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Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
1997 FINAL SAFETY ANALYSIS REPORT UPDATE

Attached are ten copies of the changed pages for the 1997 update to our final safety analysis report (FSAR). These pages are being transmitted to you according to the provisions of 10 CFR 50.71(e). Instructions for incorporating the update are included with each copy.

In addition to vertically barring the specific change, changed pages have been dated "July 1997" in the lower right corner.

We hereby certify that the information contained in this FSAR update, to our knowledge, accurately presents changes made to the plant from January 22, 1996, through January 22, 1997. We note that we are in the process of performing a revalidation of the FSAR to the plant design and operations. This effort is expected to be completed in late 1998.

Sincerely,

A handwritten signature in cursive script, reading "E. E. Fitzpatrick".

E. E. Fitzpatrick
Vice President

vlb

Attachments

c: A. A. Blind
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. R. Padgett

050007

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Fig. 14.1.6-2	Fig. 14.1.6-2

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Fig. 14.1.6-4	Fig. 14.1.6-4
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Fig. 14.1.6-8	Fig. 14.1.6-8
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14.2.1-10	14.2.1-10
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14.2.4-4	14.2.4-4
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14.3.6-30	14.3.6-30
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14.4.2-5	14.4.2-5 (Reissue)
14.4.2-6	14.4.2-6
14.4.2-15	14.4.2-15 (Reissue)
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14.4.3-1	14.4.3-1
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14.4.4-5	14.4.4-5 (Reissue)
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14.1.9-4	14.1.9-4
14.2.2-1	14.2.2-1
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14.2.2-3	14.2.2-3
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Fig. 2.5-1	1982
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Fig. 2.5-3d	1982
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Fig. 2.5-3f	1982
Fig. 2.5-3g	1982
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Fig. 2.5-3j	1982
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		4.1-6	1982
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		4.1-10	1982
		4.1-11	1982
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		4.1-22	1997
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Fig. 4.2-6	1982
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Fig. 4.2-8	1982
Fig. 4.2-9	1982
Fig. 4.2-9 Ref.. (4pp)	1982
4.3-1	1982
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Fig. 4.3-1	1982
Fig. 4.3-2	1982
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Fig. 4.3-4	1982
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Fig. 4.3-7	1990
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Fig. 4.5-1	1982
Fig. 4.5-2	1982
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Fig. 5.2-2	1982
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Fig. 5.2-4	1982
Fig. 5.2-5	1988
Fig. 5.2.2-1	1982
Fig. 5.2.2-1A	1982
Fig. 5.2.2-2	1982
Fig. 5.2.2-2A	1982
Fig. 5.2.2-3	1982
Fig. 5.2.2-4	1982
Fig. 5.2.2-4A	1982
Fig. 5.2.2-4B	1982
Fig. 5.2.2-5	1982
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Fig. 5.2.2-6C	1982
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Fig. 5.2.2-7	1982
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Fig. 5.2.2-14	1982
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Fig. 5.2.2-17	1982
Fig. 5.2.2-18	1982
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Fig. 5.2.2-23	1982
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Fig. 5.2.2-26	1982
Fig. 5.2.2-27	1982
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Fig. 5.2.2-30	1982
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Fig. 5.2.2-32	1982
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Fig. 5.2.2-34	1982
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Fig. 5.2.2-36	1982
Fig. 5.2.2-37	1982
Fig. 5.2.2-38	1982
Fig. 5.2.2-39	1982
Fig. 5.2.2-40	1982
Fig. 5.2.2-41	1982
Fig. 5.2.2-42	1982
Fig. 5.2.2-43	1982
Fig. 5.2.2-44	1982
Fig. 5.2.2-45	1982
Fig. 5.2.2-46	1982
Fig. 5.2.2-47	1982
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Fig. 5.2.2-49	1982
Fig. 5.2.2-50	1982
Fig. 5.2.2-51	1982
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Fig. 5.2.2-51B	1982
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Fig. 5.2.2-56A	1982
Fig. 5.2.2-57	1982
Fig. 5.2.2-57A	1982
Fig. 5.2.2-58	1982
Fig. 5.2.2-58A	1982
Fig. 5.2.2-59	1982
Fig. 5.2.2-59A	1982
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Fig. 5.2.2-59C	1982
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Fig. 5.2.2-59E	1982
Fig. 5.2.2-60	1991
Fig. 5.2.2-60A	1991
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Fig. 5.2.2-60C	1982
Fig. 5.2.2-61	1982
Fig. 5.2.2-62	1982
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Fig. 5.2.2-64	1982
Fig. 5.2.2-65	1982
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5.3-9	1982
5.3-10	1997
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5.3-13	1997
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5.3-16	1997
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5.3-32	1982
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5.3-36	1997
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Fig. 5.3-3	1997
Fig. 5.3-4	1982
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Fig. 5.5-3	1982
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6.2-19	1982
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6.2-22	1990
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Fig. 6.2-2	1982
Fig. 6.2-3	1982
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Fig. 6.2-5	Deleted 1997
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7.2-68	1991
7.2-69	1997
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Fig. 7.2-1a Deleted	1996
Fig. 7.2-1b Deleted	1996
Fig. 7.2-1c Deleted	1996
Fig. 7.2-1d Deleted	1996
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		7.5-17	1982
		7.5-18	1991
		7.5-19	1982
		7.5-20	1997
		7.5-21	1989
		7.5-22	1989
		7.5-23	1989
	Fig. 7.5-1		1982
	Fig. 7.5-2		1982
	Fig. 7.5-3		1982
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	7.7-7		1991
	7.7-8		1991
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		7.8-9	1994
		7.8-10	1993
		7.8-11	1992
		7.8-12	1992
		7.8-13	1992
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14.1-17	1997
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14.1-19	1997
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Fig. 14.1-1	1997
Fig. 14.1-2	1997
Fig. 14.1-3	1997
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Fig. 14.1-6	1992
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Fig. 14.1.1-2	1997
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Fig. 14.1.2-1	1997
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TABLE 1.2-1

COMPARISON OF DESIGN PARAMETERS **

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
THERMAL AND HYDRAULIC DESIGN PARAMETERS						
1	Total Primary Heat Output, MWe	3250	3250	2758	1518.5	2200
2	Total Core Heat Output, Btu/hr	$11,090 \times 10^6$	$11,090 \times 10^6$	9413×10^6	5181×10^6	7479×10^6
3	Heat Generated in Fuel, %	97.4	97.4	97.4	97.4	97.4
4	Maximum thermal Overpower	12%	12%	12%	12%	12%
5	System Pressure, Nominal, psia	2250	2250	2250	2250	2250
6	System Pressure, Minimum Steady State, psia	2220	2220	2220	2220	2220
Hot Channel Factors						
7	Heat Flux, F_q	2.79	2.79	3.23	2.80	3.23
8	Enthalpy Rise, $F_{\Delta H}$	1.60	1.60	1.77	1.60	1.77
9	DNB Ratio at Nominal Operating Conditions	1.97	2.02	2.00	2.11	1.81
10	Minimum DNBR for Design Transients	1.30	1.30	1.30	1.30	1.30
Coolant Flow						
11	Total Flow Rate, lb/hr	135.6×10^6	133.0×10^6	136.3×10^6	66.7×10^6	101.5×10^6
12	Effective Flow Rate for Heat Transfer, lb/hr	129.5×10^6	128.9×10^6	130×10^6	63.6×10^6	97.0×10^6
13	Effective Flow Area for Heat Transfer, ft ²	51.4×10^3	51.4×10^3	51.4×10^3	51.4×10^3	51.4×10^3
14	Average Velocity Along Fuel Rods, ft/sec	15.5	15.3	15.4	15.0	14.3
15	Average Mass Velocity, lb/hr-ft ²	2.53×10^6	2.52×10^6	2.53×10^6	2.37×10^6	2.32×10^6
Coolant Temperature, °F						
16	Design Nominal Inlet	536.3	530.2	543	552.5	546.2
17	Maximum Inlet Due to Instrumentation Error and Deadband, °F	540.3	534.2	547	556.5	550.2
18	Average Rise in Vessel, °F	63.0	64.1	53.0	57.6	55.9
19	Average Rise in Core	65.7	66.8	55.5	60.0	58.3
20	Average in Core	570.3	564.8	571.0	582.5	575.4
21	Average in Vessel	567.8	563.2	569.5	581.3	574.2
22	Nominal Outlet of Hot Channel	667.5	631.7	633.5	642.9	642
23	Average Film Coefficient, Btu/hr-ft ² -°F	5850	5800	5790	5600	5400

** This table is retained for historical purposes only. It compares original Cook Plant parameters to other similar nuclear plants.

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TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
24	Average Film Temperature Difference, °F	35.4	35.6	30.3	31.0	31.8
	Heat Transfer at 100% Power					
25	Active Heat Transfer Surface Area,	52,200	52,200	52,200	28,715	42,460
26	ft ²	207,900	207,900	175,600	175,800	171,600
27	Average Heat Flux, Btu/hr-ft ²	579,600	579,600	567,300	491,000	554,200
28	Maximum Heat Flux, Btu/hr-ft ²	6.7	6.7	5.7	5.7	5.5
29	Average Thermal Output, kw/ft	18.8	18.8	18.4	16.0	17.0
	Maximum Thermal Output, kw/ft					
30	Maximum Clad Surface Temperature at Nominal Pressure, °F	657	657	657	657	657
	Fuel Central Temperature, °F					
31	Maximum at 100% Power	4250	4250	4090	3750	4030
32	Maximum at Overpower	4500	4500	4380	4000	4300
33	Thermal Output, kw/ft at Maximum Overpower	21.1	21.1	20.6	17.9	20.0
	CORE MECHANICAL DESIGN PARAMETERS					
	Fuel Assemblies					
34	Design	RCC Canless 15x15	RCC Canless 15x15	RCC Canless 15x15	RCC Canless 14x14	RCC Canless 15x15
35	Rod Pitch, in.	0.563	0.563	0.563	0.556	0.563
36	Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426	7.763 x 7.763	8.426 x 8.426
37	Fuel Weight (as UO ₂), pounds	216,600	216,600	216,000	120,130	176,200
38	Total Weight, pounds	276,000	276,000	276,000	154,519	226,200
39	Number of Grids per Assembly	7	7	9	7	7
	Fuel Rods					
40	Number	39,372	39,372	39,372	21,659	32,028
41	Outside Diameter, in.	0.422	0.422	0.422	0.422	0.422
42	Diametral Gap, in. (Region 1, 2)	0.0075	0.0075	0.0065	0.0065	0.0065
	(Region 3)	0.0085	0.0085			
43	Clad Thickness, in	0.0243	0.0243	0.0243	0.0243	0.0243
44	Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
	Fuel Pellets					
45	Material	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered
46	Density (% of Theoretical)	94-93-92	94-93-92	94-92-91	94-92-91	94-92-91
47	Diameter, in. (Region 1, 2)	0.3659	0.3659	0.3669	0.3669	0.3669
	(Region 3)	0.3649	0.3649			
48	Length, in.	0.6000	0.6000	0.6000	0.6000	0.6000
	Rod Cluster Control Assemblies					
49	Neutron Absorber	5% Cd-15% In-80%Ag	5% Cd-15% In-80%Ag	5% Cd-15% In-80%Ag	5% Cd-15% In-80%Ag	5% Cd-15% In-80%Ag
50	Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
51	Clad Thickness, in.	0.019	0.019	0.019	0.019	0.019
52	Number of Cluster	53	53	53	37	53
53	Number of Control Rods per Cluster	20	20	20	16	20

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.	DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BRACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
	Core Structure				
54	Core Barrel I.D./O.D., in.	148.0/152.15	148.0/152.5	148.0/152.5	109.0/112.5
55	Thermal Shield I.D./O.D., in.	158.5/164.0	158.5/164.0	158.5/164	115.3/122.5
	FINAL NUCLEAR DESIGN DATA				
	Structural Characteristics				
56	Fuel Weight (As UO ₂), lbs	216,600	216,600	216,000	120,130
57	Clad Weight, lbs	44,547	44,547	44,600	24,260
58	Core Diameter, in (Equivalent)	132.7	132.7	132.5	96.5
59	Core Height, in. (Active Fuel)	144	143.4	144	144
	Reflector Thickness and Composition				
60	Top - Water plus Steel, in.	10	10	10	10
61	Bottom - Water plus Steel, in.	10	10	10	10
62	Side - Water plus Steel, in.	15	15	15	15
63	H ₂ O/U ₂ (Cold volume Ratio)	4.09	4.09	4.18	4.20
64	Number of Fuel Assemblies	193	193	193	121
65	UO ₂ Rods per Assembly	204	204	204	179
	Performance Characteristics				
66	Loading Technique	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform
	Fuel Discharge Burnup, MWD/MTU				
67	Average First Cycle	14,000	14,000	14,200	15,100
68	Equilibrium Core Average	21,800	21,800	24,700	33,000
	Feed Enrichments, weight %				
69	Region 1	2.25	2.25	2.2	2.27
70	Region 2	2.80	2.80	2.7	3.03
71	Region 3	3.30	3.30	3.2	3.40
	Equilibrium	3.2	3.2	-	3.40
	Control Characteristics				
	Effective Multiplication (Beginning of Life)				
72	Cold, No Power, Clean	1.183	1.183	1.257	1.211
73	Hot, No Power, Clean	1.154	1.154	1.999	1.167
74	Hot, Full Power, Xe and Sm Equilibrium	1.092	1.092	1.152	1.113

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TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
	Rod Cluster Control Assemblies					
75	Material	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag
76	Number of RCC Assemblies	53	53	53	37	53
77	Number of Absorber Rods per RCC Assembly	20	20	20	16	20
78	Total Rod Worth	See Table 3.2.1-3	See Table 3.2.1-3	See Table 3.2.1-3	See Table 3.2.1-3	See Table 3.2.1-3
	Boron Concentration					
	To shut reactor down with no rods					
79	Inserted, clean ($k_g = .99$) Cold/hot, ppm/ppm	1408/1265	1408/1265	1480/1370	1598/1676	1250/1210
	To control at power with no rods inserted,					
80	clean/equilibrium xenon and samarium,	1168/850	1168/850	1200/780	1465/1007	1000/920
81	ppm/ppm	1% $\Delta k/k/85$ ppm	1% $\Delta k/k/85$ ppm	1% $\Delta k/k/89$ ppm	1% $\Delta k/k/130$ ppm	7.3 $\Delta k/k$
82	Boron worth, Hot Boron worth, Cold	1% $\Delta k/k/70$ ppm	1% $\Delta k/k/70$ ppm	1% $\Delta k/k/72$ ppm	1% $\Delta k/k/98$ ppm	5.6 $\Delta k/k$
	Kinetic Characteristics					
83	Moderator Temperature, Coefficient, $\Delta k/k/^\circ F$	-0.3×10^{-4} -3.2×10^{-4}	-0.3×10^{-4} -3.2×10^{-4}	-0.3×10^{-4} -3.0×10^{-4}	$+0.3 \times 10^{-4}$ -3.5×10^{-4}	$+0.3 \times 10^{-4}$ -3.5×10^{-4}
84	Moderator Pressure Coefficient, $\Delta k/k/psi$	$+0.3 \times 10^{-4}$ $+4.0 \times 10^{-6}$	$+0.3 \times 10^{-4}$ $+4.0 \times 10^{-6}$	-0.3×10^{-4} $+3.0 \times 10^{-6}$	-0.3×10^{-4} 3.5×10^{-6}	-0.3×10^{-4} 3.5×10^{-6}
85	Moderator Density Coefficient $\Delta k/k/g/cm^3$	-0.1×10^{-3} $+0.8 \times 10^{-3}$	-0.1×10^{-3} -0.8×10^{-3}	+0.03 to -0.30	-0.10 to -0.30	$+0.5 \times 10^{-3}$ -2.5×10^{-3}
86	Doppler Coefficient, $\Delta k/k/^\circ F$	-1.0×10^{-3} -1.7×10^{-3}	-1.0×10^{-3} -1.7×10^{-3}	-1.1×10^{-3} -1.8×10^{-3}	-1×10^{-3} -1.6×10^{-3}	-1×10^{-3} -1.6×10^{-3}
	REACTOR COOLANT SYSTEM CODE REQUIREMENTS					
87	Component Reactor Vessel	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A

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TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
88	Steam Generator					
89	Tube Side	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
	Shell Side	ASME III Class C*	ASME III Class C*	ASME III Class C*	ASME III Class C*	ASME III Class C*
90	Pressurizer	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
91	Pressurizer Relief Tank	ASME III Class C	ASME III Class C	ASME III Class C	ASME Class C	ASME Class C
92	Pressurizer Safety Valves	ASME III	ASME III	ASME III	ASME III	
93	Reactor Coolant Piping	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1
	PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM					
94	Reactor Primary Heat Output, MWT	3250	3250	2758	1518.5	2200
95	Reactor Primary Heat Output, Btu/hr	11,090 x 10 ⁶	11,090 x 10 ⁶	9413 x 10 ⁶	5181 x 10 ⁶	7508 x 10 ⁶
96	Operating Pressure, psig	2235	2235	2235	2235	2235
97	Reactor Inlet Temperature	536.3	530.2	543	552.5	546.2
98	Reactor Outlet Temperature	599.3	594.3	596.0	610.0	602.1
99	Number of Loops	4	4	4	2	3
100	Design Pressure, psig	2485	2485	2485	2485	
101	Design Temperature, °F	650	650	650	650	650
102	Hydrostatic Test Pressure (Cold), psig	3107	3107	3110	3110	3110
103	Coolant Volume, including pressurizer, cu. ft.	12,612	12,710	12,600	6450	9088
104	Total Reactor Flow, gpm	350,000	350,000	178,000	268,500	
104A	Total Reactor Flow lb/sec	37,765	31,765			
	PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL					
105	Material	Same as others See Table 4.2-1	Same as others See Table 4.2-1	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel
106	Design Pressure, psig	2485	2485	2485	2485	2485
107	Design Temperature, °F	650	650	650	650	650
108	Operating Pressure, psig	2235	2235	2235	2235	2235
109	Inside Diameter of Shell, in.	173	173	173	132.0	155.5
110	Outside Diameter Across Nozzles, in.	262-7/16	262-7/16	262-7/16	244 1/16	236
111	Overall Height of Vessel & Enclosure Head, ft-in.	43-9 11/16	43-9 23/32 (Unit 1) 43-9 15/16 (Unit 2)	43 9-11/16	39-0	41-6
112	Minimum Clad Thickness, in.	5/32	5/32	5/32	5/32	5/32

* The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

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TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS						
113	Number of Units	4	4	4	2	3
114	Type	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator
115	Tube Material	Inconel	Inconel	Inconel	Inconel	Inconel
116	Shell Material	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel
117	Tube Side Design Pressure, psig	2485	2485	2485	2485	2485
118	Tube Side Design Temperature, °F	650	650	650	650	650
119	Tube Side Design Flow, lb/hr	33.9 x 10 ⁶	33.8 x 10 ⁶	34.1 x 10 ⁶	33.4 x 10 ⁶	33.9 x 10 ⁶
120	Shell Side Design Pressure, psig	1085	1085	1085	1085	1085
121	Shell Side Design Temperature, °F	600	600	556	556	556
122	Operating Pressure, Tube Side, Nominal psig	2235	2235	2235	2235	2235
123	Operating Pressure, Shell Side, Max, psig	1085 (design)	1085 (design)	1105.3	1020	1020
124	Maximum Moisture at Outlet at Full Load, %	1/4	1/4	1/4	1/4	1/4
125	Hydrostatic Test Pressure, Tube Side (cold), psig	3107	3107	3110	3110	3110
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS						
126	Number of Units	4	4	4	4	4
127	Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge
128	Design Pressure, psig	2485	2485	2485	2485	2485
129	Design Temperature, °F	650	650	650	650	650
130	Operating Pressure, Nominal, psig	2235	2235	2235	2235	2235
131	Suction Temperature, °F	539	539	556	551.5	546.5
132	Design Capacity, gpm	88,500	87,500	80,000	80,000	88,500
133	Design Head, ft.	277	277	252	259	261
134	Hydrostatic Test Pressure (cold), psig	3107	3107	3110	3110	3110
135	Motor Type	AC Induction single speed	AC Induction single speed air cooled	A-C Induction single speed	AC Induction single speed air cooled	AC Induction single speed air cooled 6000 HP
136	Motor Rating (nameplate)	6000 HP	6000 HP	6000 HP	6000 HP	
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING						
137	Material	See Table 4.2-1	See Table 4.2-1	Austenitic SS	Austenitic SS	Austenitic SS
138	Hot Leg - I.D., in.	29	29	29	29	29
139	Cold Leg - I.D., in.	27-1/2	27-1/2	27-1/2	27-1/2	27-1/2
140	Between Pump and Steam generator - I.D., in.	31	31	31	31	31
141	Design Pressure, psig	2485	2485	2485	2485	2485

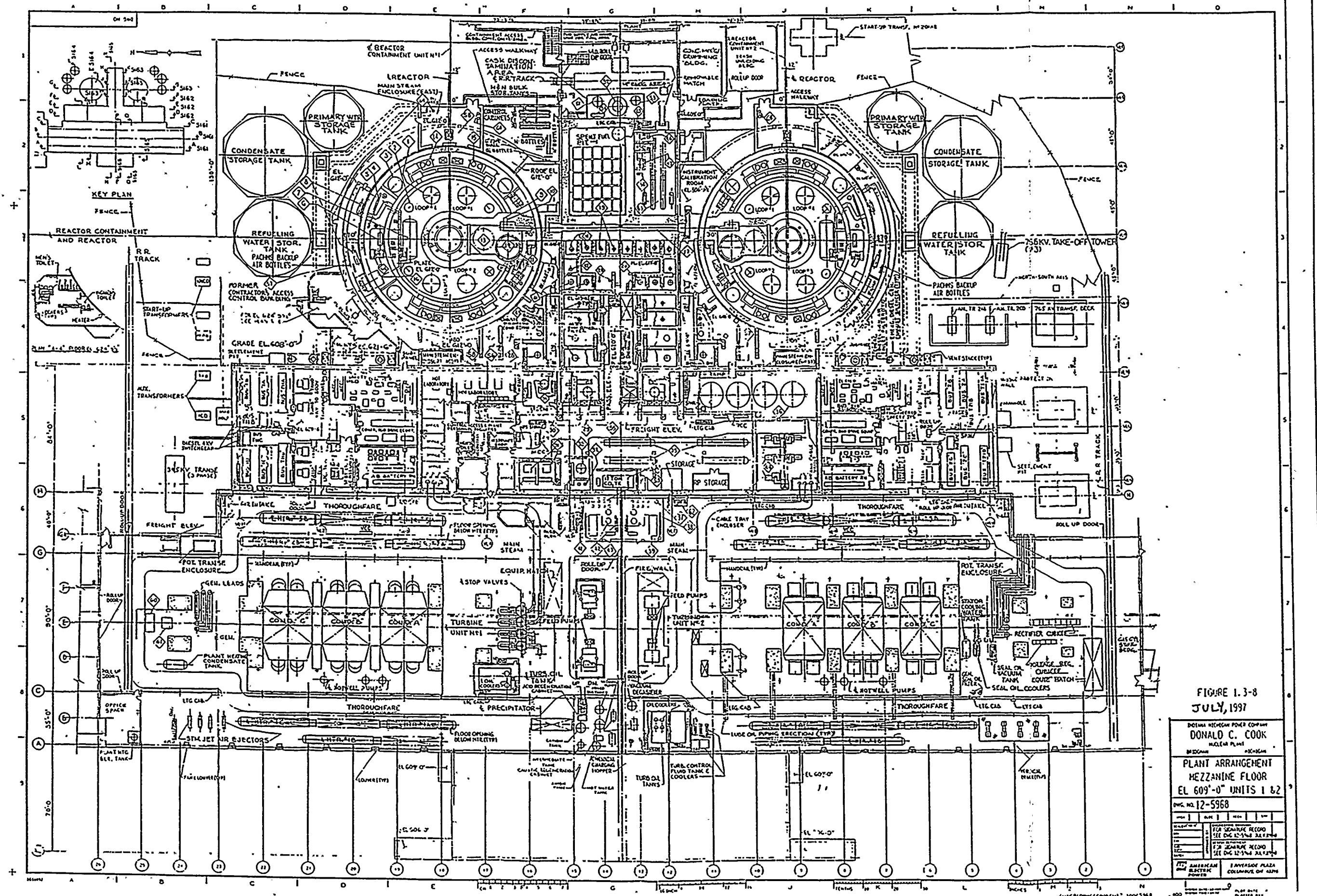


FIGURE 1.3-8
JULY, 1997

DOINA NICHAN POWER COMPANY	
DONALD C. COOK	
MECHANICAL PLANT	
PLANT ARRANGEMENT	
MEZZANINE FLOOR	
EL 609'-0" UNITS 1 & 2	
DWG. NO. 12-5968	
DATE	REV
12-5968	1
12-5968	2
12-5968	3
12-5968	4
12-5968	5
12-5968	6
12-5968	7
12-5968	8
12-5968	9
12-5968	10
12-5968	11
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12-5968	98
12-5968	99
12-5968	100

The plant was designed and constructed by the American Electric Power Service Corporation (AEPSC) which performed the function of Architect-Engineer and Constructor for Indiana Michigan Power Company (I&M). Westinghouse Electric Corporation designed and supplied the Nuclear Steam Supply Systems including the initial fuel assemblies for both Units 1 and 2 of the Donald C. Cook Nuclear Plant. Subsequent reload fuel assemblies for these units have been and will be procured from qualified suppliers such as Westinghouse.

