.49	0.805	119.00	557.76	0.670	133.20	486.14	0.585
74.80	669.43	0.802	119.20	556.40	0.669	133.40	485.39	0.584
76.00	666.98	0.799	119.40	555.07	0.667	133.80	483.90	0.582
78.20	662.52	0.794	119.60	553.74	0.666	134.20	482.44	0.581
80.40	658.10	0.789	119.80	552.43	0.664	134.40	481.71	0.580
82.40	654.13	0.784	120.00	551.14	0.662	134.80	480.28	0.578
84.60	649.79	0.779	120.20	549.85	0.661	135.20	478.87	0.576
86.80	645.51	0.774	120.40	548.58	0.659	135.60	477.48	0.575
89.00	641.26	0.769	120.60	547.33	0.658	136.00	476.11	0.573
91.20	637.06	0.764	120.80	546.08	0.656	136.40	474.76	0.571
93.40	632.91	0.759	121.00	544.85	0.655	136.80	473.43	0.570
95.40	629.16	0.755	121.20	543.63	0.653	137.20	472.12	0.568
97.60	625.09	0.750	121.40	542.42	0.652	137.40	471.47	0.567
99.80	621.06	0.745	121.60	541.22	0.651	137.80	470.19	0.566
102.00	617.06	0.740	121.80	540.04	0.649	138.40	468.31	0.564
104.20	613.12	0.736	122.20	537.70	0.646	138.80	467.07	0.562
106.40	609.21	0.731	122.60	535.40	0.644	139.40	465.25	0.560
108.40	605.70	0.727	123.00	533.14	0.641	139.80	464.06	0.559
110.60	601.89	0.723	123.40	530.92	0.638	140.40	462.31	0.557
111.60	600.17	0.720	123.60	529.83	0.637	141.00	460.60	0.554
112.80	598.12	0.718	124.00	527.67	0.635	141.60	458.92	0.552
114.00	596.09	0.716	124.40	525.54	0.632	142.20	457.28	0.550
114.20	595.62	0.715	124.80	523.45	0.629	142.60	456.20	0.549
114.40	593.24	0.712	125.20	521.39	0.627	143.20	454.62	0.547
114.60	591.49	0.710	125.40	520.37	0.626	143.60	453.59	0.546

TABLE 6.2.1-61 (SHEET 3 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
114.80	589.76	0.708	125.80	518.36	0.623	144.20	452.06	0.544
115.00	588.06	0.706	126.20	516.38	0.621	144.60	451.06	0.543
115.20	586.38	0.704	126.40	515.40	0.620	145.20	449.59	0.541
115.40	584.72	0.702	126.80	513.46	0.618	145.60	448.63	0.540
115.60	583.08	0.700	127.20	511.55	0.615	146.20	447.21	0.538
115.80	581.46	0.698	127.60	509.68	0.613	146.80	445.82	0.537
116.00	579.85	0.696	128.00	507.83	0.611			
147.40	444.46	0.535	172.20	407.41	0.491	218.80	387.01	0.466
147.80	443.57	0.534	172.80	406.86	0.490	221.00	386.69	0.466
148.40	442.26	0.532	173.40	406.32	0.489	223.20	386.41	0.465
149.00	440.97	0.531	174.40	405.44	0.488	225.40	386.15	0.465
149.60	439.71	0.529	175.40	404.60	0.487	227.40	385.93	0.465
150.20	438.48	0.528	176.40	403.79	0.486	229.60	385.72	0.465
150.80	437.28	0.527	177.00	403.32	0.486	231.80	385.52	0.464
151.40	436.10	0.525	177.60	402.85	0.485	234.00	385.34	0.464
151.80	435.33	0.524	178.20	402.40	0.485	236.20	385.19	0.464
152.40	434.19	0.523	178.80	401.96	0.484	238.40	385.04	0.464
152.80	433.45	0.522	179.80	401.25	0.483	240.40	384.93	0.464
153.40	432.35	0.521	180.80	400.56	0.482	242.60	384.81	0.463
154.00	431.27	0.519	181.40	400.16	0.482	244.80	384.71	0.463
154.60	430.22	0.518	182.00	399.77	0.481	247.00	384.62	0.463
155.20	429.19	0.517	182.60	399.38	0.481	249.20	384.54	0.463
155.60	428.52	0.516	183.20	399.01	0.481	251.40	384.47	0.463
156.20	427.52	0.515	184.20	398.40	0.480	255.60	384.36	0.463
156.60	426.87	0.514	185.20	397.82	0.479	260.00	384.28	0.463
157.20	425.91	0.513	186.40	397.15	0.478	264.20	384.22	0.463
157.60	425.28	0.512	187.40	396.62	0.478	268.60	384.18	0.463
158.20	424.35	0.511	188.40	396.10	0.477	273.00	384.16	0.463
158.60	423.75	0.510	189.40	395.61	0.476	277.60	384.15	0.463
159.20	422.85	0.509	190.60	395.03	0.476	283.80	384.16	0.463
159.80	421.98	0.508	191.80	394.49	0.475	290.00	384.19	0.463
160.40	421.12	0.507	192.80	394.05	0.475	296.00	384.23	0.463
160.80	420.55	0.506	193.80	393.64	0.474	302.20	384.29	0.463
161.40	419.72	0.505	195.00	393.15	0.474	314.60	384.41	0.463
162.00	418.90	0.504	196.20	392.69	0.473	376.40	385.10	0.464
162.60	418.11	0.504	197.20	392.33	0.473	401.00	385.36	0.464
163.20	417.32	0.503	198.60	391.84	0.472	425.80	385.61	0.464
163.80	416.56	0.502	198.80	391.65	0.472	592.20	387.24	0.466
164.40	415.81	0.501	200.40	391.13	0.471	611.40	387.41	0.467
164.80	415.32	0.500	201.40	390.82	0.471	623.80	387.51	0.467
165.40	414.60	0.499	202.40	390.52	0.470	636.00	387.59	0.467
165.80	414.13	0.499	203.60	390.18	0.470	648.40	387.66	0.467

TABLE 6.2.1-61 (SHEET 4 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
166.40	413.43	0.498	204.80	389.85	0.470	654.60	387.69	0.467
167.00	412.75	0.497	205.80	389.59	0.469	667.00	387.72	0.467
167.60	412.08	0.496	206.80	389.34	0.469	686.80	387.73	0.467
168.20	411.42	0.495	208.00	389.05	0.469	692.40	387.71	0.467
168.60	410.99	0.495	209.20	388.78	0.468	698.00	387.69	0.467
169.20	410.36	0.494	210.20	388.56	0.468	703.80	387.66	0.467
169.60	409.95	0.494	211.20	388.35	0.468	716.80	387.56	0.467
170.20	409.34	0.493	212.40	388.11	0.467	720.80	387.52	0.467
170.60	408.94	0.493	213.40	387.92	0.467	732.20	387.40	0.467
171.20	408.36	0.492	215.40	387.56	0.467	743.60	387.25	0.466
171.60	407.97	0.491	216.60	387.35	0.467	755.00	387.07	0.466
766.20	386.88	0.466	868.00	370.26	0.446	878.80	333.01	0.401
777.60	386.66	0.466	868.40	369.25	0.445	879.00	332.11	0.400
789.00	386.42	0.465	868.80	368.22	0.444	879.20	331.20	0.399
800.20	386.18	0.465	869.00	367.70	0.443	879.40	330.28	0.398
811.60	385.91	0.465	869.40	366.62	0.442	879.60	329.36	0.397
823.00	385.63	0.464	869.80	365.53	0.440	879.80	328.43	0.396
834.40	385.34	0.464	870.00	364.97	0.440	880.00	327.50	0.394
858.80	384.69	0.463	870.40	363.83	0.438	880.20	326.55	0.393
859.00	384.68	0.463	870.80	362.67	0.437	880.40	325.60	0.392
859.20	384.62	0.463	871.20	361.47	0.435	880.80	323.68	0.390
859.40	384.52	0.463	871.40	360.87	0.435	881.20	321.74	0.387
859.60	384.41	0.463	871.80	359.63	0.433	881.60	319.77	0.385
859.80	384.27	0.463	872.00	359.00	0.432	882.00	317.77	0.383
860.00	384.11	0.463	872.20	358.36	0.432	882.20	316.77	0.381
860.20	383.95	0.462	872.40	357.72	0.431	882.60	314.74	0.379
860.40	383.77	0.462	872.60	357.07	0.430	883.00	312.69	0.377
860.60	383.57	0.462	872.80	356.41	0.429	883.20	311.66	0.375
860.80	383.37	0.462	873.00	355.74	0.428	883.60	309.59	0.373
861.00	383.16	0.461	873.20	355.07	0.428	884.00	307.50	0.370
861.20	382.94	0.461	873.40	354.39	0.427	884.40	305.39	0.368
861.40	382.71	0.461	873.60	353.70	0.426	884.80	303.26	0.365
861.60	382.46	0.461	873.80	353.00	0.425	885.00	302.20	0.364
861.80	382.21	0.460	874.00	352.30	0.424	885.40	300.06	0.361
862.00	381.95	0.460	874.20	351.58	0.423	886.00	296.83	0.357
862.20	381.68	0.460	874.40	350.86	0.423	886.40	294.66	0.355
862.40	381.40	0.459	874.60	350.14	0.422	886.80	292.49	0.352
862.60	381.11	0.459	874.80	349.40	0.421	887.40	289.23	0.348
862.80	380.81	0.459	875.00	348.65	0.420	889.00	280.51	0.338
863.00	380.51	0.458	875.20	347.90	0.419	889.60	277.25	0.334
863.20	380.19	0.458	875.40	347.14	0.418	890.20	274.01	0.330
863.40	379.87	0.458	875.60	346.38	0.417	890.60	271.86	0.327
863.60	379.53	0.457	875.80	345.60	0.416	891.00	269.72	0.325

TABLE 6.2.1-61 (SHEET 5 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
863.80	379.19	0.457	876.00	344.82	0.415	891.40	267.59	0.322
864.00	378.84	0.456	876.20	344.02	0.414	891.80	265.47	0.319
864.20	378.48	0.456	876.40	343.22	0.413	892.00	264.42	0.318
864.40	378.11	0.455	876.60	342.42	0.412	892.40	262.33	0.316
864.60	377.74	0.455	876.80	341.60	0.411	892.80	260.25	0.313
864.80	377.36	0.455	877.00	340.78	0.410	893.20	258.19	0.311
865.20	376.57	0.454	877.20	339.94	0.409	893.60	256.15	0.308
865.60	375.75	0.453	877.40	339.10	0.408	893.80	255.13	0.307
866.00	374.90	0.452	877.60	338.26	0.407	894.00	254.12	0.306
866.20	374.47	0.451	877.80	337.40	0.406	894.20	253.12	0.305
866.60	373.58	0.450	878.00	336.54	0.405	894.40	252.12	0.303
867.00	372.66	0.449	878.20	335.67	0.404	894.60	251.13	0.302
867.40	371.72	0.448	878.40	334.79	0.403	894.80	250.14	0.301
867.80	370.75	0.447	878.60	333.90	0.402	895.00	249.16	0.300
895.20	248.18	0.299	904.40	210.60	0.253	913.80	185.77	0.223
895.40	247.22	0.297	904.60	209.93	0.252	914.00	185.37	0.222
895.60	246.25	0.296	904.80	209.28	0.251	914.20	184.97	0.222
895.80	245.30	0.295	905.00	208.63	0.251	914.40	184.57	0.221
896.00	244.35	0.294	905.20	207.99	0.250	914.60	184.18	0.221
896.20	243.41	0.293	905.40	207.35	0.249	914.80	183.79	0.221
896.40	242.47	0.292	905.60	206.72	0.248	915.00	183.41	0.220
896.60	241.54	0.291	905.80	206.10	0.248	915.20	183.03	0.220
896.80	240.62	0.289	906.00	205.48	0.247	915.40	182.66	0.219
897.00	239.71	0.288	906.20	204.87	0.246	915.60	182.29	0.219
897.20	238.80	0.287	906.40	204.27	0.245	915.80	181.92	0.218
897.40	237.90	0.286	906.60	203.67	0.245	916.00	181.56	0.218
897.60	237.00	0.285	906.80	203.08	0.244	916.20	181.20	0.217
897.80	236.11	0.284	907.00	202.49	0.243	916.40	180.85	0.217
898.00	235.23	0.283	907.20	201.91	0.242	916.60	180.50	0.217
898.20	234.36	0.282	907.40	201.34	0.242	916.80	180.16	0.216
898.40	233.49	0.281	907.60	200.77	0.241	917.00	179.81	0.216
898.60	232.63	0.280	907.80	200.20	0.240	917.20	179.48	0.215
898.80	231.78	0.279	908.20	199.09	0.239	917.40	179.14	0.215
899.00	230.93	0.278	908.40	198.54	0.238	917.60	178.81	0.214
899.20	230.09	0.277	908.60	198.00	0.238	917.80	178.49	0.214
899.40	229.26	0.276	908.80	197.46	0.237	918.00	178.16	0.214
899.60	228.43	0.275	909.00	196.93	0.236	918.20	177.85	0.213
899.80	227.62	0.274	909.20	196.40	0.236	918.40	177.53	0.213
900.00	226.80	0.273	909.40	195.88	0.235	918.60	177.22	0.213
900.20	226.00	0.272	909.60	195.37	0.235	918.80	176.91	0.212
900.40	225.20	0.271	909.80	194.86	0.234	919.20	176.31	0.211
900.60	224.41	0.270	910.00	194.35	0.233	919.40	176.01	0.211
900.80	223.62	0.269	910.20	193.85	0.233	919.80	175.43	0.210

TABLE 6.2.1-61 (SHEET 6 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
901.00	222.84	0.268	910.40	193.36	0.232	920.00	175.14	0.210
901.20	222.07	0.267	910.60	192.87	0.232	920.40	174.57	0.209
901.40	221.30	0.266	910.80	192.39	0.231	920.80	174.02	0.209
901.60	220.54	0.265	911.00	191.91	0.230	921.00	173.75	0.208
901.80	219.79	0.264	911.20	191.44	0.230	921.40	173.22	0.208
902.00	219.04	0.263	911.40	190.97	0.229	921.80	172.70	0.207
902.20	218.30	0.262	911.60	190.51	0.229	922.00	172.45	0.207
902.40	217.57	0.261	911.80	190.05	0.228	922.20	172.19	0.206
902.60	216.84	0.261	912.00	189.60	0.228	922.60	171.70	0.206
902.80	216.12	0.260	912.20	189.15	0.227	923.00	171.21	0.205
903.00	215.41	0.259	912.40	188.71	0.226	923.20	170.98	0.205
903.20	214.70	0.258	912.60	188.27	0.226	923.60	170.51	0.204
903.40	214.00	0.257	912.80	187.84	0.225	924.00	170.05	0.204
903.60	213.31	0.256	913.00	187.41	0.225	924.20	169.83	0.204
903.80	212.62	0.255	913.20	187.02	0.224	924.60	169.39	0.203
904.00	211.94	0.255	913.40	186.60	0.224	925.00	168.96	0.203
904.20	211.26	0.254	913.60	186.18	0.223	925.40	168.54	0.202
925.80	168.13	0.202	945.00	156.66	0.188	1001.60	151.86	0.182
926.00	167.93	0.201	945.80	156.42	0.187	1004.40	151.83	0.182
926.40	167.53	0.201	946.40	156.24	0.187	1007.40	151.82	0.182
926.80	167.15	0.200	947.00	156.07	0.187	1010.00	151.80	0.182
927.20	166.77	0.200	947.80	155.86	0.187	1015.80	151.78	0.182
927.60	166.40	0.199	948.40	155.70	0.186	1021.40	151.76	0.182
928.00	166.04	0.199	949.20	155.51	0.186	1027.20	151.75	0.182
928.40	165.69	0.199	950.00	155.32	0.186	1038.40	151.74	0.182
928.80	165.35	0.198	950.60	155.18	0.186	1070.80	151.73	0.182
929.00	165.18	0.198	951.20	155.05	0.186	1800.00	151.74	0.182
929.40	164.85	0.198	952.00	154.89	0.185	1800.20	151.78	0.182
929.80	164.53	0.197	952.80	154.73	0.185	1800.40	152.30	0.182
930.00	164.37	0.197	953.40	154.62	0.185	1800.60	152.41	0.182
930.40	164.06	0.197	954.20	154.48	0.185	1800.80	152.36	0.182
930.80	163.76	0.196	955.00	154.34	0.185	1801.00	152.17	0.182
931.00	163.61	0.196	955.80	154.21	0.185	1801.20	151.84	0.182
931.40	163.32	0.196	956.40	154.12	0.185	1801.40	151.40	0.181
931.80	163.03	0.195	957.00	154.03	0.184	1801.60	150.84	0.181
932.20	162.75	0.195	957.80	153.91	0.184	1801.80	150.19	0.180
932.60	162.48	0.195	958.40	153.83	0.184	1802.00	149.44	0.179
933.00	162.22	0.194	959.20	153.72	0.184	1802.20	148.61	0.178
933.60	161.83	0.194	960.00	153.62	0.184	1802.40	147.70	0.177
934.20	161.46	0.193	961.20	153.48	0.184	1802.60	146.72	0.176
934.60	161.22	0.193	962.00	153.40	0.184	1802.80	145.69	0.174
935.00	160.98	0.193	962.80	153.31	0.184	1803.00	144.58	0.173
935.40	160.75	0.193	963.40	153.25	0.184	1803.20	143.43	0.172

TABLE 6.2.1-61 (SHEET 7 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
935.80	160.53	0.192	964.80	153.12	0.183	1803.40	142.22	0.170
936.40	160.21	0.192	966.20	153.00	0.183	1803.60	140.93	0.169
937.00	159.89	0.192	967.60	152.89	0.183	1803.80	139.04	0.166
937.40	159.69	0.191	969.00	152.79	0.183	1804.00	137.19	0.164
937.80	159.49	0.191	970.40	152.70	0.183	1804.20	135.43	0.162
938.40	159.21	0.191	971.80	152.62	0.183	1804.40	133.68	0.160
938.80	159.02	0.190	973.20	152.54	0.183	1804.60	131.94	0.158
939.20	158.84	0.190	974.60	152.47	0.183	1804.80	130.22	0.156
939.60	158.66	0.190	976.00	152.40	0.182	1805.00	128.51	0.154
940.00	158.49	0.190	977.40	152.35	0.182	1805.20	126.82	0.151
940.60	158.24	0.190	979.00	152.29	0.182	1805.40	125.14	0.149
941.20	158.00	0.189	980.40	152.24	0.182	1805.60	123.48	0.147
941.60	157.84	0.189	981.80	152.19	0.182	1805.80	121.84	0.145
942.00	157.69	0.189	983.20	152.15	0.182	1806.00	120.22	0.143
942.40	157.54	0.189	984.60	152.12	0.182	1806.20	118.61	0.142
942.80	157.39	0.189	987.40	152.05	0.182	1806.40	117.03	0.140
943.40	157.18	0.188	990.20	152.00	0.182	1806.60	115.46	0.138
943.80	157.05	0.188	993.00	151.95	0.182	1806.80	113.92	0.136
944.20	156.92	0.188	996.00	151.91	0.182	1807.00	112.39	0.134
944.60	156.79	0.188	998.80	151.88	0.182	1807.20	110.89	0.132
1807.40	109.42	0.130	1816.60	60.30	0.071	1826.80	31.41	0.037
1807.60	107.97	0.129	1816.80	59.52	0.070	1827.00	30.99	0.036
1807.80	106.55	0.127	1817.00	58.75	0.069	1827.20	30.57	0.036
1808.00	105.14	0.125	1817.20	58.00	0.068	1827.40	30.15	0.035
1808.20	103.76	0.124	1817.40	57.25	0.068	1827.60	29.75	0.035
1808.40	102.39	0.122	1817.60	56.52	0.067	1827.80	29.35	0.034
1808.60	101.05	0.120	1817.80	55.80	0.066	1828.00	28.95	0.034
1808.80	99.73	0.119	1818.00	55.10	0.065	1828.20	28.57	0.033
1809.00	98.45	0.117	1818.20	54.40	0.064	1828.40	28.19	0.033
1809.20	97.17	0.116	1818.40	53.72	0.063	1828.60	27.83	0.032
1809.40	95.92	0.114	1818.60	53.04	0.062	1828.80	27.47	0.032
1809.60	94.69	0.113	1818.80	52.38	0.062	1829.00	27.11	0.032
1809.80	93.48	0.111	1819.00	51.73	0.061	1829.20	26.77	0.031
1810.00	92.28	0.110	1819.20	51.09	0.060	1829.40	26.44	0.031
1810.20	91.11	0.108	1819.40	50.46	0.059	1829.60	26.11	0.030
1810.40	89.96	0.107	1819.60	49.85	0.059	1829.80	25.79	0.030
1810.60	88.83	0.106	1819.80	49.24	0.058	1830.00	25.49	0.030
1810.80	87.72	0.104	1820.00	48.66	0.057	1830.20	25.19	0.029
1811.00	86.64	0.103	1820.20	48.08	0.057	1830.40	24.89	0.029
1811.20	85.57	0.102	1820.40	47.49	0.056	1830.60	24.59	0.029
1811.40	84.53	0.100	1821.20	45.15	0.053	1830.80	24.30	0.028
1811.60	83.46	0.099	1821.60	43.97	0.052	1831.00	24.00	0.028
1811.80	82.42	0.098	1821.80	43.39	0.051	1831.20	23.71	0.028

TABLE 6.2.1-61 (SHEET 8 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
1812.00	81.38	0.097	1822.00	42.81	0.050	1831.40	23.41	0.027
1812.20	80.35	0.095	1822.20	42.23	0.050	1831.60	23.13	0.027
1812.40	79.34	0.094	1822.40	41.66	0.049	1831.80	22.84	0.027
1812.60	78.32	0.093	1822.60	41.11	0.048	1832.00	22.55	0.026
1812.80	77.32	0.092	1822.80	40.55	0.048	1832.20	22.30	0.026
1813.00	76.33	0.090	1823.00	40.00	0.047	1832.40	22.06	0.026
1813.20	75.34	0.089	1823.20	39.46	0.046	1832.60	21.81	0.025
1813.40	74.37	0.088	1823.40	38.92	0.046	1832.80	21.47	0.025
1813.60	73.40	0.087	1823.60	38.39	0.045	1833.00	21.17	0.025
1813.80	72.45	0.086	1823.80	37.87	0.044	1833.20	20.89	0.024
1814.00	71.51	0.085	1824.00	37.54	0.044	1833.40	20.62	0.024
1814.20	70.58	0.084	1824.20	37.13	0.043	1833.60	20.35	0.024
1814.40	69.66	0.082	1824.40	36.71	0.043	1833.80	20.08	0.023
1814.60	68.75	0.081	1824.60	36.28	0.042	1834.00	19.82	0.023
1814.80	67.85	0.080	1824.80	35.81	0.042	1834.20	19.56	0.023
1815.00	66.96	0.079	1825.00	35.37	0.041	1834.60	19.04	0.022
1815.20	66.09	0.078	1825.20	34.93	0.041	1834.80	18.79	0.022
1815.40	65.22	0.077	1825.40	34.49	0.040	1835.00	18.53	0.021
1815.60	64.37	0.076	1825.80	33.60	0.039	1835.20	18.26	0.021
1815.80	63.53	0.075	1826.00	33.16	0.039	1835.40	18.00	0.021
1816.00	62.71	0.074	1826.20	32.71	0.038	1835.60	17.73	0.021
1816.20	61.89	0.073	1826.40	32.28	0.038	1835.80	17.46	0.020
1816.40	61.09	0.072	1826.60	31.84	0.037	1836.00	17.19	0.020
1836.20	16.92	0.020						
1836.40	16.64	0.019						
1836.60	16.35	0.019						
1836.80	16.06	0.019						
1837.00	15.77	0.018						
1837.20	15.47	0.018						
1837.40	15.17	0.018						
1837.60	14.86	0.017						
1837.80	14.55	0.017						
1838.00	14.24	0.016						
1838.20	13.91	0.016						
1838.40	13.59	0.016						
1838.60	13.25	0.015						
1838.80	12.91	0.015						
1839.00	12.57	0.014						
1839.20	12.22	0.014						
1839.40	11.86	0.014						
1839.60	11.49	0.013						
1839.80	11.12	0.013						
1840.00	10.73	0.012						

TABLE 6.2.1-61 (SHEET 9 OF 9)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
1840.20	10.34	0.012						
1840.40	9.93	0.011						
1840.60	9.51	0.011						
1840.80	9.08	0.010						
1841.00	8.62	0.010						
1841.20	8.13	0.009						
1841.40	7.59	0.009						
1841.60	6.98	0.008						
1841.80	6.17	0.007						
1842.00	0.00	0.000						
1900.00	0.00	0.000						

TABLE 6.2.1-62 (SHEET 1 OF 8)

MASS AND ENERGY RELEASE DATA FOR CASE 13 - PEAK CALCULATED CONTAINMENT TEMPERATURE FOR MSLB

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
0.0000	0.0000	0.0000	10.60	1318.0	1.574	23.00	1850.0	2.201
0.2000	1485.0	1.770	10.80	1314.0	1.570	23.50	1844.0	2.194
0.4000	1485.0	1.770	11.00	1314.0	1.570	24.00	1841.0	2.190
0.6000	1485.0	1.770	11.20	1309.0	1.564	24.50	1837.0	2.186
0.8000	1485.0	1.770	11.40	1309.0	1.564	25.00	1833.0	2.182
1.000	1480.0	1.764	11.60	1305.0	1.559	25.50	1829.0	2.177
1.200	1475.0	1.758	11.80	1300.0	1.559	26.00	1826.0	2.173
1.400	1470.0	1.752	12.00	1596.0	1.554	26.50	1822.0	2.169
1.600	1465.0	1.746	12.20	1590.0	1.907	27.00	1818.0	2.164
1.800	1460.0	1.741	12.40	1604.0	1.900	27.50	1814.0	2.160
2.000	1455.0	1.735	12.60	1613.0	1.916	28.00	1810.0	2.155
2.200	1451.0	1.730	12.80	1622.0	1.927	28.50	1806.0	2.150
2.400	1446.0	1.725	13.00	1631.0	1.937	29.00	1801.0	2.145
2.600	1442.0	1.719	13.20	1640.0	1.947	29.50	1797.0	2.140
2.800	1437.0	1.714	13.40	1648.0	1.957	30.00	1793.0	2.135
3.000	1433.0	1.709	13.60	1657.0	1.968	30.50	1788.0	2.130
3.200	1429.0	1.704	13.80	1665.0	1.977	31.00	1784.0	2.125
3.400	1425.0	1.699	14.00	1674.0	1.987	31.50	1779.0	2.119
3.600	1421.0	1.695	14.20	1682.0	1.997	32.00	1774.0	2.113
3.800	1417.0	1.690	14.40	1690.0	2.007	32.50	1769.0	2.108
4.000	1413.0	1.685	14.60	1698.0	2.016	33.00	1764.0	2.103
4.200	1409.0	1.681	14.80	1706.0	2.025	33.50	1760.0	2.097
4.400	1405.0	1.676	15.00	1714.0	2.035	34.00	1755.0	2.092
4.600	1405.0	1.676	15.20	1722.0	2.044	34.50	1750.0	2.086
4.800	1400.0	1.671	15.40	1730.0	2.053	35.00	1745.0	2.080
5.000	1395.0	1.665	15.60	1737.0	2.062	35.50	1741.0	2.075
5.200	1391.0	1.661	15.80	1744.0	2.070	36.00	1736.0	2.069
5.400	1387.0	1.656	16.00	1751.0	2.079	36.50	1731.0	2.064
5.600	1387.0	1.656	16.20	1758.0	2.087	37.00	1726.0	2.058
5.800	1383.0	1.650	16.40	1765.0	2.095	37.50	1721.0	2.053
6.000	1378.0	1.645	16.60	1772.0	2.103	38.00	1717.0	2.047
6.200	1374.0	1.641	16.80	1778.0	2.110	38.50	1712.0	2.042
6.400	1374.0	1.641	17.00	1784.0	2.118	39.00	1707.0	2.037
6.600	1370.0	1.635	17.20	1795.0	2.125	39.50	1703.0	2.031
6.800	1366.0	1.631	17.40	1803.0	2.138	40.00	1698.0	2.026
7.000	1366.0	1.631	17.60	1809.0	2.147	40.50	1693.0	2.020
7.200	1361.0	1.625	17.80	1815.0	2.154	41.00	1689.0	2.015
7.400	1357.0	1.620	18.00	1821.0	2.161	41.50	1684.0	2.009
7.600	1357.0	1.620	18.20	1826.0	2.167	42.00	1679.0	2.004
7.800	1352.0	1.615	18.40	1830.0	2.173	42.50	1675.0	1.999
8.000	1348.0	1.610	18.60	1834.0	2.178	43.00	1670.0	1.993
8.200	1348.0	1.610	18.80	1837.0	2.182	43.50	1666.0	1.988
8.400	1344.0	1.605	19.00	1837.0	2.186	44.00	1661.0	1.983
8.600	1344.0	1.605	19.20	1840.0	2.186	44.50	1656.0	1.978
8.800	1339.0	1.600	19.40	1843.0	2.190	45.00	1652.0	1.972
9.000	1335.0	1.595	19.60	1845.0	2.193	45.50	1648.0	1.967
9.200	1335.0	1.595	19.80	1847.0	2.195	46.00	1643.0	1.962
9.400	1331.0	1.590	20.00	1851.0	2.198	46.50	1639.0	1.957
9.600	1331.0	1.590	20.50	1851.0	2.202	47.00	1634.0	1.952
9.800	1326.0	1.585	21.00	1852.0	2.202	47.50	1630.0	1.947
10.000	1326.0	1.585	21.50	1852.0	2.204	48.00	1626.0	1.942
10.20	1322.0	1.579	22.00	1852.0	2.204	48.50	1621.0	1.937
10.40	1322.0	1.580	22.50	1852.0	2.204	49.00	1617.0	1.932

TABLE 6.2.1-62 (SHEET 2 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
49.50	1613.0	1.927	76.00	943.2	1.136	105.0	866.6	1.044
50.00	1608.0	1.922	76.50	939.6	1.131	106.0	866.0	1.043
50.50	1604.0	1.917	77.00	936.1	1.127	107.0	865.3	1.042
51.00	1621.0	1.938	77.50	932.8	1.123	108.0	864.7	1.041
51.50	1570.0	1.878	78.00	929.7	1.120	109.0	864.2	1.041
52.00	1538.0	1.840	78.50	926.7	1.116	110.0	863.6	1.040
52.50	1507.0	1.804	79.00	923.9	1.113	111.0	863.1	1.040
53.00	1478.0	1.770	79.50	921.1	1.109	112.0	862.6	1.039
53.50	1450.0	1.737	80.00	918.5	1.106	113.0	862.1	1.038
54.00	1423.0	1.706	80.50	916.0	1.103	114.0	861.7	1.038
54.50	1399.0	1.677	81.00	913.6	1.100	115.0	861.2	1.037
55.00	1375.0	1.649	81.50	911.3	1.097	116.0	860.8	1.037
55.50	1353.0	1.623	82.00	909.1	1.095	117.0	860.4	1.036
56.00	1332.0	1.598	82.50	907.1	1.092	118.0	860.0	1.036
56.50	1312.0	1.574	83.00	905.1	1.090	119.0	859.7	1.035
57.00	1293.0	1.552	83.50	903.2	1.088	120.0	859.3	1.035
57.50	1275.0	1.531	84.00	901.3	1.085	121.0	859.0	1.035
58.00	1258.0	1.511	84.50	899.6	1.083	122.0	858.6	1.034
58.50	1241.0	1.491	85.00	897.9	1.081	123.0	858.3	1.034
59.00	1226.0	1.473	85.50	896.3	1.079	124.0	858.0	1.033
59.50	1211.0	1.455	86.00	894.8	1.078	125.0	857.7	1.033
60.00	1197.0	1.438	86.50	893.4	1.076	126.0	857.4	1.033
60.50	1183.0	1.422	87.00	892.0	1.074	127.0	857.1	1.032
61.00	1170.0	1.406	87.50	890.7	1.073	128.0	856.8	1.032
61.50	1157.0	1.391	88.00	889.4	1.071	129.0	856.6	1.032
62.00	1145.0	1.377	88.50	888.2	1.070	130.0	856.3	1.031
62.50	1133.0	1.363	89.00	887.0	1.068	131.0	856.1	1.031
63.00	1122.0	1.350	89.50	885.9	1.067	132.0	855.8	1.031
63.50	1111.0	1.337	90.00	884.8	1.066	133.0	855.6	1.030
64.00	1101.0	1.324	90.50	883.8	1.064	134.0	855.3	1.030
64.50	1091.0	1.313	91.00	882.8	1.063	135.0	855.1	1.030
65.00	1081.0	1.301	91.50	881.9	1.062	136.0	854.9	1.030
65.50	1072.0	1.290	92.00	881.0	1.061	137.0	854.7	1.029
66.00	1063.0	1.279	92.50	880.1	1.060	138.0	854.5	1.029
66.50	1055.0	1.269	93.00	879.3	1.059	139.0	854.3	1.029
67.00	1046.0	1.259	93.50	878.5	1.058	140.0	854.1	1.029
67.50	1038.0	1.249	94.00	877.7	1.057	141.0	853.9	1.028
68.00	1030.0	1.240	94.50	877.0	1.056	142.0	853.7	1.028
68.50	1023.0	1.231	95.00	876.3	1.055	143.0	853.5	1.028
69.00	1016.0	1.223	95.50	875.6	1.055	144.0	853.3	1.028
69.50	1009.0	1.215	96.00	874.9	1.054	145.0	853.1	1.028
70.00	1003.0	1.207	96.50	874.3	1.053	146.0	853.0	1.027
70.50	996.3	1.199	97.00	873.7	1.052	147.0	852.8	1.027
71.00	990.3	1.192	97.50	873.1	1.052	148.0	852.6	1.027
71.50	984.6	1.185	98.00	872.6	1.051	149.0	852.5	1.027
72.00	979.1	1.179	98.50	872.0	1.050	150.0	852.3	1.027
72.50	973.9	1.172	99.00	871.5	1.050	151.0	852.2	1.026
73.00	968.9	1.166	99.50	871.0	1.049	152.0	852.0	1.026
73.50	964.1	1.161	100.0	870.5	1.048	153.0	851.9	1.026
74.00	959.5	1.155	101.0	869.8	1.048	154.0	851.7	1.026
74.50	955.2	1.150	102.0	868.9	1.046	155.0	851.6	1.026
75.00	951.0	1.145	103.0	868.1	1.046	156.0	851.4	1.025
75.50	947.0	1.140	104.0	867.3	1.045	157.0	851.3	1.025

TABLE 6.2.1-62 (SHEET 3 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
158.0	851.2	1.025	220.0	489.7	0.5890	324.0	150.6	0.1783
159.0	851.1	1.025	222.0	525.8	0.6326	326.0	150.3	0.1779
160.0	850.9	1.025	224.0	573.5	0.6900	328.0	150.2	0.1778
161.0	850.8	1.025	226.0	545.0	0.6551	330.0	150.4	0.1780
162.0	850.7	1.025	228.0	448.8	0.5388	332.0	150.2	0.1778
163.0	850.6	1.024	230.0	404.4	0.4860	334.0	150.1	0.1776
164.0	850.5	1.024	232.0	492.3	0.5918	336.0	150.0	0.1776
165.0	850.4	1.024	234.0	488.1	0.5863	338.0	150.1	0.1777
166.0	850.3	1.024	236.0	419.5	0.5031	340.0	150.0	0.1776
167.0	850.2	1.024	238.0	360.1	0.4317	342.0	149.9	0.1775
168.0	850.1	1.024	240.0	366.8	0.4401	344.0	149.9	0.1774
169.0	850.0	1.024	242.0	401.8	0.4819	346.0	150.0	0.1775
170.0	849.9	1.024	244.0	365.6	0.4377	348.0	149.9	0.1774
171.0	849.8	1.024	246.0	302.6	0.3619	350.0	149.9	0.1774
172.0	849.7	1.023	248.0	291.9	0.3492	352.0	149.8	0.1774
173.0	849.6	1.023	250.0	315.4	0.3774	354.0	149.9	0.1774
174.0	849.6	1.023	252.0	307.5	0.3673	356.0	149.9	0.1774
175.0	849.5	1.023	254.0	251.5	0.2998	358.0	149.8	0.1773
176.0	849.4	1.023	256.0	235.1	0.2807	360.0	149.8	0.1773
177.0	849.3	1.023	258.0	271.8	0.3243	362.0	149.8	0.1773
178.0	849.3	1.023	260.0	248.6	0.2962	364.0	149.8	0.1773
179.0	849.2	1.023	262.0	213.2	0.2536	366.0	149.8	0.1773
180.0	849.1	1.023	264.0	198.2	0.2357	368.0	149.8	0.1773
181.0	849.1	1.023	266.0	206.3	0.2454	370.0	149.8	0.1773
182.0	849.0	1.023	268.0	206.7	0.2455	372.0	149.8	0.1773
183.0	843.9	1.016	270.0	184.5	0.2191	374.0	149.8	0.1773
184.0	824.8	0.9935	272.0	178.0	0.2115	376.0	149.8	0.1773
185.0	807.5	0.9726	274.0	188.2	0.2235	378.0	149.8	0.1773
186.0	785.5	0.9461	276.0	180.1	0.2137	380.0	149.8	0.1773
187.0	759.7	0.9150	278.0	168.8	0.2002	382.0	149.8	0.1773
188.0	730.0	0.8792	280.0	165.2	0.1960	384.0	149.8	0.1773
189.0	696.7	0.8389	282.0	170.6	0.2023	386.0	149.8	0.1773
190.0	660.2	0.7948	284.0	166.8	0.1977	388.0	149.8	0.1773
191.0	625.6	0.7530	286.O	159.7	0.1892	390.0	149.8	0.1773
192.0	622.4	0.7494	288.0	158.3	0.1876	392.0	149.8	0.1773
193.0	639.9	0.7704	290.0	162.0	0.1920	394.0	149.8	0.1773
194.0	659.3	0.7939	292.0	159.3	0.1887	396.0	149.8	0.1773
195.0	672.4	0.8097	294.0	155.4	0.1840	398.0	149.8	0.1773
196.0	678.8	0.8174	296.0	154.4	0.1829	400.0	149.8	0.1773
197.0	676.2	0.8143	298.0	156.3	0.1851	402.0	149.8	0.1773
198.0	663.9	0.7994	300.0	155.1	0.1836	404.0	149.8	0.1773
199.0	647.0	0.7790	302.0	152.9	0.1810	406.0	149.8	0.1773
200.0	635.1	0.7646	304.0	152.3	0.1803	408.0	149.8	0.1773
202.0	627.1	0.7550	306.0	153.3	0.1816	410.0	149.8	0.1773
204.0	631.5	0.7603	308.0	152.7	0.1808	412.0	149.8	0.1773
206.0	646.0	0.7778	310.0	151.5	0.1793	414.0	149.8	0.1773
208.0	626.7	0.7543	312.0	151.2	0.1790	416.0	149.8	0.1773
210.0	580.6	0.6988	314.0	151.7	0.1796	418.0	149.8	0.1773
212.0	573.1	0.6898	316.0	151.4	0.1792	420.0	149.8	0.1773
214.0	610.7	0.7351	318.0	150.7	0.1784	422.0	149.8	0.1773
216.0	605.4	0.7286	320.0	150.5	0.1782	424.0	149.8	0.1773
218.0	557.0	0.6699	322.0	150.9	0.1786	426.0	149.8	0.1773

TABLE 6.2.1-62 (SHEET 4 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
428.0	149.8	0.1773	532.0	149.9	0.1774	636.0	149.8	0.1774
430.0	149.8	0.1773	536.0	149.9	0.1774	638.0	149.8	0.1774
432.0	149.8	0.1773	538.0	149.9	0.1774	640.0	149.8	0.1774
436.0	149.8	0.1773	540.0	149.9	0.1774	642.0	149.8	0.1774
438.0	149.8	0.1773	542.0	149.9	0.1774	644.0	149.8	0.1774
440.0	149.8	0.1773	544.0	149.9	0.1774	646.0	149.8	0.1774
442.0	149.8	0.1773	546.0	149.9	0.1774	648.0	149.8	0.1774
444.0	149.8	0.1773	548.0	149.9	0.1774	650.0	149.8	0.1774
446.0	149.8	0.1773	550.0	149.9	0.1774	652.0	149.8	0.1774
448.0	149.8	0.1773	552.0	149.9	0.1774	654.0	149.8	0.1774
450.0	149.8	0.1773	554.0	149.9	0.1774	656.0	149.8	0.1774
452.0	149.8	0.1774	556.0	149.9	0.1774	658.0	149.8	0.1774
454.0	149.8	0.1774	558.0	149.9	0.1774	660.0	149.8	0.1774
456.0	149.8	0.1774	560.0	149.9	0.1774	662.0	149.8	0.1774
458.0	149.8	0.1774	562.0	149.9	0.1774	664.0	149.8	0.1774
460.0	149.8	0.1774	564.0	149.8	0.1774	666.0	149.8	0.1774
462.0	149.9	0.1774	566.0	149.9	0.1774	668.0	149.8	0.1774
464.0	149.9	0.1774	568.0	149.8	0.1774	670.0	149.8	0.1774
466.0	149.9	0.1774	570.0	149.8	0.1774	672.0	149.8	0.1774
468.0	149.9	0.1774	572.0	149.8	0.1774	674.0	149.8	0.1774
470.0	149.9	0.1774	574.0	149.8	0.1774	676.0	149.8	0.1774
472.0	149.9	0.1774	576.0	149.8	0.1774	678.0	149.8	0.1774
474.0	149.9	0.1774	578.0	149.8	0.1774	680.0	149.8	0.1774
476.0	149.9	0.1774	580.0	149.8	0.1774	682.0	149.8	0.1774
478.0	149.9	0.1774	582.0	149.8	0.1774	684.0	149.8	0.1774
480.0	149.9	0.1774	584.0	149.8	0.1774	686.0	149.8	0.1774
482.0	149.9	0.1774	586.0	149.8	0.1774	688.0	149.8	0.1774
484.0	149.9	0.1774	588.0	149.8	0.1774	690.0	149.8	0.1774
486.0	149.9	0.1774	590.0	149.8	0.1774	692.0	149.8	0.1774
488.0	149.9	0.1774	592.0	149.8	0.1774	694.0	149.8	0.1774
490.0	149.9	0.1774	594.0	149.8	0.1774	696.0	149.8	0.1774
492.0	149.9	0.1774	596.0	149.8	0.1774	698.0	149.8	0.1774
494.0	149.9	0.1774	598.0	149.8	0.1774	700.0	149.8	0.1774
496.0	149.9	0.1774	600.0	149.8	0.1774	702.0	149.8	0.1774
498.0	149.9	0.1774	602.0	149.8	0.1774	704.0	149.8	0.1774
500.0	149.9	0.1774	604.0	149.8	0.1774	706.0	149.8	0.1774
502.0	149.9	0.1774	606.0	149.8	0.1774	708.0	149.8	0.1774
504.0	149.9	0.1774	608.0	149.8	0.1774	710.0	149.8	0.1774
506.0	149.9	0.1774	610.0	149.8	0.1774	712.0	149.8	0.1774
508.0	149.9	0.1774	612.0	149.8	0.1774	714.0	149.8	0.1774
510.0	149.9	0.1774	614.0	149.8	0.1774	716.0	149.8	0.1774
512.0	149.9	0.1774	616.0	149.8	0.1774	718.0	149.8	0.1774
514.0	149.9	0.1774	618.0	149.8	0.1774	720.0	149.8	0.1774
516.0	149.9	0.1774	620.0	149.8	0.1774	722.0	149.8	0.1774
518.0	149.9	0.1774	622.0	149.8	0.1774	724.0	149.8	0.1774
520.0	149.9	0.1774	624.0	149.8	0.1774	726.0	149.8	0.1774
522.0	149.9	0.1774	626.0	149.8	0.1774	728.0	149.8	0.1774
524.0	149.9	0.1774	628.0	149.8	0.1774	730.0	149.8	0.1774
526.0	149.9	0.1774	630.0	149.8	0.1774	732.0	149.8	0.1774
528.0	149.9	0.1774	632.0	149.8	0.1774	734.0	149.8	0.1774
530.0	149.9	0.1774	634.0	149.8	0.1774	736.0	149.8	0.1774
						738.0	149.8	0.1774
						740.0	149.8	0.1774

TABLE 6.2.1-62 (SHEET 5 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
742.0	149.8	0.1774	848.0	149.8	0.1774	954.0	149.8	0.1774
744.0	149.8	0.1774	850.0	149.8	0.1774	956.0	149.8	0.1774
746.0	149.8	0.1774	852.0	149.8	0.1774	958.0	149.8	0.1774
748.0	149.8	0.1774	854.0	149.8	0.1774	960.0	149.8	0.1774
750.0	149.8	0.1774	856.0	149.8	0.1774	962.0	149.8	0.1774
752.0	149.8	0.1774	858.0	149.8	0.1774	964.0	149.8	0.1774
754.0	149.8	0.1774	860.0	149.8	0.1774	966.0	149.8	0.1774
756.0	149.8	0.1774	862.0	149.8	0.1774	968.0	149.8	0.1774
758.0	149.8	0.1774	864.0	149.8	0.1774	970.0	149.8	0.1774
760.0	149.8	0.1774	866.0	149.8	0.1774	972.0	149.8	0.1774
762.0	149.8	0.1774	868.0	149.8	0.1774	974.0	149.8	0.1774
764.0	149.8	0.1774	870.0	149.8	0.1774	976.0	149.8	0.1774
766.0	149.8	0.1774	872.0	149.8	0.1774	978.0	149.8	0.1774
768.0	149.8	0.1774	874.0	149.8	0.1774	980.0	149.8	0.1774
770.0	149.8	0.1774	876.0	149.8	0.1774	982.0	149.8	0.1774
772.0	149.8	0.1774	878.0	149.8	0.1774	984.0	149.8	0.1774
774.0	149.8	0.1774	880.0	149.8	0.1774	986.0	149.8	0.1774
776.0	149.8	0.1774	882.0	149.8	0.1774	988.0	149.8	0.1774
778.0	149.8	0.1774	884.0	149.8	0.1774	990.0	149.8	0.1774
780.0	149.8	0.1774	886.0	149.8	0.1774	992.0	149.8	0.1774
782.0	149.8	0.1774	888.0	149.8	0.1774	994.0	149.8	0.1774
784.0	149.8	0.1774	890.0	149.8	0.1774	996.0	149.8	0.1774
786.0	149.8	0.1774	892.0	149.8	0.1774	998.0	149.8	0.1774
788.0	149.8	0.1774	894.0	149.8	0.1774	1000.0	149.8	0.1774
790.0	149.8	0.1774	896.0	149.8	0.1774	1002.0	149.8	0.1774
792.0	149.8	0.1774	898.0	149.8	0.1774	1004.0	149.8	0.1774
794.0	149.8	0.1774	900.0	149.8	0.1774	1006.0	149.8	0.1774
796.0	149.8	0.1774	902.0	149.8	0.1774	1008.0	149.8	0.1774
798.0	149.8	0.1774	904.0	149.8	0.1774	1010.0	149.8	0.1774
800.0	149.8	0.1774	906.0	149.8	0.1774	1012.0	149.8	0.1774
802.0	149.8	0.1774	908.0	149.8	0.1774	1014.0	149.8	0.1774
804.0	149.8	0.1774	910.0	149.8	0.1774	1016.0	149.8	0.1774
806.0	149.8	0.1774	912.0	149.8	0.1774	1018.0	149.8	0.1774
808.0	149.8	0.1774	914.0	149.8	0.1774	1020.0	149.8	0.1774
810.0	149.8	0.1774	916.0	149.8	0.1774	1022.0	149.8	0.1774
812.0	149.8	0.1774	918.0	149.8	0.1774	1024.0	149.8	0.1774
814.0	149.8	0.1774	920.0	149.8	0.1774	1026.0	149.8	0.1774
816.0	149.8	0.1774	922.0	149.8	0.1774	1028.0	149.8	0.1774
818.0	149.8	0.1774	924.0	149.8	0.1774	1030.0	149.8	0.1774
820.0	149.8	0.1774	926.0	149.8	0.1774	1032.0	149.8	0.1774
822.0	149.8	0.1774	928.0	149.8	0.1774	1034.0	149.8	0.1774
824.0	149.8	0.1774	930.0	149.8	0.1774	1036.0	149.8	0.1774
826.0	149.8	0.1774	932.0	149.8	0.1774	1038.0	149.8	0.1774
828.0	149.8	0.1774	934.0	149.8	0.1774	1040.0	149.8	0.1774
830.0	149.8	0.1774	936.0	149.8	0.1774	1042.0	149.8	0.1774
832.0	149.8	0.1774	938.0	149.8	0.1774	1044.0	149.8	0.1774
834.0	149.8	0.1774	940.0	149.8	0.1774	1046.0	149.8	0.1774
836.0	149.8	0.1774	942.0	149.8	0.1774	1048.0	149.8	0.1774
838.0	149.8	0.1774	944.0	149.8	0.1774	1050.0	149.8	0.1774
840.0	149.8	0.1774	946.0	149.8	0.1774	1052.0	149.8	0.1774
842.0	149.8	0.1774	948.0	149.8	0.1774	1054.0	149.8	0.1774
844.0	149.8	0.1774	950.0	149.8	0.1774	1058.0	149.8	0.1774
846.0	149.8	0.1774	952.0	149.8	0.1774	1060.0	149.8	0.1774

TABLE 6.2.1-62 (SHEET 6 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
1062.0	149.8	0.1774	1168.0	149.8	0.1774	1274.0	149.8	0.1774
1064.0	149.8	0.1774	1170.0	149.8	0.1774	1276.0	149.8	0.1774
1066.0	149.8	0.1774	1172.0	149.8	0.1774	1278.0	149.8	0.1774
1068.0	149.8	0.1774	1174.0	149.8	0.1774	1280.0	149.8	0.1774
1070.0	149.8	0.1774	1176.0	149.8	0.1774	1282.0	149.8	0.1774
1072.0	149.8	0.1774	1178.0	149.8	0.1774	1284.0	149.8	0.1774
1074.0	149.8	0.1774	1180.0	149.8	0.1774	1286.0	149.8	0.1774
1076.0	149.8	0.1774	1182.0	149.8	0.1774	1288.0	149.8	0.1774
1078.0	149.8	0.1774	1184.0	149.8	0.1774	1290.0	149.8	0.1774
1080.0	149.8	0.1774	1186.0	149.8	0.1774	1292.0	149.8	0.1774
1082.0	149.8	0.1774	1188.0	149.8	0.1774	1294.0	149.8	0.1774
1084.0	149.8	0.1774	1190.0	149.8	0.1774	1296.0	149.8	0.1774
1086.0	149.8	0.1774	1192.0	149.8	0.1774	1298.0	149.8	0.1774
1088.0	149.8	0.1774	1194.0	149.8	0.1774	1300.0	149.8	0.1774
1090.0	149.8	0.1774	1196.0	149.8	0.1774	1302.0	149.8	0.1774
1092.0	149.8	0.1774	1198.0	149.8	0.1774	1304.0	149.8	0.1774
1094.0	149.8	0.1774	1200.0	149.8	0.1774	1306.0	149.8	0.1774
1096.0	149.8	0.1774	1202.0	149.8	0.1774	1308.0	149.8	0.1774
1098.0	149.8	0.1774	1204.0	149.8	0.1774	1310.0	149.8	0.1774
1100.0	149.8	0.1774	1206.0	149.8	0.1774	1312.0	149.8	0.1774
1102.0	149.8	0.1774	1208.0	149.8	0.1774	1314.0	149.8	0.1774
1104.0	149.8	0.1774	1210.0	149.8	0.1774	1316.0	149.8	0.1774
1106.0	149.8	0.1774	1212.0	149.8	0.1774	1318.0	149.8	0.1774
1108.0	149.8	0.1774	1214.0	149.8	0.1774	1320.0	149.8	0.1774
1110.0	149.8	0.1774	1216.0	149.8	0.1774	1322.0	149.8	0.1774
1112.0	149.8	0.1774	1218.0	149.8	0.1774	1324.0	149.8	0.1774
1114.0	149.8	0.1774	1220.0	149.8	0.1774	1326.0	149.8	0.1774
1116.0	149.8	0.1774	1222.0	149.8	0.1774	1328.0	149.8	0.1774
1118.0	149.8	0.1774	1224.0	149.8	0.1774	1330.0	149.8	0.1774
1120.0	149.8	0.1774	1226.0	149.8	0.1774	1332.0	149.8	0.1774
1122.0	149.8	0.1774	1228.0	149.8	0.1774	1334.0	149.8	0.1774
1124.0	149.8	0.1774	1230.0	149.8	0.1774	1336.0	149.8	0.1774
1126.0	149.8	0.1774	1232.0	149.8	0.1774	1338.0	149.8	0.1774
1128.0	149.8	0.1774	1234.0	149.8	0.1774	1340.0	149.8	0.1774
1130.0	149.8	0.1774	1236.0	149.8	0.1774	1342.0	149.8	0.1774
1132.0	149.8	0.1774	1238.0	149.8	0.1774	1344.0	149.8	0.1774
1134.0	149.8	0.1774	1240.0	149.8	0.1774	1346.0	149.8	0.1774
1136.0	149.8	0.1774	1242.0	149.8	0.1774	1348.0	149.8	0.1774
1138.0	149.8	0.1774	1244.0	149.8	0.1774	1350.0	149.8	0.1774
1140.0	149.8	0.1774	1246.0	149.8	0.1774	1352.0	149.8	0.1774
1142.0	149.8	0.1774	1248.0	149.8	0.1774	1354.0	149.8	0.1774
1144.0	149.8	0.1774	1250.0	149.8	0.1774	1356.0	149.8	0.1774
1146.0	149.8	0.1774	1252.0	149.8	0.1774	1358.0	149.8	0.1774
1148.0	149.8	0.1774	1254.0	149.8	0.1774	1360.0	149.8	0.1774
1150.0	149.8	0.1774	1256.0	149.8	0.1774	1362.0	149.8	0.1774
1152.0	149.8	0.1774	1258.0	149.8	0.1774	1364.0	149.8	0.1774
1154.0	149.8	0.1774	1260.0	149.8	0.1774	1366.0	149.8	0.1774
1158.0	149.8	0.1774	1262.0	149.8	0.1774	1368.0	149.8	0.1774
1160.0	149.8	0.1774	1264.0	149.8	0.1774	1370.0	149.8	0.1774
1162.0	149.8	0.1774	1268.0	149.8	0.1774	1372.0	149.8	0.1774
1164.0	149.8	0.1774	1270.0	149.8	0.1774	1374.0	149.8	0.1774
1166.0	149.8	0.1774	1272.0	149.8	0.1774	1376.0	149.8	0.1774
						1378.0	149.8	0.1774
						1380.0	149.8	0.1774

TABLE 6.2.1-62 (SHEET 7 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)	Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
1382.0	149.8	0.1774	1488.0	149.8	0.1774	1594.0	149.8	0.1774
1384.0	149.8	0.1774	1490.0	149.8	0.1774	1596.0	149.8	0.1774
1386.0	149.8	0.1774	1492.0	149.8	0.1774	1598.0	149.8	0.1774
1388.0	149.8	0.1774	1494.0	149.8	0.1774	1600.0	149.8	0.1774
1390.0	149.8	0.1774	1496.0	149.8	0.1774	1602.0	149.8	0.1774
1392.0	149.8	0.1774	1498.0	149.8	0.1774	1604.0	149.8	0.1774
1394.0	149.8	0.1774	1500.0	149.8	0.1774	1606.0	149.8	0.1774
1396.0	149.8	0.1774	1502.0	149.8	0.1774	1608.0	149.8	0.1774
1398.0	149.8	0.1774	1504.0	149.8	0.1774	1610.0	149.8	0.1774
1400.0	149.8	0.1774	1506.0	149.8	0.1774	1612.0	149.8	0.1774
1402.0	149.8	0.1774	1508.0	149.8	0.1774	1614.0	149.8	0.1774
1404.0	149.8	0.1774	1510.0	149.8	0.1774	1616.0	149.8	0.1774
1406.0	149.8	0.1774	1512.0	149.8	0.1774	1618.0	149.8	0.1774
1408.0	149.8	0.1774	1514.0	149.8	0.1774	1620.0	149.8	0.1774
1410.0	149.8	0.1774	1516.0	149.8	0.1774	1622.0	149.8	0.1774
1412.0	149.8	0.1774	1518.0	149.8	0.1774	1624.0	149.8	0.1774
1414.0	149.8	0.1774	1520.0	149.8	0.1774	1626.0	149.8	0.1774
1416.0	149.8	0.1774	1522.0	149.8	0.1774	1628.0	149.8	0.1774
1418.0	149.8	0.1774	1524.0	149.8	0.1774	1630.0	149.8	0.1774
1420.0	149.8	0.1774	1526.0	149.8	0.1774	1632.0	149.8	0.1774
1422.0	149.8	0.1774	1528.0	149.8	0.1774	1634.0	149.8	0.1774
1424.0	149.8	0.1774	1530.0	149.8	0.1774	1636.0	149.8	0.1774
1426.0	149.8	0.1774	1532.0	149.8	0.1774	1638.0	149.8	0.1774
1428.0	149.8	0.1774	1534.0	149.8	0.1774	1640.0	149.8	0.1774
1430.0	149.8	0.1774	1536.0	149.8	0.1774	1642.0	149.8	0.1774
1432.0	149.8	0.1774	1538.0	149.8	0.1774	1644.0	149.8	0.1774
1434.0	149.8	0.1774	1540.0	149.8	0.1774	1646.0	149.8	0.1774
1436.0	149.8	0.1774	1542.0	149.8	0.1774	1648.0	149.8	0.1774
1438.0	149.8	0.1774	1544.0	149.8	0.1774	1650.0	149.8	0.1774
1440.0	149.8	0.1774	1546.0	149.8	0.1774	1652.0	149.8	0.1774
1442.0	149.8	0.1774	1548.0	149.8	0.1774	1654.0	149.8	0.1774
1444.0	149.8	0.1774	1550.0	149.8	0.1774	1656.0	149.8	0.1774
1446.0	149.8	0.1774	1552.0	149.8	0.1774	1658.0	149.8	0.1774
1448.0	149.8	0.1774	1554.0	149.8	0.1774	1660.0	149.8	0.1774
1450.0	149.8	0.1774	1556.0	149.8	0.1774	1662.0	149.8	0.1774
1452.0	149.8	0.1774	1558.0	149.8	0.1774	1664.0	149.8	0.1774
1454.0	149.8	0.1774	1560.0	149.8	0.1774	1666.0	149.8	0.1774
1456.0	149.8	0.1774	1562.0	149.8	0.1774	1668.0	149.8	0.1774
1458.0	149.8	0.1774	1564.0	149.8	0.1774	1670.0	149.8	0.1774
1460.0	149.8	0.1774	1566.0	149.8	0.1774	1672.0	149.8	0.1774
1462.0	149.8	0.1774	1568.0	149.8	0.1774	1674.0	149.8	0.1774
1464.0	149.8	0.1774	1570.0	149.8	0.1774	1676.0	149.8	0.1774
1466.0	149.8	0.1774	1572.0	149.8	0.1774	1678.0	149.8	0.1774
1468.0	149.8	0.1774	1574.0	149.8	0.1774	1680.0	149.8	0.1774
1470.0	149.8	0.1774	1576.0	149.8	0.1774	1682.0	149.8	0.1774
1472.0	149.8	0.1774	1578.0	149.8	0.1774	1684.0	149.8	0.1774
1474.0	149.8	0.1774	1580.0	149.8	0.1774	1686.0	149.8	0.1774
1476.0	149.8	0.1774	1582.0	149.8	0.1774	1688.0	149.8	0.1774
1478.0	149.8	0.1774	1584.0	149.8	0.1774	1690.0	149.8	0.1774
1480.0	149.8	0.1774	1586.0	149.8	0.1774	1692.0	149.8	0.1774
1482.0	149.8	0.1774	1588.0	149.8	0.1774	1694.0	149.8	0.1774
1484.0	149.8	0.1774	1590.0	149.8	0.1774	1696.0	149.8	0.1774
1486.0	149.8	0.1774	1592.0	149.8	0.1774	1698.0	149.8	0.1774
						1700.0	149.8	0.1774

TABLE 6.2.1-62 (SHEET 8 OF 8)

Time (s)	Mass Flowrate lbm/s	Energy Flowrate Btu/s (E+06)
1702.	149.8	0.1774
1704.	149.8	0.1774
1706.	149.8	0.1774
1708.	149.8	0.1774
1710.	149.8	0.1774
1712.	149.8	0.1774
1714.	149.8	0.1774
1716.	149.8	0.1774
1718.	149.8	0.1774
1720.	149.8	0.1774
1722.	149.8	0.1774
1724.	149.8	0.1774
1726.	149.8	0.1774
1728.	149.8	0.1774
1730.	149.8	0.1774
1732.	149.8	0.1774
1734.	149.8	0.1774
1736.	149.8	0.1774
1738.	149.8	0.1774
1740.	149.8	0.1774
1742.	149.8	0.1774
1744.	149.8	0.1774
1746.	149.8	0.1774
1748.	149.8	0.1774
1750.	149.8	0.1774
1752.	149.8	0.1774
1754.	149.8	0.1774
1756.	149.8	0.1774
1758.	149.8	0.1774
1760.	149.8	0.1774
1762.	149.8	0.1774
1764.	149.8	0.1774
1766.	149.8	0.1774
1768.	149.8	0.1774
1770.	149.8	0.1774
1772.	149.8	0.1774
1774.	149.8	0.1774
1776.	149.8	0.1774
1778.	149.8	0.1774
1780.	149.8	0.1774
1782.	149.8	0.1774
1784.	149.8	0.1774
1786.	149.8	0.1774
1788.	149.8	0.1774
1790.	149.8	0.1774
1792.	149.8	0.1774
1794.	149.8	0.1774
1796.	149.8	0.1774
1798.	149.8	0.1774
1800.	149.8	0.1774
1802.	150.4	0.1781
1804.	154.4	0.1824
1806.	118.5	0.1395
1808.	102.4	0.1204
1810.	89.18	0.1045
1812.	79.05	0.0925
1814.	70.62	0.0824
1816.	52.93	0.0617
1818.	8.932	0.0104
1820.	1.206	0.0014
1822.	0.1610	0.0002
1824.	0.0214	0.0000
1826.	0.0028	0.0000
1828.	0.0004	0.0000
1830.	0.0001	0.0000
1832.	0.0000	0.0000
1834.	0.0000	0.0000
1836.	0.0000	0.0000
1838.	0.0000	0.0000
1840.	0.0000	0.0000
1842.	0.0000	0.0000
1844.	0.0000	0.0000
1846.	0.0000	0.0000
1848.	0.0000	0.0000
1850.	0.0000	0.0000
1852.	0.0000	0.0000
1854.	0.0000	0.0000
1856.	0.0000	0.0000
1858.	0.0000	0.0000
1860.	0.0000	0.0000
1862.	0.0000	0.0000
1864.	0.0000	0.0000
1866.	0.0000	0.0000
1868.	0.0000	0.0000
1870.	0.0000	0.0000
1872.	0.0000	0.0000
1874.	0.0000	0.0000
1876.	0.0000	0.0000
1878.	0.0000	0.0000
1880.	0.0000	0.0000
1882.	0.0000	0.0000
1884.	0.0000	0.0000
1886.	0.0000	0.0000
1888.	0.0000	0.0000
1890.	0.0000	0.0000
1892.	0.0000	0.0000
1894.	0.0000	0.0000
1896.	0.0000	0.0000
1898.	0.0000	0.0000
1900.	0.0000	0.0000

TABLE 6.2.1-63

DELETED.