In the design and construction of these units, AEPSC employed various contractors and sub-contractors; however, the ultimate responsibility for all work performed was assumed by AEPSC. AEPSC and I&M are responsible for the implementation of all functions associated with the operation, maintenance, modification and control of the Donald C. Cook Nuclear Plant.

2.1 SITE DESCRIPTION

2.1.1 SUMMARY

The site is located in a region devoted primarily to agriculture. There are no continuously occupied residences within 2160 feet of the reactor containment structures. The distances from the reactor containment structures to the areas defined in the Rules and Regulations Title 10 Chapter I Part 100 are as follows:

Exclusion area	650 acres
Minimum distance to exclusion area	2000 feet
Outer boundary of low population zone	2 miles
Population center distance	8 miles

The closest population center is the twin cities of Benton Harbor-St. Joseph, Michigan. The site, therefore, provides excellent isolation as well as low population densities over a wide area.

2.1.2 LOCATION

The site is located in Lake Township, Berrien County, Michigan, about 11 miles south-southwest of the center of Benton Harbor, Michigan. The axis point of the Cook Nuclear Plant is latitude 41° 58'32.07" and longitude 86° 33'54.87". Figure 2.1-1 shows the regional features of the area up to 60 miles from the site, while Figure 2.1-2 indicates the features within about 15 miles of the facility.

The site consists of about 650 acres along the eastern shore of Lake Michigan, with approximately 4350 feet of Lake frontage, and extends an average of about one and one quarter miles eastward from the lake.

The entire site, with the exception of the right of way for Interstate Route 94, about 400 feet from the eastern site boundary, is controlled by the Indiana Michigan Power Company (I&M). No residence is permitted inside the site boundaries.

Figure 2.1-3 shows a map of the plant site defining the plant property lines.

- (1) The boundary lines inside of which I&M exercises exclusive control of access are the property lines which are to the west of the Interstate 94. These property lines are also the boundary lines at which gaseous effluent limits apply. The line in the area of Lake Michigan is the shore line El 580'-0" extended by 100 feet toward the lake, up to which I&M exercises rights, besides those obtained to install, maintain and operate the condenser cooling water intake and discharge pipes. Riparian rights extend to the low water line which in consideration of the lake bottom movement is approximately 100 feet outward from the elevation 580' line.
- (2) The points on the plant structure from which gaseous effluent containing, or potentially containing, radioactivity will be released; and the distance of each from the nearest boundary line have been shown and tabulated on the map (see Figure 2.5-1a.) Points 3, 4, 5 and 6 may release radioactivity effluents only during conditions of primary to secondary leakage.
- (3) There will be no residential housing on site. Free access along the beach in front of the plant is permitted. On site non-plant related activities, such as beach activities or picnics (excluding the Visitor Center activities) are evaluated for impact on the operation of the plant prior to granting permission for such activities.

D There are no military installations, missile sites, or industrial facilities located beyond the Donald C. Cook Nuclear Plant Site boundaries at which an accident might cause interaction with the plant so as to affect public health and safety.

The plant is located along the lake shore approximately midway between the northern and southern boundaries of the site. The distance from either of the reactor containment structures to the nearest site boundary is 2000 feet.

Figure 2.1-3 indicates the topographical details of the site and the location of the plant. Figure 2.1-4 is an aerial photograph of the site and its immediate environs before plant construction began.

2.1.3 TOPOGRAPHY

The site consists primarily of heavily wooded rugged sand dunes. A sandy beach slopes gently upwards for about 200 feet from the lake before rising sharply into the dunes. The peaks of the highest dunes reach an elevation of about 120 feet above the lake's surface; depressions between the dunes are as low as 10 feet above lake level.

Figure 2.1-4a shows modifications in the site topography due to plant structures. Figure 2.1-4b shows views of the site from the minimum exclusion radius to major plant structures showing the topography of the site in relation to major plant structures.

2.1.4 ACCESS

The site area is accessible by air, rail, and road.

The Pere Marquette Line, runs in an approximately north-south direction about 1600 feet east of the site's eastern boundary. A corridor between the site and the railroad has been purchased to permit construction of a rail spur, and a bridge spanning Thorton Drive and Interstate Route 94 has been erected to provide direct rail access to the plant.

Interstate Route 94 runs through the eastern portion of the site in a north-south direction, while the Red Arrow Highway runs along the eastern boundary in the same general direction. Thorton Drive, a local roadway, runs parallel to Interstate 94 and slightly to the west of it, while Livingston Road, also a local thoroughfare, forms the southern site boundary.

Within the 15-mile vicinity of the Donald C. Cook Nuclear Plant there are two airports: Southwest Michigan Regional Airport located approximately 12 miles NE of the plant on the NE edge of Benton Harbor and Andrews University Airport located approximately 10 miles East of the plant near Berrien Springs. For airports beyond this 15-mile radius, the orientation of runways and normal flight patterns are not in the direction of the plant or the normal glide path heights are not within the plant vicinity so that aircraft utilizing the facilities of these airports would not normally fly over the plant site.

Southwest Michigan Regional Airport has three runways all 100 feet wide, paved and lighted;

<u>Direction</u>	<u>Length</u>
East-West	5100 feet
North-South	3200 feet
NW-SE	3750 feet

For 1971 there were 67,690 operations (take-off or landings) resulting in 33,845 flights or an average of 93 flights per day of which only 9 were scheduled by commercial airplanes.

Weight load of aircraft using this field is limited to 30,000 pounds per single wheel load which is the design specification for the concrete runways. Three classifications of airplanes utilize the airport facilities: corporate, private, and commercial.

Due to the North-Easterly location of the airport and the orientation of the runways, normal glide paths would not approach the vicinity of the plant. There are no specified glide path heights since erection of structures taller than 500 feet are not permitted within a 10 mile radius of the airport. Neither is there a glide slope. However, the East-West runway, which handles most of the traffic because of the prevailing winds, is the only runway having the localizer portion of the Instrument Landing System. This indicates only the aim of the airplane.

Andrews University Airport has two runways:

<u>Direction</u>	<u>Length</u>	<u>Characteristics</u>
310° & 130°	3100 feet	Paved & Lighted
210° & 30°	2500 feet	Sod & Unlighted

There are no records maintained concerning the number of flights. The airport manager has estimated that there are approximately 70 flights some days and none during inclement weather conditions for a yearly total of 4,000 to 6,000.

The maximum weight load allowed is 12,500 pounds. There are no commercial flights; only corporate and private aircraft operate from this field.

There is no height, length, or orientation specified for a normal glide path.

2.1.5 POPULATION

The population data quoted in this section are a mixture of original analysis data, data obtained during an evacuation time estimate study performed during 1991-1992, and the demographic analysis performed during 1993. The evacuation time estimate study also provided updated information regarding schools and businesses near the Cook Nuclear Plant site. The demographic analysis projected future population in the Cook Nuclear Plant for the years 2000 and 2037.

Some of the population projections for individual sectors near the plant are greater than were anticipated in the original analysis. However, the referenced evacuation time estimate study shows that the population living near the plant could leave the area in a reasonable amount of time in the unlikely event of an ordered evacuation. Therefore, the combination of these time estimates and the fact that the total 10 mile emergency planning zone (EPZ) population has not exceeded projections indicate that there is no adverse impact on the EPZ population.

The area within 60 miles of the site which encompasses portions of Southwestern Michigan, Northwestern Indiana, and Northeastern Illinois is a region of moderate population that contained approximately 4,073,369 people in 1990. The population of this area from 1975 to 1990 has declined 12%. This decline in total population is attributed to the steady decline of the Chicago area. It is projected from the year 1990 to 2000, the population will increase approximately 3.3%. From 2000 to 2037, it is expected the population will increase by another 3.9%. The projected population distribution information for the years 1990, 2000, and 2037 is located in Tables 2.1-6 and 2.1-6a, 2.1-7 and 2.1-7a, and 2.1-8 and 2.1-8a, respectively.

The closest population center is the twin cities of Benton Harbor-St. Joseph with a combined 1990 population of 22,032. The closest population center boundary is the southern edge of St. Joseph, about eight miles north-northeast of the plant. All population centers within 60 miles of the site are indicated in Table 2.1-3.

The closest continuously occupied residence to the plant lies about 2160 feet to the north.

Figures 2.1-6, -6a, and -6b shows the 1990 population distribution around the site up to a distance of 60 miles. The Low Population Zone is identified in Figure 2.1-6 as the zone included within the 2-mile radius. Figure 2.1-6 divides the region from 0 to 5 miles from the plant into concentric circles and sectors of $22\frac{1}{2}^{\circ}$, where as Figure 2.1-6a and -6b divides the region from 5 to 60 miles and 10 to 60 miles, respectively. Similar data for the years 2000 and 2037 are included in Figures 2.1-7, -7a, -7b and 2.1-8, -8a, -8b. Population data are presented in tabular form in Tables 2.1-1 through 2.1-8b.

Thirty-four public and parochial schools existed within a ten mile radius with 625 teachers and a student population of 11,621. (1992)

Data collection to provide forecasts for the 21 counties entirely or partially within the 60 mile radius of the Donald C. Cook Nuclear Plant site was performed. This data was processed with the U.S. Census TIGER digital maps to apportion population forecasts for the years 2000 and 2037 for the radial distances and sectors as presented in the tables and figures. This analysis included the assignment of population forecasts for cities and towns within Berrien County, Michigan by one mile increments for the 0 to 5 mile area, and a five mile increment for the 5 to 10 mile area for forecast years 2000 and 2037. Similar forecasts were developed for the 10 to 60 mile area by ten mile increments for the sixteen $22\frac{1}{2}^{\circ}$ compass sectors.

The "best available data" regarding population growth during this study was obtained from the following sources:

- For the State of Michigan, initial population forecast data was obtained through telephone conversations with the State Demographer, State of Michigan, Department of State Planning and Commerce. (It should be noted that the existing State forecasts are based on pre-1990 census data and are subject to change when the new projections are released.) Based on highly variable trends in population growth over the past few decades, it was suggested that it is difficult to determine what the long range growth to the year 2037 will be. Thus, for the purposes of this analysis, population forecasts for the years 2000 and 2037 were derived using adjusted growth factors based on 1990 census data. More detailed local data for cities and towns in Berrien County was obtained through numerous informal communications with the Southwestern Michigan Commission.
- For the State of Indiana, county population forecasts were obtained through telephone conversations and from subsequent data provided by the State of Indiana reporting the results of the Business Research Center estimates for population growth through 2030.
- For Cook County Illinois, population forecasts were obtained through telephone conversations with the Northeast Illinois Planning Commission, which cited pre-1990 forecasts from the Illinois Bureau of the Budget, Illinois Population Trends - 1990.

In addition to the permanent resident population, Berrien County experiences an influx of approximately 3000 to 4000 summer residents each year. The great majority of the summer homes and cottages are located along the Lake Michigan Beach and in the Paw Paw Lake region in the north-eastern portion of the county.

The closest summer colony to the plant is the Rosemary Beach Association just north of the site boundary. Rosemary Beach is virtually uninhabited during the Fall, Winter and Spring and has a population of up to 150 during the peak of the summer season.

During the late summer and fall fruit harvest, substantial numbers of migrant farm workers are employed in Berrien County. The maximum number recorded in 1976 was 6,800.

Table 2.1-8b and Figure 2.1-10 represent the seasonal transient population out to the Population Center Distance of 8 miles with the 0-2 mile population figures representing the Low Population Zone.

The Work Employment Security Commission supplied data for total migrant workers in Berrien County in 1971 working as transient crop pickers. This total, consisting of 8355 workers, was uniformly averaged over the entire county rural area resulting in an average total of 1263 migrants within the Population Center Distance distributed evenly over the rural area. Some migrant workers arrive in the Spring to cut asparagus but most of them begin to arrive in the latter part of August, building up to a peak in the Fall during the peach and apple crop-picking periods. After the crops have been harvested, they leave the area.

The number of summer homes were supplied by the Twin Cities Chamber of Commerce (St. Joseph - Benton Harbor) and the Berrien County Clerks Office, and the number of people occupying the various beach areas were estimated from visual observations in 1971. Most of the summer vacationists begin to arrive in June when school ends and leave in late August when school recommences; although, a few remain into the Fall as long as favorable weather conditions exist. These vacationists are located mostly along the lake shore front.

Although, there is an overlapping of the seasonal transient population towards the end of summer; in general, there are two reasons: the summer months consisting of vacationists and the fall months consisting of migrant crop pickers.

The trend is towards a decreasing number of transients within Berrien County and hence within the Population Center Distance.

The migrant workers in the county decreased from a total of 11,100 in 1966 to 8,355 in 1971. This decline is attributed mainly to automation in the crop-picking industry and to a reduced apple market since the cost of picking will not support the apple market price.

Warren Dunes State Park lies along the lake about six miles south of the site. On a peak summer day in 1992, an attendance of 20,881 was recorded at the park of which 1,600 were overnight campers. In 1969, the park was enlarged somewhat to accommodate more daily visitors with increased camping facilities.

While the Warren Dunes State Park has changed from a 1976 summer peak of 23,958 days visitors of which 1300 were overnight campers to a 1992 day peak of 20,881 visitors of which 1600 were overnight campers, there has been a decline in the number of people occupying summer homes over the years with a decrease from 4,000 in 1964 to 3,000 in 1971 due to the high cost of home maintenance. Hence, the potential for a significant increase in transient population over the 40-year life of the plant does not seem probable especially within the Low Population Zone which comprises about 3 miles of lake shore front already containing four beach areas.

The area surrounding the site is devoted primarily to agricultural pursuits. Over 60% of the land in the three counties, Berrien, Cass, and Van Buren, surrounding the site is devoted to farming. The major crops produced are apples, cherries, grapes, peaches, feed grains, livestock and dairy products. Agricultural statistics are summarized in Table 2.1-9.

Figure 2.1-9 illustrates the number of farms with dairy cattle, the number of dairy cattle per farm, and their distance from the plant within a 10 mile radius (as of 1972).

In 1990, the low population zone contained approximately 764 permanent residents with no more than 174 in any 22½° sector.

Industrial activities in the area are centered around Benton Harbor and Niles, Michigan.

The primary industries are home appliances, metal casting and electronic and audio equipment.

Updated information on hospitals is given in Table 2.1-12.

Lake Michigan water in the vicinity of the plant site is not used for irrigation. Lake Michigan is however used for swimming, fishing, boating, domestic water supply and sewage.

Only crab fishing in water over 30 fathoms is permitted commercially in Michigan waters.

TABLE 2.1-12

HOSPITALS IN BERRIEN COUNTY, 1997

<u>Hospital</u>	Distance from		<u>Capacity</u>
	<u>Location</u>	<u>Station, miles</u>	
Lakeland Medical Center	Berrien Center	14	250
Lakeland Medical Center	St. Joseph	10	300
Lakeland Medical Center	Niles	18	174
Community	Watervliet	22	70



2.2.2 GENERAL METEOROLOGY

Southwestern Michigan is typical of the northern lake regions of the United States in most respects. The flat terrain and the frequent passage of well-developed extra-tropical storms create a consistently strong wind flow, as well as rapid changes in both dispersion conditions and wind direction. Some of the meteorological statistics are useful primarily for general planning of the facilities and are therefore reported with a minimum of description. Other data are important in the assessment of safety and these are discussed fully.

High Winds

Strong winds are the most important meteorological hazard to the facilities. The region is frequented by relatively strong, gusty winds, usually accompanying the passage of squall lines or thunderstorms and the maximum wind associated with these phenomena is 90 mph on a 100 year recurrency interval.

The tornado presents a very specialized type of hazard involving both violent winds and extremely large, rapid changes in barometric pressure.

The storms are small, unpredictable in detail and rather infrequent, but they undoubtedly represent one of the few environmental factors that could, if ignored in plant design, inflict direct major damage on the facility. Typically, the tornado is a narrow funnel, often only a few hundred yards wide, in which winds may briefly reach 300 mph. Almost instantaneous changes in barometric pressure occur, reaching 3 psi and causing explosion of vulnerable structures. Because of the severity of the phenomena, very few reliable measurements of tornado intensities exist. It is therefore difficult to dissociate wind and pressure effects, but the estimates given above are considered fairly reliable maximum values. This portion of Michigan has a significant tornado probability, as is apparent in the map shown in Figure 2.2-2. Berrien County has had 25 tornadoes between 1950 and 1989.

Ice Storms

Far less destructive, but far more probable, are the ice storms that frequent the north central states. Michigan lies in the belt where such storms are common and in the years from 1970 to 1989, 6 significant ice storms have been reported in this area.

2.2.3 DISPERSION METEOROLOGY

Worst Case X/Q values

X/Q values for 1992 (a typical year) were calculated from data from the main tower's 10 meter instruments using the MIDAS computer code. The data show a worst case X/Q value at the site boundary as $1.06E-05$ sec/m' during 1992. The worst case ever computed by the present meteorological system through 1994 is $1.13E-05$ sec/m'. Both of these values are well below the established worst case X/Q value of $3.15E-04$ sec/m'. Table 2.2-3 shows the X/Q values for all sectors at 10 distances.

Atmospheric Stability

The atmospheric stability for the area is now classified according to the Pasquill categories for use in dispersion calculations. These categories range from A to G, with the G category being the most stable. Joint frequency tables for 1992 (a typical year) have been compiled and are shown in Tables 2.2-4 through 2.2-11. The data show that a large percentage (33%) of the year is devoted to Category D. A rather substantial portion of the year (23%) shows an extremely unstable classification (Category A). There is only a small portion of time (7%) devoted to the extremely stable conditions of Category G.

Wind Speed and Direction

Wind speeds were moderate in 1992 (a typical year). The predominant wind speed range is 4-7 mph category. The wind speed exceeded 14 mph less than 4% of the time. The wind direction at the main tower varied,

Sediment Chemistry

Sediment chemistry studies were conducted to determine if the Cook Nuclear Plant vicinity in particular and Lake Michigan in general were contaminated from anthropogenic trace element sources and to establish the background levels of the radioactive and non-radioactive elements so these concentrations could be compared with levels measured after plant operation began.

Water Chemistry

Water chemistry studies were conducted to determine if the Cook Nuclear Plant vicinity in particular and Lake Michigan in general were contaminated from anthropogenic trace element sources and to establish the background concentrations of radioactive and non-radioactive elements so these concentrations could be compared with levels measured after plant operation began.

2.6.2.3 Initial Study Results

Physical Limnology

Figure 2.6-1 is a plot of the bottom of the lake adjacent to the site. It is characterized by gentle and regular topography. The 100-foot depth isopleth lies about six miles from shore. Isopleths are generally regular and parallel to the shoreline. Two sand bars lie close to shore along the entire length of the site property. The inner bar averages about 500 feet from the shoreline while the outer bar runs approximately 1000 feet from the shoreline. Maximum water depth of five to six feet is present between the inner bar and the shore. Twelve to thirteen feet of depth is the greatest measured between the bars. The depth over the crest of the inner bar is about four feet, while the outer bar peaks at eight to nine feet beneath the surface.

A number of studies of bottom stability along the east shore of Lake Michigan have been made in the past decade or two. Lake Michigan has what appears to be very stable conditions near shore despite severe storms and winter icing. Present evidence indicates that the nearshore sandbars fluctuate in position but maintain a fairly consistent average position, with fairly consistent water depths over their crests. Though bottom contours remain relatively stable, the littoral transport of sand has been estimated to be 100,000 cubic yards per year moving generally southward along the Michigan shore.

Although all of the currents of Lake Michigan are not thoroughly understood, certain of the larger features have been found with a surprising degree of constancy. There is a general outflow current along the Michigan shore from Little Sable Point northward toward the Straits of Mackinac, and there is a large eddy near the eastern shore near Benton Harbor, Michigan. Figure 2.6-2 indicates the results of several studies made of lake currents. In addition to the gross current features, there appears to be a thin, elongated, counterclockwise eddy close to the shore between Michigan City, Indiana and Benton Harbor (indicated by X on Figure 2.6-2). Some discussion on natural cyclic lake level fluctuation is warranted.