TABLE 6.2.1-64 (SHEET 1 OF 3)

SPECIFIC PLANT DESIGN INPUT FOR MSLB ANALYSIS

		1	2	3	4	5	6
Initial steam generator inventory (lbm)	Faulted	115,000	129,000	154,700	186, 300	114,500	129,000
	Intact	107,700	123,500	142,700	186, 300	107,500	123,500
Initial steam pressure (psia)		1003	1051	1112	1102	1003	1051
Mass added by feedwater pumping (lbm)		15,900	7,800	4,500	9,600	14,000	10,900
Mass added by feedwater flashing (lbm)		20,800	21,500	22,500	0	20,800	21,500
Unisolatable steam line volume (ft^3)		470	470	470	470	470	470
Auxiliary feedwater addition rate (lbm/h) (to faulted steam generator)		5.48E+05	5.48E+05	5.48E+05	5.48E+05	5.48E+05	5.48E+05
Main steam line isolation time (s)		11.2	11.0	10.9	10.9	11.4	11.8
Main feedwater line isolation time (s)		8.2	8.2	7.9	7.9	8.4	8.8
Termination of auxiliary feedwater addition (s)		1800	1800	1800	1800	1800	1800

TABLE 6.2.1-64 (SHEET 2 OF 3)

Case		7	8	9	10	11	12
Initial steam generator inventory (lbm)	Faulted	154,700	186,400	115,000	129,000	154,700	186,400
	Intact	142,800	154,500	107,700	123,500	142,800	154,500
Initial steam pressure (psia)		1112	1102	1003	1051	1112	1102
Mass added by feedwater pumping (lbm)		11,200	23,300	16,000	18,700	11,600	29,200
Mass added by feedwater flashing (lbm)		22,600	24,900	20,800	21,500	22,600	24,900
Unisolatable steamline volume (ft^3)		470	470	470	470	470	470
Auxiliary feedwater addition rate (lbm/h)		5.48E+05	5.48e+05	5.48E+05	5.48E+05	5.48E+05	5.48E+05
Main steam line isolation time (s)		61.6	280.9	14.2	63.1	92.1	188.1
Main feedwater line isolation signal reached (s)		16.4	112.7	11.2	18.3	22.2	185.1
Termination of auxiliary feedwater addition (s)		1800	1800	1800	1800	1800	1800

TABLE 6.2.1-64 (SHEET 3 OF 3)

Case		13	14	15	16
Initial steam generator inventory (lbm)	Faulted	116,000	130,100	155,900	186,300
	Intact	107,700	123,500	142,800	186,300
Initial steam pressure (psia)		1008	1051	1112	1102
Mass added by feedwater pumping (lbm)		25,100	17,600	10,700	4800
Mass added by feedwater flashing (lbm)		20,800	21,500	22,600	0
Unisolatable steam line volume (ft^3)		470	470	470	470
Auxiliary feedwater addition rate (lbm/h)		5.41E+05	5.41E+05	5.41E+05	5.47E+05
Main steam line isolation time (s)		50.1	48.0	46.5	114.0
Main feedwater line isolation time (s)		17.1	15.5	14.3	23.8
Termination of auxiliary feedwater addition (s)		1800	1800	1800	1800

TABLE 6.2.1-65

SUMMARY OF RESULTS FOR MSLB CONTAINMENT PRESSURE - TEMPERATURE ANALYSIS

Case No.	Power [(a)] Level (%)	Break Size (ft^2)	Break Type	Max. Press. at *Time (psig at Time)	Max. Vapor Temp *at Time (°F at s)	Dryout Time (s)	6.29 psig *at Time (s)	17.19 psig *at Time (s)	25.79 psig *at Time (s)
1	102	Full	Double ended	27.6 at 124	232 at 125	1822	2.9	28.6	94.5
2	70	Full	Double ended	27.5 at 1809	232 at 157	1822	2.4	27.5	107.6
3	30	Full	Double ended	30.4 at 197	238 at 197	1834	2.0	31.3	129.0
4	0	Full	Double ended	30.2 at 173	238 at 173	1808	1.6	10.5	109.0
5	102	0.60	Double ended	27.1 at 1808	271 at 107	1844	6.4	61.1	201.0
6	70	0.53	Double ended	28.3 at 1833	252 at 107	1864	6.8	80.3	409.2
7	30	0.36	Double ended	32.4 at 1820	260 at 95	1912	9.4	52.1	537.4
8	0	0.20	Double ended	22.9 at 6339	226 at 119	6283	17.6	133.1	-
9	102	0.33	Double ended	28.5 at 1831	249 at 111	1898	11.0	136.5	854.1
10	70	0.32	Double ended	30.8 at 1836	273 at 71	1920	11.2	53.6	442.9
11	30	0.22	Double ended	32.7 at 2177	262 at 97	2155	15.1	83.7	1085.7
12	0	0.10	Double ended	17.0 at 8603	220 at 135	8600	33.7	-	-
13	102	0.86	Split	31.9 at 190	303 at 109	1832	10.1	40.4	89.7
14	70	0.908	Split	31.3 at 226	301 at 109	1834	8.5	38.2	93.8
15	30	0.944	Split	31.8 at 327	299 at 107	1856	7.3	36.7	96.7
16	0	0.40	Split	36.5 at 898	260 at 117	1842	16.8	103.8	484.6

a. % Power Level of 3579 MWt.

TABLE 6.2.1-66

SEQUENCE OF EVENTS FOR CASE 16 -
PEAK CALCULATED CONTAINMENT PRESSURE FOR MSLB

Time (s)	Event
0.0	Break occurs, blowdown from all steam generators.
16.8	Containment pressure setpoint for isolation of main feedwater lines reached (6.29 psig).
23.8	Main feedwater line isolation valves closed.
103.8	Containment pressure setpoint for isolation of main steam lines reached (17.19 psig).
114	Main steam line isolation valves closed, blowdown from broken loop steam generator and unisolated steam piping only.
116.8	Air coolers start.
117.0	Peak containment vapor temperature of 260°F is reached.
484.6	Containment pressure setpoint for actuation of containment sprays reached (25.79 psig).
584.6	Containment sprays start.
897.6	Peak containment pressure of 36.5 psig is reached.
1800.0	Auxiliary feedwater addition is terminated.
1842	Dryout occurs, steam generator dry following auxiliary feedwater termination.

TABLE 6.2.1-67

SEQUENCE OF EVENTS FOR CASE 13 -
PEAK CALCULATED CONTAINMENT TEMPERATURE FOR MSLB

Time (s)	Event
0.0	Break occurs, blowdown from all four steam generators.
10.1	Containment pressure setpoint for isolation of main feedwater lines reached (6.29 psig).
22.0	Main feedwater line isolation valves closed.
110.2	Air coolers start.
40.4	Containment pressure setpoint for isolation of main steam lines reached (17.19 psig).
50.1	Main steam line isolation valves closed, blowdown from broken loop steam generator and unisolated steam piping only.
89.7	Containment pressure setpoint for actuation of containment sprays reached (25.79 psig).
109.0	Peak containment vapor temperature of (303.1°F) is reached.
190.3	Containment sprays start.
190.4	Peak containment pressure of 31.9 psig is reached.
1800.0	Auxiliary feedwater addition is terminated.
1832.0	Dryout occurs, steam generator dry following auxiliary feedwater termination.

TABLE 6.2.1-68

BLOWDOWN MASS/ENERGY RELEASES FOR LIMITING CASES
DECLG (C_D=0.6, Low T_{AVG})

Time (s)	Mass Flow Rate (lbm/s)	Energy Release Rate (Btu/s)
0	65184	34575122
1	65184	34575122
2	56193	30344429
3	41256	22657374
4	35492	19926677
5	28425	16498775
6	27812	16227833
7	25996	15413707
8	23030	14325843
9	19958	12961246
10	17589	11668789
11	15609	10428740
12	14522	9583417
13	13691	8944095
14	12766	8368203
15	11960	7867605
16	10654	7135285
17	8188	5941968
18	6797	5107798
19	6620	4742754
20	7356	4747262
21	7008	4062956
22	6965	3625279
23	6952	3234900
24	6121	2631796
25	4973	1945657
26	4443	1644792
27	3827	1333195
28	3141	1068429
29	2673	855329
30	2296	700974
31	1928	548382
32	1605	458978
33	1379	324043
33.39	878	186500

TABLE 6.2.1-69

REFLOOD MASS/ENERGY RELEASES FOR LIMITING CASE
DECLG (C_D=0.6, Low T_{AVG}, MIN SI)

Time (s)	Mass Flow Rate (lbm/s)	Energy Release Rate (Btu/s)
45.9	35	37993
55.9	72	82068
65.9	105	122137
75.9	122	144865
85.9	120	144282
95.9	119	143294
120.9	160	156786
145.9	324	195848
170.9	404	221510
195.9	406	221418
220.9	398	224504
245.9	389	230303
270.9	375	225886
295.9	378	226935

TABLE 6.2.1-70

BROKEN LOOP INJECTION SPILL DURING BLOWDOWN
(C_D=0.6, LOW T_{AVG})

Time (s)	Mass Flow Rate (lbm/s)
0	3138
1	2995
2	2798
3	2637
4	2503
5	2388
6	2288
7	2201
8	2122
9	2051
10	1988
11	1930
12	1877
13	1828
14	1783
15	1741
16	1702
17	1665
18	1631
19	1599
20	1569
21	1541
22	1514
23	1489
24	1466
25	1444
26	1423
27	1402
28	1383
29	1365
30	1347

TABLE 6.2.1-71

ACTIVE HEAT SINKS FOR MINIMUM
CONTAINMENT PRESSURE ANALYSIS

Containment Spray Parameters

Number of pumps operating	2
Maximum Spray Flow (gal/min)	6748
Fastest post-LOCA initiation of spray pumps, assuming offsite power loss and no diesel failure(s)	74.0

Containment Fan Coolers

Number of fan coolers	8
Maximum CCWS flow (gal/min)	8975
Maximum NSCW water flow (gal/min)	9625
Fastest post-LOCA initiation of fan coolers assuming offsite power loss and no diesel failure(s)	41.1

TABLE 6.2.1-72 (SHEET 1 OF 2)

PASSIVE HEAT SINKS[a]

Wall	Material	Thermal Conductivity (Btu/h-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)	Thickness (ft)	Area (ft^2)
1	Epoxy	3.5	20.0	0.00025	32340
	Zinc coating	1.5	20.0	0.00020833	
	Steel	27.0	58.8	0.0208333	
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	5.5	
2	Zinc Coating	1.5	20.0	0.00020833	72043
	Steel	27.0	58.8	0.020833	
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	5.5	
3	Steel	27.0	58.8	0.015833	11442
	Epoxy	3.5	20.0	0.001125	
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	6.3333	
4	Epoxy	3.5	20.0	0.0015417	15086.5
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	2.25	
	Steel	27.0	58.8	0.020833	
	Concrete	0.92	22.62	10.5	
5	Zinc Coating	1.5	20.0	0.00020833	4957
	Steel	27.0	58.8	0.020833	
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	7.5	
6	Zinc Coating	1.5	20.0	0.00020833	595.7
	Steel	27.0	58.8	0.125	
7	Galvanization	65.0	41.0	0.00020833	316975
	Steel	27.0	58.8	0.004375	
8	Zinc Coating	1.5	20.0	0.00020833	159120.9
	Steel	27.0	58.8	0.055	
9	Epoxy	3.5	20.0	0.00025	30030
	Zinc Coating	1.5	20.0	0.00020833	
	Steel	27.0	58.8	0.020833	

TABLE 6.2.1-72 (SHEET 2 OF 2)

Wall	Material	Thermal Conductivity (Btu/h-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)	Thickness (ft)	Area (ft^2)
10	Zinc Coating	1.5	20.0	0.00020833	159134
	Steel	27.0	58.8	0.020833	
11	Zinc coating	1.5	20.0	0.00020833	101266.4
	Steel	27.0	58.8	0.041667	
12	Zinc	1.5	20.0	0.00020833	30474.2
	Coating	27.0	58.8	0.10417	
	Steel				
13	Steel	27.0	58.8	0.0083333	75041
14	Steel	27.0	58.8	0.09375	232
15	Steel	27.0	58.8	0.04425	1259.4
16	Epoxy	3.5	20.0	0.001125	44202
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	6.8333333	
17	Epoxy	3.5	20.0	0.000875	19550.3
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	3.5	
18	Zinc	1.5	20.0	0.00020833	1546.8
	Coating	27.0	58.8	0.083333	
	Steel	0.92	22.62	0.5	
	Concrete	0.92	22.62	4.0	
	Concrete				
19	Epoxy	3.5	20.0	0.001125	2215.4
	Concrete	0.92	22.62	0.5	
	Concrete	0.92	22.62	4.0	
20[(a)]	Steel	27.0	58.8	0.0083333	100000

a. Wall 20 is the additional 357,900 lbm of metalmass allowance in the containment which is documented in Westinghouse SECL-99-021, Rev. 2. The effects of this are included in the LB LOCA PCT in table 15.6.5-4.

TABLE 6.2.1-73

FORCES AND MOMENTS FOR BROKEN COLD LEG WITH SUPPORT

	Elevation/Nodes					
	187.00'	190.56'	184.64'	179.06'	172.61'	Total
Loads	1-16	17-24	25-32	33-40	41-48	1-48
F_x(E-W)(lbf)	$-6.76x10^5$	$-1.10x10^6$	$-1.09x10^6$	$-1.02x10^6$	$-4.96x10^5$	$-41.9x10^5$
F_y(N-S)(lbf)	$-2.59x10^5$	$-5.54x10^5$	$-5x42x10^5$	$-5.61x10^5$	$-2.43x10^5$	$-20.6x10^5$
$F_{Lateral}$(lbf)	$7.24x10^5$	$1.23x10^6$	$1.22x10^6$	$1.63x10^6$	$5.52x10^5$	$46.7x10^5$
M_x(ft-lbf)	0	$-1.94x10^6$	$1.27x10^6$	$4.49x10^6$	$3.50x10^6$	$77.3x10^5$
M_y(ft-lbf)	0	$3.84x10^6$	$-2.56x10^6$	$-8.15x10^6$	$-7.14x10^6$	$-149x10^5$

TABLE 6.2.1-74

UPLIFT FORCES AND MOMENTS FOR BROKEN COLD LEG WITH SUPPORT

	Loads	
Nodes	F_Z (lbf)	M_Z (ft-lbf)
1-16	0	-1.27×10^5
17-24	8.45×10^5	-1.40×10^3
25-32	-4.76×10^4	-1.40×10^3
49	4.71×10^5	0
50	2.62×10^5	0
Total	7.40×10^5	-1.23×10^5

TABLE 6.2.2-1

CONTAINMENT FAN COOLER DESIGN DATA

Characteristic	Quantity
Number of units	8
Containment atmosphere design inlet conditions, post LOCA	
Design temperature, saturation (°F)	270
Design pressure (psia)	60
Steam partial pressure (psia)	42
Air partial pressure (psia)	18
Cooling water (nuclear service cooling water)	
Minimum flowrate (gal/min/unit)	700
nlet temperature post-LOCA (°F)	95
Peak outlet temperature (°F)	257
Design post-LOCA heat removal rate at 270°F containment temperature (Btu/h/unit)	56.3×10^6
Fan flowrate (actual ft^3/min)	43,500 (slow speed)
Static head (in. WG)	0.85
Motor horsepower	62.5

TABLE 6.2.2-2

CONTAINMENT FAN COOLING HEAT REMOVAL
CAPACITY (POST-LOCA MODES)

(NSCW Temperature 40°F)

Containment Temperature (°F)	Capacity (Btu/h)
120	16.39E+06
150	25.92E+06
180	39.65E+06
210	53.57E+06
240	66.09E+06
270	77.97E+06

(NSCW Temperature 95°F)

Containment Temperature (°F)	Capacity (Btu/h)
120	5.80E+06
145	12.96E+06
170	21.23E+06
195	29.92E+06
220	38.76E+06
245	47.54E+06
270	56.30E+06

TABLE 6.2.2-3 (SHEET 1 OF 5)

CONTAINMENT COOLING SYSTEM FAILURE MODE AND EFFECTS ANALYSIS

Item No.	Description of Component	Safety Function	Plant Operating Mode[(a)]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
1	Breakers for high-speed operation) on 1AB04, 480-V switchgear, 1E bus, train A, normally closed (NC) No. 04 breaker for A7-001 motor and fan. No. 08 breaker for A7-002 motor and fan. No. 12 breaker for A7-005 motor and fan. No. 16 breaker for A7-006 motor and fan.	Provide continuity and protection for item 3 motor and fan	A	Inadvertent open, one breaker	Switchgear alarm Motor indicating lights	None; loss of train A;[(b)] train B available.
2	Breakers (for low-speed operation) on 1AB04, 480-V switchgear, 1E bus, train A, normally open (NO) No. 05 breaker for A7-001 motor and fan. No. 09 breaker for A7-002 motor and fan. No. 13 breaker for A7-005 motor and fan. No. 17 breaker for A7-006 motor and fan.	Provide continuity and protection for item 3 motor and fan	A	Inadvertent open, one breaker	Switchgear alarm Motor indicating lights	None; loss of train A; train B available
			B	Fail to close, one breaker	Sequencer alarm Motor indicating lights	None; loss of train A; train B available
3	Containment building (CTB) cooling unit motor and fan, train A, normally energized (NE) 1-1501-A7-001-M01 1-1501-A7-002-M01 1-1501-A7-005-M01 1-1501-A7-006-M01	Provide motive power for circulating air in the CTB	A	Fail to operate, one motor and fan	(C)	None; loss of train A; train B available
			B	Fail to restart and operate, one motor and fan	(C)	None; loss of train A; train B available

TABLE 6.2.2-3 (SHEET 2 OF 5)

Item No.	Description of Component	Safety Function	Plant Operating Mode[(a)]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
4	Breakers, 480-V MCC, 1E bus, train A, NC No. 26 breaker on 1ABE for A7-001 damper (HV2582A) No. 27 breaker on 1ABE for A7-002 damper (HV2582B) No. 07 breaker on 1ABC for A7-005 damper (HV2584A) No. 08 breaker on 1ABC for A7-006 damper (HV2584B)	Provide continuity and protection for item 6, damper	A	Inadvertent open, one breaker	Motor control center (MCC) alarm Position indicating lights	None; no loss of train A. NO damper will remain open.
			B	Inadvertent open, one breaker	MCC alarm Position indicating lights	None; no loss of train A. NO damper will remain open.
5	Motor starters for item 6, train A NO No. 26 motor starter on 1ABE for A7-001 damper No. 27 motor starter on 1ABE for A7-002 damper No. 07 motor starter on 1ABC for A7-005 damper No. 08 motor starter on 1ABC for A7-006 damper	Provide continuity to one damper, item 6	A	Inadvertent closed, one motor starter	Position indicating lights	None; loss of train A; train B available
			B	Fail to close	Position indicating lights	None; no loss of train A. NO damper will remain open.
6	Motor-operated on-off dampers, train A, NO HV2582A damper on 1ABE for A7-001 HV2582B damper on 1ABE for A7-002 HV2584A damper on 1ABC for A7-005 HV2584B damper on 1ABC for A7-006	Allow flow of air to the containment building and prevent backflow	A	Inadvertent closed, one damper	Position indicating lights	None; loss of train A; train B available
			B	Inadvertent closed, one damper	Position indicating lights	None; loss of train A; train B available

TABLE 6.2.2-3 (SHEET 3 OF 5)

Item No.	Description of Component	Safety Function	Plant Operating Mode[(a)]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
7	001, 003 backflow damper, train A, NO	Allow flow of air to the containment building and prevent backflow	A	Inadvertent close, one damper		None; loss of train A; train B available.
			B	Fail to close, one damper	(C)	None; no loss of train A. NO damper will remain open.
				Fail to reopen, one damper		None; loss of train A; train B available.
8	Breakers, (for high speed operation) on 1BB06, 480-V switchgear, 1E bus, train B, NC No. 04 breaker for A7-003 motor and damper No. 08 breaker for A7-004 motor and damper No. 12 breaker for A7-007 motor and damper No. 16 breaker for A7-008 motor and damper	Provide continuity and protection for item 10 motor and fan	A	Inadvertent open, one breaker	Switchgear alarm Motor indicating lights	None; loss of train B; train A available.
9	Breakers, (for low-speed operation) on 1BB06, 480-V switchgear, 1E bus, train B NO No. 05 breaker for A7-003 motor and damper No. 09 breaker for A7-004 motor and damper No. 13 breaker for A7-007 motor and damper No. 17 breaker for A7-008 motor and damper	Provide continuity and protection for item 10	A	Inadvertent open, one breaker	Switchgear alarm Motor indicating lights	None; loss of train B; train A available.
			B	Fail to close, one breaker	Sequencer alarm Motor indicating lights	None; loss of train B; train A available.

TABLE 6.2.2-3 (SHEET 4 OF 5)

Item No.	Description of Component	Safety Function	Plant Operating Mode[a]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
10	CTB cooling unit motor and fan, train B, NE 1-1501-A7-003-M01 1-1501-A7-004-M01 1-1501-A7-007-M01 1-1501-A7-008-M01	Provide motive power for circulating air in the containment building	A	Fail to operate, one motor and fan	(C)	None; loss of train B; train A available
			B	Fail to restart and operate, one motor and fan	(C)	None; loss of train B; train A available
11	Breakers, 480-V MCC, 1E bus, train B, NC No. 26 breaker on 1BBE for A7-003 damper (HV2583A) No. 27 breaker on 1BBE for A7-004 damper (HV2583B) No. 07 breaker on 1BBC for A7-007 damper (HV2585A) No. 08 breaker on 1BBC for A7-008 damper (HV2585B)	Provide continuity and protection for item 13 damper	A	Inadvertent open, one breaker	MCC alarm Position indicating lights	None; no loss of train B. NO damper will remain open.
			B	Inadvertent open, one breaker	Switchgear alarm Position indicating lights	None, no loss of train B. NO damper will remain open.
12	Motor starter for item 13, train B NO No. 26 motor starter on 1BBE for A7-003 damper No. 27 motor starter on 1BBE for A7-004 damper No. 07 motor starter on 1BBC for A7-007 damper No. 08 motor starter on 1BBC for A7-008 damper	Provide continuity to one damper, item 13	A	Inadvertent close, one motor starter	Position indicating lights	None; loss of train B; train A available
			B	Fail to close	Position indicating lights	None; no loss of train B. NO damper will remain open.
13	Motor-operated on-off dampers, train B, NO HV2583A damper on 1BBE for A7-003 damper HV2583B damper on 1BBE for A7-004 damper HV2585A damper on 1BBC for A7-007 damper HV2585B damper on 1BBC for A7-008 damper	Allow flow of air to the containment building and prevent backflow	A	Inadvertent close, one damper	Position indicating lights	None; loss of train B; train A available
			B	Inadvertent close, one damper	Position indicating lights	None; loss of train B; train A available

TABLE 6.2.2-3 (SHEET 5 OF 5)

Item No.	Description of Component	Safety Function	Plant Operating Mode[a]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
14	002, 004 backflow damper train B, NO	Allow flow of air to the containment building and prevent backflow	A	Inadvertent close, one damper		None; loss of train B; train A available.
			B	Fail to close, one damper	(C)	None; no loss of train B; NO damper will remain open.
				Fail to reopen, one damper		None; loss of train B; train A available.
15	Fans and dampers for train A: 1-1501-A7-001-000 1-1501-A7-002-000 1-1501-A7-005-000 1-1501-A7-006-000	Provide circulation of air inside containment building	A	Mechanical failure	Motor indicating light	None, loss of train A; train B available
16	Fans and dampers for train B: 1-1501-A7-003-000 1-1501-A7-004-000 1-1501-A7-007-000 1-1501-A7-008-000	Provide circulation of air inside containment building	A	Mechanical failure	Motor indicating light	None; loss of train B; train A available
17	Cooling coils for train A: 1-1501-A7-001-000 1-1501-A7-002-000 1-1501-A7-005-000 1-1501-A7-006-000	Provide cooling and heat removal inside containment building	A	Leakage in cooling coil	Flow indication Temperature alarm high Leak detector level alarm	None; loss of train A; train B available
18	Cooling coils for train B: 1-1501-A7-003-000 1-1501-A7-004-000 1-1501-A7-007-000 1-1501-A7-008-000	Provide cooling and heat removal inside containment building	A	Leakage in cooling coil	Flow indication Temperature alarm high Leak detector level alarm	None; loss of train B; train A available

a. A - Normal, four fans in train A or four fans in train B operating; B - safety injection, same as normal except all fans are sequenced at low speed. Four fans required (load shed occurs only under loss of offsite power).
b. Loss of one breaker or fan degrades the performance of that train. The other train remains available.
c. Flow indication inferred by motor indicating lights and switchgear alarm.

TABLE 6.2.2-4

CONTAINMENT SPRAY SYSTEM COMPONENT DESIGN PARAMETERS

Containment Spray Pump	
Type	Horizontal centrifugal
Quantity	2
Design pressure (psig)	300
Design temperature (°F)	250
Design flowrate (gal/min)	2600
Design head (ft)	450
Material	Stainless steel
Containment Spray Nozzle	
Quantity	342[a]
Type	Spraco 1713A or equivalent
Flow per nozzle at 40 psi Δp (gal/min)	15.2
Material	Stainless steel
Refueling Water Storage Tank	
Quantity	1
Nominal volume (gal)	715,000
Boric acid concentration (ppm)	2400-2600
Design pressure	Hydraulic head
Design temperature (°F)	150
Operating pressure	Hydraulic head
Material	Stainless steel-lined concrete
Eductors[b]	
Quantity	2
Eductor inlet (spray water)	
Operating fluid	Borated water
Operating temperature	Ambient
Design flow (gal/min)	39.3
NaOH concentration (wt percent)	30-32
Design temperature (°F)	300
Design pressure (psig)	300
Material	Stainless steel

a. On Unit 1, one nozzle is currently obstructed to prevent flow through the nozzle.

b. The spray additive subsystem has been abandoned in place. The eductors currently serve only to maintain the spray system pressure boundary integrity.

TABLE 6.2.2-5 (SHEET 1 OF 2)

FAILURE MODE AND EFFECTS ANALYSIS - CONTAINMENT SPRAY SYSTEM - ACTIVE COMPONENTS

Component	Failure Mode	CS Operation Phase	Effect on System Operation[a]	Failure Detection Method[b]	Remarks
1. Motor-operated gate valve 1-9001A (1-9001B analogous)	Fails to open on demand	Containment spray injection and recirculation phases	Failure blocks flow of spray coolant to nozzles of spray header of train A of containment spray system, which reduces redundancy of spray system. No safety effect on system operation. Minimum containment spray requirements will be met by the flow of containment spray coolant from the operation of train B (train A).	Valve position indication (closed to open position change) at CB. Valve monitor light and alarm (closed position) for group monitoring of components at CB.	Valve is normally closed during power operations. Valve opens on actuation by a CS signal.
2. Containment spray pump 1 (pump 2 analogous)	Fails to deliver working fluid	Containment spray injection and recirculation phases	Failure reduces the redundancy of providing coolant spray to the containment. Fluid flow from CS pump 1 (pump 2) will be lost. Minimum flow requirements for containment spray will be met by CS pump 2(pump 1) delivering working fluid to spray header in train B (train A).	Open pump switchgear circuit breaker indication at CB. Circuit breaker close position monitor light and alarm for group monitoring of components at CB common breaker trip alarm at CB.	Pump circuit breaker is aligned to close on actuation by a CS signal.

TABLE 6.2.2-5 (SHEET 2 OF 2)

	Component	Failure Mode	CS Operation Phase	Effect on System Operation[a]	Failure Detection Method[b]	Remarks
3.	Motor-operated gate valve 1-9002A or 1-9003A (1-9002B or 1-9003B analogous)	Fails to open on demand	Containment spray recirculation phase	Failure blocks flow of coolant from the containment sump to the suction of CS pump 1 (pump 2). Coolant flow from CS pump 1 (pump 2) will be lost which reduces the redundancy of spray system. No safety effect on system operation. Minimum flow requirements for containment spray will be met by CS train B (train A).	Valve position indication (closed to open position change) at CB. Monitor light and alarm (valve open) for group monitoring of components at CB.	Valves are opened by operator during the switchover from the RWST to the containment sumps (paragraph 6.2.2.2).
4.	Motor-operated gate valve 1-9017A (1-9017B analogous)	Fails to close on demand	Containment spray recirculation phase	Failure reduces the redundancy of isolation provided to prevent coolant flow from containment sump to RWST. No safety effect on system operation. Check valve 001 (008) provides isolation.	Valve position indication) open to closed position change) at CB. Monitor light and alarm (valve closed) for group monitoring of components at CB.	Valves are closed by the operator during the switchover from the RWST to the containment sumps (paragraph 6.2.2.2).

5. Deleted.

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a. List of abbreviations and acronyms:

CB	-	Control board
CI	-	Containment isolation
CS	-	Containment spray
RWST	-	Refueling water storage tank

b. As part of plant operation, periodic tests, surveillance inspections, and instrument calibrations are made to monitor equipment and performance. Failure may be detected during such monitoring of equipment, in addition to detection methods noted.

TABLE 6.2.4-1 (SHEET 1 OF 17)

CONTAINMENT PENETRATION/ISOLATION VALVE INFORMATION

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
1	57[i]	Main steam and steam to auxiliary feed-water pump driver	Secondary coolant	29.5	Yes	1X4DB159-3	11	HV-3006A	Out	A	20'-9"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	Out
				29.5		2X4DB159-3		HV-3006B	Out		28'-4"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				8				PV-3000	Out		45'-10"	Globe	E/H	E	Auto	Remote man.	C	C	C	FC	Process/Signal	NA	A	
				6				PSV-3001	Out		20'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3002	Out		20'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3003	Out		20'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3004	Out		20'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3005	Out		20'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				4				HV-3009	Out		10'-1"	Gate	Elec. motor	E	Remote man.	None	O	O	O	FAI	Remote man.	NA	B (dc)	
				2				176	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				088	Out		7'-10"	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				356	Out		29'-8"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-192	Out		28'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				HV-13005B	Out		42'-4"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				1				X-117	Out		23'-3"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-119	Out		30'-10"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				HV-13005A	Out		26'-6"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	
				1				X-209[o]	Out		10'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-211	Out		45'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-429[p]	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-438	Out		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
2	57[i]	Main steam and steam to auxiliary feed-water pump driver	Secondary coolant	29.5	Yes	1X4DB159-3	11	HV-3016A	Out	A	28'-4 9/16"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	Out
				29.5		2X4DB159-3		HV-3016B	Out		36'-11 9/16"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				8				PV-3010	Out		16'-3"	Globe	E/H	E	Auto	Remote man.	C	C	C	FC	Process/Signal	NA	B	
				6				PSV-3011	Out		27'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3012	Out		27'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3013	Out		27'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3014	Out		27'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3015	Out		27'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				4				HV-3019	Out		5'-9"	Gate	Elec. motor	E	Remote man.	None	O	O	O	FAI	Remote man.	NA	A (dc)	
				2				178	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				082	Out		4'-7"	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-196	Out		14'-1"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-207[0]	Out		27'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-162	Out		26'-11"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-121	Out		31'-11"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-123	Out		40'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				358	Out		10'-8"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				HV-13007A	Out		51'-6"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	
				4				HV-13007B	Out		57'-10"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				1				X-215[p]	Out		46'-5"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-166	Out		26'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-430	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 2 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
3	57(l)	Main steam line	Secondary coolant	29.5	Yes	1X4DB159-1	11	HV-3026A	Out	A	27'-4 3/4"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	Out
				29.5		2X4DB159-1		HV-3026B	Out		35'-1 3/4"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				8				PV-3020	Out		15'-3"	Globe	E/H	E	Auto	Remote man.	C	C	C	FC	Process/Signal	NA	B	
				6				PSV-3021	Out		26'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3022	Out		26'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3023	Out		26'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3024	Out		26'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3025	Out		26'-9"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				2				218	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				093	Out		3'-7"	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				352	Out		13'-11"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-198	Out		26'-9"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-125	Out		30'-11"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-127	Out		39'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				HV-13008A	Out		50'-6"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	
				4				HV-13008B	Out		56'-10"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				1				X-217	Out		57-'1"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-431	In		-	Globe	manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
4	57(l)	Main steam line	Secondary coolant	29.5	Yes	1X4DB159-1	11	HV-3036A	Out	A	20'-0"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	Out
				29.5		2X4DB159-1		HV-3036B	Out		27'-4"	Gate	E/H	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				8				PV-3030	Out		49'-3"	Globe	E/H	E	Auto	Remote man.	C	C	C	FC	Process/Signal	NA	A	
				6				PSV-3031	Out		20'-9 3/4"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3032	Out		20'-9 3/4"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3033	Out		20'-9 3/4"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3034	Out		20'-9 3/4"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				6				PSV-3035	Out		20'-9 3/4"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				2				220	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				152	Out		6'-10"	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-213(o)	Out		28'-5"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-194	Out		27'-10"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				354	Out		27'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-129	Out		22'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-131	Out		29'-10"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				4				HV-13006A	Out		26'-6"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	A	
				4				HV-13006B	Out		42'-4"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	SLI	5	B	
				1				X-432	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-440(p)	Out		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
5	N/A	Eddy current/sludge lancing	N/A	0.75	NO	1X4DB159-1	56(k)	X-017	Out	B	-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	-
				10		2X4DB159-1			In		-	Flange	NA	N	NA	None	C	C	C	NA	NA	NA	NA	-
7	57(l)	Steam generator blowdown	Secondary coolant	3	No	1X4DB159-3	34	HV-7603A	Out	A	10'	Globe	Air	N	Auto	Remote man.	O	C	C	FC	AFS	15	A,B	Out
				1.5		2X4DB159-3		126	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				335	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				409	In		-	Globe	manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
8	57(l)	Steam generator blowdown	Secondary coolant	3	No	1X4DB159-3	34	HV-7603B	Out	A	1'-0"	Globe	Air	N	Auto	Remote man.	O	C	C	FC	AFS	15	A,B	Out
				1.5		2X4DB159-3		129	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				336	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-157	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-164	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				410	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 3 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
9	57[i]	Steam generator blowdown	Secondary coolant	3 1.5 0.75 1 1	No	1X4DB159-1 2X4DB159-1	34	HV-7603C 132 337 X-155 407	Out In In In In	A	1'-0" - - - -	Globe Globe Globe Globe Globe	Air Manual Manual Manual Manual	N N N N N	Auto Manual Manual Manual Manual	Remote man. None None None None	O C C C C	C C C C C	C C C C C	FC NA NA NA NA	AFS NA NA NA NA	15 NA NA NA NA	A,B NA NA NA NA	Out
10	57[i]	Steam generator blowdown	Secondary coolant	3 1.5 0.75 1	No	1X4DB159-1 2X4DB159-1	34	HV-7603D 135 338 408	Out In In In	A	1'-0" - - -	Globe Globe Globe Globe	Air Manual Manual Manual	N N N N	Auto Manual Manual Manual	Remote man. None None None	O C C C	C C C C	C C C C	FC NA NA NA	AFS NA NA NA	15 NA NA NA	A,B NA NA NA	Out
11A	54	Chemical addition	Secondary coolant chemicals	0.5 0.5 0.5	No	1X4DB159-1 2X4DB159-1	3A	HV-5280 676 081	Out In In	C	2'-0" - -	Globe Globe Globe	Air Manual Manual	N N N	Remote man. Manual Manual	None None None	C LC C	C LC C	C LC C	FC NA NA	Remote man.[h] NA NA	NA NA NA	NA NA NA	In
11B	57[i]	Steam generator secondary side sample	Secondary coolant	0.5 0.5 0.5 0.5 0.375	No	1X4DB159-3 2X4DB159-3	35	HV-9451 HV-9553A HV-9553B 047 043	Out In In Out In	A	2'-6" - - 3'-3" -	Globe Globe Globe Globe Globe	Solenoid Solenoid Solenoid Manual Manual	N N N N N	Auto Remote man. Remote man. Manual Manual	Remote man. None None None None	O C O C C	C C C C C	C C C C C	FC FC FC NA NA	AFS Remote man. Remote man. NA NA	15 NA NA NA NA	A NA NA NA NA	Out
11C	57[i]	Steam generator secondary side sample	Secondary coolant	0.5 0.5 0.5 0.5 0.375	No	1X4DB159-3 2X4DB159-3	35	HV-9452 HV-9554A HV-9554B 048 044	Out In In Out In	A	2'-6" - - 3'-3" -	Globe Globe Globe Globe Globe	Solenoid Solenoid Solenoid Manual Manual	N N N N N	Auto Remote man. Remote man. Manual Manual	Remote man. None None None None	O C O C C	C C C C C	C C C C C	FC FC FC NA NA	AFS Remote man. Remote man. NA NA	15 NA NA NA NA	B NA NA NA NA	Out
12A	54	Chemical addition	Secondary coolant chemicals	0.5 0.5 0.5	No	1X4DB159-1 2X4DB159-1	3A	HV-5281 677 084	Out In In	C	2'-0" - -	Globe Globe Globe	Air Manual Manual	N N N	Remote man. Manual Manual	None None None	C LC C	C LC C	C LC C	FC NA NA	Remote man.[h] NA NA	NA NA NA	NA NA NA	In
12B	57[i]	Steam generator secondary side sample	Secondary coolant	0.5 0.5 0.5 0.5 0.375 0.5 0.5	No	1X4DB159-1 2X4DB159-1	35	HV-9453 HV-9555A HV-9555B 049 045 X-185 X-423[o]	Out In In Out In In In	A	2'-6" - - 3'-3" - - -	Globe Globe Globe Globe Globe Globe Globe	Solenoid Solenoid Solenoid Manual Manual Manual Manual	N N N N N N N	Auto Remote man. Remote man. Manual Manual Manual Manual	Remote man. None None None None None None	O C O C C C C	C C C C C C C	C C C C C C C	FC FC FC NA NA NA NA	AFS Remote man. Remote man. NA NA NA NA	15 NA NA NA NA NA NA	B NA NA NA NA NA NA	Out
12C	57[i]	Steam generator secondary side sample	Secondary coolant	0.5 0.5 0.5 0.5 0.375		1X4DB159-1 2X4DB159-1	35	HV-9454 HV-9556A HV-9556B 050 046	Out In In Out In	A	2'-6" - - 3'-3" -	Globe Globe Globe Globe Globe	Solenoid Solenoid Solenoid Manual Manual	N N N N N	Auto Remote man. Remote man. Manual Manual	Remote man. None None None None	O C O C C	C C C C C	C C C C C	FC FC FC NA NA	AFS Remote man. Remote man. NA NA	15 NA NA NA NA	A NA NA NA NA	Out

TABLE 6.2.4-1 (SHEET 4 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
13A	56	Containment air radioactivity monitor inlet	Containment atmosphere	1 1 1	No	1X4DB213-2 2X4DB213-2	36	HV-12975 HV-12976 X-001	In Out In	C C	- 3'-0" -	Gate Gate Globe	Solenoid Solenoid Manual	N N N	Auto Auto Manual	Remote man. Remote man. None	O O C	C C C	C C C	FC FC NA	CVI CVI NA	15 15 NA	A B NA	Out
13B	56	Containment air radioactivity monitor outlet	Containment atmosphere	1 1 1	No	1X4DB213-2 2X4DB213-2	45	HV-12977 HV-12978 X-003	Out In In	C C	2'-0" - -	Globe Globe Globe	Solenoid Solenoid Manual	N N N	Auto Auto Manual	Remote man. Remote man. None	O O C	C C C	C C C	FC FC NA	CVI CVI NA	15 15 NA	B A NA	In
13C	1.141	Containment pressure detector	DC 702 silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
14A	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
14B	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
14C	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
15	54	Purification water supply to refueling cavity	Borated water	3 3	No	1X4DB130 2X4DB130	37	050 051	Out In	C	1'-6" -	Dia Dia	Manual Manual	N N	Manual Manual	None None	LC LC	LC(c) LC(c)	LC LC	NA NA	NA NA	NA NA	NA NA	In
18	57(i)	Feedwater	Secondary coolant	16 1	No	1X4DB168-3 2X4DB168-3	12	HV-5229 X-031	Out Out	A	11'-0" 4'-1"	Gate Globe	E/H Manual	N N	Auto Manual	Remote man. NA	O C	C C	C C	FC NA	FI NA	5 NA	A,B	In
19	57(i)	Feedwater	Secondary coolant	16 1 1	No	1X4DB168-3 2X4DB168-3	12	HV-5228 X-036(p) X-037(o)	Out Out Out	A	11'-0" 4'-5" -	Gate Globe Globe	E/H Manual Manual	N N N	Auto Manual Manual	Remote man. NA NA	O C C	C C C	C C C	FC NA NA	FI NA NA	5 NA NA	A,B NA NA	In
20	57(i)	Feedwater	Secondary coolant	16 1	No	1X4DB168-3 2X4DB168-3	12	HV-5230 X-073	Out Out	A	8'-0" 3'-0"	Gate Globe	E/H Manual	N N	Auto Manual	Remote man. NA	O C	C C	C C	FC NA	FI NA	5 NA	A,B NA	In
21	57(i)	Feedwater	Secondary coolant	16 1 1	No	1X4DB168-3 2X4DB168-3	12	HV-5227 X-075(o) X-076(p)	Out Out Out	A	8'-9" 3'-0" -	Gate Globe Globe	E/H Manual Manaul	N N N	Auto Manual Manual	Remote man. NA NA	O C C	C C C	C C C	FC NA NA	FI NA NA	5 NA NA	A,B NA NA	In
22	54	Demineralized water supply	Demin. water	2 2 1.5 1.0 1.0	No	AX4DB190-2	38	005 038 PSV-17589 X-065 X-950	Out In In In In	C	2'-2" - - - -	Globe Check Relief Globe Globe	Manual Self Self Manual Manual	N N N N N	Manual Auto Auto Manual Manual	None None None None None	LC - C LC LC	LC - C LC LC	LC - C LC LC	NA NA NA NA NA	NA NA NA NA NA	NA NA NA NA NA	NA NA NA NA NA	In

TABLE 6.2.4-1 (SHEET 5 OF 17)

Penetra-tion Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrange-ment Fig. 6.2.4-1	Valve Number	Location Relative to Con-tainment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
23	54	Breathing air supply	Compressed air	1.5	No	1X4DB186-1	53	211	Out	C	2'-9"	Gate	Manual	N	Manual	None	LC	LC	LC	NA	NA	NA	NA	In
				1.5		2X4DB186-1		184	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				226	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
24	55	Hot leg sample line	Primary coolant	0.5	No	1X4DB140	52	HV-3502	Out	C	8"	Globe	Air	N	Auto	Remote man.	O	O	C	FC	CIA	15	A	Out
				0.5		2X4DB140		HV-3548	In		-	Globe	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	CIA	20	B	
28	54	ACCW supply	Water with corrosion inhibitors	10	No	1X4DB138-2	40	HV-1978	In	C	-	B-fly	Elec. motor	E[g]	Remote man.	Manual	O	O	C[g]	FAI	Remote man.	NA	B	In
				10		1X4DB138-1		HV-1979	Out		1'-6"	B-fly	Elec. motor	E	Remote man.	Manual	O	O	C[g]	FAI	Remote man.	NA	A	
				0.75	No	1X4DB138-2 2X4DB138-2	40	PSV-1978	In		1'-6"	Relief	Self	N	Auto	None	C	C	C	N/A	N/A	N/A	N/A	In
29	54	ACCW return	Water with corrosion inhibitors	10	No	1X4DB138-2	24	HV-1974	In	C	-	B-fly	Elec. motor	E[g]	Remote man.	Manual	O	O	C[g]	FAI	Remote man.	NA	B	Out
				10		1X4DB138-1		HV-1975	Out		1'-6"	B-fly	Elec. motor	E	Remote man.	Manual	O	O	C[g]	FAI	Remote man.	NA	A	
				0.75				113	In			Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
30	55	Safety injection to cold leg	Borated water	4	Yes	1X4DB121	19	HV-8835	Out	A	2'-0"	Gate	Elec. motor	E	Remote man.	Manual	O	O	O	FAI	Remote man.	NA	A	In
				2		2X4DB121		143	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				144	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				145	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				146	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				113	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				HV-8823	In		-	Globe	Air	N	Auto	Remote	C	C	C	FC	CIA	15	B	
				0.75				X-119	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-120	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-121	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-122	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-123	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-124	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-125	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-126	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-181	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-239	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-241	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-346	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-348	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
31	55	Safety injection to hot leg	Borated water	4	Yes	1X4DB121	20	HV-8802B[illegible]	Out	A	2'-6"	Gate	Elec. motor	E	Remote man.	Manual	C	C	O[b]	FAI	Remote man.	NA	B	In
				0.75		2X4DB121		HV-8824	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				2				122	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				123	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				063	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-288	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-289	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-290	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-291	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-286	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-310	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 6 OF 17)

Penetra-tion Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrange-ment Fig. 6.2.4-1	Valve Number	Location Relative to Con-tainment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
32	55	Boron injection line to cold leg	Borated water	4	Yes	1X4DB119	25	HV-8801A	Out	A	16'-5"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O	FAI	SI	NA	A	In
				4		2X4DB119		HV-8801B	Out		16'-3"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O	FAI	SI	NA	B	
				3				013	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				HV-8843	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CI	NA	NA	
				0.75				173	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
33	55	Safety injection to hot leg	Borated water	4	Yes	1X4DB121	20	HV-8802A$^{(l)}$	Out	A	1'-8"	Gate	Elec. motor	E	Remote man.	Manual	C	C	O$^{(b)}$	FAI	Remote man.	NA	A	
				0.75		2X4DB121		HV-8881	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				2				120	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				121	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				290	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-292	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-293	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-294	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-295	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
34	56	Containment spray supply	Borated water	8	Yes	1X4DB131	26	HV-9001B	Out	C	2'-0"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O	FAI	CS	NA	B	In
				8		2X4DB131		016	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				014	Out		2'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				046	Out		2'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
35	56	Containment spray supply	Borated water	8	Yes	1X4DB131	26	HV-9001A	Out	C	7'-5"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O	FAI	CS	NA	A	In
				8		2X4DB131		015	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				2				013	Out		2'-1"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				047	Out		2'-1"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
36	56	RHR emergency sump suction	Borated water	14	Yes	1X4DB122 2X4DB122	27	HV-8811B$^{(l)}$	Out	A	6'-6"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O$^{(b)}$	FAI	SI$^{(e)}$	NA	B	Out
37	56	RHR emergency sump suction	Borated water	14	Yes	1X4DB122 2X4DB122	27	HV-8811A$^{(l)}$	Out	A	6'-6"	Gate	Elec. motor	E	Auto	Remote man.	C	C	O$^{(b)}$	FAI	SI$^{(e)}$	NA	A	Out
38	56	Containment spray emergency sump suction	Borated water	12	Yes	1X4DB131 2X4DB131	23	HV-9002B$^{(l)}$	Out	A	6'-9"	Gate	Elec. motor	E	Remote man.	Manual	C	C	O$^{(b)}$	FAI	Remote man.	NA	B	Out

TABLE 6.2.4-1 (SHEET 7 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
39	56	Containment spray emergency sump suction	Borated water	12	Yes	1X4DB131 2X4DB131	23	HV-9002A(l)	Out	A	6'-9"	Gate	Elec. motor	E	Remote man.	Manual	C	C	O(b)	FAI	Remote man.	NA	A	Out
40	54	Fire protection water	Well water	4	No	1X4DB174-4 2X4DB174-4	41	HV-27901	Out	C	2'-6"	Gate	Air	N	Auto	Remote man.	C	O	C	FC	CIA	20	A,B	In
				6				036	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				1				018	In		-	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
41	54	Accumulator test and drain line	Borated water	0.75	No	1X4DB121 2X4DB121	1	HV-8871	In	C	3'-7"	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	A	Out
				0.75				HV-8964	Out		2'-7"	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				0.75				HV-8888	Out		2'-9"	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				0.75				016	Out		2'-9"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-165	Out		2'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-444(o)	Out		6'-8"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				X-835(o)	Out		1'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				293(p)	Out		1'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				324(p)	Out		5'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-8871	In		3'-7"	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	Out
42	54	Nitrogen supply to accumulator	N_2	1	No	1X4DB120 2X4DB120	3	HV-8880	Out	C	2'-6"	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	A	In
				1				017	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				013	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
43	57	NSCW supply to reactor cavity coolers	Treated well water	8	No	1X4DB135-1 2X4DB135-1	17	HV-2134	Out	A	3'-0"	B-fly	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	SI	40	A	In
				0.75				PSV-11673	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				225	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-658	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-185	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-186	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-181	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-182(p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				309	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				315	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				317	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				323	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				325	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				346	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				348	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-206(o)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 8 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
44	57	NSCW return from	Treated	8	No	1X4DB135-1	46	HV-2138	Out	A	2'-0"	B-fly	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	SI	40	A	Out
		reactor cavity	well	1		2X4DB135-1		X-183	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
		coolers	water	1				X-184$^{(p)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-187	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-188	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				198	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-2136	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				313	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				311	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				321	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				319	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				327	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				350	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				352	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-222$^{(o)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
45	57	NSCW supply to	Treated	8	No	1X4DB135-2	17	HV-2135	Out	A	1'-8"	B-fly	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	SI	40	B	In
		reactor cavity	well	1		2X4DB135-2		X-062$^{(o)}$	Out		2'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
		coolers	water	0.75				159	Out		2'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-2137	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				022	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-654	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-565	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-389	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-490	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-388	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.1				X-405	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.1				X-431	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-391	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-392	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				427	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				443	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				447	In		-	Globe	Manual	N	Manual	None	C	O	C	NA	NA	NA	NA	
				1				455	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				461	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				429	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
46	57	NSCW return from	Treated	8	No	1X4DB135-2	46	HV-2139	Out	A	1'-8"	B-fly	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	SI	40	B	Out
		reactor cavity	well	1		2X4DB135-2		X-040	Out		1'-6"	Globe	Manual	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
		coolers	water	1				164	In		-	Globe	Manual	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
				1				X-491$^{(o)}$	In		-	Globe	Manual	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
				1				X-568	In		-	Globe	Manual	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
				1				X-390$^{(o)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-387	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-11772	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				431	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				433	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				445	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				449	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				453	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				457	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				461$^{(p)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-816$^{(o)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-241$^{(p)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-676$^{(p)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				451	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				459$^{(o)}$	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 9 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
48	54	Normal letdown line	Primary coolant	3	No	1X4DB114	7	HV-8160	In	C	-	Globe	Air	N	Auto	Remote man.	O	O	C	FC	CIA	15	A	Out
				3		2X4DB114		HV-8152(l)(o)	Out		4'-5"	Globe	Air	N	Auto	Remote man.	O	O	C	FC	CIA	15	B	
				0.75				502	Out		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
49	54	Excess letdown and seal water leakoff	Primary coolant	2	No	1X4DB114	31	HV-8100	Out	C	2'-4"	Globe	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	CIA	15	B	Out
				2		2X4DB114		HV-8112	In		-	Globe	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	CIA	15	A	
				0.75				021	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
50	54	Normal charging line	Primary coolant	3	No	1X4DB114 2X4DB114	33	HV-8105	Out	C	3'-0"	Gate	Elec. motor	N	Auto	Remote man.	O	O	C	FAI	SI	17(m)	B	In
				3				032	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				465	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
51	54	Reactor coolant pump seal water supply (pump loop No. 4)	Primary coolant	1.5	Yes	1X4DB114	32	HV-8103D	Out	A	1'-3"	Globe	Elec. motor	E	Remote man.	Manual	O	O	O	FAI	Remote man.	NA	B	In
				1.5		2X4DB114		355	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				452	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
52	54	Reactor coolant pump seal water supply (pump loop No. 3)	Primary coolant	1.5	Yes	1X4DB114	32	HV-8103C	Out	A	1'-3"	Globe	Elec. motor	E	Remote man.	Manual	O	O	O	FAI	Remote man.	NA	B	In
				1.5		2X4DB114		354	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				451	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
53	54	Reactor coolant pump seal water supply (pump loop No. 2)	Primary coolant	1.5	Yes	1X4DB114	32	HV-8103B	Out	A	1'-4"	Globe	Elec. motor	E	Remote man.	Manual	O	O	O	FAI	Remote man.	NA	B	In
				1.5		2X4DB114		353	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				450	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
54	54	Reactor coolant pump seal water supply (pump loop No. 1)	Primary coolant	1.5	Yes	1X4DB114	32	HV-8103A	Out	A	1'-4"	Globe	Elec. motor	E	Remote man.	Manual	O	O	O	FAI	Remote man.	NA	B	In
				1.5		2X4DB114		004	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				449	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
55	NA	Eddy current/sludge lancing	NA	0.75	No	1X4DB159-1	56(k)	X-018	Out	B	-	Globe(q)	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	-
				10		2X4DB159-1			In		-	Flange	NA	N		None	C	C	C	NA	NA	NA		NA
56	55	RHR pump discharge to hot leg	Borated water	12	Yes	1X4DB121	21	HV-8840(l)	Out	A	3'-0"	Gate	Elec. motor	E	Remote man.	Manual	C	C	O(b)	FAI	Remote man.	NA	B	In
				8		2X4DB121		128	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				8				129	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				HV-8825(o)	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				0.75				112	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-435(p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				296	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
57	55	RHR loop into cold leg	Borated water	8	Yes	1X4DB121	30	HV-8809A	Out	A	2'-0"	Gate	Elec. Motor	E	Remote man.	Manual	O	O	O(b)	FAI	Remote man.	NA	A	In
				6		2X4DB121		147	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				6				148	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				111	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				HV-8890A	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
58	55	RHR loop into cold leg	Borated water	8	Yes	1X4DB121	30	HV-8809B	Out	A	1'-10"	Gate	Elec. motor	E	Remote man.	Manual	O	O	O(b)	FAI	Remote man.	NA	B	In
				6		2X4DB121		149	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				6				150	In		-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				110	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				HV-8890B	In		-	Globe	Air	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	
				1				X-410	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 10 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
59	55	RHR suction from hot leg	Primary coolant	12 3	Yes	1X4DB122 2X4DB122	4	HV-8701A PSV-8708A	In In	A	- -	Gate Relief	Elec. motor Self	E N	Remote man. Auto	Manual None	C C	O C	C C	FAI NA	Remote man. NA	NA NA	A NA	Out
60	55	RHR suction from hot leg	Primary coolant	12 3	Yes	1X4DB122 2X4DB122	4	HV-8702A PSV-8708B	In In	A	- -	Gate Relief	Elec. motor Self	E N	Remote man. Auto	Manual None	C	O C	C C	FAI NA	Remote man. NA	NA NA	B NA	Out
62	54	Pressurizer relief tank sample to waste gas compressor suction and N supply	Waste gas/N	1 1	No	1X4DB112 2X4DB112	2	HV-8033 HV-8047	Out In	C	1'-6" -	Dia Dia	Air Air	N N	Auto Auto	Remote man. Remote man.	C C	C C	C C	FC FC	CIA CIA	15 15	B A	Out/In
63	54	Pressurizer relief tank makeup water supply	Demin. water	3 3 1	No	1X4DB112 2X4DB112	39	HV-8028 112 020	Out In In	C	1'-6" - -	Dia Check Globe	Air Self Manual	N N N	Auto Auto Manual	Remote man. None None	O - LC	O - LC	C - LC	FC NA NA	CIA NA NA	15 NA NA	B NA NA	In
64A	NA	Flow verification and pressure sensing piping	Containment atmosphere	0.5 1	No	1X4DB132 2X4DB132	51	119 NA	In In	B	 -	Globe Flange	Manual NA	N N	Manual NA	None None	C NA	C NA	C NA	NA NA	NA NA	NA NA	NA NA	
64B	NA	Flow verification and pressure sensing piping	Containment atmosphere	0.5 1	No	1X4DB132 2X4DB132	51	120 NA	In In	B	 -	Globe Flange	Manual NA	N N	Manual NA	None None	C NA	C NA	C NA	NA NA	NA NA	NA NA	NA NA	
67A	55	Pressurizer steam sample line	Primary coolant	0.5 0.5	No	1X4DB140 2X4DB140	7	HV-3514 HV-3513	Out In	C	4'-0" -	Globe Globe	Air Air	N N	Auto Auto	Remote man. Remote man.	C C	C C	C C	FC FC	CIA CIA	15 15	A B	Out
67B	55	Pressurizer liquid sample line	Primary coolant	0.5 0.5	No	1X4DB140 2X4DB140	7	HV-3507 HV-3508	In Out	C	- 3'-0"	Globe Globe	Air Air	N N	Auto Auto	Remote man. Remote man.	C C	C C	C C	FC FC	CIA CIA	15 15	B A	Out
67C	1.141	Containment pressure detector	DC 702 Silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48		-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
68	NA	Containment leak rate test	Containment atmosphere	0.75 8 0.75	No	1X4DB132 2X4DB132	18	018[o] NA 019[p]	In In In	B	- - -	Globe Flange Globe	Manual NA Manual	N N N	Manual NA Manual	None None None	C NA C	C NA C	C NA C	NA NA NA	NA NA NA	NA NA NA	NA NA NA	
69A	54	Chemical addition	Secondary coolant chemicals	0.5 0.5 0.5	No	1X4DB159-3 2X4DB159-3	3B	HV-5278 678 087	Out In In	C	1'-3" - -	Globe Globe Globe	Air Manual Manual	N N N	Remote Manual Manual	None None None	C LC C	C LC C	C LC C	FC NA NA	Remote man.[h] NA NA	NA NA NA	NA NA NA	In
69B	54	Chemical addition	Secondary coolant chemicals	0.5 0.5 0.5	No	1X4DB159-3 2X4DB159-3	3B	HV-5279 679 090	Out In In	C	1'-6" - -	Globe Globe Globe	Air Manual Manual	N N N	Remote Manual Manual	None None None	C LC C	C LC C	C LC C	FC NA NA	Remote man.[h] NA NA	NA NA NA	NA NA NA	In

TABLE 6.2.4-1 (SHEET 11 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
69C	1.141	Containment pressure detector	DC 702 silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
70A	56	Containment H_2 monitor suction	Containment atmosphere	0.75 0.75 0.75	Yes	1X4DB213-2 2X4DB213-2	14	HV-2791A HV-2790A HV-2790B	Out In In	C	2'-9" - -	Globe Globe Globe	Solenoid Solenoid Solenoid	E E E	Remote man. Remote man. Remote man.	Remote man.[(n)] None None	C C C	C C C	O O O	FC FC FC	Remote man. Remote man. Remote man.	NA NA NA	A B B	Out
70B	56	Containment H_2 monitor discharge	Containment atmosphere	0.75 0.75 0.75	Yes	1X4DB213-2 2X4DB213-2	8	HV-2793A 001 039	Out In In	C	1'-2" - -	Globe Check Globe	Solenoid Self Manual	E E N	Remote man. Auto Manual	Remote man.[(n)] None None	C - C	C - C	O - C	FC NA NA	Remote man. NA NA	NA NA NA	A NA NA	In
70C	1.141	Containment pressure detector	DC 702 silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
71A	56	Containment H_2 monitor suction	Containment atmosphere	0.75 0.75 0.75	Yes	1X4DB213-2 2X4DB213-2	14	HV-2792B HV-2791B HV-2792A	In Out In	C	- 1'-9" -	Globe Globe Globe	Solenoid Solenoid Solenoid	E E E	Remote man. Remote man. Remote man.	None Remote man.[(n)] None	C C C	C C C	O O O	FC FC FC	Remote man. Remote man. Remote man.	NA NA NA	A B A	Out
71B	56	Containment H_2 monitor discharge	Containment atmosphere	0.75 0.75 0.75	Yes	1X4DB213-2 2X4DB213-2	8	HV-2793B 002 040	Out In In	C	2'-3" - -	Globe Check Globe	Solenoid Self Manual	E E N	Remote man. Auto Manual	Remote man.[(n)] None Manual	C - C	C - C	O - C	FC NA NA	Remote man. NA NA	NA NA NA	B NA NA	In
71C	1.141	Containment pressure detector	DC 702 silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
72A	54	Accumulator sample line	Borated water	0.75 0.75 0.75	No	1X4DB120 2X4DB120	28	HV-10950 159 178	In Out In	C	- 2'-9" -	Globe Globe Globe	Solenoid Manual Manual	N N N	Auto Manual Manual	Remote man. None None	C LC C	C LC C	C LC C	FC NA NA	CIA NA NA	15 NA NA	A NA NA	Out
72B	54	Accumulator sample line	Borated water	0.75 0.75 0.75	No	1X4DB120 2X4DB120	28	HV-10952 161 183	In Out In	C	- 2'-4" -	Globe Globe Globe	Solenoid Manual Manual	N N N	Auto Manual Manual	Remote man. None None	C LC C	C LC C	C LC C	FC NA NA	CIA NA NA	15 NA NA	A NA NA	Out
73A	54	Accumulator sample line	Borated water	0.75 0.75 0.75	No	1X4DB120 2X4DB120	28	HV-10951 160 181	In Out In	C	- 2'-9" -	Globe Globe Globe	Solenoid Manual Manual	N N N	Auto Manual Manual	Remote man. None None	C LC C	C LC C	C LC C	FC NA NA	CIA NA NA	15 NA NA	B NA NA	Out
73B	54	Accumulator sample line	Borated water	0.75 0.75 0.75	No	1X4DB120 2X4DB120	28	HV-10953 162 185	In Out In	C	- 2'-4" -	Globe Globe Globe	Solenoid Manual Manual	N N N	Auto Manual Manual	Remote man. None None	C LC C	C LC C	C LC C	FC NA NA	CIA NA NA	15 NA NA	B NA NA	Out
77	54	Reactor coolant drain tank pump discharge	Primary coolant	3 3 1 1 1 1 1 1 1 1 0.75	No	1X4DB127 2X4DB127	29	HV-7699 HV-7136 X-186[(o)] X-189[(o)] X-218[(o)] X-220[(o)] X-153[(p)] X-154[(p)] X-173[(p)] X-229[(p)] PSV-7699	In Out In In In In In In In In In	C	- 1'-0" - - - - - - - - 1'-0"	Dia Dia Globe Globe Globe Globe Globe Globe Globe Globe Relief	Air Air Manual Manual Manual Manual Manual Manual Manual Manual Self	N N N N N N N N N N N	Auto Auto Manual Manual Manual Manual Manual Manual Manual Manual Auto	Remote man. Remote man. None None None None None None None None None	O O C C C C C C C C C	O O C C C C C C C C C	C C C C C C C C C C C	FC FC NA NA NA NA NA NA NA NA N/A	CIA CIA NA NA NA NA NA NA NA NA N/A	15 15 NA NA NA NA NA NA NA NA N/A	A B NA NA NA NA NA NA NA NA N/A	Out Out