The speed and direction of local water currents in the site vicinity control the movement and dispersal of plant thermal plume. Studies (4) indicated that alongshore currents are established and controlled by interactions between local winds and the regional current pattern. Local winds are the dominant factors in establishing alongshore currents.

Lake levels tend to follow cycles. Over the past fifty years, periods of high lake levels and erosion were experienced from 1951-55, 1969-1975, and 1983-1987. The all-time high water level was recorded in 1986 at 581.10 ft. International Great

Lake, Datum (IGLD)*. The current period appears to have begun at the beginning of 1996 at a water level of 578.5 and ended the year up 1.5 ft. to 580 ft. IGLD. The current predication is that lake levels will continue to increase into 1997, and could meet or exceed the 1986 high. (2)

An Illinois State Geological Survey report (3) cites that where lake levels are rising above the 579 ft. IGLD level, well-developed beaches will delay the onset of maximum bluff erosion until they are depleted. After beaches have been depleted, bluff erosion from wave attack progresses fairly rapidly. Bluff erosion generally does not immediately decrease with decreasing lake levels, even when they fall below the 579 ft. level. Commonly, there is a lag effect by which recession rates are maintained or accelerated because slopes remain exposed until vegetation can become firmly established.

The Cook Nuclear Plant is protected by a sheet piling wall which runs the entire length of its lake frontage from the north to the south property lines. A second row of sheet piling runs parallel and 35 ft. west of the first line of piling and spans the length of the protected area.

Figure 2.6-3 is a plot of surface water temperatures in Lake Michigan during the relatively cool year of 1965 and the relatively warm year of 1966. Temperatures rise abruptly from a 32°F icing condition in winter to a peak in July and August and then decrease linearly to ice-water temperatures by late December.

A number of southwestern Michigan municipalities use Lake Michigan as their potable water source. These intakes and their

* U.S. Army Corps of Engineers' reports are expressed in IGLD. Cook Nuclear Plant elevations are expressed in National Geodetic Vertical Datum (NGVD). $NGVD = IGLD + 1.56 \text{ ft.}$

approximate distances from the plant discharge are as follows:

Northward

South Haven	32 miles
Benton Harbor	11 miles
St. Joseph	9 miles

Southward

Lake Township	0.4 miles
Bridgman	2.5 miles
Orchard Beach	7 miles
New Buffalo	16 miles
Grand Beach	18 miles
Michigan	19 miles
Unknown	22 miles
Michigan City, Indiana	25 miles

To the north, the outflow of the St. Joseph River interposes a physical and dynamic barrier to further progress of effluent northward along the shore. The plant effluent plume could reach the water intakes to the south at Lake Township, Bridgman and Orchard Beach. These intakes are also of the infiltration type. However, the prevailing winds of summer, when the worst dilution conditions (minimum wind and wave section) exist, are expected to carry the plume north from the plant and away from these water intakes.

Seiches are oscillations in the level of lakes and similar bodies of water caused by the passage of squall lines across the body of water. In Lake Michigan, these squalls have their fronts oriented NE to SW and are accompanied by an abrupt increase in barometric pressure and local high winds. Although seiches occur frequently in the Great Lakes, the great majority are only a few inches in amplitude. A large seiche occurred on June 26, 1954 and caused water level increases of up to 10 feet

at North Avenue in Chicago, Illinois. The greatest level increase recorded on the lake's eastern shore was 6 feet at Michigan City, Indiana.

The maximum recorded amplitude of an open lake seiche was 4.2 feet observed at the Wilson Avenue Crib in Chicago on July 6, 1954. A previous seiche on June 26, 1954, which resulted in a rise of 3.2 feet at Wilson Avenue Crib, caused the rise estimated at less than 6 feet in the Michigan City yacht basin, a point approximately 25 miles south of the plant site in an area where seiche effects are considered more severe than those farther to the north. Taking these values in proportion, one can postulate the maximum seiche producing a water level increase of as much as 8 feet in the Michigan City yacht basin.

To determine the plant elevation necessary to protect the plant from flooding due to seiches, the characteristics of the lake shore at the plant, historical meteorological conditions, and mathematical modeling were used to estimate a maximum seiche of 8 feet. However, the estimate was arbitrarily increased to 11 feet as an extra safety margin.

Wind generated waves are limited in their dimensions by wind velocity, duration and fetch. The greatest Lake Michigan fetch for the plant site is 265 miles to the north. The maximum deep waterwave is approximately 23 feet, and would require a sustained north wind of about 26 knots for over 19 hours. The runup of such a wave on the site shore, discounting the effects of the off-shore sandbars, has been calculated as 3.7 feet. This figure is overly conservative, however, since the large wave approaching the beach would be tripped by each of the sand bars.

The coincidental occurrence of maximum wave and maximum seiche was evaluated and determined not to be a possible event.

Elimination System Permit issued to the plant by the Michigan Water Resources Commission. The majority of the research was conducted by the University of Michigan with the private consulting firm ETA Engineering, Inc. and the Cook Nuclear Plant staff conducting the thermal plume mapping and bathymetric surveys.

Many studies begun during the initial phase were continued during this phase. These studies provide the plant operational data to compare with the pre-operational data gathered during the initial phase studies. Table 2.6-2 is a bibliography of the reports published during this phase of study at the Cook Nuclear Plant.

2.6.3.1 Study Groupings

Physical Limnology Studies

These studies include the shore ice formation and melt studies, the lake current and temperature study using in situ monitors, the study of the effects of the thermal plume on local meteorology, the thermal plume mapping studies and the bathymetric studies.

Biological Studies

These studies include continuations of the periphyton, phytoplankton, zooplankton and benthos studies of species composition and abundance. Fish studies of population size and species composition were initiated.

Sediment Chemistry Studies

Sediment chemistry studies included the non-radiological elemental composition of the sediments. Radiological elements

were also monitored for the increase in certain radioactive isotopes.

Water Chemistry Studies

The water chemistry studies included analyses for pH, hardness, conductivity, phosphorous, total nitrogen, sulfate, ammonia and trace metals.

2.6.3.2 Purpose of Technical Specification, Appendix B Studies

The Technical Specification Appendix B is part of the Cook Nuclear plant operating license that regulated the radiological and non-radiological environmental monitoring, aquatic ecological studies of the post-startup impacts to Lake Michigan, and regulated plant effluents, both radiological and non-radiological. Radiological issues are now addressed in the Offsite Dose Calculation Manual. The non-radiological issues remain in the Appendix B Technical Specifications, whose objectives are to:

- (1) Verify that the station is operated in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments.
- (2) Coordinate NRC requirements and maintain consistency with other Federal, State, and local requirements for environmental protection.
- (3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by way of the Plant's NPDES permit.

Physical Limnological Studies

The ice studies were continued to determine how the thermal discharge affected the shore ice and the floating ice in front of the plant. Winter storms could potentially cause severe beach erosion if the thermal plume melted the floating ice and the ice foot (the ice frozen to the bottom at the water/substrate interface) exposing the beach to winter storm generated waves.

A five-year meteorological study was conducted at Cook Nuclear Plant to determine if the operation of the once-through lake water cooling system would significantly effect the natural temperature, moisture, precipitation and fog conditions inland from the plant and, if so, how and to what extent these climatic conditions are affected. This investigation was undertaken because of the absence of quantitative information on the meteorological effects of near-shore warm water plumes. The

local economy is heavily dependent on agriculture, especially fruit crops. Changes to the local weather conditions could have a serious impact on the local economy.

Thermal plume mapping studies were conducted to determine the aerial extent of the 3F° and 1F° isotherms and the 3F° plume volume. This information was needed to determine if the plume would impact potable water intakes north and south of the plant, if the thermal plume would sink in the winter and impact the benthos and if the thermal discharge would comply with the 570-acre areal limit imposed by the state issued NPDES Permit. The lake current and temperature studies were used to help interpret the thermal plume mapping studies and evaluate the accuracy of the mathematical plume dispersion model. Knowing the size of the thermal plume also helped aquatic biologists determine how much aquatic habitat was impacted by the plume. Knowing the plume dimensions and location also helped the research team from the University of Michigan evaluate causes of changes in fish populations near the plant.

The bathymetric studies were conducted to determine if the lake water intake and discharge structures caused sediment erosion outside of the rip-rap aprons around these structures.

Biological Studies

The biological studies of the abundance and distribution of periphyton, phytoplankton, zooplankton and benthos were continued from the initial phase studies to provide the pre-operational and operational data comparisons. Fish studies were initiated in 1973 and were fully implemented in 1974. These studies were conducted to determine the impact of the Cook Nuclear Plant from construction and operation. Construction related impacts include the habitat alteration resulting from increased silt run-off from the construction site, placement of rip-rap around the intake and discharge structures and burying

of the intake and discharge tunnels in the lake bed. Operational impacts could result from oil and chemical spills, thermal discharges, the impingement of fish and benthos (crayfish) on traveling screens and the entrainment of planktonic organisms (phytoplankton, zooplankton, benthos and fish eggs and larvae) through the cooling water system. The combined impacts of the plant construction and operation were studied by analyzing the biological community structure for changes in species diversity and abundance. Primary production was estimated using the C-14 method; a measure of the effect of plant effluents on algae cell function. The health of algae cells entrained through the plant was assessed by measuring chlorophyll to phaeophytin ratios. Zooplankton could be impacted by the Cook Nuclear Plant thermal plume or by entrainment through the cooling system. Heat and mechanical damage caused by turbulent water flow through the system are the major effects.

Benthos were studied to determine the impacts of heat, habitat alteration, impingement on travelling water screens and plant entrainment caused by Cook Nuclear Plant operation. Changes in species composition and abundance was the measure used to determine effects.

Fish were studied to determine the effects of the thermal plume on adult and juvenile fish distributions, the impact of adult and juvenile fish impingement on travelling water screens, and the effects of fish egg and larvae entrainment through the power plant.

Sediment Chemistry Studies

The sediment chemistry studies were conducted to determine the changes in sediment chemical composition due to chemical discharges from the plant and from the possible build-up of organic material due to the settling and decomposition of

aquatic biota killed by the Cook Nuclear Plant thermal plume or plant entrainment.

Water Chemistry Studies

Water samples were collected and analyzed for phosphorus, dissolved silica, nitrate, nitrite, chloride, sulfate, oxygen saturation, alkalinity, pH, conductivity and these trace metals - Ba, Ca, Co, Cr, Cu, Fe, K, Mg, Mn, Mo, Na, Ni, Sr and Zn. In addition, a detailed study of the thermal bar was conducted to determine if it is a barrier to mixing of onshore with offshore water and, if so, how great did the chemical gradient become before the thermal bar moved far offshore.

2.6.3.3 Results of Technical Specification, Appendix B Studies

All results reported below are from Publication 22 from the University of Michigan unless noted otherwise.

Physical Limnology Studies

Ice studies were conducted over a ten year period from the winter of 1969-1970 through 1979-1980. A method of photographing the ice formation and analyzing the photographs was developed so the distance from the camera and the elevation of the object could be determined with reasonable accuracy. The conclusions of the ice study were:

1. The data show that the offshore ice ridges, offshore breakers and breaker zones, three characteristic features of the Lake Michigan shoreline in front of Cook Nuclear Plant, are coincident.

Ice ridges appear to be grounded features of the near shore ice complex and they serve a dual role. They

protect the beaches from incoming wave energy when present and, during the breakup of the complex, may modify the topography in the offshore bar vicinity.

2. The stages of ice development appear not to be controlled by any single meteorological variable but by a complex interrelationship between ice development and meteorological conditions. Air temperatures below freezing were found to be a necessary condition for initiation of the ice foot. Growth of the ice complex was associated with westerly winds and deterioration with easterly winds.
3. The plant's thermal plume produced a melthole that ranged from 0.1 to 0.5 square miles in size. The melthole was restricted to the vicinity of the discharge area. The ice ridges closest to the shoreline were minimally affected by the melthole and the effectiveness of the "ice ridge" complex as a wave energy dissipator to protect the beach was not significantly altered.
4. North and south of the melthole there was no noticeable change in the normal ice complex of ridges and lagoons and the nearshore ice complex was not discernibly altered due to the presence of the plant thermal plume.

Lake current studies were used to help analyze the thermal plume mapping data. Lake currents near the plant were the single most important physical parameter affecting the position, size and trajectory of the thermal plume. The lake current data were needed to determine the probability of the plume influencing other water intakes, recirculation to the Cook Nuclear Plant intakes and contacting the beaches north or south of the plant. The in situ current meters were moored about 1m from the bottom of the lake in four locations (see Figure 2.6-4). Surface drogues were used on the days thermal

month and year-to-year changes in phytoplankton assemblages" (2).

2.6.4 Ongoing Study Phase (1983 to Present)

Asiatic clams (Corbicula fluminea) and zebra mussels (Dreissena polymorpha) have been introduced to the Cook Nuclear Plant area as well as other locations in Lake Michigan. An Asiatic clam shell was found at the plant in 1983 and zebra mussels were discovered in the plant intake forebay in 1990.

Asiatic clams have caused serious clogging problems in water intake systems in the southern United States over the past 30 years or so. The Nuclear Regulatory Commission issued a bulletin requiring nuclear plants to monitor for Asiatic clam infestation in 1982. Asiatic clams are heat tolerant and cold intolerant. Water temperatures at the plant will prevent this species from becoming a serious biofouling organism at Cook Nuclear Plant.

Monitoring to ensure the Asiatic clam population remained low was begun in 1982 and has been conducted annually since then. Larval Asiatic clams (veligers) are monitored in filtered intake water samples and plant raw water systems are carefully inspected during routing maintenance. One live clam and about a dozen shell halves have been found in eight years of monitoring. No veligers have been collected.

Zebra mussels have been the cause of serious biofouling problems in Europe and Russia for many years (5). Water intakes for drinking water supplies and power plants have been clogged by zebra mussels in Lake Erie since they were first discovered in the St. Clair River in 1988. Zebra mussels are

cold adapted animals and are considered a potentially serious biofouling problem at the Cook Nuclear Plant.

No Asiatic clams have been found since April 1991, when half-shells were found during a Clam-trol flush of the fire protection system. This system has been placed on Lake Township water since the Spring of 1993. No Asiatic clams or zebra mussels have been reported in the Fire Protection System since it has been placed on the Lake Township water system. There is a consensus that Asiatic Clams do not pose a threat to Cook Nuclear Plant as they are a warm water species. They are no longer a part of the monitoring program.

Biocides supplemented by mechanical cleaning and design changes including strainers, filters, screens, and chemical delivery systems, work to protect plant systems. A zebra mussel monitoring program utilizing side-stream and artificial substrate monitors, along with diver and heat exchanger inspections, is used to evaluate the effectiveness of chemical and physical control measures.

References

1. Donald C. Cook Nuclear Plant, Supplement to Environmental Report, November 8, 1971
2. Monthly Bulletin of Lake Levels for the Great Lakes, March 1997, U.S. Army Corps of Engineers
3. Bluff Erosion, Recession Rates, and Volumetric Losses on the Lake Michigan Shore in Illinois, Richard C. Berg and Charles Collins, Illinois State Geological Survey, Environmental Geology Notes No. 76, July 1976
4. Baker, D. L. 1993. Report on the 1992 Zebra Mussel Control Program. Letter from D. L. Baker (I&M) to F. P. Morley (Michigan Department of Natural Resources).

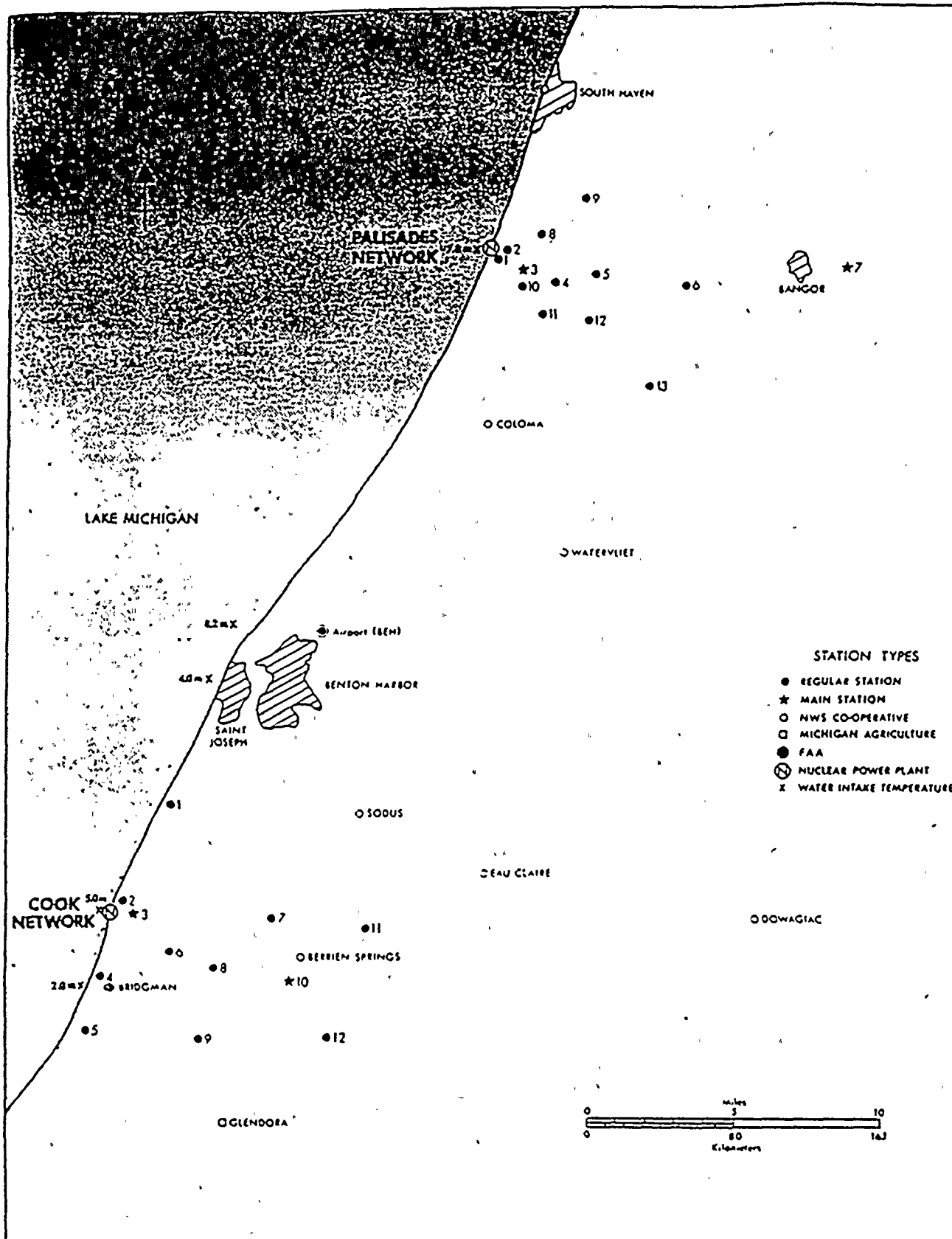


Figure 2.6-7 Donald C. Cook and Palisades Nuclear Plants meteorological networks. Network sites are given by numbers. Main sites are C03A, C10A, P03A, and P07A. Open circles are other locations with meteorological information.

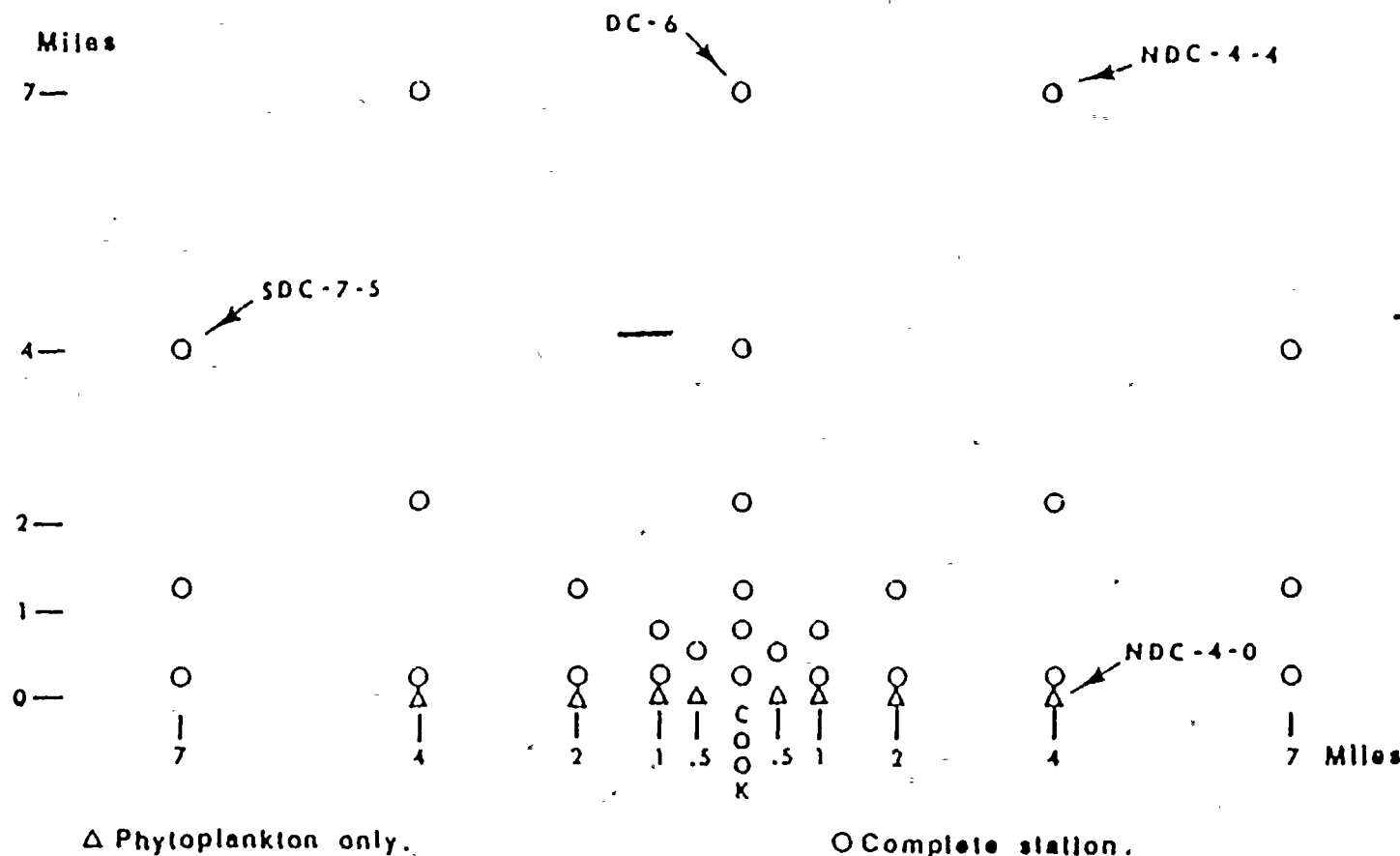


Figure 2.6-8

The 36-station Cook Plant sampling grid, used after April of 1972. The stations are designated as follows: SDC stations are located south of the plant, NDC stations are north of the plant, and DC stations are directly offshore of the plant. The first number of the designation is the number of miles north or south of the plant. The second number is the serial number of the station from shore lakeward. The serial number of the phytoplankton-only stations is 0.

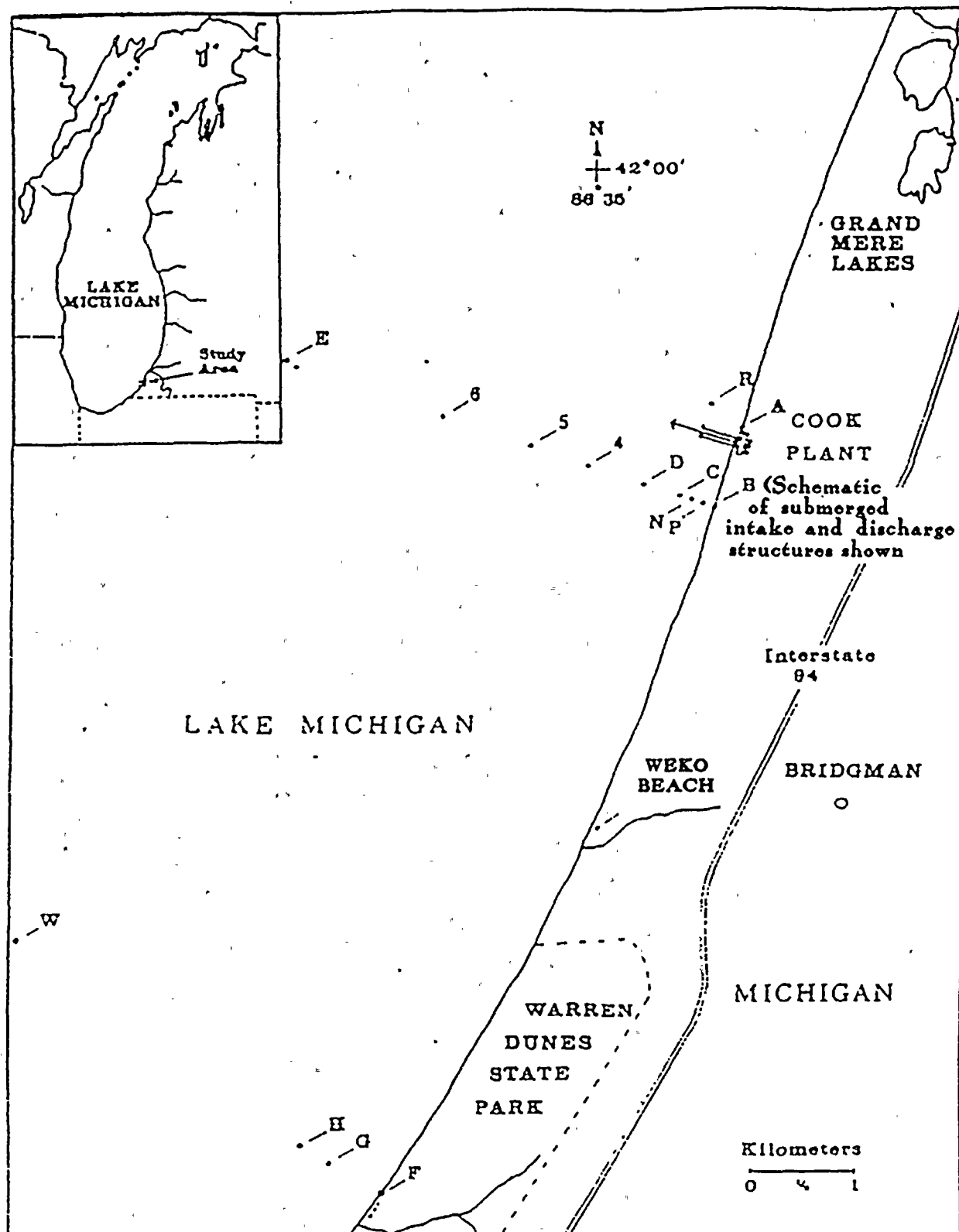


Figure 2.6-11. Map of southeastern Lake Michigan, showing locations of the Donald C. Cook Plant and our field fish larvae sampling stations:

JULY, 1997

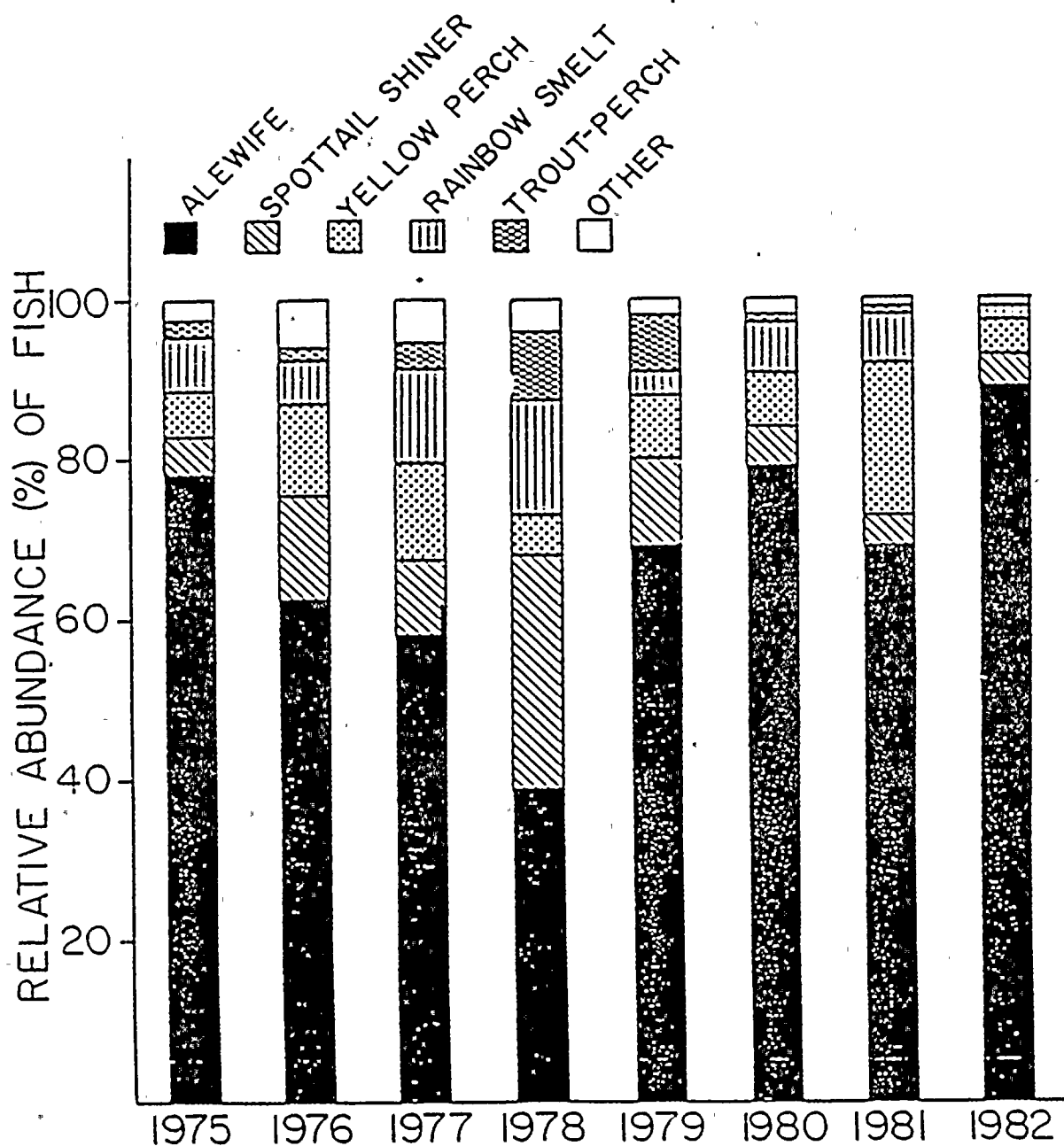


Figure 2.6-12. Species composition of the total number of fish impinged each year during 1975-1982 at the Donald C. Cook Plant, southeastern Lake Michigan.

2.7 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

2.7.1 PURPOSE OF THE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP)

The purpose of the REMP is to establish baseline radiation and radioactivity concentrations in the environs prior to reactor operations, to monitor critical environmental exposure pathways, and to determine the radiological impact, if any, caused by the operation of the Donald C. Cook Nuclear Plant upon the local environment.

The first purpose of the REMP was completed prior to the initial operation of either of the two nuclear units at the Cook Nuclear Plant Site. The second and third purposes of the REMP are an on-going operation and as such various environmental media and exposure pathways are examined. A complete and technical representation of the REMP is set forth in the Donald C. Cook Off-site Dose Calculation Manual (ODCM).

2.7.2 PREOPERATIONAL STUDY

The preoperational portion of the REMP was started 12 - 18 months before fuel was loaded into Unit 1. During this period, equipment was tested, sampling stations and sample media were determined, analytical procedures were tested, and some data was accumulated and examined for statistical variability. Modifications that were necessary to attain reliable and coherent data were made during this period.

2.7.3 SUMMARY OF PREOPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

There were several different types of environmental samples collected and analyzed during the preoperational sampling phase. Results from these samples are listed below.

The average monthly LiF thermoluminescent dosimeter (TLD) readings of August 1971 through December 1971, on-site, varied from $3.9 \pm$

1.3 mrem to 11.7 ± 0.8 mrem and off-site from 3.9 ± 1.2 mrem to 13.3 ± 1.1 mrem.

Initial water samples were taken in Lake Michigan and at water treatment facilities located in Bridgman, St. Joseph, Benton Harbor, and New Buffalo. These showed tritium concentrations of 562 ± 36 pCi/l to 583 ± 36 pCi/l. Gross beta at the above sampling stations showed 0.0 ± 2.0 pCi/l to 6.8 ± 1.0 pCi/l.

The determination of gross beta in the on-site airborne particulate samples was 0.01 ± 0.01 to 0.24 ± 0.1 pCi/m³. The same values for off-site stations are 0.01 ± 0.01 to 0.24 ± 0.1 pCi/m³.

3.0 REACTOR

3.1 SUMMARY DESCRIPTION

All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life. The fuel rods are cold worked, partially annealed Zircaloy tubes containing slightly enriched uranium dioxide fuel.

The fuel assembly is a canless type with the basic assembly consisting of the rod cluster control guide thimbles fastened to the grids, and to the top and bottom nozzles. The fuel rods are supported at several points along their length by the spring-clip grids.

Full length rod cluster control assemblies (RCCAs) are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the RCCAs are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes.

The control rod drive mechanisms for the full length RCCAs assemblies are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the RCCAs are released and fall by gravity to shut down the reactor.

The reactor was initially supplied with fuel from Westinghouse Electric Corp. (W). Reload fuel for Cycles 2 through 7 was supplied by Exxon Nuclear Co (ENC). Cycles 8 through 15 reload fuel was supplied by Westinghouse Electric Corp. The latest information regarding the current fuel cycle may be found in Sub-Chapter 3.5.

In addition to this summary description, this chapter contains: a description of the mechanical components of the reactor and reactor core, including Cycle 1 W fuel assemblies, reactor internals and control rod mechanisms (Sub-Chapter 3.2); a description of the Cycle 1 nuclear design for the W fuel (Sub-Chapter 3.3); a description of the Cycle 1 thermohydraulic design (Sub-Chapter 3.4); and a description of the current core design (Sub-Chapter 3.5).

The information contained in this chapter is principally concerned with the nuclear fuel and reactor internals design and therefore does not necessarily reflect the same information as that used in the safety analysis. For information concerning safety analysis, Chapter 14 should be consulted.

3.1.1 Performance Objectives

The current licensed thermal power limit is 3250 MWt. Calculations indicate that hot channel factors are considerably less than those used for design purposes in this application. The thermal and hydraulic design, and accident analyses (except large break LOCA) in Chapter 14, were performed at 3411 MWt for Cycle 8. These analyses identify design/safety limits for a potential uprating.

The turbine-generator and plant heat removal systems have been designed for a thermal rating of 3391 MWt. The portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power rating of 3391 MWt, as have the evaluations of activity release and radiation exposure.

The initial reactor core fuel loading was designed to yield the first cycle nominal burnup of 16,666 MWD/MTU, and the Cycle 2 through 7 reload designs yield an nominal cycle burnup of 10,000 MWD/MTU. Reload designs

3.3 NUCLEAR DESIGN

3.3.1 NUCLEAR DESIGN AND EVALUATION

This section presents the nuclear characteristics of the initial core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under operational modes, including both transient and steady state, is demonstrated. Power distribution limits have been updated in the current Technical Specifications which applies to cores with W OFA reloads. These current limits are incorporated in Section 3.5. Nuclear characteristics of the current cycles reload fuel are discussed in Section 3.5.

Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics for the initial core is presented in Table 3.3.1-1.

Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with changes in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions; (2) changes in reactivity associated with changes in the fission product poisons xenon and samarium; (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium); and (4) changes in reactivity due to burnable poison burnup.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during Cycle 1 refueling has been established as shown in Table 3.3.1-1, line 29. This concentration, together with the control rods, provides approximately 10 per cent shutdown margin for these operations. The concentration was also sufficient to maintain the core shutdown without any control rods during refueling. For cold shutdown, at the beginning of Cycle 1 core life, a concentration (shown in Table 3.3.1-1, line 37) was sufficient for one per cent shutdown with all but the highest worth rod inserted. The boron concentration (Table 3.3.1-1, line 29) for Cycle 1 refueling was equivalent to less than two per cent by weight boric acid (H_3BO_3) and was well within solubility limits at ambient temperature. This concentration was also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial Cycle 1 full power boron concentration without equilibrium xenon and samarium was 1152 ppm. As these fission product poisons were built up, the boron concentration was reduced to 838 ppm.

This initial boron concentration was that which permitted the withdrawal of the control banks to their operational limits. The xenon-free hot, zero power shutdown ($k = 0.99$) with all but the highest worth rod inserted, was maintained with a boron concentration of 734 ppm. This concentration was less than the full power operating value with equilibrium xenon.

This section describes the thermal and hydraulic design of the Unit 1 core with Westinghouse (W) fuel. The current cycles' thermal/hydraulic design is discussed in Section 3.5.3.

3.4.1 THERMAL AND HYDRAULIC EVALUATION FOR THE INITIAL CORE

Thermal and Hydraulic Characteristics of the Design

Thermal Data

Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains the maximum cladding surface temperature below about 657°F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap^{(1) (2)} and may be calculated by the following equation:

$$h = 0.6 P + \frac{k}{f(14.4 \times 10^{-6})}$$

where:

h is conductance in Btu/hr-ft²-°F

P is contact pressure in psi

k is the thermal conductivity of the gas mixture in the rod

f is the correction factor for the accommodation coefficient.

The thermal-hydraulic design assures that the temperature of the center of the hottest fuel pellet is below the melting point of the UO_2 . (Melting point of 5080°F ^[7] unirradiated and reducing by 58°F per 10,000 MWD/MTU.) The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. The pellet surface temperature is governed by the cladding temperature and the thermal conductance of the fuel pellet-cladding gap.

The thermal conductivity of uranium dioxide was evaluated from data reported by Howard, et al.^[21]; Lucks, et al.^[22]; Daniel, et al.^[23]; Feith^[24]; Vogt, et al.^[25]; Nishijima, et al.^[26]; Wheeler, et al.^[27]; Godfrey, et al.^[3]; Stora, et al.^[28]; Bush^[29]; Asamoto, et al.^[30]; Kruger^[31]; and Gyllander^[32]. An examination of the UO_2 thermal conductivity data, Figure 3.4.1-1 shows that at temperatures between 0°C and 1600°C there is little variation in the data, while above 1600°C the scatter increases considerably.

At the higher temperatures, thermal conductivity is best obtained 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 for the value of $\int_0^{2800^\circ\text{C}} k dT$ is 93 watts/cm. This conclusion is based on the integral values reported by Gyllander^[32]; Lyons, et al.^[33]; Coplin, et al.^[34]; Duncan^[5]; Bain^[35]; and Stora^[36].

The design curve for the thermal conductivity is shown in Figure 3.4.1-1. The section of the curve at temperatures between 0°C and 1300°C is in excellent agreement with the recommendation of the IAEA panel^[37]. The section of the curve above 1300°C is normalized to an integral value of 93 watts/cm^[5, 32-36] from 0 to 2800°C .

3.5 Current Westinghouse OFA Reload Fuel

Starting with Cycle 8 operation (startup November 1983), the Cook Nuclear Plant Unit 1 has been refueled with Westinghouse (W) fuel assemblies of the 15x15 optimized fuel assembly (OFA) design. This chapter evaluates the mechanical, nuclear, and thermal hydraulic design of the OFAs. Unit 1 Cycle 15 fuel will be principally used as an example of Westinghouse OFA Reload Fuel for current cycles. Information for other reloads will be used when a significant design change is introduced.

The design of the W OFA (15x15 fuel rod array) is similar to the W 15x15 LOPAR (low parasitic) fuel which was used in the Cook Nuclear Plant Unit 1, Cycle 1 core. W LOPAR fuel also has had substantial operating performance in a number of nuclear plants.⁽²⁾ The major difference introduced by the W 15x15 OFA design is the use of five intermediate Zircaloy grids, replacing five intermediate Inconel grids for the LOPAR fuel. The 15x15 Zircaloy grid design is similar to the W 17x17 OFA grid design. The W 17x17 OFA design has been generically approved by the NRC via their review of the W 17x17 OFA Reference Core Report.⁽³⁾ Prior to the insertion of 15x15 OFAs in Cook Nuclear Plant Unit 1, operating experience had been obtained in other plants for six demonstration 17x17 OFAs which contain Zircaloy intermediate grids.⁽²⁾

Eight of the eighty-four feed fuel assemblies for Cook Unit 1 Cycle 15 were fabricated with Intermediate Flow Mixer (IFM) and Low Pressure Drop (LPD) grids. These eight feed assemblies will be placed in peripheral locations in Cycle 15 as Lead Test Assemblies (LTAs).

IFM grids are considered non-structural upper assembly grids which contain mixing vanes similar to zircaloy structural grids. Specifically, the IFM grids are located between mid-grids 4 and 5, 5 and 6, and finally 6 and 7. Based on a proven design, the 15x15 IFM grids have less than one third the mass of the current 15x15 structural grids. These IFM grids will have virtually no effect on neutron economy.

As expected, adding the IFM grids alone to the fuel assembly design would increase the pressure drop across the assembly and have a significant impact on the safety analysis of record. In order to alleviate this impact, Westinghouse has developed, previously licensed, and utilized the Low Pressure Drop (LPD) structural grids in conjunction with these IFM grids. These LPD grids are reduced in height with chamfered upstream edges and have an innovative diagonally-oriented grid spring design. While these modifications do not detract from the structural capabilities of the grid, they significantly reduce the pressure drop of the grids and therefore allow adding IFM grids with minimal impact.

Additionally, all fresh fuel beginning with Cycle 15 are fabricated with the Performance + features which are described below.

Protective Grid

The protective grid is a partial height grid fabricated of Inconel without mixing vanes. It is positioned directly above the top plate of the bottom nozzle. Rods are positioned close to the bottom and are modified with a slightly longer bottom end plug. This grid provides added protection against debris induced fretting by trapping debris below the fuel stack. In addition, the protective grid provides improved resistance to grid-rod fretting by means of the additional support at the bottom of the rod.

Debris Mitigating Bottom End Plug

A 0.81" long, debris mitigating fuel rod bottom end plug (0.38" longer than the current design) is used in conjunction with the protective grid. The rods are positioned close to the bottom nozzle at the beginning of life (BOL), and the spring dimples of the protective grid contact the fuel rod only on the solid end plug. The active fuel stack length remains at 144".

3.5.1 Fuel Mechanical Design

Each W OFA consists of 204 fuel rods, 20 guide thimble tubes, and 1 instrumentation thimble tube arranged within a supporting structure. The instrumentation thimble is located in the center position and provides a channel for insertion of an incore neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a neutron source assembly, a WABA assembly, or a thimble plug assembly, depending on the position of the particular fuel assembly in the core. The fuel rod pitch is maintained by two Inconel end grids and five Zircaloy-4 intermediate grids. The Zircaloy-4 guide tubes are mechanically attached to the OFA top and bottom nozzles. The guide tubes, nozzles and grids form the structural skeleton of the fuel bundle. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzle. Figure 3.5.1-1 shows an OFA fuel length schematic view, and Table 3.5.1-1 shows OFA design values. Figure 3.5.1-1 shows the difference for certain dimensions between the OFA and the ENC fuel, which had been used in previous cycles.

Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate, which is fitted with alignment pins to locate and orient the assembly. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against the holddown springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

The 15x15 OFA design meets the same basic mechanical design requirements and criteria stated for the 17x17 OFA in WCAP-9500-A.⁽⁶⁾ Design values for the properties of materials which comprise the fuel rod, fuel assembly, and core components are given in Reference 7.

The mechanical design of the fuel assemblies starting with Region 15 is essentially the same as the Region 14 fuel assemblies with the exception of:

- (1) Debris Filter

Bottom Nozzle (DFBN), (2) increased radius bottom end plug, (3) anti-snap top, mid and bottom grids, (4) Integral Fuel Burnable Absorbers (IFBA), and (5) reconstitutible top nozzles (Region 16). These modifications described below meet all fuel assembly/rod design criteria as presented.

The bottom nozzle is designed to inhibit debris from entering the active fuel region of the core and thereby improves fuel performance by minimizing debris related fuel failures. The DFBN is a low profile bottom nozzle design made of stainless steel. The DFBN is structurally and hydraulically equivalent to the existing low profile bottom nozzle.

The bottom end plug has an increased radius in the transition between the chamfer and the end of the plug. There are no changes in the critical dimensions of the bottom end plug or the pressure drop from the previous region. Therefore the fuel rod performance and core safety considerations are not adversely affected.

The top, mid and bottom snap-resistant grids contain outer grid straps which are modified to help prevent assembly hangup due to grid strap interference during fuel assembly removal. This was accomplished by changing the grid strap corner geometry and adding guide tabs on the outer grid strap. The inner grid straps were also modified slightly. Core safety considerations are not adversely affected.

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating on the pellet cylindrical surface along the central portion of the fuel stack length. IFBAs provide power peaking and moderator temperature coefficient control. Details of the IFBA design are given in Section 2.0 Reference 15. IFBA description is summarized in Section 3.5.1.3.