TABLE 6.2.4-1 (SHEET 12 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
78	54	Normal containment sump pumps discharge	Drains	3	No	1X4DB143	10	HV-780	In	C	-	Gate	Air	N	Auto	Remote man.	O	C	C	FC	CIA	15	A	Out
				3				HV-781	Out		3'-9"	Gate	Air	N	Auto	Remote man.	O	C	C	FC	CIA	15	B	
				0.75		2X4DB143		PSV-0780	In		3'-9"	Relief	Self	N	Auto	None	C	C	C	N/A	N/A	N/A	N/A	Out
79	54	Reactor coolant drain tank vent and H_2 supply	Gas	0.75	No	1X4DB127	9	HV-7126	In	C	-	Dia	Air	N	Auto	Remote man.	O	C	C	FC	CIA	15	A	Out
				0.75		2X4DB127		HV-7150	Out		1'-9"	Dia	Air	N	Auto	Remote man.	O	C	C	FC	CIA	15	B	
80	56	Service air and post-LOCA purge air supply	Compressed air	4	No	1X4DB186-1	6	HV-9385	Out	C	2'-0"	Gate	Air	N	Auto	Remote man.	C	C	C	FC	CIA	20	A,B	In
				4		2X4DB186-1		034	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				228	Out		1'-3"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				229	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
81	54	Instrument air	Compressed air	2	No	1X4DB186-4	5	HV-9378	Out	C	1'-6"	Globe	Air	N	Auto	Remote man.	O	O	C	FC	CIA	15	A,B	In
				2		2X4DB186-4		049	In		-	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.75				256	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
83	56	Normal containment purge supply and equalizing	Containment atmosphere	24	No	1X4DB213-1	15	HV-2626A	In	C	-	B-fly	Elec. motor	N	Auto	Remote man.	LC	O	C	FAI	CVI	10	A	In
				24		2X4DB213-1		HV-2627A	Out		7'-0"	B-fly	Elec. motor	N	Auto	Remote man.	LC	O	C	FAI	CVI	10	B	
				14				HV-2626B	In		-	B-fly	Air	N	Auto	Remote man.	C	C	C	FC	CVI	5	A	
				14				HV-2627B	Out		6'-0"	B-fly	Air	N	Auto	Remote man.	C	C	C	FC	CVI	5	B	
				0.75				001	Out		2'-3"	Gate	Manual	N	Manual	None	LC	LC	LC	NA	NA	NA	NA	
84	56	Normal containment purge exhaust and equalizing	Containment atmosphere	24	No	1X4DB213-1	47	HV-2628A	In	C	-	B-fly	Elec. motor	N	Auto	Remote man.	LC	O	C	FAI	CVI	10	A	Out
				24		2X4DB213-1		HV-2629A	Out		13'-0"	B-fly	Elec. motor	N	Auto	Remote man.	LC	O	C	FAI	CVI	10	B	
				14				HV-2628B	In		-	B-fly	Air	N	Auto	Remote man.	C	C	C	FC	CVI	5	A	
				14				HV-2629B	Out		7'-0"	B-fly	Air	N	Auto	Remote man.	C	C	C	FC	CVI	5	B	
				0.75				001	Out		2'-3"	Gate	Manual	N	Manual	None	LC	LC	LC	NA	NA	NA	NA	
85C	1.141	Containment pressure detector	DC 702 silicone oil	Tubing	Yes	1X4DB131 2X4DB131	48	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
86A	56	Post-accident sampling	Containment atmosphere	1.0	No	1X4DB110	45	HV-8211	In	C	-	Globe	Solenoid	N	Auto	Remote man.	C	C	C	FC	CIA	15	B	In
				1.0		2X4DB110		HV-8212	Out		8'-6"	Globe	Solenoid	N	Auto	Remote man.	C	C	C	FC	CIA	15	A	
				0.5				X-002	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
87	NA	Containment leak rate test	Containment atmosphere	0.75	No	1X4DB132	18	019[(o)]	In	B	-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	-
				8.0		2X4DB132		NA	In		-	Flange	NA	N	NA	None	NA	NA	NA	NA	NA	NA	NA	
				0.75				018[(p)]	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
88A	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
88B	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-

TABLE 6.2.4-1 (SHEET 13 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/ Outside	Type Tests	Length of Pipe (ft-in.)	Valve: Type	Valve: Operator	Essential or Nonessential	Actuation Mode: Primary	Actuation Mode: Secondary	Valve Position: Normal	Valve Position: Shutdown	Valve Position: Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
88C	1.141	Reactor vessel water level instrumentation	Water	Tubing	Yes	1X4DB113 2X4DB113	54	None	-	A	-	-	-	-	-	-	-	-	-	-	-	-	-	-
89	NA	Transfer tube	Refueling pool water	20	No	1X4DB130 2X4DB130	13	NA	In	B	-	Flange	NA	N	NA	None	C	C(c)	C	NA	NA	NA	NA	-
				0.50				090	In			Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
90	NA	Eddy current/sludge lancing	NA	0.75	No	1X4DB159-1 2X4DB159-1	56(k)	X-019	Out	B	-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	-
				10					In		-	Flange		N		None	NA	NA	NA	NA	NA	NA	NA	
91	57	NSCW supply to containment coolers	Treated well water	8	Yes	1X4DB135-2 2X4DB135-2	55	HV-1809	Out	A	2'-6"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	B	In
				1				019	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-1817	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				X-005	Out		1'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-652	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-486	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-394	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-395	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-393	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-397	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-398	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-396	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-487	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-11774	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				365	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				369	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				375	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				377	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				387	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				381	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				439	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				441	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
92	57	NSCW supply to containment coolers	Treated well water	8	Yes	1X4DB135-2 2X4DB135-2	44	HV-1807	Out	A	2'-6"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	B	In
				3				001	In		-	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				018	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-1815	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				X-077	Out		1'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-656	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-400	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-401	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-500(o)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-245	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-403	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-404	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				393	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				397	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				403	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				405	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				409	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				415	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				435	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				437	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-399(p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

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Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
93	57	NSCW supply to containment coolers	Treated well water	8	Yes	1X4DB135-1 2X4DB135-1	44	HV-1806	Out	A	2'-0"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	A	In
				3				002	In		-	Gate	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-1814	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				X-127	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-196	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-197	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-199	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-120	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-121	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-212[o]	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-11672	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				285	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				291	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				293	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				297	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				303	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				305	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				338	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				340	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				482	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-198[p]	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
94	57	NSCW supply to containment coolers	Treated well water	8	Yes	1X4DB135-1 2X4DB135-1	55	HV-1808	Out	A	3'-0"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	A	In
				1				015	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-1816	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				X-259	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-193	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-194	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-192	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-189	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-190	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				X-191	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-11671	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				261	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				267	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				269	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				273	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				279	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				281	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				330	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				332	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

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Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
95	57	NSCW return from	Treated	8	Yes	1X4DB135-2	16	HV-1831	Out	A	2'-0"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	B	Out
		containment coolers	well	1				X-046	Out		1'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
			water	1				034	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-488	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-489	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-383	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-384	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				367	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				373	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				371	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				379	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				385	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				383	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				389	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				391	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
96	57	NSCW return from	Treated	8	Yes	1X4DB135-2	16	HV-1823	Out	A	2'-6"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	B	Out
		containment coolers	well	1		2X4DB135-2		X-052	Out		1'-6"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
			water	1				032	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-501 (o)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-386	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.75				PSV-11773	In		-	Relief	Self	N	Auto	None	C	C	C	NA	NA	NA	NA	
				1				395 (o)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				401	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				399	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				407	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				413	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				411	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				417 (o)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				419	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-385 (p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				396 (p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-813 (p)	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
97	57	NSCW return from	Treated	8	Yes	1X4DB135-1	16	HV-1830	Out	A	2'-0"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	A	Out
		containment coolers	well	1		2X4DB135-1		017	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
			water	1				X-177	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-178	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				263	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				265	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				271	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				277	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				275	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				283	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				334	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				336	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

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Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
98	57	NSCW return from containment coolers	Treated well water	8	Yes	1X4DB135-1 2X4DB135-1	16	HV-1822	Out	A	3'-0"	B-fly	Elec. motor	E	Auto	Remote man.	O	O	O	FAI	SI	NA	A	Out
				1				016	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-179	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				289	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				287	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				295	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				301	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				299	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				307	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				342	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				344	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-180	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
100	56	Post-accident air exhaust	Containment atmosphere	4	No	1X4DB213-1 2X4DB213-1	22	HV-2624A	In	C	-	B-fly	Elec. motor	N	Auto	Remote man.	C	C	C	FAI	CVI	NA[j]	A	Out
				4				HV-2624B	In		-	B-fly	Elec. motor	N	Auto	Remote man.	C	C	C	FAI	CVI	NA[j]	B	
				4				012	Out		6'-3"	Gate	Manual	N	Manual	None	LC	LC	LC	NA	NA	NA	NA	
				0.75				001	Out		6'-6"	Gate	Manual	N	Manual	None	LC	LC	LC	NA	NA	NA	NA	
101	57[i]	Auxiliary feedwater	Secondary coolant	6	Yes	1X4DB168-3 2X4DB168-3	49	128	In	A	-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	In
				6				120	Out		5'-8"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				4				115	Out		3'-3"	Stop	Self	E	Auto	None	LO	LO	LO	NA	NA	NA	NA	
				6				HV-15198	Out		8'-3"	Gate	Air	N	Auto	Remote man.	O	C	C	FC	FI	NA	A,B	
				1				X-194[o]	Out		1'-8"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				136	Out		2'-2"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.5				HV-5196	Out		5'-2"	Globe	Air	N	Remote man.	None	C	C	C	FC	Remote man.[h]	NA	NA	
				1				X-186	Out		18'-9"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-191	Out		31'-4"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-193[p]	Out		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-241	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
102	57[i]	Auxiliary feedwater	Secondary coolant	6	Yes	1X4DB168-3 2X4DB168-3	49	126	In	A	-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	In
				6				118	Out		6'-4"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				4				114	Out		6'-6"	Stop	Self	E	Auto	None	LO	LO	LO	NA	NA	NA	NA	
				6				HV-15197	Out		9'-0"	Gate	Air	N	Auto	Remote man.	O	C	C	FC	FI	NA	A,B	
				1				X-195	Out		4'-7"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				134	Out		2'-9"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.5				HV-5195	Out		8'-0"	Globe	Air	N	Remote man.	None	C	C	C	FC	Remote Man.[h]	NA	NA	
				1.0				X-188	Out		20'-0"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-197	Out		30'-7"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-237	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	

TABLE 6.2.4-1 (SHEET 17 OF 17)

Penetration Number	GDC or RG	System Name	Fluid	Line Size (in.)	ESF or Support Systems	Drawing Number	Valve Arrangement Fig. 6.2.4-1	Valve Number	Location Relative to Containment Inside/Outside	Type Tests	Length of Pipe (ft-in.)	Valve Type	Valve Operator	Essential or Nonessential	Actuation Mode Primary	Actuation Mode Secondary	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Power Failure	Actuation Signal	Valve Closure Time (s)	Power Source 1E Bus A or B	Normal Direction of Flow
103	57[i]	Auxiliary feedwater	Secondary coolant	6	Yes	1X4DB168-3 2X4DB168-3	49	127	In	A	-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	In
				6				119	Out		13'-0"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				4				116	Out		10'-9"	Stop	Self	E	Auto	None	LO	LO	LO	NA	NA	NA	NA	
				6				HV-15199	Out		15'-10"	Gate	Air	N	Auto	Remote man.	O	C	C	FC	FI	NA	A,B	
				1				X-180[p]	Out		5'-2"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				135	Out		7'-1"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.5				HV-5197	Out		7'-10"	Globe	Air	N	Remote man.	None	C	C	C	FC	Remote Man.[h]	NA	NA	
				1				X-215	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				1				X-181[o]	Out		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
104	57[i]	Auxiliary feedwater	Secondary coolant	6	Yes	1X4DB168-3 2X4DB168-3	49	125	In	A	-	Check	Self	E	Auto	None	-	-	-	NA	NA	NA	NA	In
				6				117	Out		3'-9"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				4				113	Out		9'-1"	Stop	Self	E	Auto	None	LO	LO	LO	NA	NA	NA	NA	
				6				HV-15196	Out		17'-2"	Gate	Air	N	Auto	Remote man.	O	C	C	FC	FI	NA	A,B	
				1				X-178	Out		5'-2"	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
				0.5				133	Out		7'-1"	Check	Self	N	Auto	None	-	-	-	NA	NA	NA	NA	
				0.5				HV-5194	Out		7'-10"	Globe	Air	N	Remote man.	None	C	C	C	FC	Remote Man.[h]	NA	NA	
				1				X-233	In		-	Globe	Manual	N	Manual	None	C	C	C	NA	NA	NA	NA	
	56	Equipment hatch	Containment atmosphere	NA	No	NA	42	NA	NA	B	NA	None	None	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
	56	Personnel locks	Containment atmosphere	NA	No	NA	43	NA	NA	B	NA	None	None	N	Manual	Manual	C	C	C	NA	NA	NA	NA	
	56	Emergency doors	Containment	NA	No	NA	43	NA	NA	B	NA	None	None	N	Manual	Manual	C	C	C	NA	NA	NA	NA	

a. The following is a list of abbreviations:

GDC	-	General Design Criteria	SLI	-	steam line isolation
RG	-	NRC Regulatory Guides	FI	-	feedwater isolation
Dia	-	diaphragm	CVI	-	containment ventilation isolation
B-fly	-	butterfly	CIA	-	containment isolation phase A
O	-	open	CIB	-	containment isolation phase B
C	-	closed	SI	-	safety injection signal
LC	-	locked closed	CS	-	containment spray signal
FC	-	failed closed	E/H	-	electrohydraulic
Self	-	actuated by the fluid pressure	FAI	-	fail as is
NA	-	not applicable	AFS	-	auxiliary feedwater automatic start signal
Auto	-	automatic			

b. These valves are required to function for long term cooldown and recirculation.
c. These valves and/or flanges are opened for the refueling operation.
d. Valves identified in this table are shown with an asterisk (*) in figure 6.2.4-1.

e. Opens coincident on SI and RWST low-low.

f. Valve closure time is defined in SRP 6.2.4 paragraph II.6.N.

g. The ACCW penetrations are not part of an engineered safety features system nor are they required for safe shutdown; however, the penetrations are classified as essential due to the importance of maintaining cooling water to the reactor coolant pump. These valves remain open post-accident and are only closed for certain accident conditions.

h. Power to the solenoid shall be removed during all modes of operation, except during wet layup operation and periodic testing of the valves, by moving the slide links of the states terminal blocks, located in the terminal box, to the open position.

i. These penetrations are associated with the secondary side of the steam generators. The present design isolation provisions for secondary system lines meet the intent of GDC 57 (i.e., one isolation valve capable of remote manual operation is provided), but GDC 57 was not the design basis. The steam generator and associated secondary system piping form the primary barrier to the outside environment, much the same as the containment liner Plate. The valves associated with these penetrations do not receive a containment isolation signal and are not credited with affecting containment isolation in the safety analysis. The verification of the integrity of this barrier is accomplished in the type A test.

j. The stroke time of these values is 20 s. This penetration is verified closed by the Technical Specifications.

k. These penetrations are associated with the steam generator eddy current and sludge lancing operation. During modes 5 and 6, the blind flanges may be removed and replaced with a fixture that allows cables and hoses to pass through. The cables are sealed, and manually operated valves are provided on the hoses both inside and outside containment to provide the capability for containment closure during fuel movement as required by Technical Specifications.

l. Valve disk is provided with bonnet vent on the containment side of disk.

m. See table 6.3.2-3 for other stroke time information.

n. For LOCA conditions with a loss of one train of dc power, the secondary actuation mode for valves HV-2791A/B and HV-2793A/B is accomplished by providing an alternate train power connection in the QPCP panel per plant procedures.

o. Unit 1 only.

p. Unit 2 only.

q. Unit 2 is a gate valve.

TABLE 6.2.4-2 (SHEET 1 OF 15)

CONTAINMENT ISOLATION VALVES[(a)]

Valve Number	Function	Valve Closure Time(s)
1. Containment Isolation Phase "A"		
1HV-3502	Hot leg sample line	≤ 15
1HV-3548	Hot leg sample line	≤ 20
2HV-3502	Hot leg sample line and gross failed fuel detector	≤ 15
2HV-3548	Hot leg sample line and gross failed fuel detector	≤ 20
HV-8823	Safety injection pump discharge to cold leg	≤ 15
HV-8824	Safety injection pump discharge to hot leg	≤ 15
HV-8843	Boron injection line to cold leg	≤ 15
HV-8881	Safety injection pump discharge to hot leg	≤ 15
HV-27901	Fire protection water	≤ 20
HV-8871	Accumulator test and drain line	≤ 15
HV-8964	Accumulator test and drain line	≤ 15
HV-8888	Accumulator test and fill line	≤ 15
HV-8880	Nitrogen supply to accumulator	≤ 15
HV-8160	Normal letdown line	≤ 15

TABLE 6.2.4-2 (SHEET 2 OF 15)

Valve Number	Function	Valve Closure Time(s)
1. Containment Isolation Phase "A" (continued)		
HV-8152	Normal letdown line	≤ 15
HV-8100	Excess letdown and seal water leakoff	≤ 15
HV-8112	Excess letdown and seal water leakoff	≤ 15
HV-8825	RHR pump discharge to hot leg	≤ 15
HV-8890A	RHR pump discharge to cold leg	≤ 15
HV-8890B	RHR pump discharge to cold leg	≤ 15
HV-8033	Pressurizer relief tank sample to waste gas compressor suction	≤ 15
HV-8047	Pressurizer relief tank sample to waste gas compressor suction	≤ 15
HV-8028	Pressurizer relief tank makeup water supply	≤ 15
HV-3514	Pressurizer steam sample line	≤ 15
HV-3513	Pressurizer steam sample line	≤ 15
HV-3507	Pressurizer liquid sample line	≤ 15
HV-3508	Pressurizer liquid sample line	≤ 15
HV-10950	Accumulator sample line	≤ 15
HV-10952	Accumulator sample line	≤ 15
HV-10951	Accumulator sample line	≤ 15
HV-10953	Accumulator sample line	≤ 15
HV-7699	Reactor coolant drain tank pump discharge	≤ 15

TABLE 6.2.4-2 (SHEET 3 OF 15)

Valve Number	Function	Valve Closure Time(s)
1. Containment Isolation Phase "A" (continued)		
HV-7136	Reactor coolant drain tank pump discharge	≤ 15
HV-0780	Normal containment sump pumps discharge	≤ 15
HV-0781	Normal containment sump pumps discharge	≤ 15
HV-7126	Reactor coolant drain tank vent and H_2 supply	≤ 15
HV-7150	Reactor coolant drain tank vent and H_2 supply	≤ 15
HV-9385	Service air and post-LOCA purge air supply	≤ 20
HV-9378	Instrument air	≤ 15
HV-8211	Post-accident sampling	≤ 15
HV-8212	Post-accident sampling	≤ 15
2. Containment Ventilation Isolation		
HV-12975	Containment air radioacitvity monitor inlet	≤ 15
HV-12976	Containment air radioactivity monitor inlet	≤ 15
HV-12977	Containment air radioactivity monitor outlet	≤ 15
HV-12978	Containment air radioactivity monitor outlet	≤ 15
HV-2626A	Containment pre-access purge supply and equalizing	≤ 10
HV-2627A	Containment pre-access purge supply and equalizing	≤ 10

TABLE 6.2.4-2 (SHEET 4 OF 15)

Valve Number	Function	Valve Closure Time(s)
2.	Containment Ventilation Isolation (continued)	
HV-2626B	Containment mini-purge supply and equalizing	≤ 5
HV-2627B	Containment mini-purge supply and equalizing	≤ 5
HV-2628A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2629A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2628B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2629B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2624A	Post-accident air exhaust	N/A
HV-2624B	Post-accident air exhaust	N/A
3.	Safety Injection	
HV-8811B[d]	RHR emergency sump suction	N/A
HV-8811A[d]	RHR emergency sump suction	N/A
HV-2134[e]	NSCW supply to reactor cavity coolers	≤ 40
HV-2138[e]	NSCW return from reactor cavity coolers	≤ 40
HV-2135[e]	NSCW supply to reactor cavity coolers	≤ 40
HV-2139[e]	NSCW return from reactor cavity coolers	≤ 40
HV-8105	Normal charging line	≤ 17[j]
HV-1809[e]	NSCW supply to containment coolers	N/A

TABLE 6.2.4-2 (SHEET 5 OF 15)

	Valve Number	Function	Valve Closure Time(s)
3.	Safety Injection (continued)		
	HV-1807[e]	NSCW supply to containment coolers	N/A
	HV-1806[e]	NSCW supply to containment coolers	N/A
	HV-1808[e]	NSCW supply to containment coolers	N/A
	HV-1831[e]	NSCW return from containment coolers	N/A
	HV-1823[e]	NSCW return from containment coolers	N/A
	HV-1830[e]	NSCW return from containment coolers	N/A
	HV-1822[e]	NSCW return from containment coolers	N/A
	HV-8801A[d]	Boron injection line to cold leg	N/A
	HV-8801B[d]	Boron injection line to cold leg	N/A
4.	Check Valves		
	1418-U4-038	Demineralized water supply	N/A
	2401-U4-184	Breathing air supply	N/A
	1217-U4-113	ACCW return	N/A
	1204-U4-143	Safety injection to cold leg	N/A
	1204-U4-144	Safety injection to cold leg	N/A
	1204-U4-145	Safety injection to cold leg	N/A
	1204-U4-146	Safety injection to cold leg	N/A
	1204-U4-122	Safety injection to hot leg	N/A
	1204-U4-123	Safety injection to hot leg	N/A
	1204-U6-013	Boron injection to cold leg	N/A

TABLE 6.2.4-2 (SHEET 6 OF 15)

Valve Number	Function	Valve Closure Time(s)
4. Check Valves (continued)		
1204-U4-120	Safety injection to hot leg	N/A
1204-U4-121	Safety injection to hot leg	N/A
1206-U6-016	Containment spray supply	N/A
1206-U6-015	Containment spray supply	N/A
2301-U4-036	Fire protection water	N/A
2402-U4-017	Nitrogen supply to accumulator	N/A
1208-U4-021	Excess letdown and seal water leakoff	N/A
1208-U6-032	Normal charging line	N/A
1208-U4-355	Reactor coolant pump seal water supply	N/A
1208-U4-354	Reactor coolant pump seal water supply	N/A
1208-U4-353	Reactor coolant pump seal water supply	N/A
1208-U4-004	Reactor coolant pump seal water supply	N/A
1204-U6-128	RHR pump discharge to hot leg	N/A
1204-U6-129	RHR pump discharge to hot leg	N/A
1204-U6-147	RHR loop into cold leg	N/A
1204-U6-148	RHR loop into cold leg	N/A
1204-U6-149	RHR loop into cold leg	N/A
1204-U6-150	RHR loop into cold leg	N/A
1201-U6-112	Pressurizer relief tank makeup water supply	N/A
1513-U4-001	Containment H_2 monitor discharge	N/A

TABLE 6.2.4-2 (SHEET 7 OF 15)

Valve Number	Function	Valve Closure Time(s)
4. Check Valves (continued)		
1513-U4-002	Containment H_2 monitor discharge	N/A
2401-U4-034	Service air and post-LOCA purge air supply	N/A
2420-U4-049	Instrument air	N/A
1302-U4-126[k]	Auxiliary feedwater	N/A
1302-U4-118[k]	Auxiliary feedwater	N/A
1302-U4-114[k]	Auxiliary feedwater	N/A
1302-U4-134[k]	Auxiliary feedwater	N/A
1302-U4-128[k]	Auxiliary feedwater	N/A
1302-U4-120[k]	Auxiliary feedwater	N/A
1302-U4-115[k]	Auxiliary feedwater	N/A
1302-U4-136[k]	Auxiliary feedwater	N/A
1302-U4-127[k]	Auxiliary feedwater	N/A
1302-U4-119[k]	Auxiliary feedwater	N/A
1302-U4-116[k]	Auxiliary feedwater	N/A
1302-U4-135[k]	Auxiliary feedwater	N/A
1302-U4-125[k]	Auxiliary feedwater	N/A
1302-U4-117[k]	Auxiliary feedwater	N/A
1302-U4-113[k]	Auxiliary feedwater	N/A
1302-U4-133[k]	Auxiliary feedwater	N/A

TABLE 6.2.4-2 (SHEET 8 OF 15)

	Valve Number	Function	Valve Closure Time(s)
5.	Remote Manual		
	HV-5280[b]	Chemical Addition	N/A
	HV-5281[b]	Chemical Addition	N/A
	HV-1978	ACCW Supply	N/A
	HV-1979	ACCW Supply	N/A
	HV-1974	ACCW Return	N/A
	HV-1975	ACCW Return	N/A
	HV-8835	Safety injection to cold leg	N/A
	HV-8802B	Safety injection to hot leg	N/A
	HV-8802A	Safety injection to hot leg	N/A
	HV-9002B[f]	Containment spray emergency sump suction	N/A
	HV-9002A[f]	Containment spray emergency sump suction	N/A
	HV-8103D	Reactor coolant pump seal water supply	N/A
	HV-8103B	Reactor coolant pump seal water supply	N/A
	HV-8103C	Reactor coolant pump seal water supply	N/A
	HV-8103A	Reactor coolant pump seal water supply	N/A
	HV-8840	RHR pump discharge to hot leg	N/A
	HV-8809A	RHR loop into cold leg	N/A
	HV-8809B	RHR loop into cold leg	N/A
	HV-8701A	RHR suction from hot leg	N/A
	HV-8702A	RHR suction from hot leg	N/A

TABLE 6.2.4-2 (SHEET 9 OF 15)

	Valve Number	Function	Valve Closure Time(s)
5.	Remote Manual (continued)		
	HV-5278[(b)]	Chemical addition	N/A
	HV-5279[(b)]	Chemical addition	N/A
	HV-2791A[(i)]	Containment H_2 monitor suction	N/A
	HV-2790A[(i)]	Containment H_2 monitor suction	N/A
	HV-2790B[(i)]	Containment H_2 monitor suction	N/A
	HV-2793A[(i)]	Containment H_2 monitor discharge	N/A
	HV-2792B[(i)]	Containment H_2 monitor suction	N/A
	HV-2791B[(i)]	Containment H_2 monitor suction	N/A
	HV-2792A[(i)]	Containment H_2 monitor suction	N/A
	HV-2793B[(i)]	Containment H_2 monitor discharge	N/A
	HV-5194[(b)(k)]	Auxiliary feedwater	N/A
	HV-5197[(b)(k)]	Auxiliary feedwater	N/A
	HV-5195[(b)(k)]	Auxiliary feedwater	N/A
	HV-5196[(b)(k)]	Auxiliary feedwater	N/A
	HV-9556A[(k)]	Steam generator secondary side sample	N/A
	HV-9556B[(k)]	Steam generator secondary side sample	N/A
	HV-9555A[(k)]	Steam generator secondary side sample	N/A
	HV-9555B[(k)]	Steam generator secondary side sample	N/A
	HV-9554A[(k)]	Steam generator secondary side sample	N/A
	HV-9554B[(k)]	Steam generator secondary side sample	N/A

TABLE 6.2.4-2 (SHEET 10 OF 15)

	Valve Number	Function	Valve Closure Time(s)
5.	Remote Manual (continued)		
	HV-9553A[k]	Steam generator secondary side sample	N/A
	HV-9553B[k]	Steam generator secondary side sample	N/A
	HV-3009[h]	Main steam to auxiliary feedwater pump driver	N/A
	HV-3019[h]	Main steam to auxiliary feedwater pump driver	N/A
6.	Manual		
	1213-U6-050[c]	Purification water supply to refueling cavity	N/A
	1213-U6-051[c]	Purification water supply to refueling cavity	N/A
	1418-U4-005[c]	Demineralized water supply	N/A
	2401-U4-211[c]	Breathing air supply	N/A
	1411-U4-676[c]	Chemical addition	N/A
	1411-U4-677[c]	Chemical addition	N/A
	1411-U4-678[c]	Chemical addition	N/A
	1411-U4-679[c]	Chemical addition	N/A
	1204-U4-159[c]	Accumulator sample line	N/A
	1204-U4-161[c]	Accumulator sample line	N/A
	1204-U4-160[c]	Accumulator sample line	N/A
	1204-U4-162[c]	Accumulator sample line	N/A
	1202-U4-001	NSCW supply to containment fire protection	N/A
	1202-U4-002	NSCW supply to containment fire protection	N/A
	1508-U4-012[c]	Post-accident air exhaust	N/A

TABLE 6.2.4-2 (SHEET 11 OF 15)

	Valve Number	Function	Valve Closure Time(s)
7.	Containment Spray		
	HV-9001A	Containment spray supply	N/A
	HV-9001B	Containment spray supply	N/A
8.	Pressure Relief Valves		
	PSV-17589	Plant demineralized water to containment	N/A
	PSV-11673	NSCW supply to reactor cavity coolers	N/A
	PSV-2136	NSCW from reactor cavity coolers	N/A
	PSV-2137	NSCW supply to reactor cavity coolers	N/A
	PSV-11772	NSCW from reactor cavity coolers	N/A
	PSV-8708A	RHR suction from hot leg	N/A
	PSV-8708B	RHR suction from hot leg	N/A
	PSV-1817	NSCW supply to containment coolers	N/A
	PSV-11774	NSCW supply to containment coolers	N/A
	PSV-1815	NSCW supply to containment coolers	N/A
	PSV-1814	NSCW supply to containment coolers	N/A
	PSV-11672	NSCW supply to containment coolers	N/A
	PSV-1816	NSCW supply to containment coolers	N/A
	PSV-11671	NSCW supply to containment coolers	N/A
	PSV-11773	NSCW supply to containment coolers	N/A
	PSV-3001	Main steam	N/A
	PSV-1978	ACCW Supply	N/A

TABLE 6.2.4-2 (SHEET 12 OF 15)

Valve Number	Function	Valve Closure Time(s)
8. Pressure Relief Valves (continued)		
PSV-8871	Accumulator Test and Drain	N/A
PSV-7699	RCDT Pump Discharge	N/A
PSV-0780	Normal Containment Sump Pump	N/A
PSV-3002	Main steam	N/A
PSV-3003	Main steam	N/A
PSV-3004	Main steam	N/A
PSV-3005	Main steam	N/A
PSV-3011	Main steam	N/A
PSV-3012	Main steam	N/A
PSV-3013	Main steam	N/A
PSV-3014	Main steam	N/A
PSV-3015	Main steam	N/A
PSV-3021	Main steam	N/A
PSV-3022	Main steam	N/A
PSV-3023	Main steam	N/A
PSV-3024	Main steam	N/A
PSV-3025	Main steam	N/A
PSV-3031	Main steam	N/A
PSV-3032	Main steam	N/A
PSV-3033	Main steam	N/A

TABLE 6.2.4-2 (SHEET 13 OF 15)

	Valve Number	Function	Valve Closure Time(s)
8.	Pressure Relief Valves (continued)		
	PSV-3034	Main steam	N/A
	PSV-3035	Main steam	N/A
9.	Other Automatic Valves		
	HV-3006A(g)	Main steam	≤5
	HV-3006B(g)	Main steam	≤5
	HV-3016A(g)	Main steam	≤5
	HV-3016B(g)	Main steam	≤5
	HV-13005A(g)	Main steam	≤5
	HV-13005B(g)	Main steam	≤5
	HV-13007A(g)	Main steam	≤5
	HV-13007B(g)	Main steam	≤5
	HV-3026A(g)	Main steam	≤5
	HV-3026B(g)	Main steam	≤5
	HV-13008A(g)	Main steam	≤5
	HV-13008B(g)	Main steam	≤5
	HV-3036A(g)	Main steam	≤5
	HV-3036B(g)	Main steam	≤5
	HV-13006A(g)	Main steam	≤5
	HV-13006B(g)	Main steam	≤5

TABLE 6.2.4-2 (SHEET 14 OF 15)

	Valve Number	Function	Valve Closure Time(s)
9.	Other Automatic Valves (continued)		
	HV-7603A[h]	Steam generator blowdown	≤15
	HV-7603B[h]	Steam generator blowdown	≤15
	HV-7603C[h]	Steam generator blowdown	≤15
	HV-7603D[h]	Steam generator blowdown	≤15
	HV-9451[h]	Steam generator secondary side sample	≤15
	HV-9452[h]	Steam generator secondary side sample	≤15
	HV-9453[h]	Steam generator secondary side sample	≤15
	HV-9454[h]	Steam generator secondary side sample	≤15
	HV-5229[g]	Feedwater	≤5
	HV-5228[g]	Feedwater	≤5
	HV-5230[g]	Feedwater	≤5
	HV-5227[g]	Feedwater	≤5
	HV-15198[g]	Feedwater	≤5
	HV-15197[g]	Feedwater	≤5
	HV-15196[g]	Feedwater	≤5
	HV-15199[g]	Feedwater	≤5
	PV-3000	Main steam	N/A
	PV-3010	Main steam	N/A
	PV-3020	Main steam	N/A
	PV-3030	Main steam	N/A

TABLE 6.2.4-2 (SHEET 15 OF 15)

a. See FSAR subsection 6.2.4 for discussion of the containment isolation system.

b. The containment isolation valves will be maintained closed by administratively controlling the air supply valve links in an open position.

c. Locked closed.

d. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead the requirements of Technical Specification 3.5.2 apply. Valve stroke times where specified will be tested pursuant to the IST program.

e. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead the requirements of Technical Specification 3.7.8 apply. Valve stroke times where specified will be tested pursuant to the IST program.

f. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead the requirements of Technical Specification 3.6.6 apply. Valve stroke times where specified will be tested pursuant to the IST program.

g. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead the requirements of Technical Specification 3.7.2 apply to the main steam isolation and bypass valves and Technical Specifications 3.3.2 and 3.7.3 apply to the main feedwater isolation valves.

h. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead the requirements of Technical Specification 3.7.5 apply. Valve stroke times where specified will be tested pursuant to the IST program.

i. These valves may be opened on an intermittent basis under administrative control.

j. See table 6.3.2-3 for other stroke time information.

k. These valves are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply; instead verification of containment isolation is accomplished in the type A leakage test which credits the steam generator tubes and associated piping as the primary barrier to the outside environment. These valves are associated with the secondary side of the steam generators and are not subject to GDC 57. These valves do not receive a containment isolation signal and are not credited with effecting containment isolation in the safety analyses. Reference UFSAR paragraph 6.2.4.2.1 and table 6.2.4-1, penetrations 102, 103, and 104 and reference note "I."