The reconstitutible top nozzle functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the rod cluster control assembly or other components. It consists of an adapter plate, enclosure, top plate, and pads. The nozzle assembly

The OFA bottom nozzle design has a reconstitutable feature, as shown in Figure 3.5.1-7, which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard W LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

Evaluation

The OFA bottom nozzle has similar design features and dimensions compared to the LOPAR nozzle, thus assuring mechanical compatibility.

3.5.1.2.4 Grids

Description

Two types of grid assemblies are used in each fuel assembly. Both types consist of individual slotted straps interlocked in an "egg-crate" arrangement. The straps contain spring fingers, support dimples, and mixing vanes. One type, consisting of five (5) inner-grids per assembly, consists of Zircaloy straps arranged as described above and permanently joined by welding at their points of intersection. Their internal straps include mixing vanes which project into the coolant stream and promote mixing of the coolant. The other grid type, two (2) located at each end of a fuel assembly, does not include mixing vanes on the internal straps. The material of these grid assemblies is Inconel-718, chosen because of its corrosion resistance and high strength. Joining of the individual straps is achieved by brazing at the points of intersection. The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core. The individual grid cells at each fuel rod location provide six-point contact with the rod; four dimples and two springs.

The attachment of the five Zircaloy inner-grid and two Inconel end-grid assemblies to the guide thimble tubes is described in Section 3.5.1.2.1.

Evaluation

The fuel rods, as shown in Figure 3.5.1-1, are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods. Each fuel rod is supported within each grid by the combination of support dimples and springs. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The top and bottom Inconel grids of the OFA are the same as the Inconel grids of a W LOPAR fuel assembly. The five intermediate OFA Zircaloy-4 grids have thicker and wider straps than the OFA Inconel grids (See Table 3.5.1-1) in order to closely duplicate the Inconel grid strength. The ENC assembly grids are bimetallic, consisting of Zircaloy-4 straps with Inconel grid springs. Both the OFA Zircaloy and ENC bimetallic grids have grid heights of 2.25 inches. The OFA Inconel grid height is 1.5 inches.

Elevation of the grids was established to ensure axial match-up during operation.

Impact tests that have been performed at 600°F to obtain the dynamic strength data verify that the Zircaloy grid strength at reactor operating conditions is acceptable. The 15x15 Zircaloy grids have approximately 7% less crush strength than the 15x15 Inconel grids at reactor operating temperatures.

The ability of the grids to withstand seismic and LOCA impact loads is shown in Section 3.5.1.1.

3.5.1.3 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow greater axial expansion at the center of the pellets.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel density changes during irradiation, thus avoiding overstressing of the cladding or seal welds. Shifting of the fuel within the cladding during handling or shipping prior to core loading is prevented by a stainless steel helical spring which bears on top of the fuel. At assembly, the pellets are stacked in the cladding to the required fuel height, the spring is then inserted into the top end of the fuel tube, and the end plugs are pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process in order to minimize compressive cladding stresses and prevent cladding flattening due to coolant operating pressures. Nominal fuel rod parameters are given in Table 3.5.1-1.

Integral Fuel Burnable Absorber (IFBA)

The IFBA coating on the fuel pellets provides partial control of the excess reactivity available during the beginning of the fuel cycle. In doing so, the burnable absorber prevents the moderator temperature coefficient from violating safety limits at normal operating conditions. The burnable absorber performs this function by reducing the requirement for soluble poison in the moderator at the beginning of the fuel cycle as described previously. For purposes of illustration, a typical IFBA pattern in the core together with the number of IFBA pins per assembly are shown in Figure 3.5.2-1. The IFBA coating on the fuel is part-length (117", offset +1.5"

relative to the fuel rod axial centerline). For Region 18 fuel the coating is 120" without offset. The burnable absorber is part-length to allow a better (flatter) axial power distribution in the core. The ZrB₃ coating on the fuel pellets is depleted with burnup, but at a sufficiently slow rate so that the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains within safety limits at all times for power operating conditions.

Design Bases

The fuel design bases and criteria for W 15x15 OFA fuel are the same as those discussed in Sections 4.2 and 4.4.1.2 of WCAP-9500⁽⁶⁾ for the W 17x17 OFA design. The bases and criteria given in Section 3.2.1.1.1 of the UFSAR for Cook Nuclear Plant Unit 2 are also applicable, but it should be noted that the region average discharge burnups considered in the Cook Nuclear Plant Unit 1 OFA fuel design are typically in the range of 38,000 MWD/MTU. These design bases and criteria are summarized below:

- a. The cladding stresses under Condition I and II events are less than the Zircaloy 0.2% offset yield stress, with due consideration of temperature and irradiation effects. While the cladding has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design basis.
- b. Cladding Tensile Strain - The total tensile creep strain is less than 1% from the unirradiated condition. The elastic tensile strain during a transient is less than 1% from the pre-transient value. This limit is consistent with proven practice.
- c. Strain Fatigue - The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice.
- d. Wear - Potential for fretting wear of the clad surface exists due to flow induced vibrations. This condition is taken into account in the design of the fuel rod support system. The clad wear depth is limited to acceptable values by the grid support dimple and spring design.

3.5.1.4 Core Components

The core components consist of the rod cluster control assemblies (RCCAs), the primary and secondary source assemblies and the thimble plug assemblies. The design of the control rod assemblies in the Cook Nuclear Plant Unit 1 core is essentially unchanged from previous cycles and is compatible with the OFA guide thimbles. Enhanced performance rod cluster control assemblies (EP-RCCAs), which use silver-indium-cadmium (Ag-In-Cd), are utilized with the standard RCCAs starting from Cycle 12. These have a thin chrome electroplate applied over the length of absorber rodlet cladding in contact with the reactor internal guides to provide increased resistance to cladding wear. In addition, the absorber diameter is reduced slightly at the lower extremity of the rodlets in order to accommodate absorber swelling and minimize cladding interaction. The absorber rod cladding material is a very high purity 10% cold worked type 304 stainless steel tubing.

New secondary source assemblies and OFA compatible plugging devices were supplied in Cycle 8, Cycle 12, and Cycle 16. As discussed in Section 3.5.1.2.1, the reduced diametral clearance compared to LOPAR guide thimble results in an increased RCCA scram time from 1.8 to 2.4 seconds which was used in all accident reanalyses.

The guide thimble plug used with the OFA has a smaller diameter (0.485") than the LOPAR thimble plug diameter, in order to maintain the same thimble plug to thimble tube diametral clearance.

The optimized assemblies, their thimble plugging devices, and source assemblies are compatible with existing handling tools.

Wet Annular Burnable Absorber (WABA)

The Wet Annular Burnable Absorber (WABA) rod design is sometimes used in the Cook Nuclear Plant Unit 1 reload cores with 15x15 W OFA fuel. The materials, mechanical thermal hydraulic, and nuclear design evaluations of the WABA rods are presented in a topical report,⁽⁵⁾ which has received NRC generic approval,⁽⁵⁾ and approval for Cook Nuclear Plant Unit 1 application⁽¹⁾ of WABAs.

The WABA design has annular aluminum oxide - boron carbide ($Al_2O_3 - B_4C$) absorber pellets contained within two concentric Zircaloy tubes with water flowing through the center tube as well as around the outer tube. The WABA design provides significantly enhanced nuclear characteristics, when compared with the W borosilicate absorber rod design. Fuel cycle benefits result from the reduced parasitic neutron absorption of Zircaloy compared to stainless steel tubes. Increased water fraction in the burnable absorber cell, and a reduced boron penalty at the end of each cycle.

Figures 3.5.1-8 and 3.5.1-9 show the design of a WABA rod, and Table 3.5.1-2 and Figure 3.5.1-9 present a comparison between the WABA rod and a W borosilicate glass absorber rod.

The WABA rods inserted into each OFA are attached at their top ends to a holddown assembly and retaining plate in the same manner as burnable absorber rods previously used in Cook Nuclear Plant Unit 1 reload cores.

Based on the materials and design evaluations in Reference 5, it is concluded that the wet annular burnable absorber rod satisfies all performance and design requirements for 18,000 effective-full-power-hours irradiated life.

3.5.1.5 PERFORMANCE + Debris Mitigation Features

The following PERFORMANCE+ fuel features are used beginning with the fresh fuel delivered for Cycle 15.

1. Protective fuel rod oxide coating
2. Debris mitigating bottom end plug
3. Protective bottom grid
4. External gripper top end plug
5. Variable pitch plenum spring

Pre-Oxidized Fuel Rod Cladding

The ZIRLO™ or Zircaloy-4 (Zr4) fuel rod cladding is pre-oxidized on the bottom 6" to 7" (beginning with the bottom end plug) for debris fretting resistance and fuel rod reliability. This fuel feature consists of a 3 to 5 micron coating applied to the fuel rod at a location below the bottom inconel structural grid and as such provides for an increase in the resistance to debris damage in this region. The Zirconium Oxide Coating is thermally grown on the cladding as part of the fuel rod manufacturing process and, according to test results, is expected to increase the wear resistance by a factor of 10 over that of the uncoated cladding.

Debris Mitigating Bottom End Plug

A 0.81" long, debris-mitigating fuel rod bottom end plug (0.38" longer than the current design) is used with the protective grid, and the rods have been positioned close (0.095" gap) to the bottom nozzle at the beginning of life (BOL). The active fuel stack length remains at 144".

Protective Bottom Grid (PBG)

A protective grid has been added at the bottom of the assembly to provide an additional debris barrier thereby improving fuel reliability. The protective grid will also provide additional fretting resistance by supporting the bottom of the fuel rod. This feature is designed to enhance the debris mitigation performance of the fuel and consists of an additional "thin" inconel grid positioned directly above the bottom nozzle.

Rods are positioned close to the bottom and are modified, as discussed below, with a slightly longer bottom end plug. This grid provides added protection against debris induced fretting by trapping debris below this grid where it can wear against the solid end plug. In addition, the grid provides improved resistance to grid-rod fretting by means of the additional support at the bottom of the rod.

External Gripper Top End Plug

The fuel rod top end plug was elongated (0.12" longer than the current design) to 0.450" and fitted with an external gripper. The external grip feature assists in positioning the fuel rod during manufacture, and also in rod removal during reconstitution.

Variable Pitch Plenum Spring

Because of the longer fuel rod end plugs and resulting decrease in internal rod void volume, a PERFORMANCE+ variable pitch (VP) plenum spring was designed for fuel rods utilizing the protective grid package. The VP spring has longer coil lengths in the middle section of the spring. The VP spring continues to meet all design and manufacturing criteria while regaining some of the plenum volume lost to the longer end plugs.

3.5.1.5.1 Evaluations

Pre-Oxidized Fuel Rod Cladding

This feature results in a oxide layer "coating" before the fuel rod is loaded into the core. Currently, the length of this coating is expected to be no longer than 6 or 7 inches, with the thickness tapering from its maximum thickness to zero beginning at approximately 5.5" from the bottom of the rod. A review of the available data reveals that bottom of the active fuel stack begins about 6.8" from the bottom of the rod (as a result of the longer bottom end plug and, in this case, axial blankets). As such, this debris coating does not affect the LOCA related transients as only the active fuel length is modeled for heat transfer characteristics in the LOCA analyses.

3.5.1.6.5.2 Small Break LOCA

With respect to the eight assemblies with LPD and IFM grids, the 0.3 psi increase in core pressure drop will have a negligible effect on results. In addition, transition core effects need not be considered for SBLOCA, as the low flows typical of a Small Break transient will not be substantially affected by a small mismatch in hydraulic resistance. As a result, the introduction of IFM/LPD assemblies will have no effect on SBLOCA.

3.5.1.6.5.3 LOCA Related Issues

3.5.1.6.5.3.1 LOCA Forces

The most recent evaluation of LOCA forces for the Donald C. Cook Units was performed in 1993 as part of the 30% Steam Generator Tube Plugging effort. This evaluation utilized, as a basis, the analysis performed in 1988 to support the T-hot Reduction/Up-rated Power effort. However, the 15x15 OFA fuel analyzed in these previous efforts did not contain IFMs. In general, the presence of IFMs would lead to higher vertical LOCA forces on the fuel associated with the area and hydraulic loss coefficients of IFMs. In this particular case, the presence of only 8 assemblies with IFMs will not result in a significant change in the calculated LOCA forces because 8 assemblies are such a small fraction of the total core composition and the change in force due to IFMs is not large in relation to the total vertical force on the fuel. Therefore, the current LOCA forces will remain bounding. There is sufficient margin in the Steamline Break Transient to cover the transition core penalty.

3.5.1.6.5.3.2 Post - LOCA Long Term Core Criticality

In order to ensure the peak temperatures from the large break LOCA are not exceeded, it is necessary to demonstrate that the core remains subcritical in the long term. This is accomplished by calculating the mixed mean boron concentration of the sump following a postulated large break LOCA and comparing it to the boron required to maintain subcriticality for the cycle of operation. This analysis is not negatively impacted by IFM/LPD grids in 8 fresh peripheral assemblies.

3.5.1.6.5.3.3 Hot Leg Switchover

As part of the requirement to maintain core coolable geometry in the long term, NUREG-0800 required that steps be taken to preclude the precipitation of boron in the vessel. This results in the requirement to switch from cold leg recirculation mode to hot leg injection mode (or simultaneous hot and cold leg injection mode). The hot leg switchover time is calculated as the minimum time at which boron solution concentration in the vessel due to boiloff comes within four percent of the solubility limit. This calculation is not negatively impacted by IFM/LPD grids in 8 fresh peripheral assemblies.

3.5.1.6.6 Transient (non-LOCA) Evaluation

The installation of 8 fresh assemblies with IFM/LPD grids into the Cook Unit 15x15 core can potentially impact the following areas of the non-LOCA safety analyses:

- the pressure drop across the core,
- the rod drop time/rod insertion vs. time curve used to generate the trip reactivity insertion curve, and
- DNB effects, e.g., core thermal limit adjustments and transition core penalties.

Section 3.5.1.6.1 provides the pressure drop information for the core configuration applicable to Cook Unit 1 Cycle 15. It can be seen that the installation of the assemblies with IFM/LPD grids results in an increase in the core pressure drop of less than 1.15%. An increase in the core pressure drop of this magnitude is very insignificant. The non-LOCA transients, with the exception of the loss of flow events, are insensitive to the RCS pressure drops.

An increase in the core pressure drop results in a slightly faster flow coastdown during a loss of RCS flow or locked reactor coolant pump rotor event (UFSAR Section 14.1.6).

However, the installation of IFM/LPD grids in 8 of the 193 fuel assemblies will have no impact since an increase in the core pressure drop of 0.3 psi will have a negligible affect on the total RCS pressure drop. (As a reference point, it should be noted that the total RCS pressure drop under 10% steam generator tube plugging conditions is on the order of 85 psi. Any increase in the steam generator tube plugging level will cause the total RCS pressure drop to increase, thereby making the 0.3 psi core pressure drop increase to be even more negligible by contrast.) Therefore, the RCS flow coastdown will not be adversely impacted. Thus, the slight increase in the core pressure drop will not invalidate the conclusions presented in the Donald C. Cook Unit 1 UFSAR for the Loss of Reactor Coolant Flow transients, nor the Locked Rotor Accident.

Since the assemblies with IFM/LPD grids will not be in RCCA locations, the rod drop time is assumed to not be affected. As such, the currently assumed rod drop time of 2.4 seconds and the normalized RCCA position versus time curve assumed in the safety analyses remains valid. Thus, the trip reactivity assumptions that have been made in the non-LOCA safety analyses are not impacted by the installation of the 8 fresh assemblies with IFM/LPD grids.

There is no impact on the calculated statepoints and/or minimum DNBR calculations performed by Transient Analyses since the core thermal limits and axial offset limits continue to remain valid and the change in the core pressure drop is insignificant. Furthermore, the applicability of the core thermal limits and axial offset limits are verified for each particular cycle during the Reload Process. Assembly-specific effects with respect to minimum DNBR (i.e., transition core penalties) due to the 8 installed fresh assemblies with IFM/LPD grids are addressed in Section 3.5.1.6.5.

Since the core pressure drop change is insignificant, such that non-LOCA transient behavior has been determined to not be impacted, and the assembly-specific DNB affects have been addressed, it can be concluded that there is no safety impact on the non-LOCA portion of the Donald C. Cook Nuclear Plant Unit 1 licensing basis.

3.5.1.6.7 Other Safety Related Areas

The following safety related areas were determined not to be impacted by the insertion of 8 fuel assemblies with IFM/LPD grids.

Mechanical and fluid systems
Instrumentation and control systems
Environmental qualification of components
Containment analysis
Radiological consequences of accidents
Steam generator tube rupture
Emergency Operating Procedures
Technical Specifications
Protection system setpoints

REFERENCES - SECTION 3.5 and 3.5.1

1. Letter from D. L. Wigginton (NRC) to John Dolan (Indiana and Michigan Electric Co.) dated September 20, 1983; Subject: Amendment 74 to Facility Operating License No. DPR-53 for Donald C. Cook Unit 1 and enclosed NRC SER, Docket No. 50-315.
2. Skaritka, J., Iorii, J. A., "Operational Experience with Westinghouse Cores (up to December 31, 1982)," WCAP-8183, Rev. 12, August 1983.
3. Letter from R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), Safety Evaluation of WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly," NRC SER letter dated May 22, 1981.
4. Bordelon, F. M., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273 (Non-Prop.), March 1978.
5. Skaritka, J., et al., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A, Revision 1 (Proprietary), October 1983. Contains NRC SER dated August 9, 1983 and W responses to NRC questions.

dimensional model which is utilized to simulate operation of the core for previous cycles.

As an example, Cycle 15 core calculations used assembly exposures calculated from the Cycle 14 burnup of 14,267 MWD/MTU.

3.5.2.2.2 Design Bases

For each cycle, the nuclear design bases are very similar to those for the example Cycle 15 core as follows:

1. At core full power, 3250 MWt (not including pump heat), nuclear peaking factors of 2.15 and 1.55 for F_Q^T and $F_{\Delta H}^N$ respectively, will not be exceeded. In addition, at any relative power level P ($0.0 \leq P \leq 1.0$), F_Q^T and $F_{\Delta H}^N$ shall not exceed the bases of the plant control and protection system.
2. The moderator temperature coefficient at operating conditions greater than 70% power level is a ramp function limited to +5.0 pcm/ $^{\circ}$ F at 70% power and 0.0 pcm/ $^{\circ}$ F at 100% power. Below 70% power level, the moderator temperature coefficient shall be less than +5.0 pcm/ $^{\circ}$ F.
3. With the most reactive control rod stuck out of the core, the remaining control rods shall be able to shut the reactor down by a sufficient reactivity to reduce the consequences of any credible accident to acceptable levels.
4. The effects of all accident situations in Cycle 15 will be acceptable and compatible with the safety bases of the Final Safety Analysis Report (FSAR), as specified in Reference 7.
5. The fuel loading specified shall be capable of generating approximately 15520 MWD/MTU at normal full power operating conditions during Cycle 15.

3.5.2.2.3 Design Description and Results

Each cycle's reactor core consists of 193 W OFA assemblies, each having a 15x15 fuel rod array. A description of the W OFAs is given in Section 3.5.1.

As an example, the Cycle 15 loading pattern is given in Figure 3.5.2-1 which shows the region number, sources, and the burnable absorber configuration. The core consists of 32 fresh W OFAs with an average enrichment of 3.117 w/o U-235, 52 fresh OFAs with an average enrichment of 3.612 w/o, 80 once burnt OFA assemblies, and 29 twice burnt OFA assemblies. A low leakage loading pattern was developed which results in the scatter-loading of the fresh OFAs throughout the interior of the core. 4304 new IFBA rods and 120 wet annular burnable absorbers (WABA) are present in a number of OFAs to control power peaking and MTC. The IFBA rods contain approximately 0.0018 gm/in of B-10. Pertinent fuel assembly parameters for the Cycle 15 fuel are given in Tables 3.5.1-1 and 3.5.2-1.

Physics Characteristics

The neutronics characteristics of a reactor core with W OFA fuel are presented in Table 3.5.2-2. These reactivity coefficients are bounded by the coefficients used in the safety analysis. For an example cycle length, Cycle 15 was projected to be 15,520 MWD/MTU at a core power of 3,250 MWt with ~ 9 ppm soluble boron remaining.

Power Distribution Considerations

Figure 3.5.2-2 shows the $K(Z)$ function (fuel height limit for normalized $F_Q(Z)$). Each cycle's core loading satisfies the envelope shown in Figure 3.5.2-2.

TABLE 3.5.2-1

FUEL ASSEMBLY DESIGN PARAMETERS
 COOK NUCLEAR PLANT UNIT 1 - CYCLE 15

<u>Region</u>	<u>15</u>	<u>15</u>	<u>16</u>	<u>16</u>	<u>17</u>	<u>17</u>
Enrichment (w/o of U 235) *	3.099	3.609	3.110	3.513	3.117	3.612
Density (percent theoretical)*	95.486	95.378	95.521	95.418	95.451	95.451
Number of Assemblies	22	11	45	31	32	52
Burnup at Beginning of Cycle 15 (MWD/MTU)**	26301	34533	17037	17166	0	0
Approximate Burnup at End of Cycle 14 (MWD.MTU)***	37715	48408	30012	31687	17604	19121

* All values are as-built.

** Based upon actual EOC 14 burnup of 14267 MWD/MTU

*** Assumes EOC burnup of 15520 MWD/MTU

TABLE 3.5.2-2

KINETICS CHARACTERISTICS
COOK NUCLEAR PLANT UNIT 1 WITH W OFA FUEL

Most Positive Moderator Temperature Coefficient (pcm/°F)**	+5.0 ≤ 70% RTP* linear ramp to 0.0 from 70 to 100% RTP
Doppler Temperature Coefficient (pcm/°F)	-0.9 to -3.2
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)	-9.55 to -6.17
Most Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)	-19.4 to -12.90
Delayed Neutron Fraction, β_{eff} (%)	0.40 to 0.70
β_{eff} (%) minimum (BOL rod ejection only)	≥ 0.5
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/sec)**	≤ 75

*RTP = Rated Thermal Power

**1 pcm = $1.0 \times 10^{-5} \Delta\rho$

3.5.3 Thermal and Hydraulic Design

Introduction

This section describes the thermal and hydraulic design of Cook Nuclear Plant Unit 1 core with Westinghouse Optimized Fuel Assemblies (OFA)

The thermal hydraulic design of the core is conservatively analyzed at 3413 MWt core power with a 578.7°F vessel average temperature. Example Cycle 15 was operated at a licensed power of 3250 MWt with a nominal vessel average temperature of 553°F. The analyses employed the Improved Thermal Design Procedure⁽¹⁾ (ITDP) and THINC IV^(2,3) computer code. The WRB-1⁽⁴⁾ DNB correlation was used in the Westinghouse 15x15 OFA analyses. Vessel temperature was increased to 556°F in Cycle 16.

Summary

The design method employed to meet the DNB design basis is the ITDP.⁽¹⁾ Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent probability that the minimum DNBR will be greater than or equal to the limit DNBR for the peak power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a design DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, the limit DNBR values are increased to values

designated as the safety analysis limit DNBRs. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR values is not required to meet the design basis.

In this application, the WRB-1 DNB correlation⁽⁴⁾ is employed in the thermal hydraulic design of the Westinghouse 15x15 OFA fuel. Due to an improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation compared to previous DNB correlations, a correlation limit DNBR of 1.17 is applicable.

The table below shows the relationships which exist between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design, using the Westinghouse Improved Thermal Design Procedure (ITDP)⁽¹⁾.

	Typical	Thimble
Correlation Limit	1.17	1.17
Design Limit	1.33	1.32
Safety Analyses Limit	1.45	1.45

For events where conditions fall outside the range of applicability of the WRB-1 correlation, the W-3^(5, 6) correlation is used.

The margin to the safety analysis DNBR limit is more than sufficient to cover the maximum rod bow penalty at full flow conditions⁽⁷⁾.

defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration although limited fuel erosion can occur. As has been shown by operating experience and extensive experimental work, the thermal design parameters conservatively account for changes in the thermal performance of the fuel elements due to pellet fracture which may occur during power operation. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. Observations from several operating Westinghouse PWR's^(2,4) has shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets could result in local and distributed gaps in the fuel rods.

An extensive analytical and experimental effort has been conducted by Westinghouse^(2,4) to characterize the fuel densification phenomenon. Fuel densification during the manufacturing process is approximately 95.3 percent theoretical fuel density.

The effects of fuel densification have been taken into account in the nuclear and thermal hydraulic design of the reactor described in Sections 3.3 and 3.4, respectively.

Metallographic examination of irradiated commercial fuel rods have shown occurrences of fuel/clad chemical interaction. Reaction layers of < 1 mil in thickness have been observed between fuel and clad at limited points around the circumference. Westinghouse metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out of pile tests have shown that in the presence of high clad tensile stresses, large concentrations of iodine can chemically attack the Zircaloy tubing and can lead to eventual clad cracking. Westinghouse has no inpile evidence that this mechanism is operative in commercial fuel.

Materials - Strength Considerations

One factor in fuel element duty is potential mechanical interaction of fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these in turn consume clad fatigue lifetime.

The reduction of fuel/clad interaction is therefore a goal of design. In order to achieve this goal and to enhance the cyclic operational capability of the fuel rod, the technology for using pre-pressurized fuel rods in Westinghouse PWR's has been developed.

Initially the gap between the fuel and clad is sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in the fuel/clad contact. During this period of fuel/clad contact, changes in power level could result in changes in clad stresses and strains. By using pre-pressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod pre-pressurization delays the time at which fuel/clad interaction and contact occur and hence significantly reduces the number and extent of cyclic stresses and strains experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability. If gaps should form in the fuel stacks, clad flattening will be prevented by the rod pre-pressurization so that the flattening time will be greater than the fuel's core life.

A two dimensional (r,h) finite element model has been established to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration, herein, is defined as the difference between the maximum clad stress in the h-direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium and as a result the stress in the clad are close to zero. In subsequent cases the pellet power is increased in steps and the resultant fuel thermal expansion imposes

Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated energy output of the fuel (MWD/MTU) and is a convenient means for quantifying fuel exposure criteria.

The core design lifetime or design discharge burnup is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Section 3.3.2) that meets all safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements (e.g., the controlling band at the "bite" position). In terms of chemical shim boron concentration this represents approximately 10 ppm with no control rod insertion.

A limitation on initial installed excess reactivity is not required other than as is quantified in terms of other design bases such as core negative reactivity feedback and shutdown margin discussed below.

3.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficient)

Basis

The fuel temperature coefficient is negative and the moderator temperature coefficient of reactivity is non-positive for power operation at 100% RTP, thereby providing negative reactivity feedback characteristics. The design basis meets GDC-11.

Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler)

associated with changing fuel temperature and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have an overall non-positive moderator temperature coefficient of reactivity at full power so that average coolant temperature or void content provides another, slower compensatory effect. Full power operation is permitted only in a range of overall non-positive moderator temperature coefficient. The non-positive moderator temperature coefficient can be achieved through use of fixed burnable absorber, integral fuel burnable absorbers and/or control rods by limiting the reactivity held down by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a non-positive moderator temperature coefficient at power operating conditions discussed above.

3.3.1.3 Control of Power Distribution

Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

1. The fuel is not to be operated at greater than 12.9 Kw/ft under normal operating conditions including an allowance of 2 percent for calorimetric error and not including power spike factor due to densification.
2. Under abnormal conditions including the maximum overpower condition, the fuel peak power does not cause melting as defined in Section 3.4.1.2.

is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods. Such oscillations are readily observable and alarmed, using the excore long ion chambers. Indications are also available from incore thermocouples and loop temperature measurements. Moveable incore detectors can be activated to provide more detailed information. In all presently proposed cores these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects designed into the core.

However, axial xenon spatial power oscillations may occur late in core life. The control banks and excore detectors are provided for control and monitoring of axial power distributions. Assurance that fuel design limits are not exceeded is provided by reactor Overpower ΔT and Overtemperature ΔT trip functions which use the measured axial power imbalance as an input.

3.3.1.7 Anticipated Transients Without Scram

In the Code of Federal Regulations, 10 CFR 50.62(c)(1) requires that each pressurized water reactor have equipment, from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an anticipated transient without scram (ATWS). Such a system has been installed at Cook Nuclear Plant, having been designed in accordance with Reference 1. This system is called "ATWS Mitigating System Actuation Circuitry" (AMSAC). This equipment will protect against reactor coolant system overpressurization in the event that a loss of normal feedwater or a loss of load transient is not accompanied by a reactor trip after having reached the reactor trip setpoint.

3.3.2 DESCRIPTION

The majority of the information in this subsection refers to the Cycle 8 reload design. Cycle 8 is selected as the example of current reload design practice when reloading the core with the Westinghouse Vantage 5 17 x 17 fuel assembly design. Additional information on Cycle 8 may be found in Reference 23.

3.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids attached to rod cluster control thimbles which are held in turn by top and bottom fittings. The fuel rods are constructed of Zircaloy cylindrical tubes containing UO_2 fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly contains a 17 x 17 rod array composed of 264 fuel rods, 24 rod cluster control thimbles and an incore instrumentation thimble. Figure 3.2-1 shows a cross sectional view of a 17 x 17 Westinghouse fuel assembly and the related rod cluster control locations. Further details of the fuel assembly are given in Section 3.2.1.

Starting with Cycle 8, fresh fuel has axial zoning of uranium enrichment; however, the enrichment in the radial direction of an assembly is still maintained at a consistent enrichment. Generally, axial zoning consists of loading the top and bottom six (6) inches of the fuel with natural uranium. Blankets, as the natural uranium regions are referred to, reduce the axial neutron leakage, thereby contributing to better fuel utilization. Figure 3.3-1 shows the axial zoning of the fuel and the axial placement of the integral fuel burnable absorber (IFBA) coated fuel pellets. The reference reloading pattern is similar to the example in Figure 3.3-2. The loading of fresh fuel in the interior of the core decreases the radial neutron leakage by reducing the power produced on the core periphery. This type of reload pattern increases the fuel utilization and is referred to as a low leakage loading pattern.

Each cycle will operate for approximately 476 EFPD (which is equal to an 18 month cycle at 95% capacity factor with a 45 day refueling outage). The exact reloading pattern, initial and final positions of assemblies and the number of fresh assemblies are dependent on the energy requirement for the cycle and power histories of the previous cycles.

The core average enrichment is determined by the amount of fissionable material required to provide the desired core lifetime and energy requirements, namely a region average discharge burnup of 48,000 MWD/MTU. The physics of the burnup process are such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons.

These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 3.3-3 for the 17 x 17 fuel assembly, which occurs due to the nonfission absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be "built" into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant and IFBA coating on fuel pellets.

The concentration of boric acid in the primary coolant is varied to provide control and to compensate for long term reactivity requirements. The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable absorber depletion, and the cold-to-operating moderator temperature change. The normal and emergency boration paths of the chemical and volume control system (CVCS) are each capable of inserting negative reactivity at a rate in excess of the peak xenon burnout rate. The rate of boration, with a single boric acid transfer pump operating, is sufficient to take the reactor from full power operation to 1 percent shutdown in the hot condition, with no rods inserted, in less than 90 minutes.

In less than 90 additional minutes, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the equilibrium operating level will not begin until approximately 25 hours after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions. Rapid transient reactivity requirements and safety shutdown requirements are met with control rods.

As the boron concentration is increased, the moderator temperature coefficient becomes less negative. The use of a soluble poison alone could result in the MTC exceeding the Technical Specifications limit. Therefore, IFBA coated fuel pellets are used to reduce the soluble boron concentration sufficiently to ensure that the MTC is negative for full power operating conditions, and within safety limits for part power operating conditions. IFBA pins contain enriched uranium pellets with a thin coating of ZrB_2 on the fuel pellet's cylindrical surface. During operation, the burnable absorber content in the IFBA coating is depleted thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product buildup. The depletion rate of the IFBA coating is not critical since chemical shim is always available and flexible enough to cover any possible deviations in the expected IFBA depletion rates. The IFBA coating is thin enough (approx. 0.6 mil) to not cause significant increases in resistance to heat conduction.

In addition to reactivity control and axial power shaping, the IFBA pins are strategically located to provide a favorable radial power distribution. Figure 3.3-4 shows example absorber distributions within a fuel assembly for several example IFBA configurations used in a 17 x 17 array. An example, IFBA core loading pattern is shown in Figure 3.3-5. Figure 3.3-6 is a graph of an example core depletion with IFBA coated fuel pellets.

Tables 3.3-1 through 3.3-3 contain a summary of the reactor core design parameters for an example fuel cycle, including reactivity coefficients, delayed neutron fraction and neutron lifetimes. Sufficient information is included to permit an independent calculation of the nuclear performance characteristics of the core.

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21. Poncelet, C. G., "LASER, A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6075, April, 1966.
22. Olhoeft, J. E., "The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," WCAP-2048, July, 1962.
23. Johansen, B. J., et. al., "Nuclear Parameters and Operations Package for the Donald C. Cook Nuclear Plant (Unit 2 Cycle 8)," WCAP-12651 (Proprietary), October 1990.
24. Meyers, R. O., "The Analysis of Fuel Densification," Division of Systems Safety, USNRC, NUREG-0085, July, 1976.
25. Meyer, C. E. and Stover, R. L., "Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors," WCAP-8498, July 1975.
26. Ford, W. E., et. al., "CSRL-V: Processed ENDF/B-V 227-Neutron Group and Pointwise Cross Section Libraries for Criticality Safety, Reactor and Shielding Studies," NUREG/CR-2306, ORNL/CSD/TM-160 (1982).

TABLE 3.3-1
REACTOR CORE DESCRIPTION

Active Core

Equivalent Diameter, in	132.7
Active Fuel Height, First Core, in	144.0
Height-to-Diameter Ratio	1.09
Total Cross Section Area, ft ²	96.06
H ₂ O/U Molecular Ratio, lattice (Cold)	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 - per assembly	1058
Zircaloy Weight, lb - per assembly	206
Number of Grids per Assembly	2-R 6-Z 3-IFM 1-P
Composition of Grids	R-Inconel 718 Z-Zircaloy 4 IFM - Zircaloy 4 P-Debris Resistent - Inconel 718
Weight of Grids (Effective in Core), lb - per assembly	20.79
Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	Zircaloy 4
Diameter of Guide Thimbles (upper part), in	0.442 I.D. x 0.474 O.D.
Diameter of Guide Thimbles (lower part), in	0.397 I.D. x 0.430 O.D.
Diameter of Instrument Guide Thimbles, in	0.440 I.D. x 0.476 O.D.

TABLE 3.3-1 (Continued)
REACTOR CORE DESCRIPTION

Fuel Rods

Number	50,952
Outside Diameter, in	0.360
Diameter Gap, in	0.0062
Clad Thickness, in	0.0225
Clad Material	Zircaloy-4

Fuel Pellers

Material	UO ₂ Sintered
Density (percent of Theoretical)	95.3
Maximum Fuel Enrichments w/o	4.95
Diameter, in	0.3088
Length, in	0.370 Enriched 0.462 Axial Blanket
Mass of UO ₂ per Foot of Fuel Rod, lb/ft	0.349

Rod Cluster Control Assemblies

Neutron Absorber	Ag-In-Cd
Composition	80%, 15%, 5%
Diameter, in	0.341
Density, lb/in ³	0.367
Cladding Material	Type 304, Cold Worked Stainless Steel
Clad Thickness, in	0.0185
Number of Clusters	
Full Length	53
Part Length	0
Number of Absorber Rods per cluster	24
Full Length Assembly Weight (dry), lb	157

TABLE 3.3-1 (Continued)

REACTOR CORE DESCRIPTIONExcess Reactivity

Maximum Fuel Assembly k (Cold, Clean, unborated Water)	1.476
Maximum Core Reactivity (Cold, Zero Power Beginning of Cycle)	1.224

Integral Fuel Burnable Absorber

Number	~8100
Material	ZrB ₂
Coating Thickness, mil	~0.2
Boron 10 Loading, mg/in	2.25

5. Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events⁽⁴⁵⁾. A portion of the margin resulting from the difference between the design and safety analysis limit DNBRs is used to counteract the rod bow penalties.

The maximum rod bow penalties ($\leq 1.3\%$) 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 $F_{\Delta H}^N$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required⁽⁴⁶⁾.

In the upper spans of the Vantage 5 fuel assembly, additional restraint is provided with the intermediate flow mixer (IFM) 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% closure. Therefore, no rod bow DNBR penalty is required in the 10 inch spans in the Vantage 5 safety analyses.

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3.4.2.4 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 RCCA could cause changes in hot channel factors; however, these events are analyzed separately in Chapter 14. This discussion will be confined to flux tilts caused by x-y xenon transients, inlet temperature mismatches, enrichment variations within tolerances and so forth.

It is assumed that the spray valve opens to admit spray water into the pressurizer once, at the design flowrate, for each design step change in plant load. Thus the number of occurrences for the spray nozzle corresponds to that shown for the other components in Table 4.1-10.

During plant cooldown, spray water is introduced into the pressurizer to cool it down. The maximum pressurizer cooldown rate is specified at 200°F per hour which is twice the rate specified for the other Reactor Coolant System components.

12. Accident Conditions

The effect of the accident loading was evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

A brief description of each accident transient considered follows. In each case one occurrence is evaluated.

a. Reactor Coolant Pipe Break

This accident involves the rupture of a Reactor Coolant System pipe resulting in a loss of primary coolant. It was conservatively assumed that the system pressure and temperature would be reduced rapidly and that the Safety Injection System would be initiated to introduce 70°F (30°F for Unit 2) water into the Reactor Coolant System. The safety injection signal will also result in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F (30°F for Unit 2) safety injection water is introduced into the system.

b. Steam Line Break

For component evaluation, the following conservative conditions were considered:

- (1) The reactor is initially in a hot, no-load, just critical condition, assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.
- (2) A steam line break occurs inside the containment resulting in a reactor and turbine trip.
- (3) Subsequent to the break, there is no return to power and the reactor coolant temperature cools down to 212°F.
- (4) The centrifugal charging pumps restore the reactor coolant pressure to 2500 psia.

The above conditions result in the most severe temperature and pressure variations which the component will encounter during a steam break accident.

c. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to a safety injection signal on low pressurizer pressure. When the accident occurs, some of the reactor coolant blows down into the affected steam generator causing the level to rise. If the level rises to a pre-selected setpoint, a high level alarm will occur and the feedwater regulating valve will close.

It is expected this accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment in so far as fatigue evaluation is concerned. Further detail about the sequence of events may be found in Section 14.2.4 (Units 1 and 2).

4.1.5 SERVICE LIFE

The service life of the Reactor Coolant System pressure containing components depends upon the end-of-life material radiation damage, unit operational thermal cycles, design and manufacturing quality standards, environmental protection, maintenance standards and adherence to established operating and maintenance procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and therefore it is the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds during service due to radiation damage effects is monitored by a radiation damage surveillance program. Details are given in Sub-Chapter 4.5.

Reactor vessel design was based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material as a result of operation.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for "A" vessels, unit operating conditions have been established

for the 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes is listed in Table 4.1-10.

4.1.6 CODES AND CLASSIFICATIONS

Pressure-containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-12. Refer to Sub-Chapter 4.5 for a discussion of Inservice Inspection.

Reactor Coolant System piping has been designed and supported in accordance with the USAS B31.1-1967 Code for Pressure Piping. The Code requirement that the piping shall be arranged and supported with consideration of vibration was met by means of variable spring hangers, rigid supports, constant support hangers, pipe anchors, guides and snubbers. The Code does not specifically require any vibrational test programs. However, during the normal course of the preoperational test program, specific attention was directed at evaluating possible vibration problems during performance of specific transients associated with the required preoperational tests. Excessive vibrations or deficiencies, determined by visual examinations, which were indicative of possible vibration problems, were investigated and corrected when necessary. This was done to verify that the piping and piping restraints within the Reactor Coolant System pressure boundary were adequately designed to withstand dynamic effects resulting from transient conditions.

TABLE 4.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40	
Number of heat transfer loops	4	
Design pressure, psig	2485	
Nominal operating pressure, psig	2085 (unit 1)/2235 (unit 2)	
Total system volume including pressurizer and surge line (ambient conditions)**, ft ³ (estimated)	12,500	
System liquid volume, including pressurizer and surge line (ambient conditions)**, ft ³	11,892	
System liquid volume, including pressurizer max. guaranteed power**, ft ³ (estimated)	11,891	
Total Reactor heat output (100% power) Btu/hr	11,089 x 10 ⁶ (Unit 1)	
	(3250 MWt)	
	11,641 x 10 ⁶ (Unit 2)	
	(3411 MWt)	

	<u>Unit 1</u>	<u>Unit 2</u>
Bounding Conditions		
for Rerating		
Lower/Upper		

Reactor vessel coolant temperature		
at full power:		
Inlet, nominal, °F	514.9/545.2	541.3
Outlet, nominal, °F	579.1/607.5	606.4
Coolant temperature rise in vessel		
at full power, avg., °F	64.2/62.3	64.8
Total coolant flow rate, lb/hr x 10 ⁶	139.0/133.9	134.6
Steam pressure at full power, psia	618/820	820
Steam Temp. @ full power, °F	489.4/521.1	521.1
Total Reactor Coolant Volume at ambient conditions**, ft ³	12,438	12,470

** These values are subject to change due to tube plugging

TABLE 4.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	<u>Pressure, psig</u>	
	<u>Unit 1</u>	<u>Unit 2</u>
Design Pressure	2485	2485
Operating Pressure	2085	2235
Safety Valves	2485	2485
Power Relief Valves*	2335	2335
Pressurizer Spray Valves (Begin to Open)	2260	2260
Pressurizer Spray Valves (Full Open)	2310	2310
Pressurizer Pressure High - Reactor Trip	2378	2378
High Pressure Alarm	2310	2310
Pressurizer Pressure Low - Reactor Trip	≥ 1865	≥ 1950
Low Pressure Alarm	2135	2135
Pressurizer Pressure Low - Safety Injection	≥ 1815	≥ 1900
Hydrostatic Test Pressure	3106	3106
Backup Heaters On	2185	2185
Proportional Heaters (Begin to Operate)	2250	2250
Proportional Heaters (Full Operation)	2220	2220

*During Start-up and Shut-down when the Reactor Coolant System temperature is below 266°F for Unit 1 and 300°F for Unit 2, a safeguard circuit is manually switched on which allows opening of that Unit's two Power Relief Valves at ≤435 psig for Unit 1 and ≤435 psig for Unit 2 for low temperature overpressure protection (LTOP) of the Reactor Vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection Against Non-ductile Failure" limits in the case of an LTOP event.

TABLE 4.1-3
REACTOR VESSEL DESIGN DATA

Design Pressure, psig	2485
Operating Pressure, psig	2085 (unit 1)/2235 (unit 2)
Hydrostatic Test Pressure, psig	3107
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in.	43-9 11/16 (Unit 1)
(Bottom Head O.D. to top of Control Rod Mechanism Adapter)	43-10 (Unit 2)
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in.	7
ID of Flange, in.	172 1/2
OD of Flange, in.	205
ID at Shell, in.	173
Inlet Nozzle ID, in.	27 1/2
Outlet Nozzle ID, in.	29
Clad Thickness, min., in.	5/32
Lower Head Thickness, min., in. (base metal)	5-3/8
Vessel Belt-Line Thickness, min., in. (base metal)	8 1/2
Closure Head Thickness, in.	6 1/2

	<u>Unit 1</u>	<u>Unit 2</u>
	Bounding Conditions for Rerating Lower/Upper	
Reactor Coolant Inlet Temperature, °F	514.9/545.2	541.27
Reactor Coolant Outlet Temperature, °F	579.1/607.5	606.35
Reactor Coolant Flow, lb/hr x 10 ⁶	139.0/133.9	134.6
Total Water Volume Below Core, ft ³		1050
Water Volume in Active Core Region, ft ³		665
Total Water Volume to Top of Core, ft ³		2352
Total Water Volume to Coolant Piping Nozzles Centerline, ft ³		2959
Total Reactor Vessel Water Volume (with core and internals in place), ft ³ (estimated)		4848

TABLE 4.1-4

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATAPressurizer

Design Pressure, psig	2485
Operating Pressure, psig	2085 (unit 1)/2235 (unit 2)
Hydrostatic Test Pressure (cold), psig	3106
Design/Operating Temperature, °F	680/653
Water Volume, Full Power*, ft ³	858 (Unit 1) - 974 (Unit 2)
Steam Volume, Full Power*, ft ³	973 (Unit 1) - 826 (Unit 2)
Total Internal Volume, ft ³	1831
Surge Line Nozzle Diameter, in.	14
Shell ID, in.	84
Electric Heater Capacity, kW	1685 (unit 1)/1523 (unit 2)
Heatup rate of Pressurizer, °F/hr	55 (approx.)
Start-up Water Solid, °F/hr	40
Hot Standby Condition, °F/hr	70
Design Spray Rate for Valves Full Open, gpm	800
Continuous Spray Rate, gpm	1

Pressurizer Relief Tank

Design Pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design Temperature, °F	340
Normal Water Temperature, °F	Containment Ambient (120°F Max.)
Normal Operating Pressure, psig	3
Normal Water Volume, ft ³	1430
Normal Gas Volume, ft ³	370
Cooling time required following design maximum discharge, hr.	Approx. 1
Number of spray nozzles	5
Total Spray Flow, gpm	150
Total Volume, ft ³	1800
Total Rupture Disc Relief Capacity, saturated steam, lb/hr	1.6 x 10 ⁶

*At current rating conditions

TABLE 4.1-5

STEAM GENERATOR DESIGN DATA* Unit 1 Unit 2

Number of Steam Generators	4	4
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107	3107
Design temperature, Reactor Coolant/Steam, °F	650/600	650/600
Reactor Coolant Flow, lb/hr	33.9×10^6	33.7×10^6
Total Heat Transfer Surface Area, ft ²	51,500	54,500
Rated Thermal Output/MWt)	812.5	852.75
Operating Parameters at 100% Load		
Primary Side:		
Heat Transfer Rate (per unit), Btu/hr	2773×10^6	2910×10^6
Coolant Inlet Temperature, °F	582.8	606.4
Coolant Outlet Temperature, °F	520.0	541.3
Flow Rate, (per unit), lb/hr	33.9×10^6	33.7×10^6
Pressure loss, psi	31.4	26.1
Secondary Side:		
Steam Temperature at full power, °F	502.0	521.1
Steam Flow, lb/hr	3.55×10^6	3.685×10^6
Steam Pressure at full power, psia	692	820
Maximum moisture carryover, wt %	0.15	0.15
Feedwater Temperature at No. 6 Heater Outlet	436.5	431.3
Fouling Factor, hr-ft ² -°F/Btu	0.0002	0.00005
Overall Height, ft-in.	67-8	67-8
Shell OD, upper/lower, in.	175.75/135	175.9/135.
Number of U-tubes	3388	3592
U-tube outer Diameter, in.	0.875	0.875
Tube Wall Thickness, (minimum), in.	0.050	0.050
Number of manways/ID, in.	4/16	4/16
Number of handholes/ID, in.	2/6	6/6
Number of inspection ports/ID, in.	0	2/4

*Quantities are for each steam generator

TABLE 4.1-5 (cont'd.)