NOTE: The valve table as shown reflects the inservice test program as modified by the second 10-year interval update. The updated IST program will be fully implemented by May 31, 1998.

TABLE 6.2.5-1

DESIGN DATA FOR HYDROGEN RECOMBINERS

Quantity	2 per unit
Power (each) (maximum/minimum) (kW)	75/50
Capacity (each) (minimum) (sf^3/min)	100
Heaters (per recombiner)	
Number	4 banks
Maximum heat flux (Btu/h-ft^2)	2850
Maximum sheath temperature (°F)	1550
Gas temperatures	
Inlet (°F)	80-155
Outlet of heater section (°F)	1150 to 1400
Exhaust (°F)	Approximately 50 above ambient
Materials	
Outer structure	Type 300 series SS
Inner structure	Incoloy 800
Heater element sheath	Incoloy 800
Base skid	Type 300 series SS
Weight (lb)	4500 lb
Codes and standards	American Society of Mechanical Engineers Section IX, Underwriters Laboratory, National Electric Manufacturers Association, National Fire Protection Association, Institute of Electrical and Electronic Engineers 279, 308, 323, 344, and 383, Safety Class 0 (See table 3.2.2-1.)

TABLE 6.2.5-2 (SHEET 1 OF 2)

CONTAINMENT BUILDING - FAILURE MODES AND EFFECTS ANALYSIS, POST-LOCA CAVITY PURGE SYSTEM

Item No.	Description of Component	Safety Function	Plant Operating Mode[(a)]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
1	No. 29 breaker on 1ABE, 480-V, 1E, MCC, train A normally closed (NC)	Provide continuity and protection to fan motor, item 3	A	Inadvertent open	Motor control center (MCC) alarm Fan motor lights Flow alarm low	None; loss of train A; train B available
2	No. 29 motor starter, for item 3, train A normally open (NO)	Provide continuity to fan motor, item 3	A	Fail to close	Fan motor lights Flow alarm low	None; loss of train A; train B available
3	1-1516-B7-001-M01, Containment building (CTB) post-LOCA cavity purge unit motor and fan train A, normally deenergized (ND)	Provide motive power by supplying air to the reactor cavity	A	Fail to start and operate	Flow alarm low	None; loss of train A; train B available
4	003 backflow damper, train A, NC	Allow flow of air to the reactor cavity and prevent backflow	A	Fail to open	Flow alarm low	None; loss of train A; train B available
5	No. 29 breaker on 1BBE, 480-V, 1E, MCC, train B, NC	Provide continuity and protection to fan motor, item 7	A	Inadvertent open	MCC alarm Fan motor lights	None; loss of train B; train A available Flow alarm low
6	No. 29 motor starter, item 7 train B, NO	Provide continuity to fan motor, item 7	A	Fail to close	Fan motor lights Flow alarm low	None; loss of train B; train A available
7	1-1516-B7-002-M01 CTB post-LOCA cavity purge unit motor and fan, train B, NO	Provide motive power by supplying air to the reactor cavity	A	Fail to start and operate	Flow alarm low	None; loss of train B; train A available

TABLE 6.2.5-2 (SHEET 2 OF 2)

Item No.	Description of Component	Safety Function	Plant Operating Mode[(a)]	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability
8	004 backflow damper, train B, NC	Allow flow of air to the reactor cavity and prevent backflow	A	Fail to open	Flow alarm low	None; loss of train B; train A available
9	Fan, fan shaft, bearing, etc., 1-1516-B7-001-000	Provide circulation of air	A	Mechanical failure	Flow alarm, low Motor indicating light	None; loss of train A; train B available
10	Fan, fan shaft, bearing, etc., 1-1516-B7-002-000	Provide circulation of air	A	Mechanical failure	Flow alarm low Motor indicating light	None; loss of train B; train A available

a. A - Safety injection for trains A and B start automatically and operate on safety injection only.

TABLE 6.2.5-3

DESIGN DATA FOR PRINCIPAL COMPONENTS OF POST-LOCA CONTAINMENT HYDROGEN PURGE SYSTEM

Containment Building Post-LOCA Purge Filter Exhaust Unit	
Quantity	1
Capacity (ft^3/min)	500
System Components	
Charcoal Adsorber	
Efficiency (%)	89.0[(a)]
Face velocity (ft/min)	40
Residence time (s/2-in. thickness)	0.25
Nominal size (Tyler mesh)	8 x 16
HEPA Filters	
Filter element	Pleated fiberglass
Size (in.)	24 x 24 x 12
Efficiency (%)	99.97% for 0.3-μm and larger particulates
Moisture Eliminator	
Separator element	Fiberglass or galvanized steel
Efficiency (%)	99% for 5 to 10 μm droplets
Electric Heating Coil	
Heater element	80% Ni, 20% Cr
Heating capacity (kW)	2.5

(a) Four-inch filter tested per Table 2 of Regulatory Guide 1.140. With bypass leakage of 1%, the charcoal adsorber section efficiency is 89%.

TABLE 6.2.5-4

DESIGN DATA FOR PRINCIPAL COMPONENTS OF POST-LOCA CAVITY PURGE SYSTEM

Fan	
Quantity	2
Type	Centrifugal
Capacity (ft^3/min)	300
Static pressure (in. WG)	30
Motor (hp)	5

TABLE 6.2.5-5

DESIGN DATA FOR PRINCIPAL COMPONENTS OF
THE CONTAINMENT HYDROGEN MONITORING SYSTEM

Hydrogen Analyzer	
Quantity	2 per unit
Type	Thermal conductivity
Range	0 to 1 and 0 to 10 volume percent
Accuracy	±5 percent of full scale
Valves (isolation)	
Quantity	10
Type	8 solenoid-operated globe valves and 2 check valves
Tubing Material	Stainless steel (Class 2)

TABLE 6.2.5-6 (SHEET 1 OF 4)

PLANT PARAMETERS USED TO CALCULATE POST-ACCIDENT HYDROGEN PRODUCTION

Core thermal power (MWt)[(a)]	3565
Containment free volume (ft^3)	2.75×10^6
Normal containment temperature (°F)	120
Weight of zirconium clad incore (lb)	45,914
Percent zirconium-water reaction (%)	1.5
Hydrogen recombiner flowrate (sf^3/min)	100
Hydrogen recombiner efficiency (%)	95

Baseline Aluminum Inventory in Containment

Component	Weight (lb)	Surface (ft^2)
Flux map drive system	183	48
Nuclear instrumentation system	244	57
Digital rod position indicators	199	241
Control rod drive mechanism (CRDM) connectors	129	68
Miscellaneous valves (nuclear steam supply system) (NSSS)	230	86
Radiation monitoring system	4	4
Containment fan cooler return bend assemblies	765	3000
Communication equipment	65	10
Miscellaneous valves (balance of plant (BOP))	11	2
Containment lighting fixture plugs	10	6
Contingency (NSSS)	250	85
Baseline Total	2,090	3,607

TABLE 6.2.5-6 (SHEET 2 OF 4)

Baseline Zinc Inventory in Containment

	Type	Weight (lb)	Surface (ft^2)
Snubbers	Zn[b]	11	555
Integrated reactor vessel (RV) head/CRDM shroud	ZBP[c]	17,734	180,100
Platform grating	GS[d]	12,453	99,625
Pressurizer grating	GS	696	5,566
Steam generator grating	GS	813	6,500
Cables and related items	Zn	4,988	56,127
Cable tray supports	Zn	1,710	13,680
Inorganic zinc-based paint	ZBP	23,476	500,691
Miscellaneous BOP items	ZBP	36	31
Baseline Total		61,917	862,875

TABLE 6.2.5-6 (SHEET 3 OF 4)

Regulatory Guide 1.7 Hydrogen Production
Calculational Assumptions

Core Cooling Solution Radiolysis

Sources:	
Percent of total halogens retained in the core	50
Percent of total noble gases retained in the core	0
Percent of other fission products retained in the core	99
Energy absorption by core cooling solution:	
Percent of gamma energy absorbed by solution	10
Percent of beta energy absorbed by solution	0
Hydrogen production:	
Molecules hydrogen produced per 100 eV energy absorbed by solution	0.50

TABLE 6.2.5-6 (SHEET 4 OF 4)

Sump Solution Radiolysis

Sources:	
Percent of total halogens released to sump solution	50
Percent of noble gases released to sump solution	0
Percent of other fission products released to sump solution	1
Energy absorption by sump solution:	
Percent of total energy (beta and gamma) which is absorbed by the sump solution	100
Hydrogen production:	
Molecules of hydrogen produced per 100 eV of energy absorbed by the sump solution	0.5

Long-Term Aluminum Corrosion Rate

Mils per year	200

a. Hydrogen generation analysis for core thermal power of 3565 MWt is bounding for MUR power uprate of 3625.6 MWt (see figure 6.2.5-7)

b. Zn - zinc metal.

c. ZBP - zinc-based paint.

d. GS - galvanized steel.

TABLE 6.2.5-7

CORE FISSION PRODUCT ENERGY AFTER 650 FULL-POWER DAYS

DELETED

TABLE 6.2.5-8

FISSION PRODUCT DECAY DEPOSITION IN SUMP SOLUTION

DELETED

TABLE 6.2.5-9

POST-ACCIDENT CONTAINMENT TEMPERATURE TRANSIENT
FOR HYDROGEN GENERATION ANALYSIS

Time Interval (s)	Temperature (°F)
0 - 1.0	rapid increase from 120 to 259
1.0 - 4 E3	259
4 E3 - 6 E3	Ramp down from 250 to 240
6 E3 - 1 E4	Ramp down from 240 to 225
1 E4 - 2 E4	Ramp down from 225 to 205
2 E4 - 4 E4	Ramp down from 205 to 190
4 E4 - 6 E4	Ramp down from 190 to 175
6 E4 - 1 E5	Ramp down from 175 to 160
1 E5 - 1.00 E6	Ramp down from 160 to 131[a]
> 1.00 E6	131

a. The long-term aluminum corrosion rate of 200 mils per year begins when the containment temperature drops below 154°F.

TABLE 6.2.5-10

CORROSION RATES USED IN THE POST-ACCIDENT CONTAINMENT HYDROGEN GENERATION ANALYSIS

Temperature (°F)	Aluminum Corrosion Rate ($lb\text{-}moles_{al}/ft^2\text{-}h$)	Zinc Corrosion Rate[a] ($lb\text{-}moles_{zn}/ft^2\text{-}h$)
100	1.19 E-5	7.53 E-9
140	-	7.16 E-8
154	1.19 E-5	-
160	1.49 E-5	2.21 E-7
180	3.15 E-5	6.80 E-7
200	6.65 E-5	2.10 E-6
220	1.40 E-4	6.47 E-6
240	2.97 E-4	1.99 E-5
260	6.27 E-4	6.15 E-5
280	1.32 E-3	1.90 E-4
300	2.80 E-3	5.84 E-4

a. Zinc corrosion rate also applies to zinc-based paint.

TABLE 6.2.6-1 (SHEET 1 OF 2)

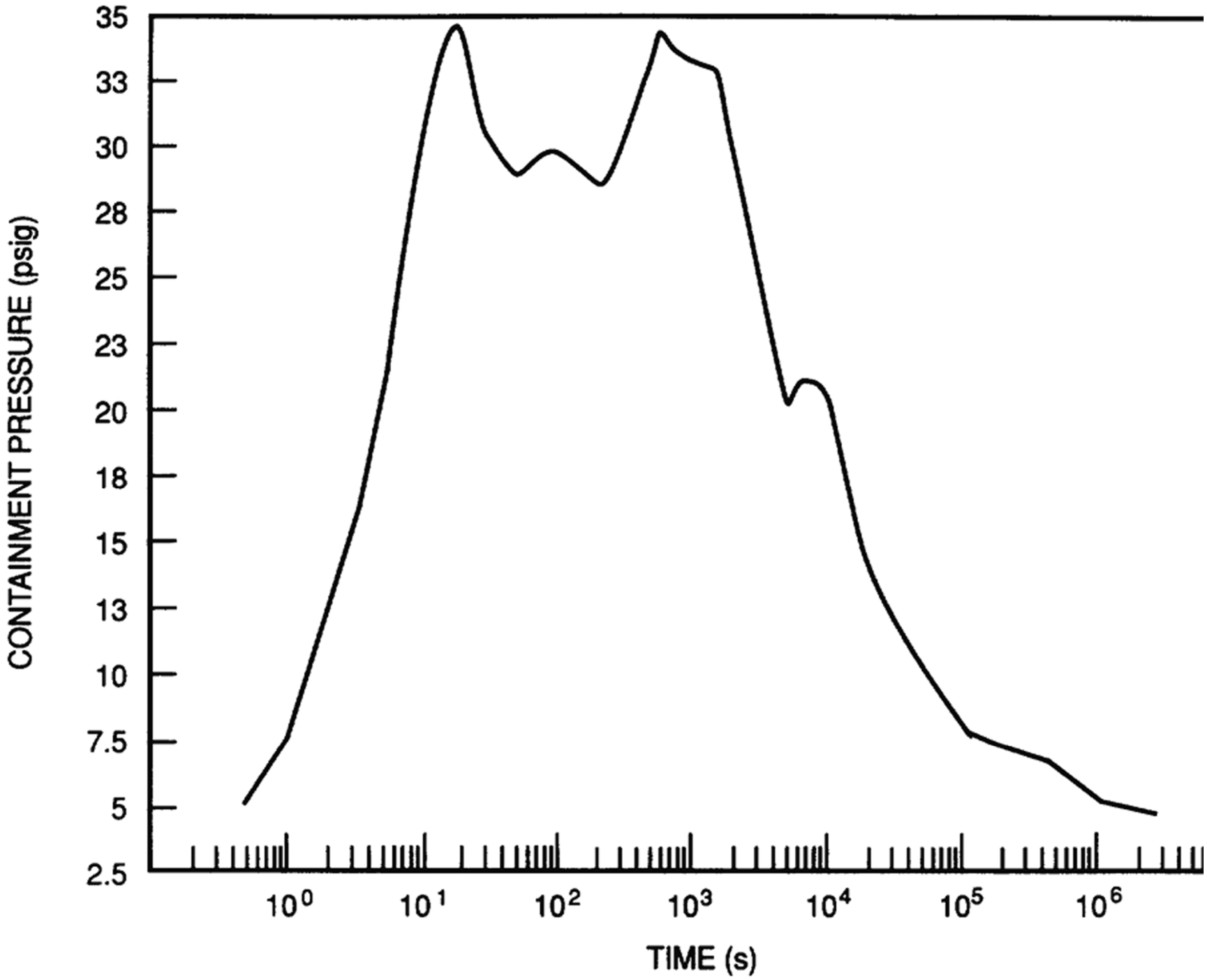
PENETRATIONS NOT VENTED TO CONTAINMENT OR DRAINED DURING TYPE A TESTING

Penetration Number[(a)]	Description[(a)]	Justification
1-4, 7-10, 11B, 11C, 12B, 12C, 18-21, 101-104	Steam generator secondary side (main steam blowdown, sampling, feedwater, and auxiliary feedwater)	The steam generator secondary side is considered an extension of the containment. The system is not part of the reactor coolant pressure boundary and does not open directly to the containment atmosphere during normal and post-accident conditions. However, since the secondary side of the steam generator is a viable leak path, the main steam lines are vented outside of containment.
30, 31, 33	Safety injection line	The system is normally filled with water from refueling water storage tank and operating under post-accident conditions.
32	Boron injection line (high head safety injection)	The system is normally filled with water from the charging pump discharge and operating under post-accident conditions.
36-39	Residual heat removal and containment spray pump suction from containment emergency sump	The system is normally filled with water from refueling water storage tank and operating under post-accident conditions. After an accident the static head between emergency sump and valve provides water seal to prevent leakage of containment atmosphere.
56, 57, 58	Residual heat removal discharge to reactor coolant system	The system is normally filled with water from the refueling water storage tank and operating under post-accident conditions.
59, 60	Residual heat removal suction from hot leg	The system is closed outside containment and constructed to ASME III, Class 2, and Seismic Category 1 standards.

TABLE 6.2.6-1 (SHEET 2 OF 2)

Penetration Number	Description	Justification
43-46, 91-98	Nuclear service cooling water supply and return	The system is a closed system inside containment per GDC 57 and thus not a potential atmospheric leak path. The system is normally filled with water and operating post-accident. The system may be operated to cool the containment atmosphere during the Type A tests.
51-54	RCP seal water supply	The system is closed outside containment and constructed to ASME III, Class 2, and Seismic Category 1 standards. The system is filled with water during all modes of plant operation (normal and post-accident) by the charging pumps.
13C, 67C, 69C, 70C, 71C, 85C	Containment pressure detectors	The system is filled with liquid and designed to satisfy the requirements of Regulatory Guide 1.141. Also the system is closed both inside and outside the containment.
14A, 14B, 14C, 88A, 88B, 88C	Reactor vessel water level instrumentation	The system is filled with liquid and designed to satisfy the requirements of Regulatory Guide 1.141. Also the system is closed both inside and outside.

a. See table 6.2.4-1 for further description.



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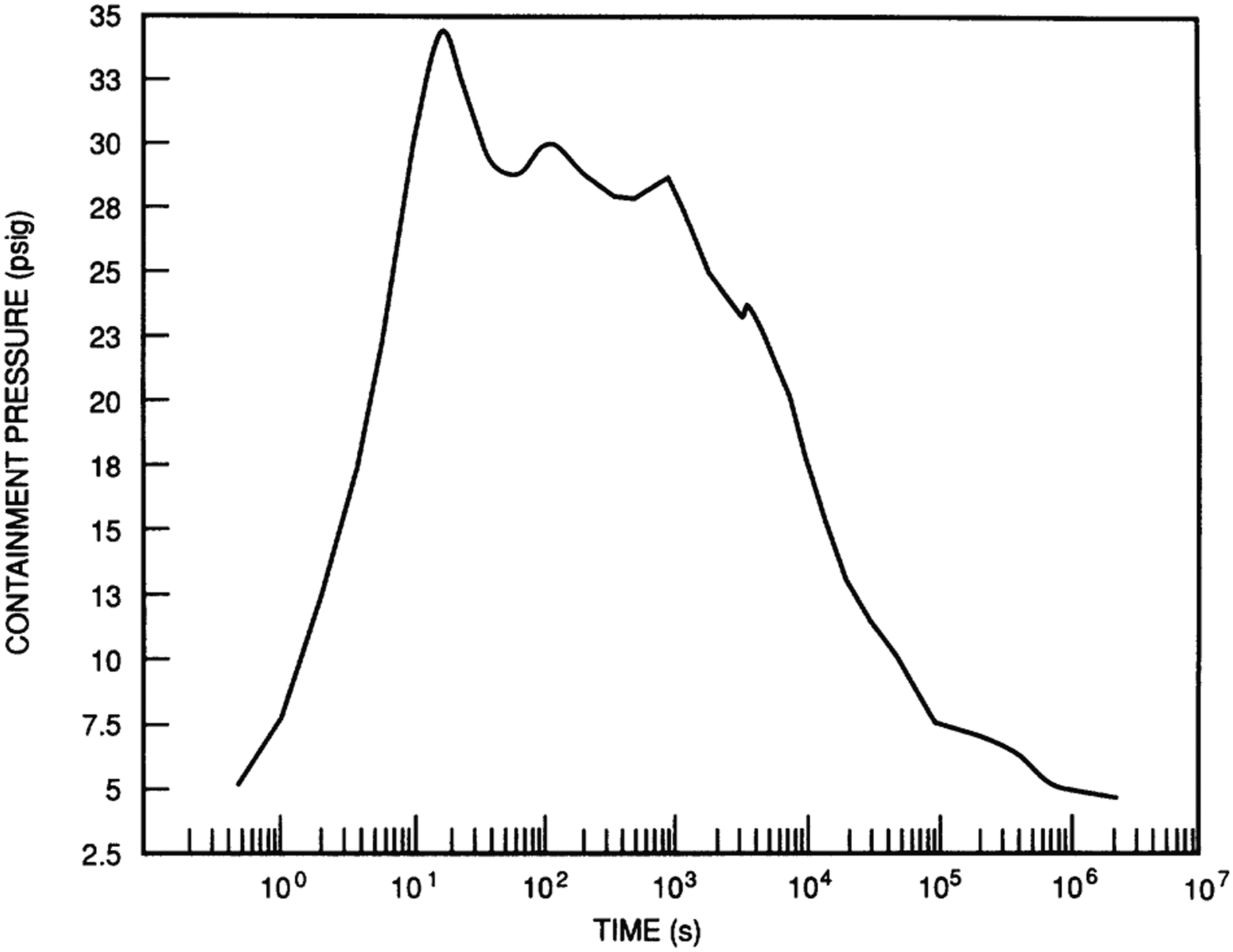
CONTAINMENT PRESSURE TRANSIENT
LOCA–DOUBLE–ENDED PUMP SUCTION–
MINIMUM SAFETY INJECTION

FIGURE 6.2.1–1



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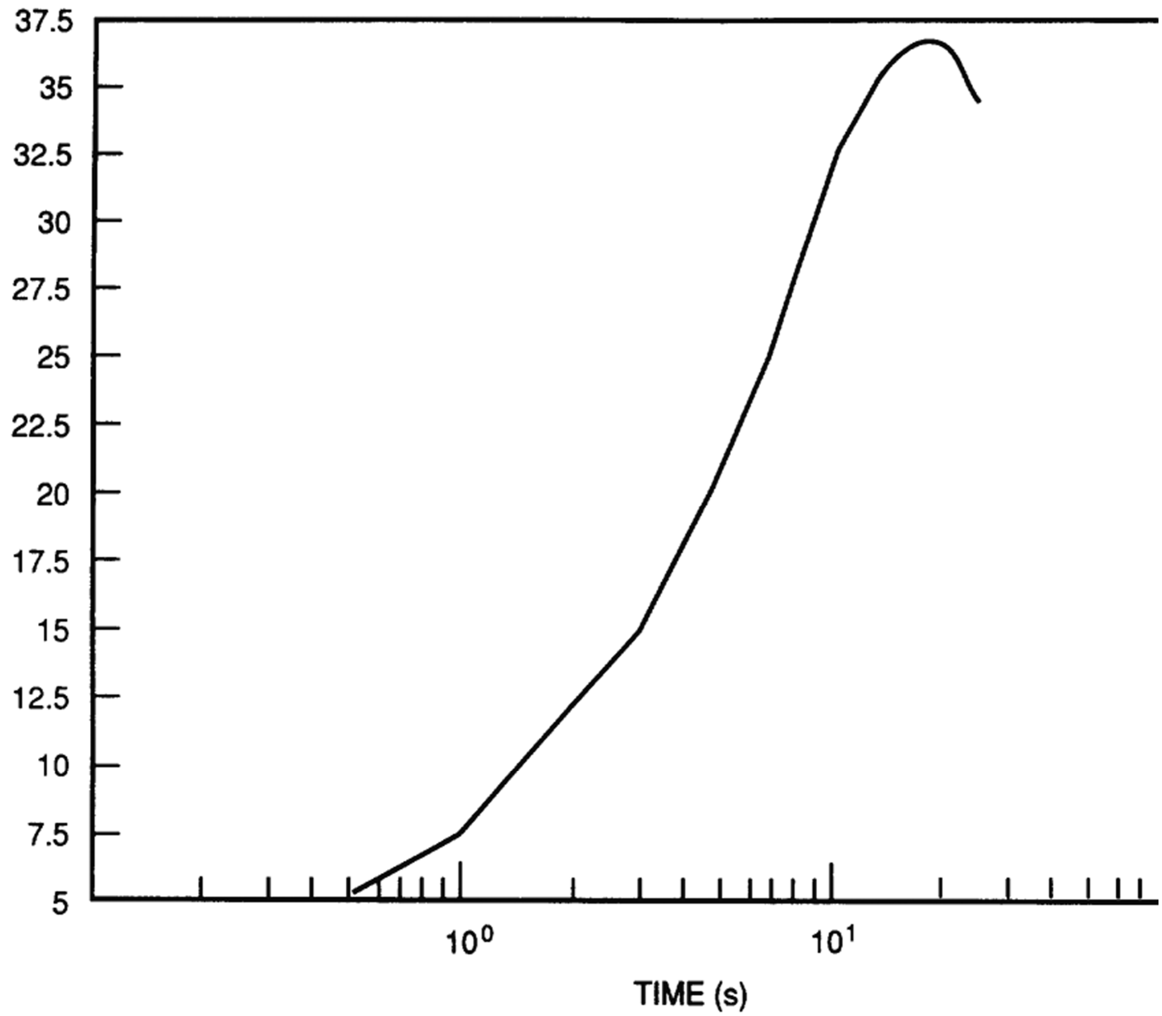
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT PRESSURE TRANSIENT LOCA–DOUBLE–ENDED PUMP SUCTION, MAXIMUM SAFETY INJECTION
		FIGURE 6.2.1–2



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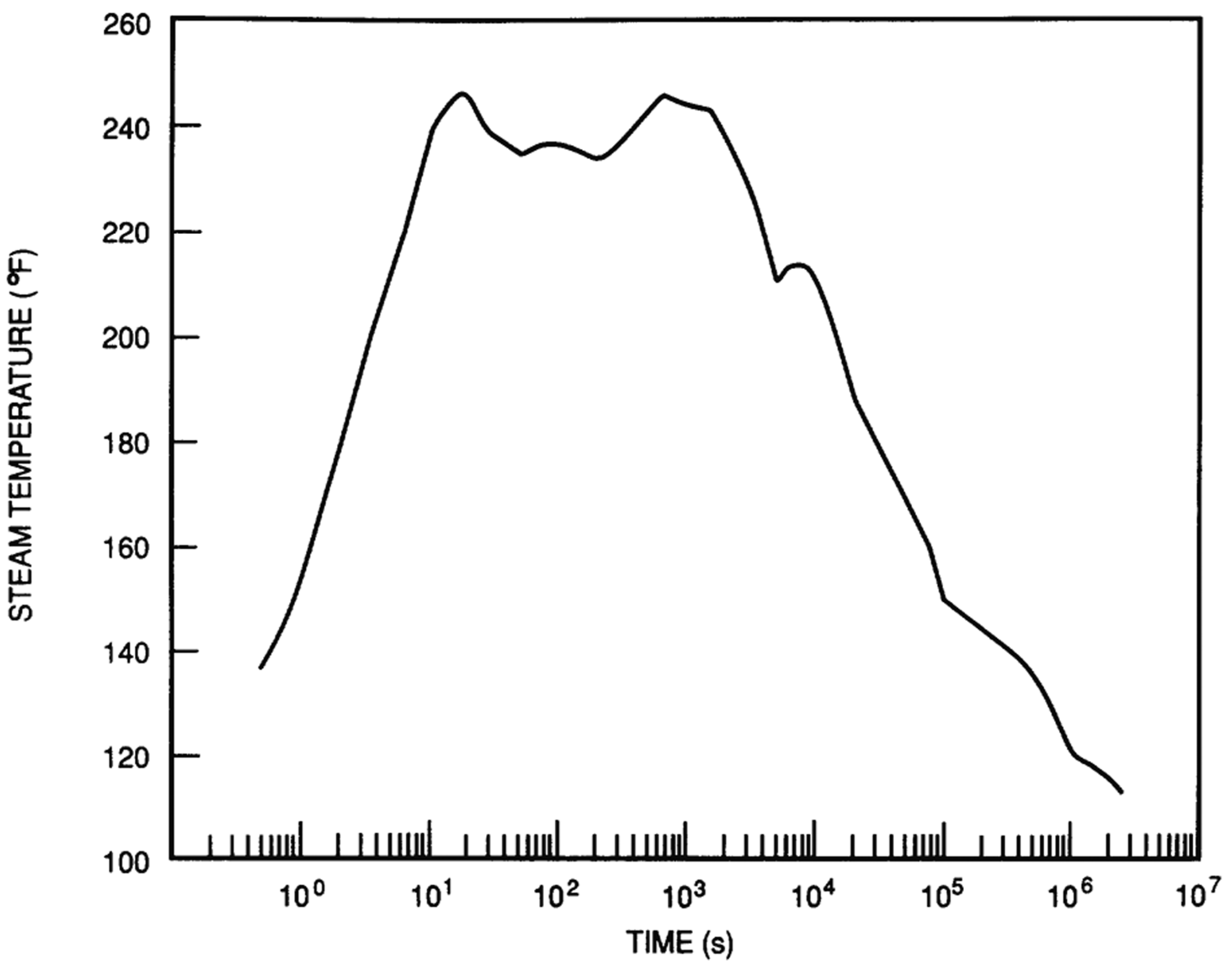
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT PRESSURE TRANSIENT LOCA–DOUBLE–ENDED HOT LEG GUILLOTINE
		FIGURE 6.2.1–3



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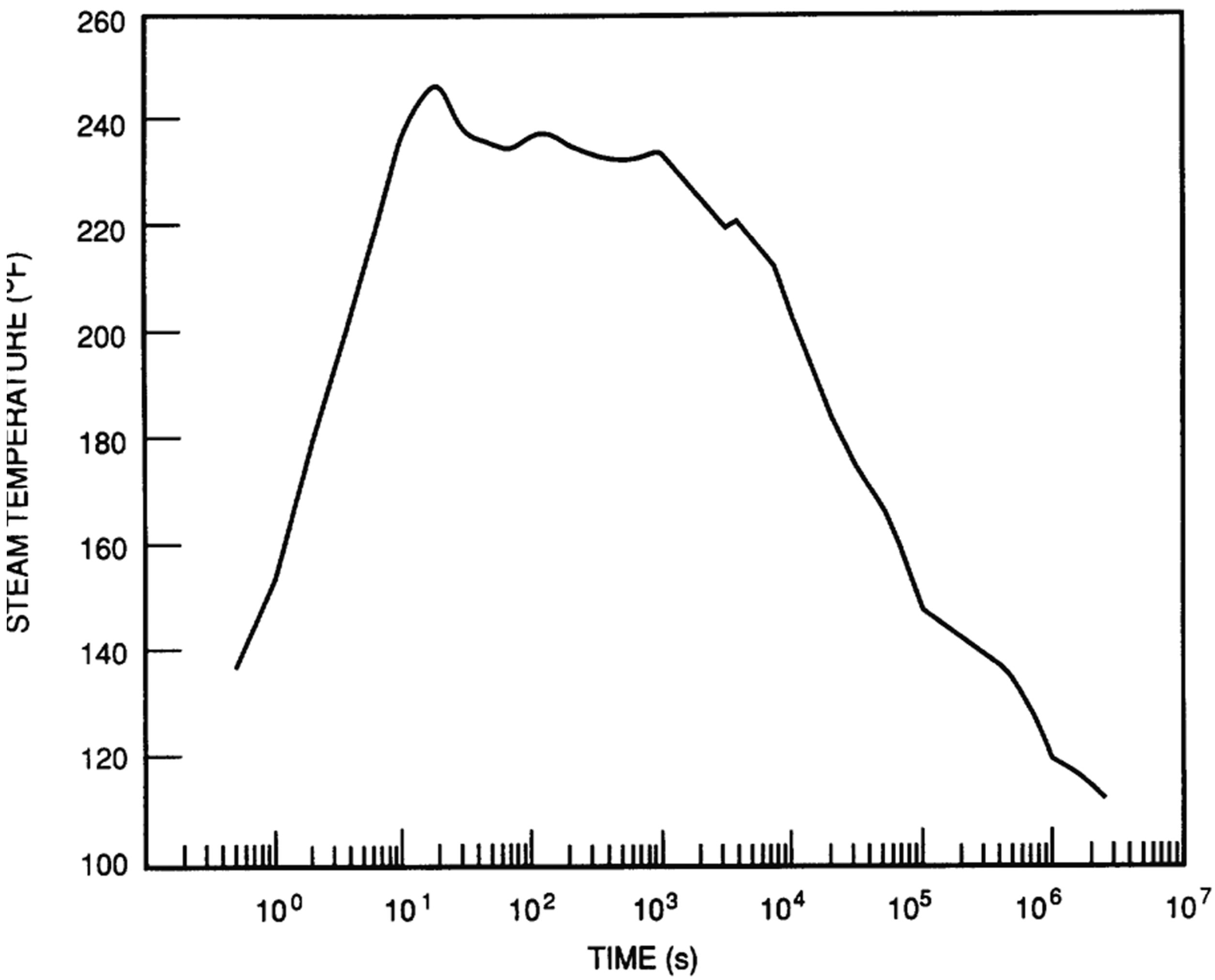
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT TEMPERATURE TRANSIENT LOCA–DOUBLE–ENDED PUMP SUCTION–MINIMUM SAFETY INJECTION
	FIGURE 6.2.1–4