STEAM GENERATOR DESIGN DATA*Unit 1

	<u>Rated Load</u>	<u>No Load</u>
Reactor Coolant Water Volume**, ft ³	1080	1080
Primary Side Fluid Heat Content, Btu	28.7×10^6	27.7×10^6
Secondary Side Water Volume, ft ³	1837	3524
Secondary Side Steam Volume, ft ³	4030	2344
Secondary Side Fluid Heat Content, Btu	5.738×10^7	9.628×10^7

Unit 2

Reactor Coolant Water Volume**, ft ³	1112	1112
Primary Side Fluid Heat Content, Btu	29.0×10^6	28.46×10^6
Secondary Side Water Volume, ft ³	2077	3351
Secondary Side Steam Volume, ft ³	3589	2315
Secondary Side Fluid Heat Content, Btu	5.18×10^7	8.44×10^7

*Quantities are for each steam generator

** Values may change subject to steam generator tube plugging.

TABLE 4.1-6

REACTOR COOLANT PUMPS DESIGN DATA*

Number of Pumps	4 Design
Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3106
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1189
Suction Temperature, °F	536.3 (Unit 1); 541.27 (Unit 2)
Required net positive suction head, ft	170
Developed Head, ft	277
Capacity, gpm	88,500
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in.	27.5
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft-in.	27-0
Water Volume, ft ³	56
Pump-Motor Moment of Inertia, lb-ft ²	82,000
Motor Data:	
Type	AC Squirrel Cage
	Induction, Single Speed,
	Water Cooled
Voltage	4160.
Insulation Class	F
Phase	3
Frequency, Hz	60
Starting	
Current, amp	4692
Input (hot reactor coolant), kw	4337
Input (cold reactor coolant), kw	5663
Power, HP (nameplates)	6000
Pump Weight, lb. (dry)	175,200

*Quantities are for each pump

TABLE 4.1-7

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor inlet piping, ID, in.	27.5
Reactor inlet piping, nominal thickness, in.	2.38
Reactor outlet piping, ID, in.	29
Reactor outlet piping, nominal thickness, in.	2.50
Coolant pump suction piping, ID, in.	31
Coolant pump suction piping, nominal thickness, in.	2.66
Pressurizer surge line piping, ID, in.	11.188
Pressurizer surge line piping, nominal thickness, in.	1.406
Design pressure, psig	2485
Operating Pressure, psig	2085 (unit 1)/2235 (unit 2)
Hydrostatic test pressure (cold), psig	3106
Design temperature, °F	650
Design temperature (pressurizer surge line), °F	680
Design pressure, pressurizer relief line, psig	(1)
Design temperature, pressurizer relief lines, °F	(1)
Water volume (all 4 loops without surge line), ft ³	1127.8
Surge line volume, ft ³	58.5

(1) From pressurizer to safety valve: 2485 psig, 650°F

From safety valve to pressurizer relief tank: 500 psig, 470°F

Indication of valve position for the pressurizer safety and power-operated relief valves is provided by a four channel acoustic flow monitor. There are four accelerometers, one strapped to the discharge of each of the three pressurizer safety valves and one on the common discharge of the three power relief valves. Flow through any of these valves produces an acoustic energy input to the respective accelerometer and this is amplified on the assigned channel of the monitor which is located in the control room. Indication on four vertical rows of light emitting diodes represents a bar graph display of relative flow through the monitored valves.

Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and with back-pressure compensation designed to prevent system pressure from exceeding the design pressure by more than 110 percent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The set pressure of the valves is 2485 psig.

The 6" pipes connecting the pressurizer nozzles to their respective safety valves are shaped in the form of a loop seal. Piping is connected to the bottom of each loop seal to drain any condensate that accumulates in the loop seal. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

Power Relief Valves

The pressurizer is equipped with 3 power-operated relief valves which limit system pressure for a large power mismatch and thus lessen the likelihood of an actuation of the fixed high-pressure reactor trip. The relief valves operate automatically or by remote manual control. The original design for 3 PORVs was to provide 100% load rejection capability. Since the load rejection capability has been reduced to 50%, the third PORV is now considered an installed spare. The operation of these valves also limits the undesirable operation of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves. An acoustic flow monitor and a temperature indicator on the common discharge of the relief valves alerts the operator to the passage of steam due to leakage or valve opening.

During startup and shutdown transient conditions, when the reactor coolant system temperature is below 266°F for Unit 1 and 300°F for Unit 2, a safeguard circuit is manually energized in the control room to allow automatic opening of that unit's two power relief valves at 435 psig for Unit 1, and 435 psig for Unit 2, for low temperature over pressure (LTOP) protection of the reactor vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection Against Nonductile Failure" limits in the case of an LTOP event.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 4.1-8.

4.2.2.9 Reactor Coolant System Supports

1. Steam Generator Support

Each steam generator is supported by a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 46½ feet apart. The vertical columns have a ball joint connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

The lower lateral support consists of an inner frame, keyed and shimmed to the four steam generator support feet to accommodate radial growth of these feet. The inner frame is surrounded by an outer frame which is embedded in both the primary shield and crane wall concrete. The connection between the inner and outer frame consists of a series of shimmed points which act as both guides and limit stops to allow for expansion from the center of the reactor. The lower lateral support restrains both torsional and translational movements.

The upper lateral support consists of a ring band which is shimmed to the steam generator at twelve locations around the circumference. Attached to this band are lugs 180° apart which are shimmed and guided to a structural framing system which is embedded in the crane wall and steam generator enclosure wall concrete. Hydraulic snubbers are also connected 180° apart on the band and tied to other embedded frames in a direction coincident with the direction of movement away from the reactor center. The upper lateral support restrains rapid translational movements in all horizontal directions.

2. Reactor Vessel Supports

The reactor vessel is supported by four of its eight nozzles by four individual weldments embedded in the primary shield concrete. Each nozzle pad bears on a shoe, that is supported by a heavy U-shaped weldment which wraps around the shoe. The U-shaped weldment is water-cooled at the junction of the outer flange and the web by two continuous welded angles on either side of the web. The U-shaped weldment bears vertically on two shims and is restrained horizontally by a series of shims and bearing plates. These bearing plates and shims are connected to an outer weldment which completely surrounds the U-shaped weldment and is embedded in the concrete.

The reactor support system allows the reactor to expand radially from its vertical centerline but resists rotational motion in all orthogonal planes. The nozzle horizontal centerlines translate in the vertical direction relative to the shoes.

3. Pressurizer Support

The pressurizer is supported on a ring girder which is in turn supported on a concrete slab. Horizontally, the vessel is restrained at two elevations approximately 27 feet apart.

The lower restraint consists of anchor bolts in slightly oversize holes in the ring girder. The upper restraint consists of four individual weldments embedded in concrete that allow the pressurizer to expand radially, but resist torsional and translational horizontal movements.

4. Reactor Coolant Pump Support

Each reactor coolant pump is supported vertically by three ball joint ended columns. This structural column system resists both overturning and vertical movement while allowing for expansion from the center of reactor. Excessive torsional and horizontal translational movements are resisted by a combination of lateral thrust columns anchored into the crane wall concrete.

4.2.3 PRESSURE-RELIEVING DEVICES

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by relief and safety valves connected to the top head of the pressurizer. The relief and safety valves are currently analyzed for steam discharge only. However, evaluations have shown that the pressurizer will not become water solid before at least 10 minutes following a spurious Safety Injection or a feedline break. The relief and safety valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the

relief devices is shown in Figure 4.2-1A, and the valve design parameters are given in Table 4.1-8. The valves are further discussed in Sub-Section 4.2.2.8.

Upon completion of the containment sub-slabs, the reactor pit was excavated and 1/4" steel plates were welded to the soldier piles to prevent the sand from sloughing into the excavation from beneath the sub slab. Some sand did slough in during the excavation and plate installation creating a void behind the plates in some areas. These voids were filled by pressure grouting after concrete work was completed within the reactor pit.

The containment structures were constructed on mat foundations founded directly on the dense beach sands. These sands were studied in detail to determine their susceptibility to liquefaction under the maximum design earthquake. The relative densities of these sands were found to be in the range of values which are not susceptible to liquefaction. The supporting data for this conclusion is contained in Appendix G of the Original FSAR. In addition, a complete settlement analysis was conducted to determine the anticipated total and differential settlement between major structures, the major portion of these settlements taking place during the construction period. Computed differential settlement does not exceed one (1) inch. The supporting data and detailed analysis is contained in Appendix G of the Original FSAR.

In order to monitor settlements of the containment structures, three permanent benchmarks were installed through each containment base slab 120 degrees apart and were positioned such that they are outside the containment building. These benchmarks extend to bedrock and are equipped with extensometers which indicate directly the amount of settlement of the containment slabs. The installation was monitored at regular intervals to substantiate the conclusion of the settlement analysis. The monitoring has confirmed the predictions of the settlement analysis and the settlement activity has virtually ceased. The periodic monitoring was discontinued in 1981.

With the exception of the Class I Tanks, all remaining Class I Structures were handled in a manner similar to the containment structures, e.g., soldier piles were driven, excavation progressed with the installation

of steel plates, sub-slabs were installed at the bottom of the excavation. Pressure grouting was also performed behind these plates to fill the small voids which developed as a result of sloughing of the sand.

The Class I Tanks were founded on compacted backfill. The areas were first excavated down to the dense beach sands and then brought back to foundation grade with controlled compacted backfill.

The Underground area of the auxiliary and containment buildings have been waterproofed by means of a PVC 40 mil thick membrane. This membrane extends at least 5 feet above the maximum known GW level. The water-proofing used provides adequate protection against flooding of areas located below the highest GW level.

Underground structures such as tunnels have been designed to articulate. Where underground structures join main facilities the joint has been made non-moment resisting. The stress criteria used in evaluating the underground structure design is the same as that used in other class I structures. Typical design sketch of connections are shown in Figure 5.2.2-5.

The estimated potential static differential settlement after connection of interlinking mechanical and electrical elements between the Containment Structure and the Auxiliary Building is 0.2" and between the Auxiliary Building and the Turbine Building is 0.1". This estimate was based on an assumption that 75% of the total settlement of the structure occurs during the construction phase.

There are no direct interconnecting building structural elements between the Auxiliary Building and the Containment Structure, nor between the Auxiliary Building and the Turbine Building. The inter-connecting element between the Auxiliary Building and the Diesel

The following information presents an overview of the Ice Condenser design. Additional detailed information is presented in the updated FSAR Appendices J and M.

The Ice Condenser, is a completely enclosed annular compartment located around approximately 300° of the perimeter of the upper compartment of the containment, but penetrating the operating deck so that a portion extends into the lower compartment of the containment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part at the Ice Condenser. These doors remain closed during normal plant operation.

In the ice condenser, ice is held in baskets arranged to promote heat transfer from steam to ice to allow the ice condenser to perform its function (see following paragraph). A refrigeration system maintains the ice in the solid state. Suitable insulation surrounding both the ice condenser volume and the refrigeration ducts serves to minimize the heat transfer through the ice condenser enclosure.

In the event of a loss-of-coolant accident or steam line break, the door panels located below the operating deck (divider barrier) open due to the pressure rise in the lower compartment. This allows the air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the ice condenser to open, which allows the air

to flow out of the condenser into the upper compartment. The ice condenser condenses the steam as the steam enters the ice condenser compartment, thus limiting the peak pressure and temperature buildup in the containment. Condensation of steam within the ice condenser results in a continual flow of steam from the lower compartment to the condensing surface of the ice, thus reducing the time that the lower compartment is at an elevated pressure. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the bottom of the ice condenser. Only a negligible amount of steam can bypass the ice condenser through the divider barrier.

5.3.1 DESIGN CONSIDERATIONS

The following is a summary of the ice condenser design considerations. The design considerations are presented in two categories: performance criteria, and structural and mechanical considerations. More specific criteria for individual components of the ice condenser are also presented.

Performance Criteria

- a) The energy absorption capacity of the ice condenser is at least twice that required to absorb all of the energy that can be released, 1) during the initial blowdown of the Reactor Coolant System for any reactor coolant pipe break sizes up to and including the hypothetical severance of the reactor coolant piping, or 2) during any steam or feedwater system pipe break size up to and including the hypothetical severance of the main steam line inside the containment, without exceeding the containment design pressure.
- b) After an accident as described in (a), the ice condenser together with the containment spray system, has sufficient remaining heat absorption capacity such that subsequent assumed

heat loads are absorbed without exceeding the containment design pressure. The subsequent heat loads considered include stored and residual heat of the reactor core and coolant system, plus a substantial margin for an undefined additional energy release.

- c) Sufficient ice heat transfer area and flow passages are provided in the ice condenser so that the magnitude of the pressure transient resulting from an accident as described in (a) does not exceed the containment design pressure.
- d) The lower (Reactor Coolant System) compartment is bounded by the divider barrier such that essentially all of the energy released in this compartment is directed through doors at the bottom of the ice condenser.
- e) The resistance to flow into the ice condenser is such that the maximum energy input into any section of the ice condenser does not exceed its design capability.
- f) The force required to open the doors of the ice condenser is sufficiently low such that the energy from any leakage of steam through the divider barrier can be readily absorbed by the containment spray system without exceeding containment design pressure.
- g) The inlet doors of the ice condenser are designed to open and distribute steam to the ice bed in accordance with design basis (e) above, for any postulated loss-of-coolant accident.
- h) Both the inlet and outlet doors of the ice condenser are designed to fail open at a slightly higher differential pressure above the design opening pressure if for any reason the doors are prevented from opening normally.

- i) Ice with a suitable concentration of sodium tetraborate is used in the ice condenser so that in case of an accident, the water resulting from the melted ice is available for cooling the core.
- j) Raising the pH of the ice by addition of boron to the ice as sodium tetraborate rather than boric acid provides for the absorption and retention of iodine released from the core.
- k) Condensation of steam in the ice condenser aids in the removal of iodine from the containment atmosphere.

Structural and Mechanical Design

- a) The ice condenser internal structures are capable of withstanding all loading combinations with the following stress limits:

<u>Loading Combinations</u>	<u>Stress Limits</u>
Normal plus Operating Basis Earthquake Loads	Within Code Allowable
Normal plus Maximum Design Basis Hypothetical Earthquake Loads	Within yield after load redistributions
Normal plus Design Basis Accident Loads	Within yield after load redistributions
Normal plus Design Basis Earthquake plus Design Basis Accident Loads	Limit Curves - WCAP-5890, Rev. 1

In addition to the stated stress limits, structural stability and deformation requirements are determined so as to ensure no loss of function under accident and design bases earthquake (DBE) loads.

- b) In particular, the structure, equipment mounting, supports and joints are designed to accommodate the maximum temperature range and gradients which will occur.
- c) Structural loads are not transmitted between the ice condenser internals and the containment shell structure.
- d) The internals of the ice condenser are designed to facilitate maintenance.
- e) The ice condenser internals are designed for a lifetime consistent with that of the plant.
- f) The materials of construction are selected to be effectively inert under all conditions of operation of the ice condenser. In particular, corrosion is prevented by inhibitors or protective coatings where necessary and non-metallic materials are stable.
- g) Materials in the ice condenser system are selected to be compatible with the general environmental conditions inside the reactor containment during normal operation or during accident conditions. In particular, the choice of materials for the ice condenser insulation panels is compatible with the containment shell. See updated Appendix M.
- h) Sufficient redundancy is incorporated in the system design to provide a high assurance degree of plant availability.
- i) Components forming the boundary of the ice condenser are continuously sealed to limit the ingress or egress of air and vapor, except where specific provision is made for venting.

Specific Design Criteria

Ice Support Structure

- a) The structure is designed to maintain the ice in the required array to maintain the integrity of performance of the ice condenser. In particular, the hydraulic diameter and heat transfer area are maintained within the limits established by test to be consistent with the containment design pressure.
- b) The structure allows loadings of the ice baskets in position, and permits lifting of complete basket columns for removal in sections. Many columns can be lifted and weighed for surveillance purposes.
- c) Any section of the ice baskets is capable of supporting the total weight of ice above and below that section.

Insulated Duct Panels and Insulation

- a) The insulation limits the maximum total heat load on the ice condenser refrigeration system to a level consistent with the installed capacity, including reasonable margin.
- b) The heat input to the ice bed is minimized so that the ice bed performance capability will be maintained for a long period of time if the refrigeration system is shutdown.

In the region of the ice condenser, increased conductivity due to humidity and compression during an accident will not detract from the performance of the ice condenser. The insulation requirements for the ice condenser (under normal operating conditions) exceed the insulation requirements to protect the

containment vessel from thermal shock under accident conditions, even allowing for increased conductivity due to compression.

- c) The galvanized sheet metal covers on the inner faces of the duct panels are continuously sealed, and the outer sheet metal covers adjacent to the crane wall and the end walls form a vapor barrier. Under normal operating conditions, the vapor barrier prevents significant loss of insulation due to humidity.
- d) At the boundaries of the ice condenser where air cooling is not incorporated, insulation is provided in a form consistent with the structural and functional requirements of those areas.
- e) The panel insulation is installed as prefabricated sections (fiberglass encapsulated in polyethylene bags) and can be removed and replaced if necessary during the lifetime of the plant, after disassembly of the ice condenser internals. Precompression of the ice condenser insulation from structural and leakage tests does not detract from its performance capabilities.
- f) The performance of the insulation is not affected by the earthquake conditions.
- g) Under accident conditions the insulation does not affect the overall performance of the ice condenser.
- h) The materials used for insulation are compatible with the other areas of the reactor containment and systems.
- i) The method of attachment of the panels and their positions relative to the ice baskets and support structure precludes displacement of the panels during an accident.

Ice Condenser Doors

Normal Operation

- a) The doors restrict the leakage of air into and out of the ice condenser to the minimum practicable limit. Such provisions as are required for venting the ice condenser are treated separately and incorporate a range of adjustment to achieve the required total vent area in conjunction with any inherent leakage.
- b) The doors restrict the local heat input in the ice condenser to the minimum practicable limit.
- c) The lower inlet doors are instrumented to provide indication of their open position.
- d) Provision is made for adequate means of inspecting and testing of the doors during reactor shutdown.

Earthquake Conditions

The doors are designed to withstand earthquake loadings so as not to affect ice condenser operation for normal and accident conditions. These loads are derived from the seismic analysis of the containment.

Accident Conditions

Lower Inlet Doors

- a) The doors open (at least partially) in the event of a primary coolant or steam leak which produces an equalization of the cold air head differential pressure across the doors of 1/2 to 1 lb/sq. ft.

- b) The inlet doors and door parts of the ice condenser are designed to distribute steam to the condenser to limit mal-distribution to less than 150 percent of any loss-of-coolant accident which causes the door to open.
- c) The inertia of the doors is low, consistent with producing a negligible effect on initial pressure.
- d) During blowdown, adequate flow area is provided for the effluent of condensate and melted ice to drain from the ice condenser, without impeding the distributed input of steam.

Intermediate and Top Deck Doors

For larger leak rates, the resulting differential pressure opens a sufficient number of doors to permit air flow into the upper compartment.

Intermediate and Top Deck Vents

Venting of the ice condenser for small leak rates is provided by permanent vents in both the intermediate and top deck. This allows the inlet doors to open into the proportioning range.

Ice Condenser Drains

1. Drains have sufficient flow capacity to minimize the time that a water level in the ice condenser could cause an additional resistance to door opening, and to limit the potential increase in containment pressure that could occur in the short time that the doors are closed.

2. Drains are provided with flapper valves to seal the ice condenser and prevent loss of cold air during normal operation. When the ice melts during a LOCA, the resulting borated water will flow through these drains to the lower containment and sumps.
3. Drain flapper valves are gravity loaded to hold shut against the cold air head (1/2 to 1 lb/sq. ft.) in the ice condenser during normal operation. The pressure required to open the drain does not exceed 36 inches of water.
4. Drains are located such as to drain the water level in the condenser to an elevation below the bottom of the doors.

Performance Capability

Because of the static nature of the ice bed, the ice condenser function is not susceptible to failure of active components and the resulting consideration of additional capability to accommodate failure. In any case, the ice condenser does have an excess of capability for both rate and quantity of energy released from the Reactor Coolant System.

The door panels and drain check valve flappers are the only elements required to move during the accident. These items are considered as passive or static elements equivalent to rupture discs rather than active components requiring an external signal and energy source to function.

Testing and Inspection

The ice condenser design includes provisions for inservice visual inspection of the ice beds, flow channels, door panels, and cooling equipment. Samples of the ice can be taken to check additive

concentrations. During periods when the reactor is shut down door panels and drain check valves can be inspected, the door opening force can be tested and compared with the design force. In addition, inspection and testing of the installed ice condenser before and after the initial ice loading was conducted prior to initial plant startup.