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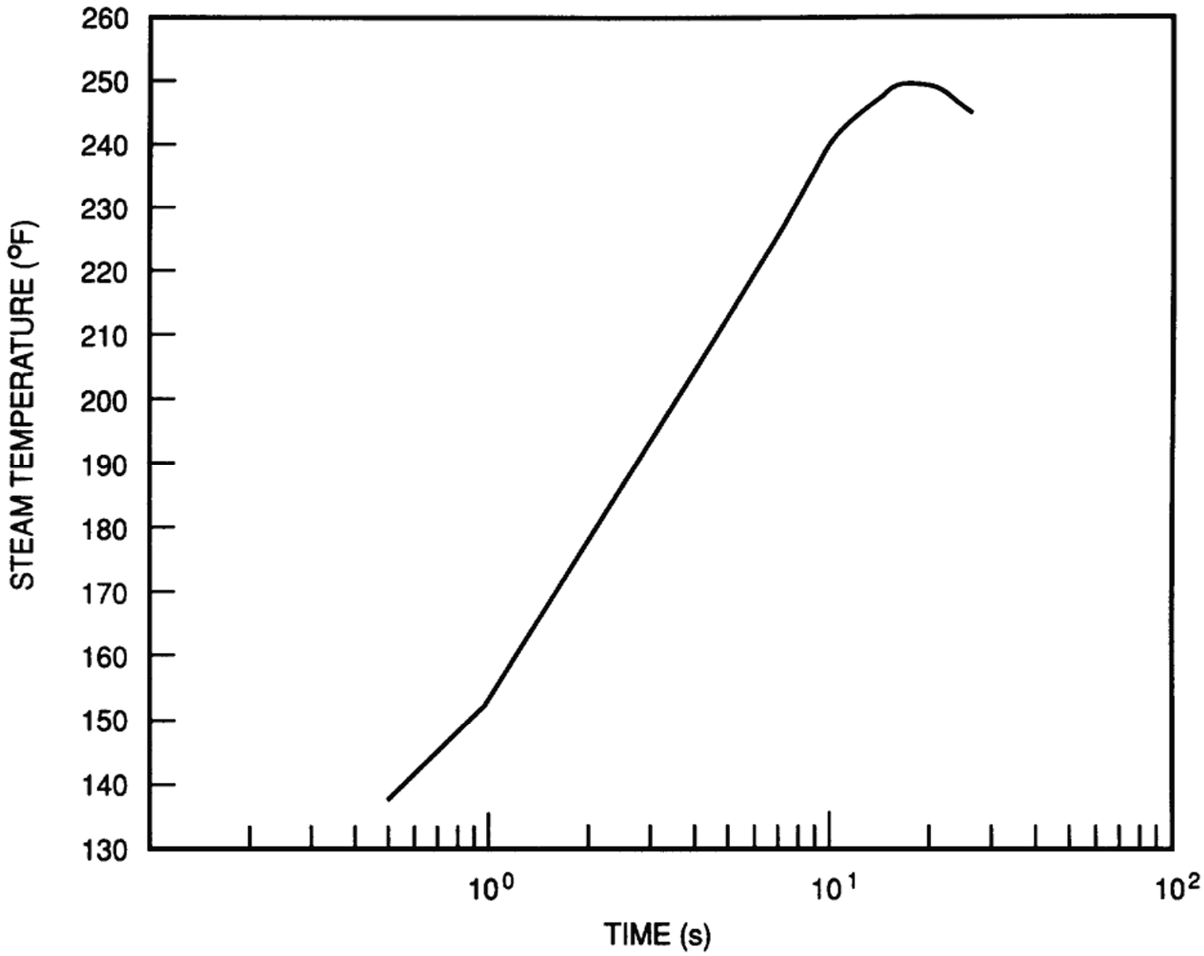
CONTAINMENT TEMPERATURE TRANSIENT
LOCA–DOUBLE–ENDED PUMP SUCTION–
MAXIMUM SAFETY INJECTION

FIGURE 6.2.1–5



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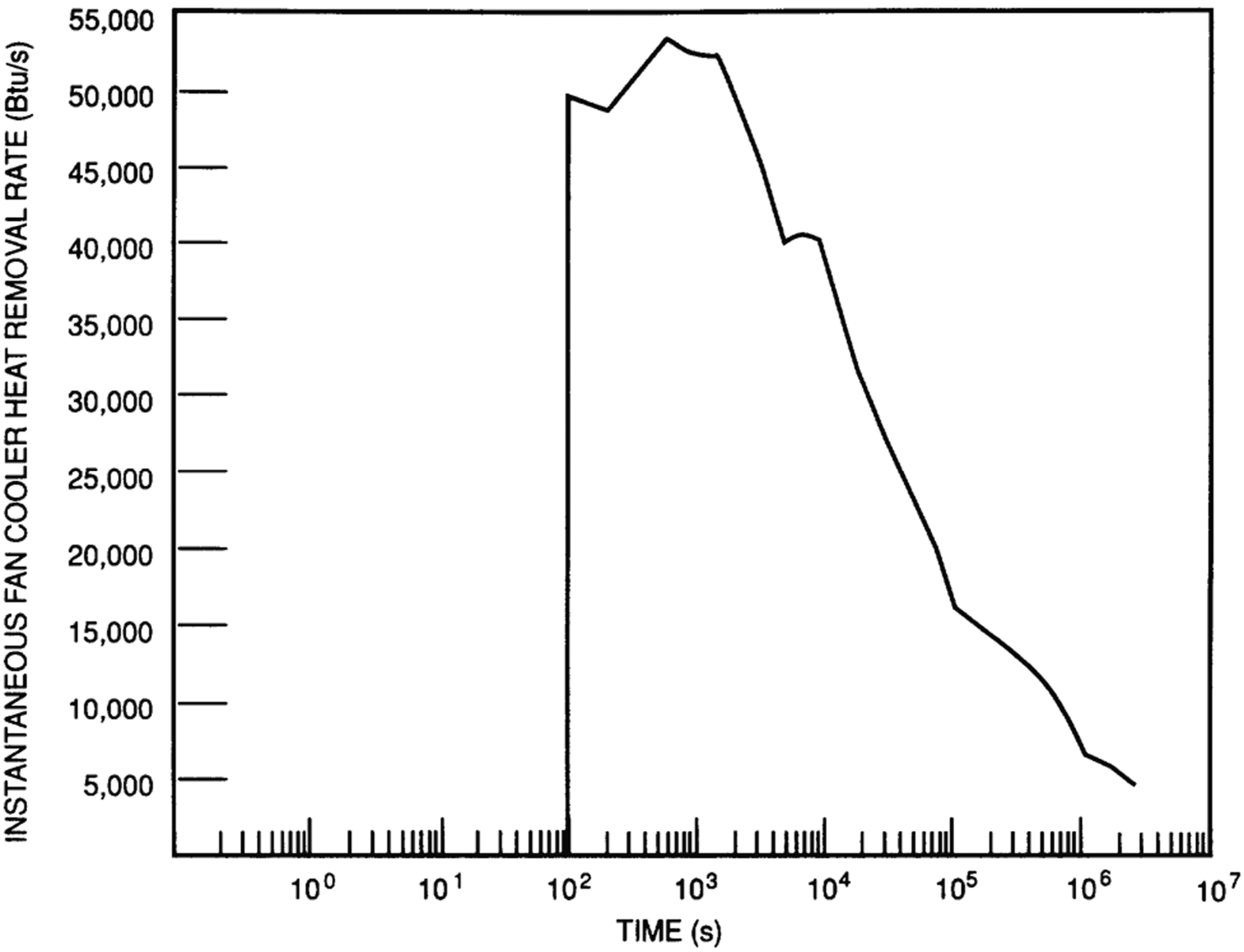
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	CONTAINMENT TEMPERATURE TRANSIENT LOCA–DOUBLE–ENDED HOT LEG GUILLOTINE
	FIGURE 6.2.1–6



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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT FAN COOLER PERFORMANCE LOCA–DOUBLE–ENDED PUMP SUCTION– MAXIMUM SAFETY INJECTION
	FIGURE 6.2.1–7

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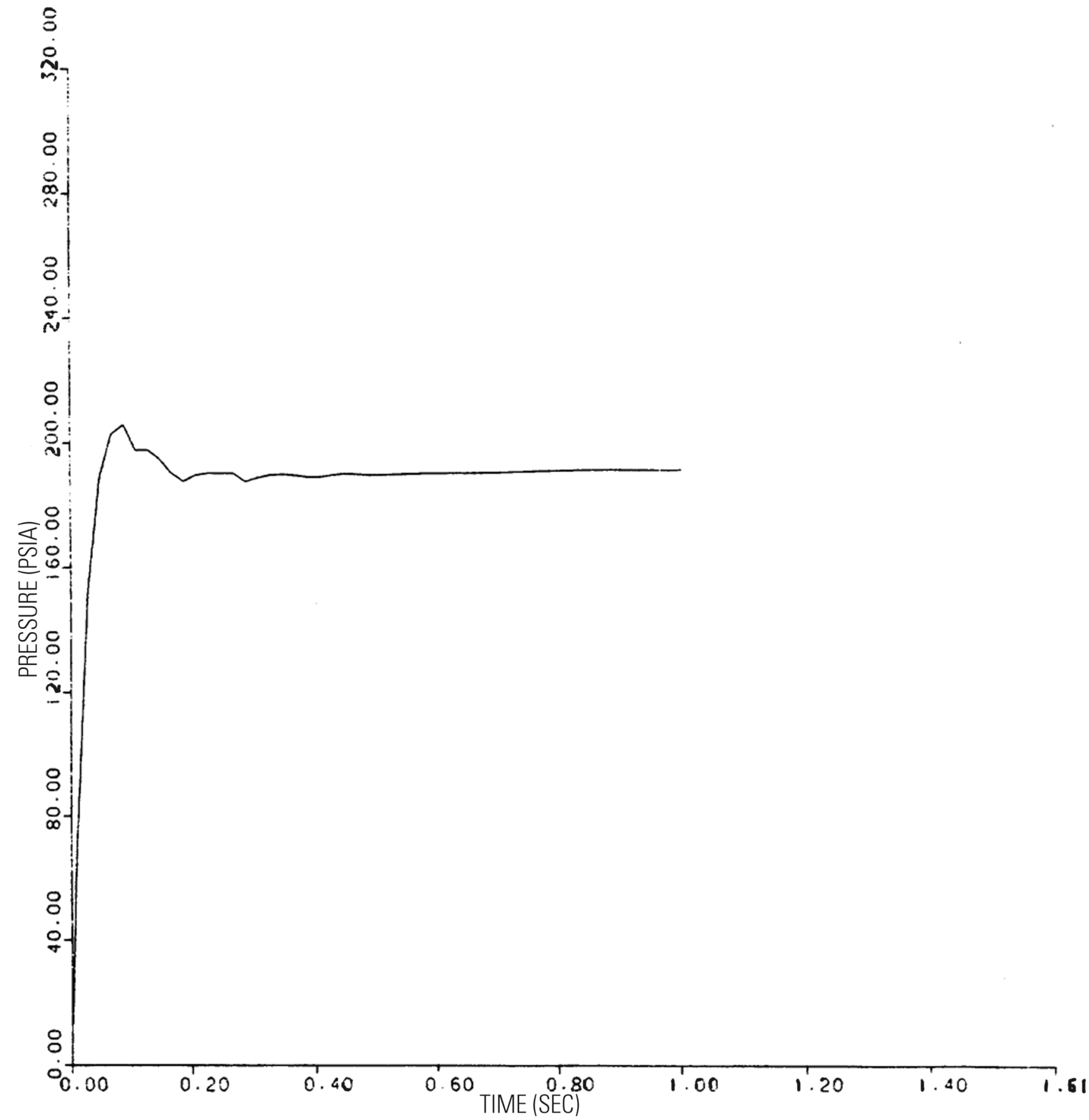
FIGURE 6.2.1–8 THROUGH
FIGURE 6.2.1–14



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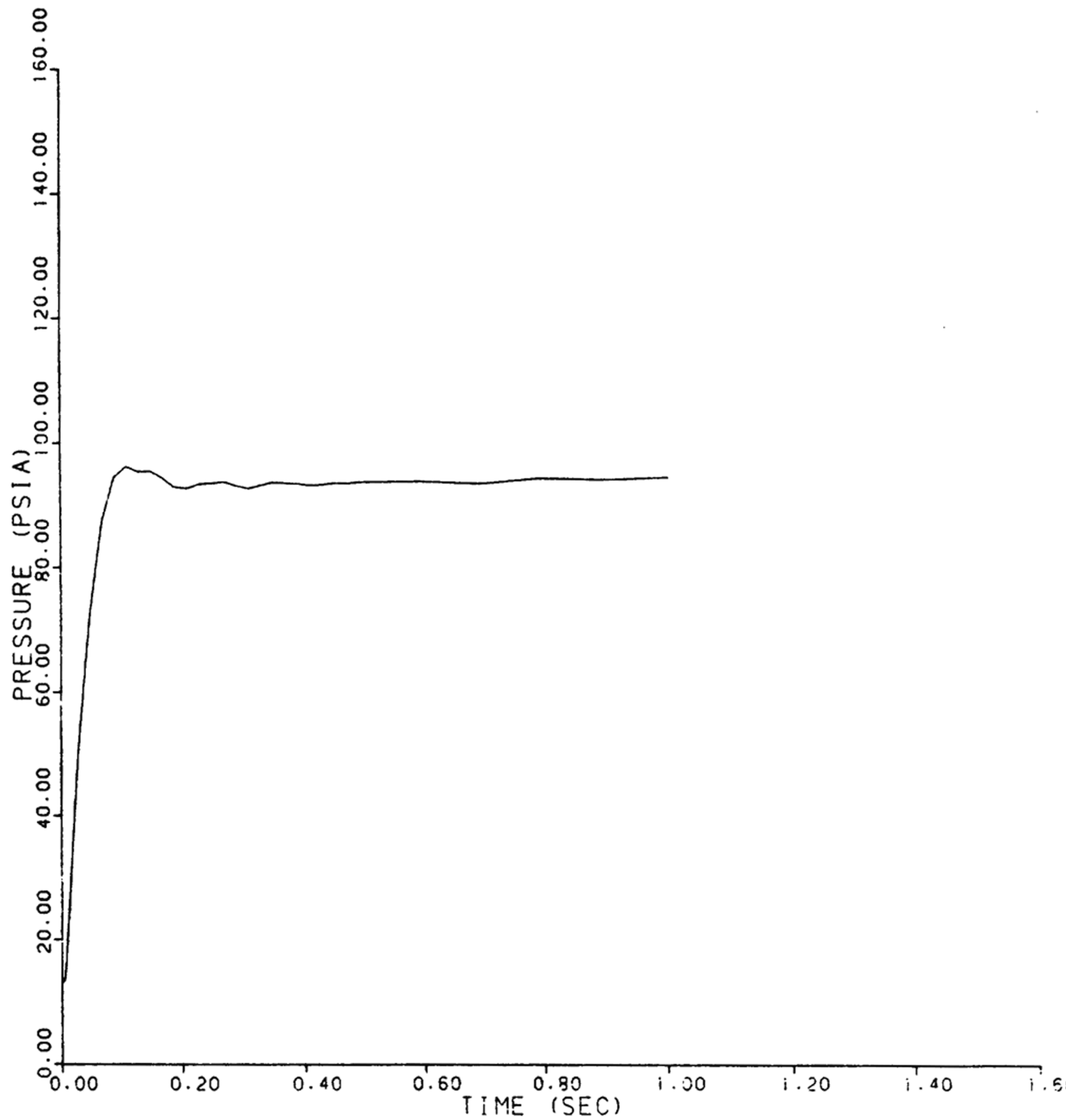
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E1
		FIGURE 6.2.1–15 (SHEET 1 OF 65)



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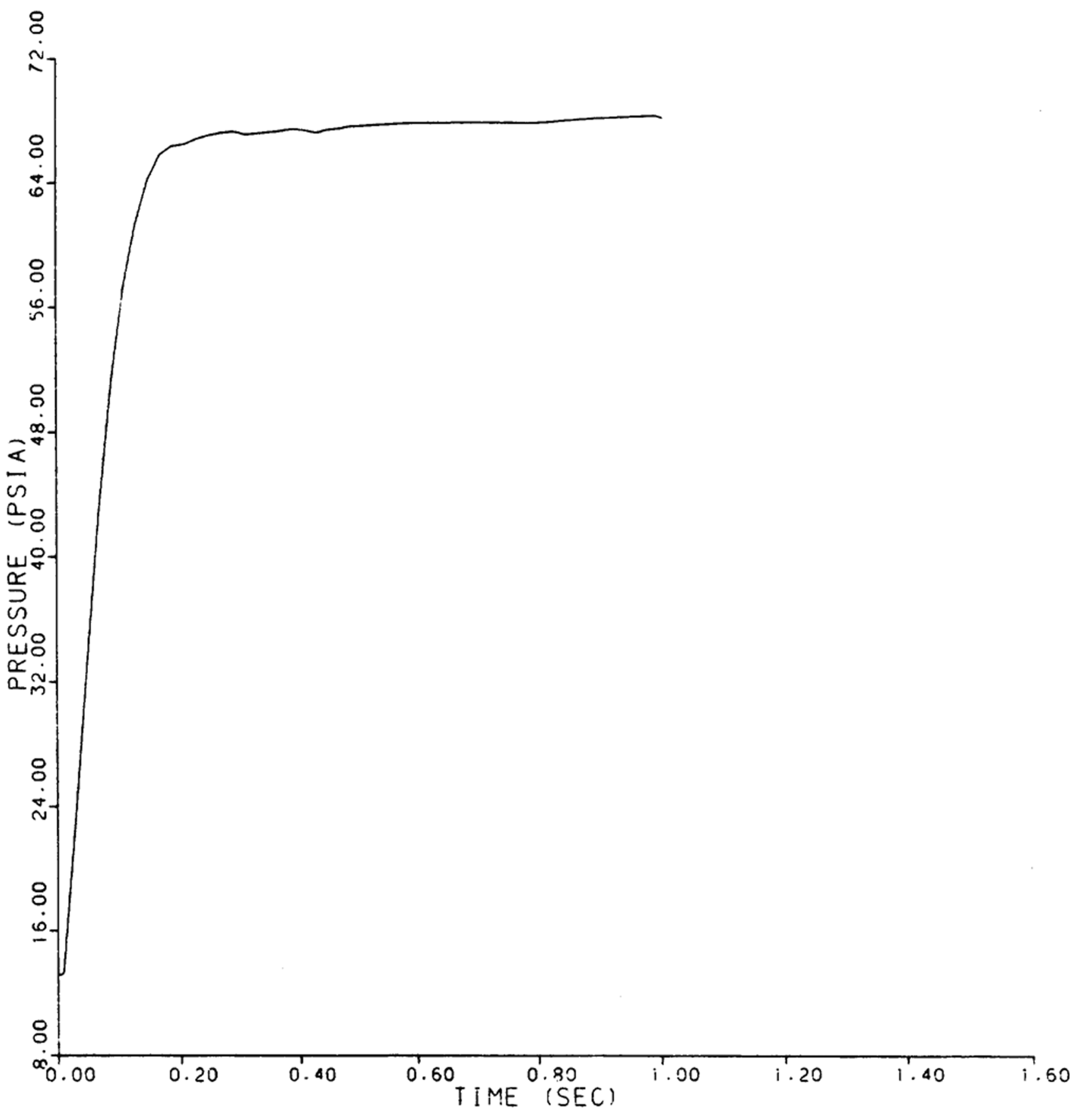
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE
RESPONSE – NODE E2

FIGURE 6.2.1–15 (SHEET 2 OF 65)



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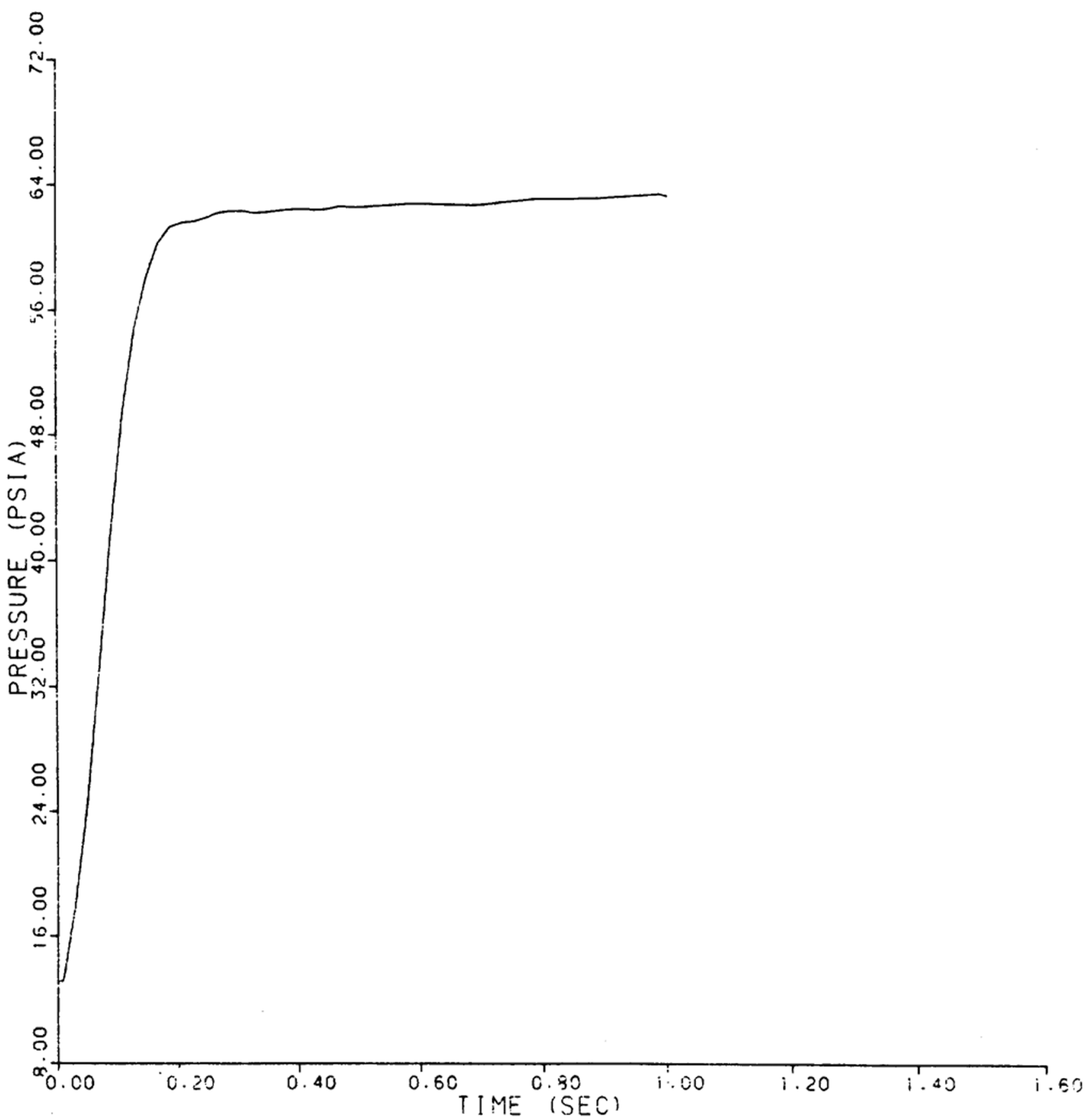
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E3
		FIGURE 6.2.1–15 (SHEET 3 OF 65)



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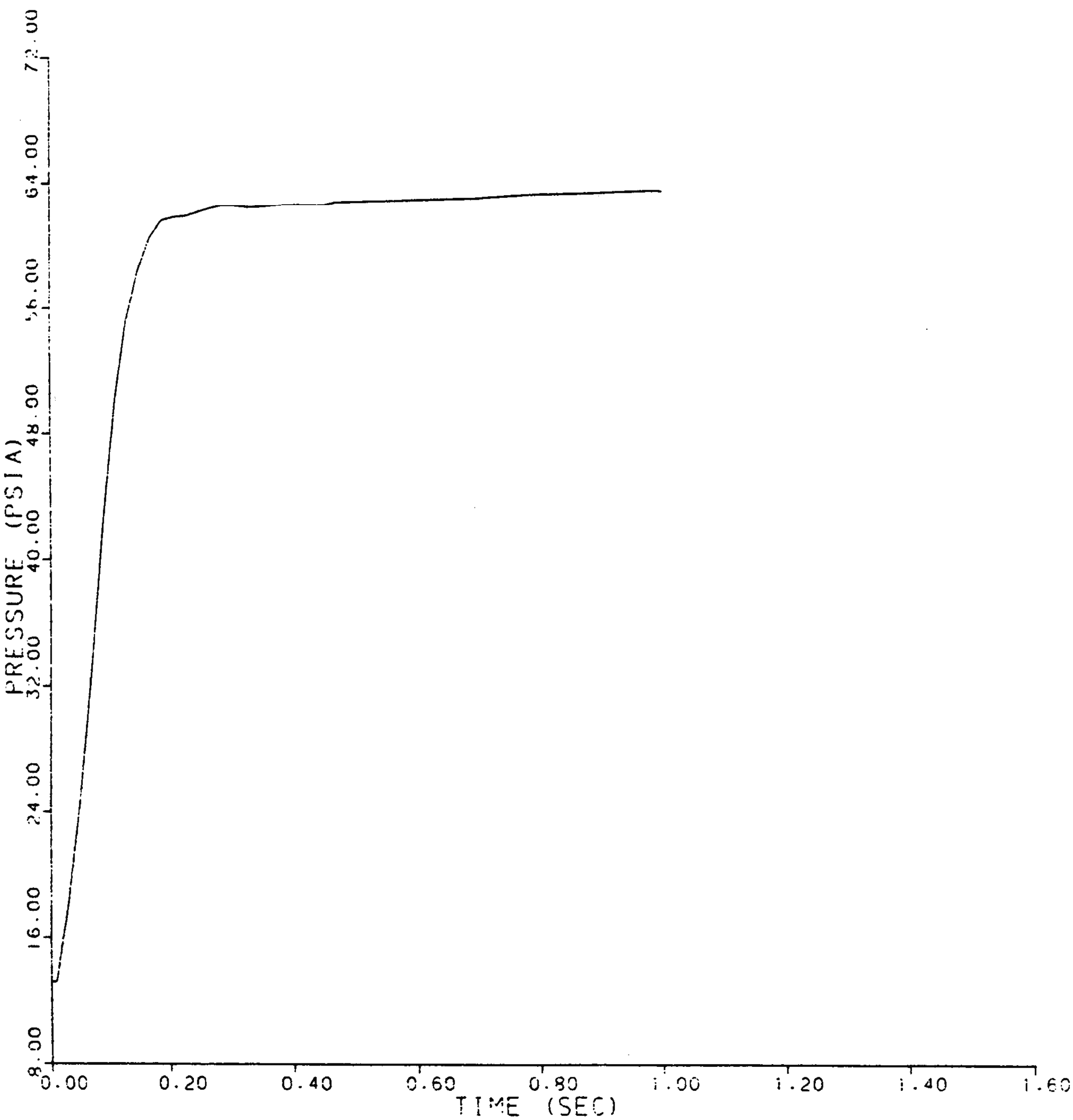
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E4
		FIGURE 6.2.1–15 (SHEET 4 OF 65)



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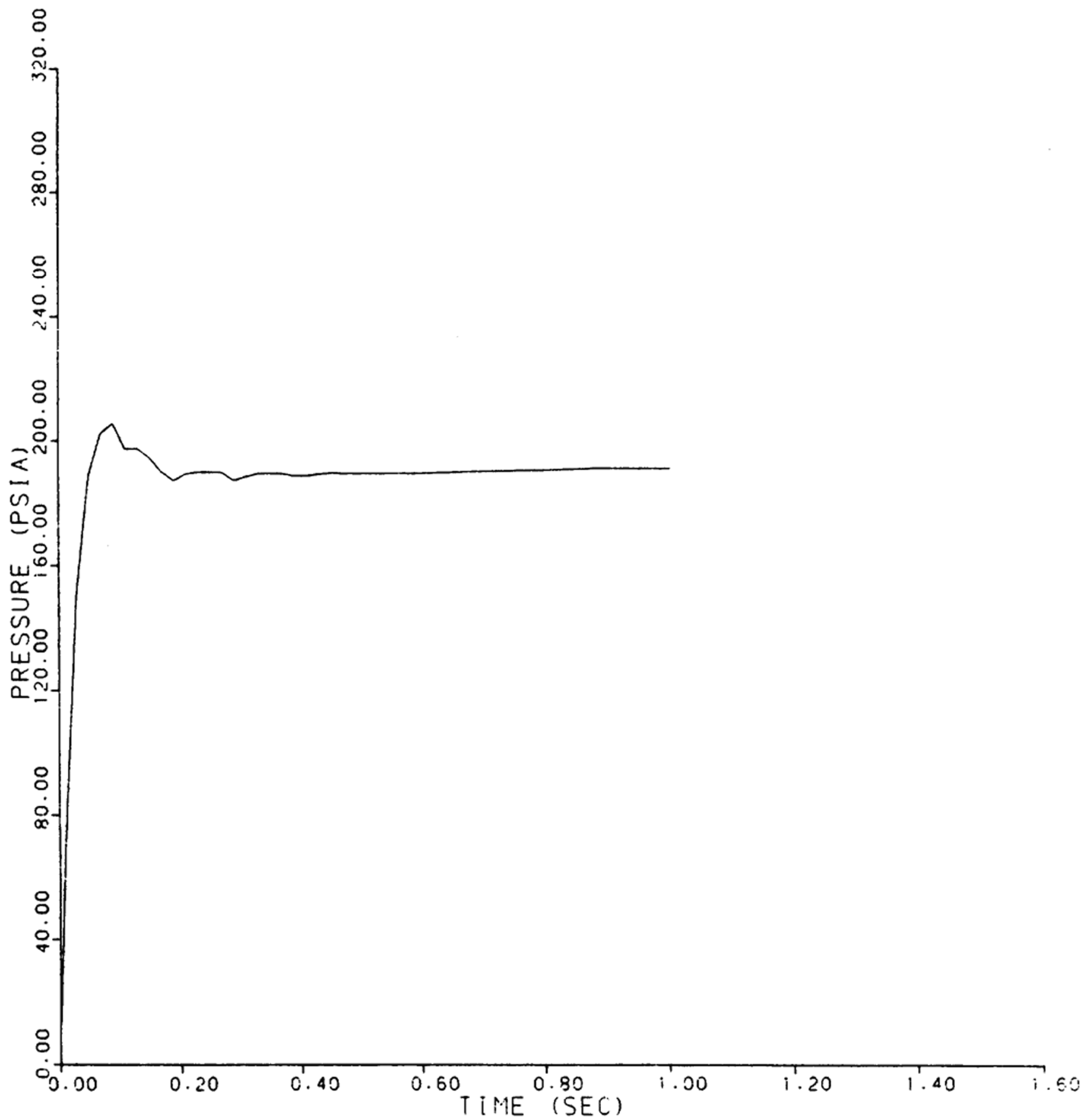
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E5 FIGURE 6.2.1–15 (SHEET 5 OF 65)



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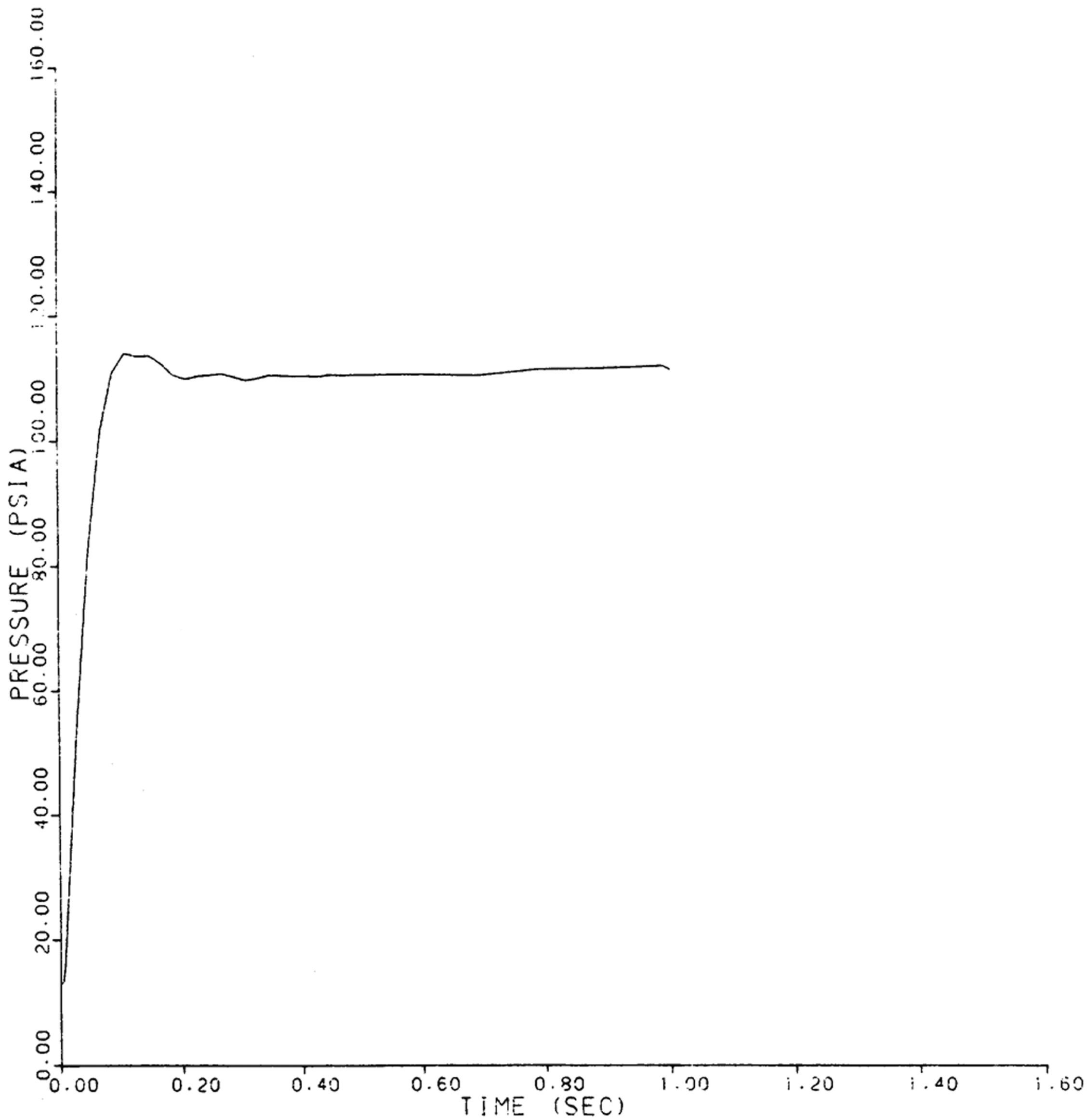
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE RESPONSE – NODE E6

FIGURE 6.2.1–15 (SHEET 6 OF 65)



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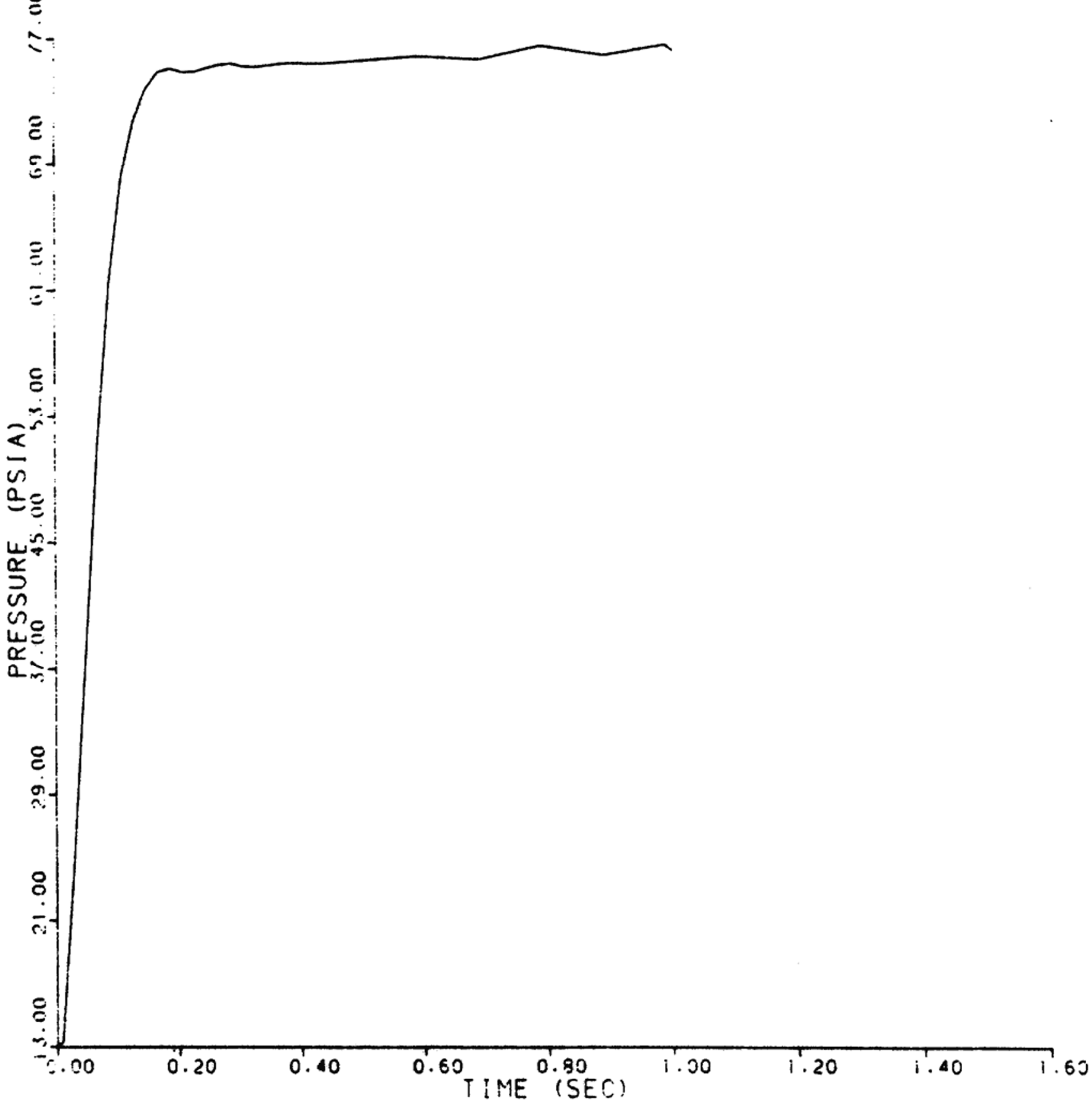
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE
RESPONSE – NODE E7

FIGURE 6.2.1–15 (SHEET 7 OF 65)



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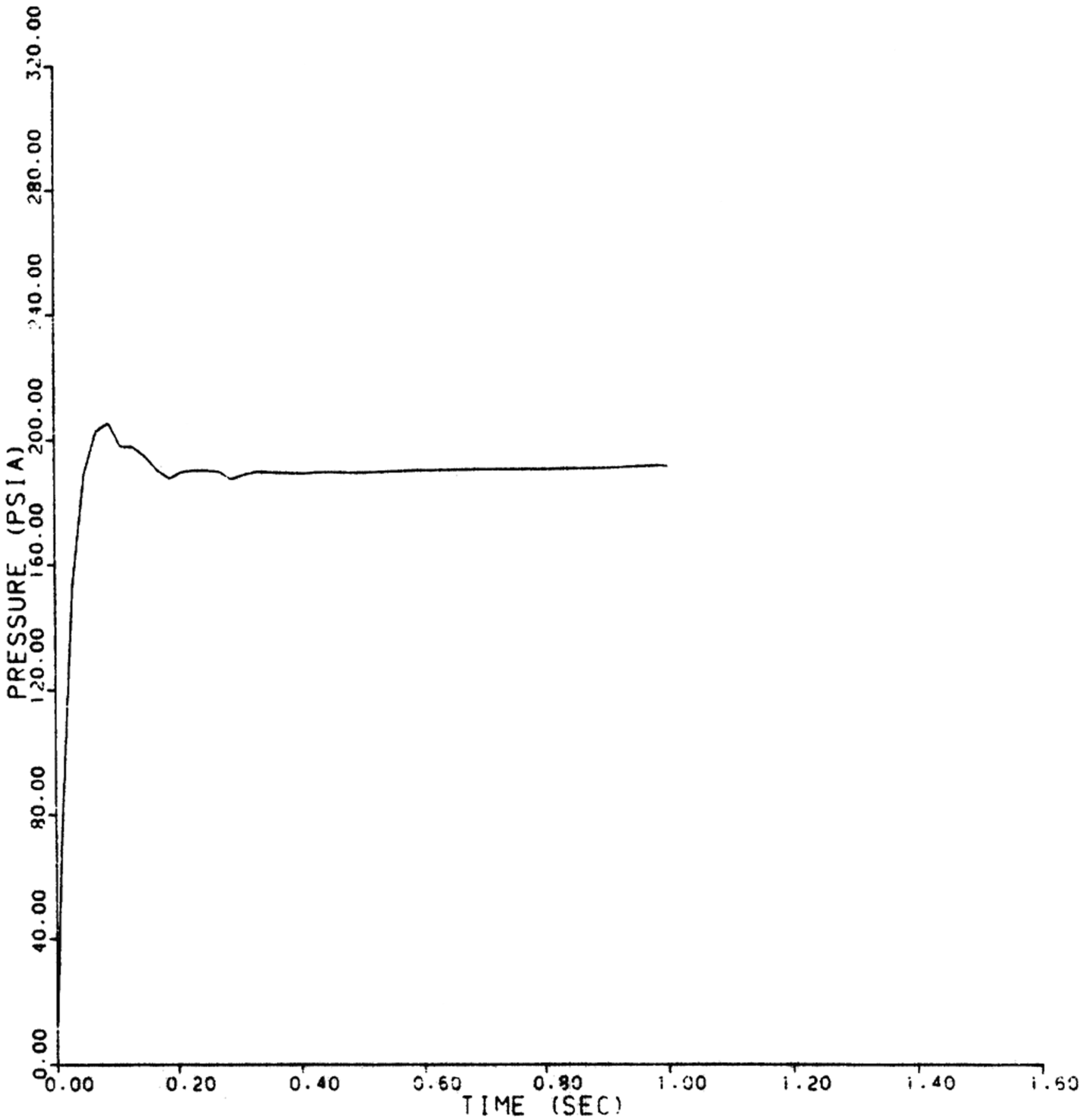
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E8
		FIGURE 6.2.1–15 (SHEET 8 OF 65)



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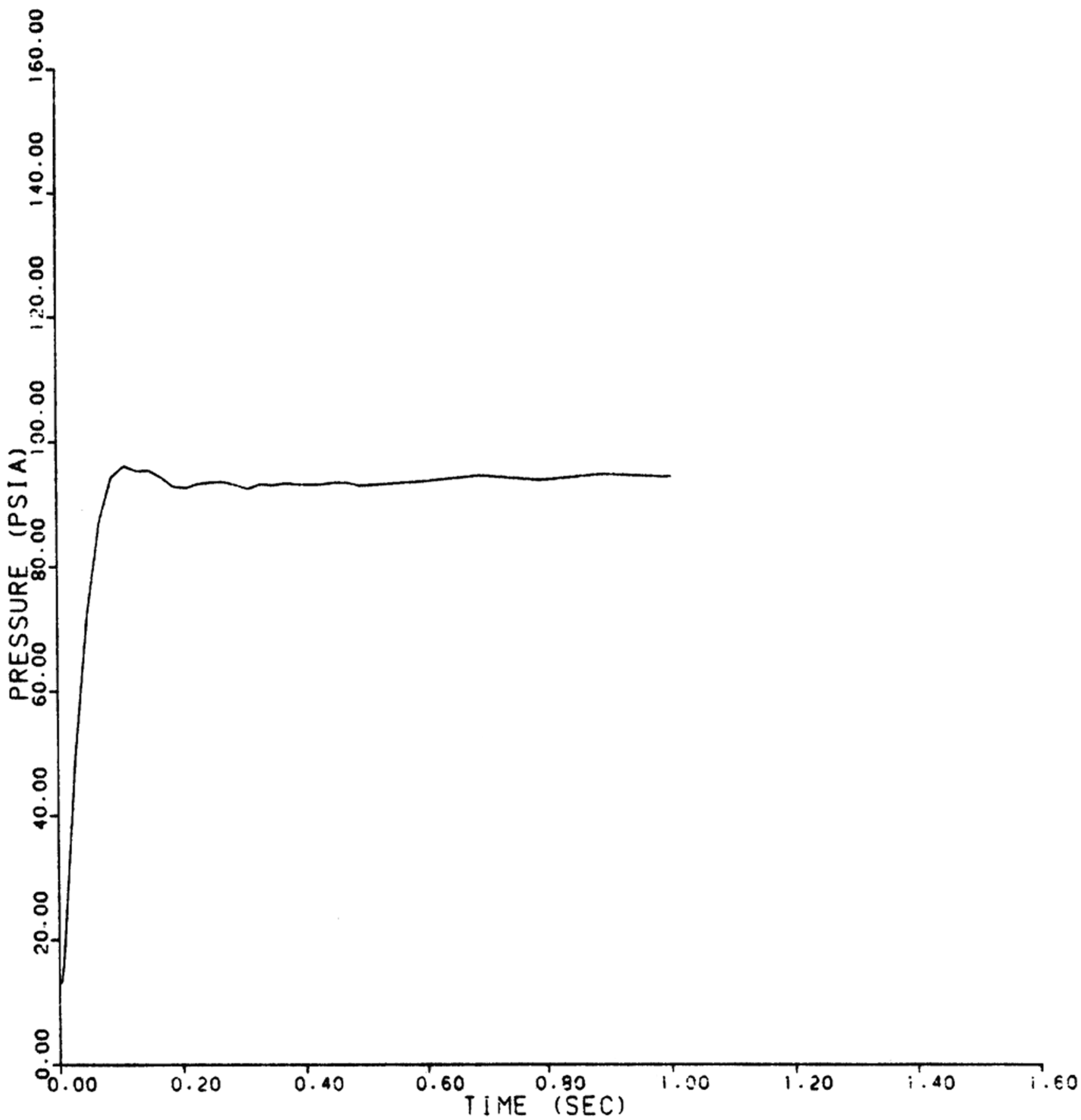
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE
RESPONSE – NODE E9

FIGURE 6.2.1–15 (SHEET 9 OF 65)



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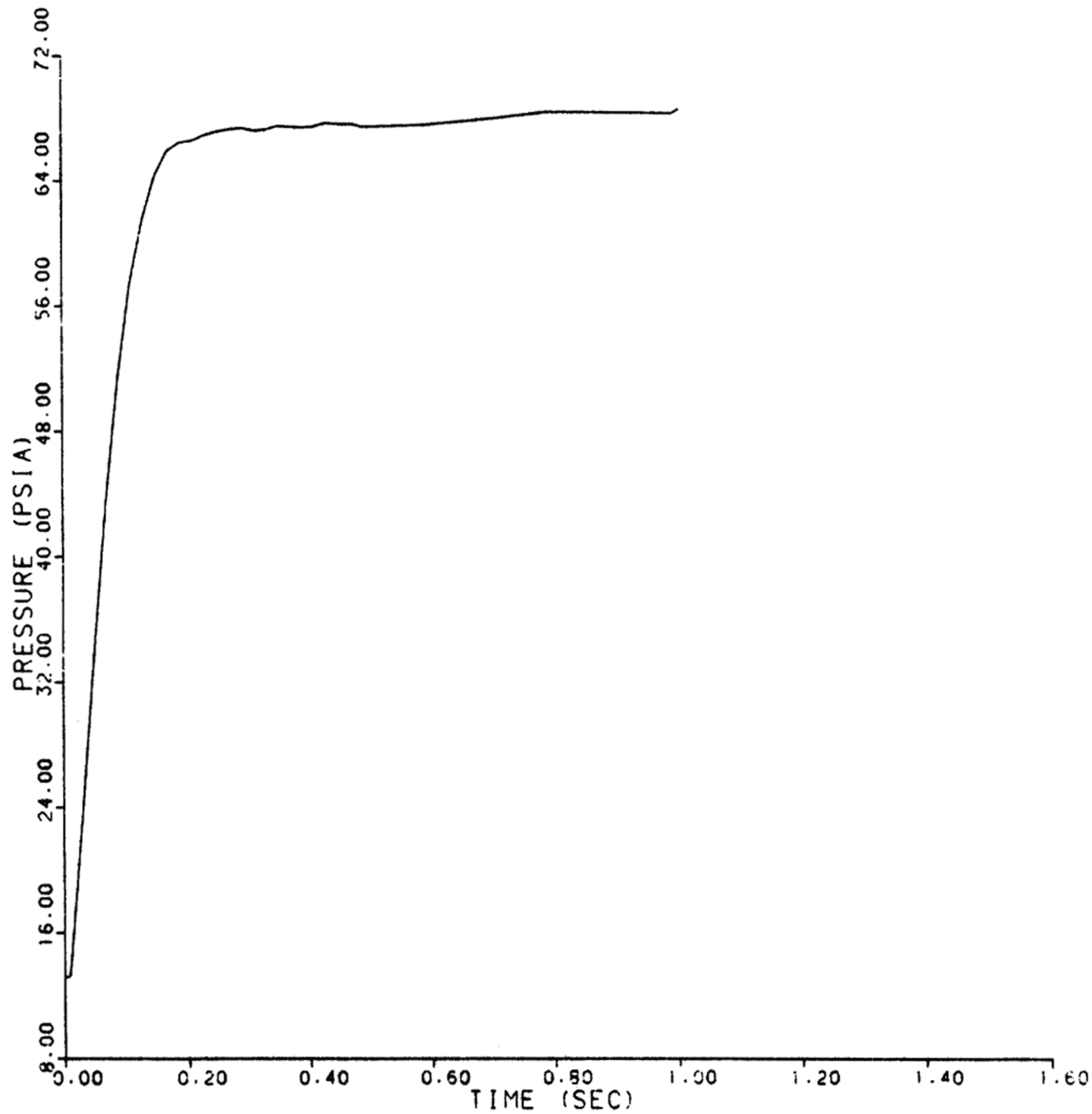
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE
RESPONSE – NODE E10

FIGURE 6.2.1–15 (SHEET 10 OF 65)



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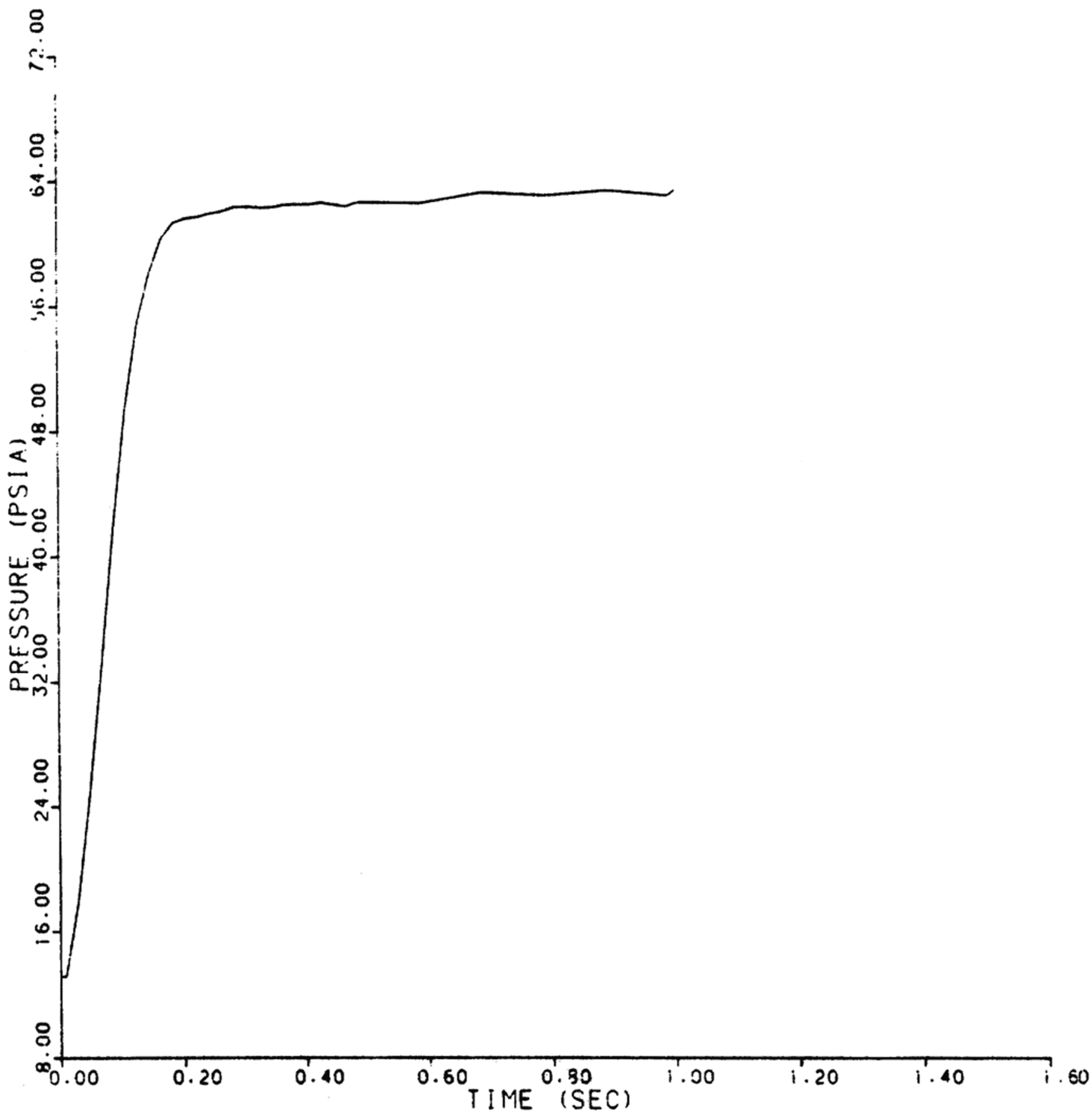
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE RESPONSE – NODE E11

FIGURE 6.2.1–15 (SHEET 11 OF 65)



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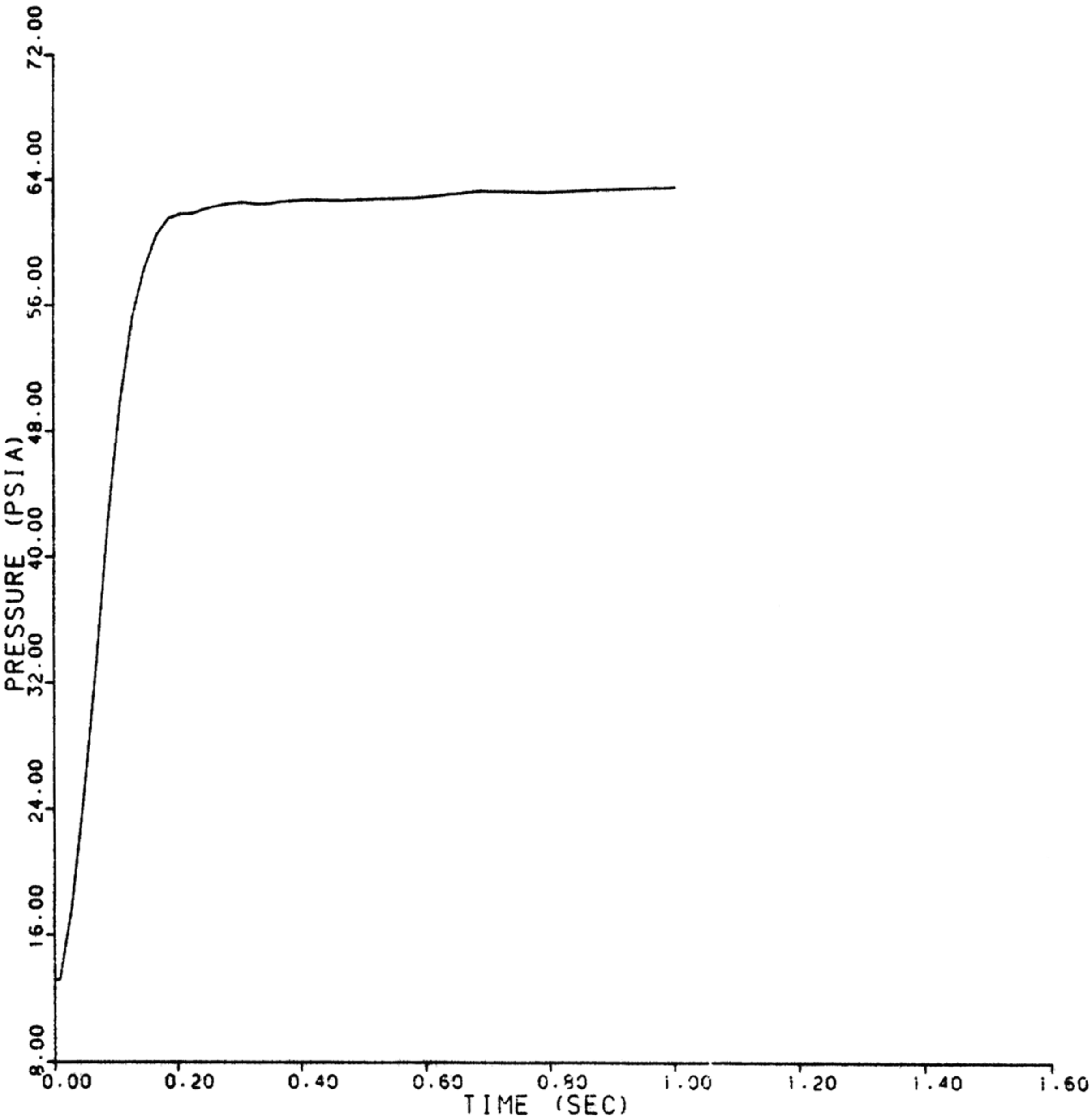
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	REACTOR CAVITY PRESSURE RESPONSE – NODE E12
		FIGURE 6.2.1–15 (SHEET 12 OF 65)



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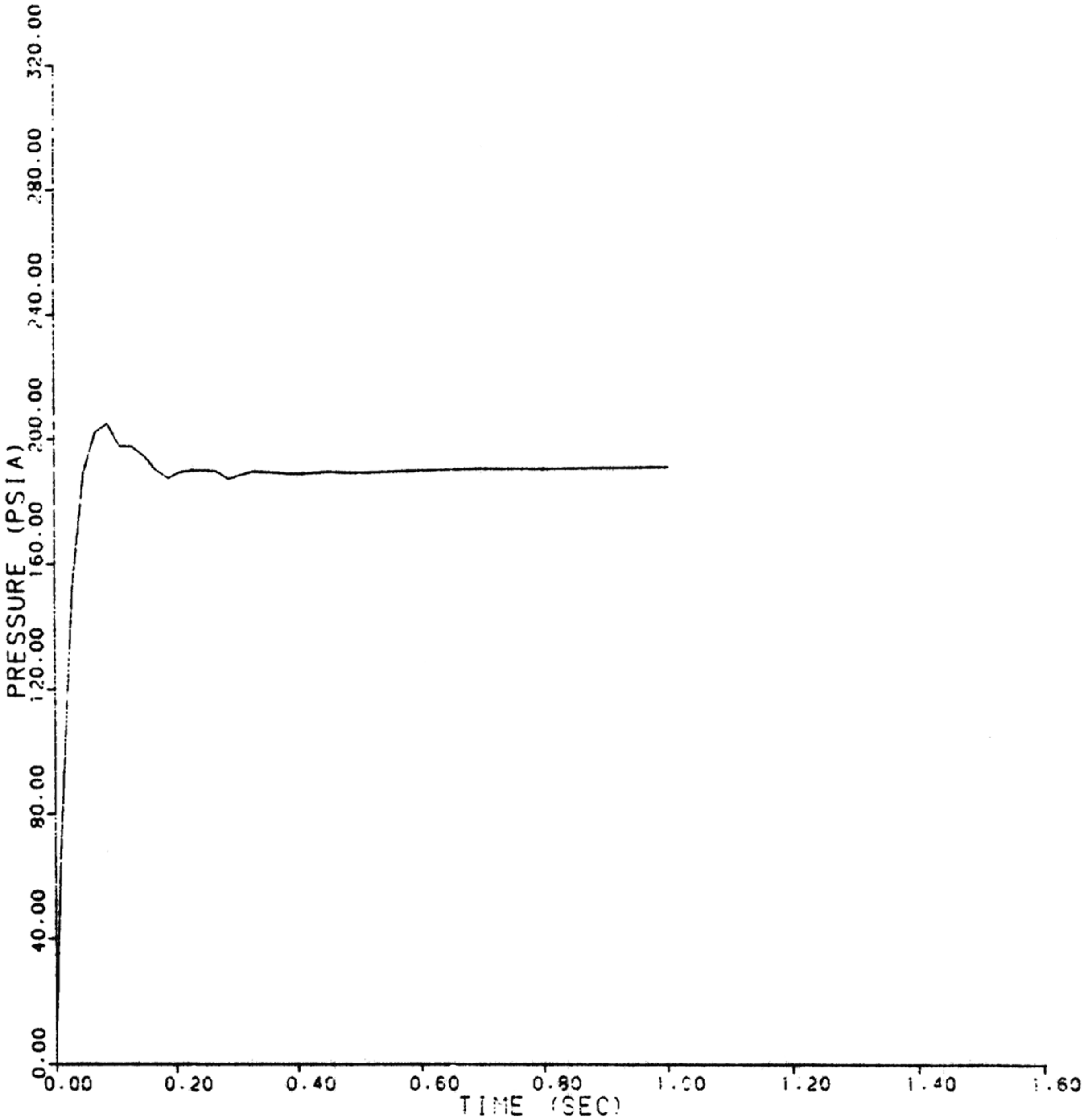
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR CAVITY PRESSURE RESPONSE – NODE E13
	FIGURE 6.2.1–15 (SHEET 13 OF 65)



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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REACTOR CAVITY PRESSURE
RESPONSE – NODE E14

FIGURE 6.2.1–15 (SHEET 14 OF 65)