5.3.2 DESCRIPTION OF ICE CONDENSER AND COMPONENTS

Included in this section are descriptions of the general arrangements of the ice condenser, the refrigeration-cooling system, the door panels at the top and bottom of the ice condenser, and the ice condenser internals which form the flow channels and ice beds. Table 5.3-1 presents principal design parameters for the Ice Condenser System. Additional information is presented in Appendices J and M.

General Arrangement

The general arrangement of the ice condenser is shown in Figure 5.3-1.

The ice condenser is essentially a well-insulated cold storage room in which ice is maintained in an array of vertical cylindrical columns. The columns are formed by perforated metal sheet baskets with the space between columns forming the flow channels for steam and air. The ice condenser is contained in the annulus formed by the containment vessel wall and the crane wall circumferentially over a 300° arc. The refueling canal and equipment hatch are located in the remaining 60° arc.

The total height of the ice condenser compartment extends from below the operating deck to the top of the crane wall. The uppermost section of

the ice condenser forms a plenum which accommodates the air cooling equipment and provides access for ice loading and maintenance. A small bridge crane is provided at the top of the compartment for construction and maintenance purposes. Below the operating deck, the inner wall of the ice condenser incorporates the inlet doors, through which the air from the lower containment volume and the discharged loop contents pass into the ice condenser. The top deck of the ice condenser is formed by doors supported from radial I-beams supported by the top of the crane wall. Intermediate deck doors form a partition between the upper plenum and the ice compartment.

The ice condenser is insulated at its external boundaries to maintain the total heat load on the cooling system to an acceptable level for the required equilibrium temperature of the ice bed, and to minimize the temperature gradients on the inner surfaces of the ice compartment. In the region of the walls of the ice compartment, the insulation incorporates cooling air ducts, by which the heat gained is absorbed and transferred to the coolers in the upper plenum. The temperature of the cooling air in the ducts and temperature of the ice is normally maintained between 10°F and 20°F. The floor of the ice compartment incorporates glycol cooling coils embedded in the concrete top wear slab to absorb and transfer the heat gained to the refrigeration coolers.

The heat absorbed by the coolers is transferred to the refrigeration units outside the reactor containment by an ethylene glycol circulation system.

An instrumentation system monitors the ice bed temperature and the open position of the inlet and personnel access doors.

Tool boxes containing tools used for maintenance of the ice condenser air handling units are located on the floor grating and seismically restrained at the 701' elevation outside of the ice condenser in each unit.

Ice Support Structure

The ice support structure consists of the lower support structure, lattice frames and columns, and ice baskets.

The floor of the ice condenser compartment is arranged to provide structural support for the internals and cooling of the ice bed. The ice condenser structure and ice is supported by the floor and lower support structure. The lower support structure supports the ice and baskets. The structure extends above the floor to provide an access area behind the inlet doors. The lower support structure is essentially a lattice of radial rectangular box beams on the centerline of the ice baskets. These radial box beams are supported by inner and outer main beam members running circumferentially around the ice condenser. The radial box beams are fabricated from structural steel channel sections and plate. The main beam members are supported by the columns from the ice condenser floor. The lower support columns straddle the ice condenser inlet doors providing a clear area for the inlet doors.

Lateral support against seismic effects is provided by structural ties to the crane wall. Clearance is maintained between the ice support structure and the primary containment vessel to ensure that there is no load transmitted between them due to seismic accelerations.

The lattice frames, of welded steel construction, locate the ice baskets in the desired array. The frames are supported from vertical columns at each corner, each column serving adjacent frames. The columns, located at the inner and outer walls of the ice condenser, are built from rectangular steel sections and are fastened to the lower support structure. The radial length of the frames is adjustable to accommodate construction clearances and permit vertical alignment of the basket locations within the frames. Cross braced box truss sections are built into the insulated duct panels at the ends and midsection. The vertical spacing of these truss sections is such that a reinforced section is provided at each lattice frame elevation. Outrigged brackets with insulating pads at the lattice frame elevations provide the structural connection between the inner duct panel and the inner lattice frame support column. As discussed before, the lattice frames are connected to the support columns.

The box truss sections of the wall panels are attached to the panel mounting angles with structural angle brackets. These brackets and the wall panel clamping bolts provide a vertically pinned connection to the crane wall that will transmit the lateral loading to the inner wall.

Groups of six lattice frames are connected together with a slip joint between groups to provide for circumferential thermal expansion and contraction. Since there is no connection between the structure and the containment wall, no special provisions are necessary for radial thermal expansion or contraction. No seismic or thermal loads from the ice support structure are restrained by the containment shell.

The ice columns are composed of part-length round basket sections, filled with pieces of ice, and formed to allow exposure to the steam.

Interconnection stiffening rings are located at each end of the basket section. Brace rings are located within the stiffening ring at the bottom of each basket to provide support for the ice in addition to the shear support provided by the baskets. The baskets are assembled into the lattice frames to form a continuous column of ice the full height of the ice bed. Only the bottom ends of the lowest basket sections are closed to prevent the ice falling through. Overall basket column length is 48 feet and is composed of baskets in 2 foot, 3 foot or 12 foot lengths. Basket sections can be assembled in any combination as long as a coupling ring or stiffening ring is located every 6 feet to coincide with the lattice frame location. The lattice frames provide only lateral ice basket support at intervals corresponding to the ends of the ice basket sections or at the midspan stiffener ring location of a 12 foot ice basket. The vertical support of the ice and ice baskets is transmitted by the basket to the lower support structure. The ice is loaded from the top of the ice bed into the completed basket assembly. The columns of ice can be lifted and removed in sections, and provision is made for lifting and weighing the whole length of selected columns for surveillance purposes.

Normal Loads

For normal load conditions the ice load is applied statically, and the only lateral load is that due to any misalignment of the basket columns and lattice frames. The lattice frame and column structure does not carry any vertical components of load from the ice baskets. Horizontal load components from misalignment of the basket columns and lattice frame locations are minimized by proper adjustment during installation.

The resultant basket structure considered for analysis comprises a pinjointed cylindrical shell, laterally supported at the joints and vertically at the base. The loading due to the ice in the vertical direction is assumed to be applied in uniform shear above any section, combined with a hydrostatic load below that section, thus accounting for the worst condition of ice support.

The lower support structure carries a total static load from the ice bed, uniformly distributed. In addition, the inner circumferential main beams take the weight of the insulated duct panels on the crane wall and the reactions at the feet of the inner columns supporting the lattice frames.

For design purposes the basket columns and structure are considered to be subjected to other modes of loading outside the scope of the Normal, Earthquake, and Accident conditions. These arise from handling during removal or weighing of the basket columns when it is necessary to consider dynamic loading of a full column of ice due to lifting by the crane. Consideration is also given to minimizing damage in the event that a basket loaded with ice is dropped or arrested while being lowered by the crane.

Earthquake Loads

In addition to the seismic effects on the ice condenser structure due to the weight of the ice, seismic effects will be transmitted to the structure through the crane wall and floor. The behavior is analyzed using a response spectrum, absolute accelerations, displacements and relative motions of the walls as determined from the dynamic analysis of the containment structure. The natural frequencies and vibration modes of the ice internals comprising the baskets, framework, and structural ties to the crane wall are determined by dynamic analysis, with the internals being supported transversely by the structural ties incorporated in the insulated duct panels at the crane wall.

The behavior of the ice structure is investigated using two mathematical models. The first of these models represents one basket or group of baskets supported laterally at the level of each frame. The support is assumed to be a linear spring representing the effect of the insulation. The second model represents a typical horizontal frame. This frame is continuous around 300 degrees of arc of the containment and bears on the crane wall.

These models give the lowest frequencies and mode shapes of the ice condenser internals. In the analysis the ice is initially assumed to have mass but zero stiffness. To represent the overall effect of the ice baskets, framework, and insulation, damping is taken as 5% critical damping for the operating basis earthquake and 10% for the design basis earthquake linear analysis. For a discussion on the nonlinear analysis performed on the ice condenser structures, refer to Appendix M.

Using the frequencies established from the models described above, the response of the internal structure is determined from the response spectra. The maximum accelerations are then applied to the structural components to determine the maximum stresses and deflections.

Experimental verification of the component design with respect to the structural ties to the crane wall has been determined by the construction of 8 feet long prototype panels for manufacturing evaluation purposes. One such prototype panel was subjected to static load tests, simulating the effect of seismic accelerations, to determine the corresponding deflections in the structure.

As mentioned before cross braced box truss sections are built into the insulated duct panels to transfer the lateral load to the crane wall.

Thus no reliance is placed on the insulation material as such to accommodate the seismic loads, and coupling between the crane wall and containment shell due to the effective stiffness of the insulation is wholly obviated by maintaining a positive clearance between the framework of the ice condenser, the insulated duct panels and the containment shell for all conditions.

Accident Loads

Accident loads are considered for the maximum postulated break size in the reactor coolant or steam system piping. This gives rise to an increase in pressure in the ice condenser together with drag forces on the baskets and support structure due to the velocity of the steam/air mixture. Consideration is given to temperature gradients between the top and bottom of the ice condenser which develop from heat transfer between the structural steelwork and steam condensed by the ice.

The maximum differential pressure between the ice condenser and the upper compartment is effectively applied across the insulated duct panels. The wall panels are designed and fastened to the walls in a manner that precludes significant steam channeling due to panel deflections, or leakage through the vapor barrier, for full accident design pressure differential.

The vapor barrier joints are mechanically fastened, sealed joints for which the fastening is designed to withstand the full pressure differential. The pressure differential is applied to the seated joints in a manner that increases the sealing pressure applied on the sealants.

Ice Condenser Cooling Ducts and Insulation

The insulated cooling ducts for the walls of the ice compartment, shown in Figure 5.3-1 and 5.3-3, are installed as prefabricated panels. At the outer wall of the condenser, the panels cover the full height of the ice compartment. At the inner wall and end walls, the lower end of the panels terminate above the inlet doorports. For ease of handling during construction the panels were made in half height sections, and the panel width is consistent with the pitch of the columns and lattice frame.

Each panel is an integral duct unit, consisting of two back to back cross braced box section for cooling air flow. Fiberglass insulation is provided on the outer side (next to the ice compartment walls). See Figure 5.3-3. Each wall duct panel contains two parallel ducts divided by an insulated wall and joined with a "U" type return section at the bottom end of the duct panel. The air flow channels are fed from the refrigeration system air handling units by an air header around the periphery of the upper plenum. The chilled air is distributed from the air header into the ice bed duct side of the wall panels. The chilled air descends to the bottom of the duct and flows up in the return duct of the wall panel and is discharged into the upper plenum. The layer of insulation provided on the outer face of the panels minimizes the heat gain to the cooling system. The supply of refrigerated air down through the inner face of the panels and the insulation provided between the inner and outer panels minimizes the temperature gradients in the ice compartment due to a differences in wall temperature in the ice condenser, thereby minimizing sublimation and mass transfer of ice from one region of the condenser to another. The inner, ice compartment side of the panels is galvanized steel sheet and the joints between panels are sealed to provide a vapor barrier between the ice compartment and the cooling air flow. The galvanized steel

sheet surfaces of the panels on the crane wall and end walls forms the vapor barrier for the compartment on those walls.

The panels are clamped to the walls by studs, clamp washers and nuts. The weight of the panels is transmitted to the floor at the outer wall of the ice condenser and to the ice lower support structure at the inner and end walls.

The fastening of the insulated duct panels to the walls, the construction incorporating sheet metal faces, and the additional constraints provided by the configuration of the ice baskets and support structure, eliminates any mechanism which would allow the insulation material to significantly impede the performance of the ice condenser during accident conditions.

The floor of the ice condenser is cooled by embedded pipe coils through which chilled glycol is circulated.

The wall panel design is such that the structure of panels resists compression of insulation with exception of slight momentary compression of the interior insulation layer between the back to back ducts. This slight compression is due to the deformation of the face and will return after the pressure is reduced.

An additional load is imposed on the refrigeration system due to the heat flow into the upper plenum. The walls of the plenum are also insulated by prefabricated fiberglass panels, but the air flow from the duct panel exhaust to the air handling units circulates in the plenum, picking up heat input through the insulation and top deck doors. This ensures that moisture leaking into the ice condenser plenum is picked up by the air and freezes on the cooler coils. Together with the vapor barrier on the inner face of the insulated duct panels, this minimizes the ingress of moisture into the ice bed.

Ice Condenser Doors

Inlet Doors

The inlet doors at the bottom of the ice condenser are suitably insulated panels mounted as vertically hinged pairs, on an angle section frame between the concrete pillars supporting the crane wall as shown in Figure 5.3-4.

The doors consist of a 1/2 inch composite panel with steel facings and a structural steel channel frame, and a 7 inch foam insulation backing that is enclosed with a stainless steel sheet metal cover.

The doors are provided with springs which produce a small force to resist door opening and to provide a positive closing force to bring the door back to its neutral position. The magnitude of the force produced by the springs when the doors are fully open is equivalent to a differential pressure of approximately 1 pound per square foot. The doors are normally held shut, against a seal mounted on the frame, by the static differential pressure due to the higher density air in the ice condenser compartment. With zero differential pressure across the doors (no cold air head), the neutral position of the spring is set so that the doors are slightly open (3/4" nominal). Thus, all doors will begin to open at the same pressure (cold air head) and, within the limits of spring tolerances, the doors will all open equal amounts.

Provision is made for a drain area of approximately 13 sq. ft. at the bottom of the ice condenser to permit water and air to flow from the ice condenser during and after the reactor coolant blowdown. This provision assures that, if doors reclose after a large break before all water drains out, or for small break or residual heat release cases where doors are not fully open, water from melted ice can drain from the compartment. The total drain area is provided by 21 individual floor drains.

Top Deck and Intermediate Doors

The intermediate doors enclosing the ice compartment and forming the floor of the upper plenum are supported by the lattice frame support columns. The door panels are comprised of a structural steel framing, and an insulation foam plastic core with bonded and mechanically fastened sheet metal facings. The doors are hinged horizontally and are normally closed. On an increase in pressure in the ice condenser compartment, these doors will open as required, allowing air to flow into the upper plenum.

The top deck doors are flexible metal encased, insulating bats. These bats are attached to the crane wall only. An increase in pressure below these flexible bats will cause them to flip open over the top of the crane wall. This will permit the air to flow out of the ice condenser into the upper compartment.

A small vent area approximately 20 sq. ft. is provided through each set of intermediate deck doors and top deck doors to equalize pressure between the ice condenser and containment volumes during normal operating pressure fluctuations and to permit small break LOCA steam/air flow into the ice bed.

Equipment doors with integral personnel access doors are provided at each end wall of the upper plenum. Personnel access is provided into the lower void space for surveillance of the inlet doors during reactor shutdown via a lower personnel access door in one of the end wall lower areas. These doors are all closed during normal operation.

For small leaks less than approximately 70 gpm to 240 gpm, a pressure difference sufficient to open the inlet doors would not be developed. This range of leakage would be quickly detected by reactor coolant system instrumentation, and plant shutdown would be initiated. For this case, the containment ventilation fan-coolers although not engineered safeguards, have capacity for removal of additional heat input. In any event, the containment pressure would be limited to a low value by the Containment Spray System.

For slightly greater leak rates, zero differential pressure would be developed across the doors (cold air head balanced), and the neutral position of the spring is set so that the doors would open slightly. Thus, all doors would begin to open at the same differential pressure (cold air head) and within the limits of spring tolerances, the doors would all open by equal amounts.

For larger break sizes, the doors will open by greater amount consistent with the spring characteristics of the doors. The one pound per square foot differential pressure required to fully open the lower doors would be developed by the steam flow from approximately an 8 inch diameter single-ended pipe break. Above this break size, maldistribution will be limited to a low amount by the size of the door ports.

At the conclusion of the reactor coolant system blowdown, the doors would tend to reclose. The provision made for a drain area at the bottom of the ice condenser allows for the flow of water and air from the ice condenser during and after the blowdown. If it is assumed that residual heat is released to the containment in the form of steam, the doors would reopen as required to allow the steam to flow into the condenser.

Refrigeration System

The refrigeration system is designed to initially cool down the ice condenser from the ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for the ice machines.

Cooling of the ice condenser is achieved by a three-stage system shown diagrammatically in Figure 5.3-2 and Figure 5.3-2A.

- First stage - Refrigerant cycle
- Second stage - Glycol cycle
- Third stage - Air cooling cycle.

First Stage - Refrigerant Cycle

Ten 25 ton capacity freon vapor compression refrigeration units (5 Twin unit skids) located outside the containment provide coolant for the two units of the plant.

Condenser cooling water is provided for the non-essential service water system.

Second Stage - Glycol Cycle

The second stage working fluid is a 50% solution of ethylene glycol and water. The cycle carries the heat absorbed from the ice condenser air handling units, floor cooling circuits or the ice machines to the evaporator/coolers of the refrigeration units. Provision is made for cross-connecting the second stage for each unit of the plant to any of the refrigeration units.

Six pumps have been provided to convey the glycol to the ten refrigeration units and air handling units and floor cooling circuits in each ice condenser. Two piping manifolds from the pump discharge conduct ethylene glycol into and out of any combination of these components. The glycol is fed to each reactor containment by single flow and return pipes and subsequently by branch feed and return lines to the air handling units along each side of the ice condenser plenum. The glycol floor cooling circuit is entirely within the containment and consists of two floor cooling pumps circulating chilled glycol through coils embedded in the ice compartment floor. The floor cooling pumps take suction from the air handler glycol return line before the line leaves the containment. Glycol from the floor cooling circuit is returned to the same line, downstream of the suction point, through a modulating valve for floor temperature control. With the above described arrangement, the systems can be refrigerated from the central source with a minimum of interaction and a high degree of redundancy.

A surge tank is provided in each containment system to accommodate volumetric expansion and contraction of the ethylene glycol solution and is equipped with level alarms.

Third Stage - Air Cooling Cycle

Thirty dual air handling units are located around the periphery of the ice condenser plenum, spaced to give an even distribution of air flow to the cooling air ducts in the ice compartment insulation panels.

Air is drawn from the upper plenum through the cooling coils by the fans and is fed to a continuous header around the plenum for distribution to the ducts. This arrangement assures that the cooling function can be maintained by adjacent units should a fan or cooler be shut down.

The ethylene glycol flow in the cooling coils is controlled by air outlet temperature, and the flow is bypassed during defrost. The coils are defrosted automatically by individual electric heaters, and flapper-type check valves prevent reverse air flow through the coil when the fans are shut off during defrost.

Handling Equipment

The bridge crane could move completely around the annulus outside the crane wall. This crane was used for construction, ice loading, weighing selected ice baskets, and general maintenance during plant construction and initial plant operation. However, since then the end wall equipment access doors have been permanently positioned in the up or closed position which prevents the crane from accessing into the ice condenser area.

The bridge crane track is supported from the beams of the plenum top deck structure which in turn are supported by radial beams supported by the crane wall.

Ice Machine

Three ice machines are installed in the auxiliary building. The machines are each capable of producing ten tons of borated ice per day, which is adequate for all recharging requirements. The ice is made in a shape and size convenient for handling, and provision is made for checking that ice loading and ice chemistry are maintained within the prescribed limits.

Instrumentation - Monitoring System

There are 96 temperature sensing elements which are distributed throughout the ice bed and are monitored and recorded in the Control Room. An annunciator panel provides an alarm for a pre-set deviation from the prescribed limits of ice bed equilibrium temperature.

Each inlet door panel operates two switches when in the closed position. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. An annunciator panel in the Control Room gives an alarm signal for the door open condition.

Similarly, ice condenser compartment equipment and personnel access doors are fitted with switches providing Control Room indication of the position of those doors. Also see Appendix M, Section 6.10.

Ice Condenser Materials

Corrosion of ice condenser components will be greatly reduced by the low temperature operation of the ice condenser. Corrosion at ice condenser operating temperature, even at saturation, is almost non-existent.

Ice bed structural steel member materials were impact tested to meet the temperature requirements of N1210 of Subsection B Section III Nuclear Vessel Code of ASME, of at least 30°F lower than the lowest service temperature for all section thicknesses in excess of 5/8 inch. For section thickness equal to or less than 5/8 inch material was either impact tested or specified to fine grain practice.

To further inhibit corrosion, galvanizing is used for baskets and metal panels. The Lower Support Structure is fabricated from Cortex Structural Steel, which forms its own protective oxide surface coating as the material weathers. The other major structural members inside the ice condenser are protected by corrosion resistant paints and are expected to last the life of the ice condenser without maintenance.

Any ice condenser equipment whose performance might be affected by corrosion employs corrosion resistant materials for critical components. Any corrosion that would develop over long term operation would not impair the performance of the ice condenser.

The ice condenser cooling system utilizes ethylene glycol as a coolant. According to published data(1) all ice condenser materials selected have good chemical resistance to ethylene glycol.

The insulation panels in the ice bed region are provided with a vapor barrier which would prevent moisture from reaching the containment vessel insulation interface region. A small proportion of the air flow bleeds from the ducts into the annulus between the duct panel and the containment. The inner surface temperature at the boundary of the ice condenser, in particular, the containment liner, varies with external ambient conditions from a maximum of 80°F down to 0°F. For all boundary temperatures above the duct air temperature, any moisture in the insulation diffuses to the cooling ducts and in addition is absorbed by the bleed flow of air. For containment liner temperature below duct air temperature, which corresponds to near zero external ambient conditions, any moisture transferred to the annulus between the containment liner and the duct panels forms as frost, but since the temperature gradients are reversed for this condition, the frost cannot detract from the performance of the cooling system.

With the above factors considered, removal of the ice condenser wall panels for containment liner inspection should not be necessary, as is the case for the analogous situation where the steel is encased in concrete. Access to the containment liner in the ice condenser region for containment inspection is provided by three special inspection ports through duct panels located approximately at the quarter points, in the lower section of the ice bed. Access to the containment liner in the plenum region is provided by removal of a plenum insulation panel.

5.3.3 ICE CONDENSER OPERATING CONSIDERATIONS

Refrigeration System

The ice condenser and associated systems provide a completely reliable, static heat sink which is instantaneously available if needed during a loss-of-coolant accident. The design assures that the quantity and configuration of the ice is maintained within the limits acceptable for the accident requirements. The insulation system minimizes the total heat gain into the compartment, and air leakage flow through the compartment, the double-walled insulation adjacent to the compartment further minimizes heat flows which would otherwise tend to promote ice sublimation and mass transfer within the ice bed. Long-term ice storage tests have shown that the ice can be stored for long periods of time without significant weight loss or physical distortion.

The ice condenser refrigeration system is provided with excess capacity to assure that the ice is maintained below the freezing temperature. The capacity of the refrigeration units exceed the maximum heat gain to the ice condenser compartment, and two standby refrigeration units are available for maintenance shutdowns, emergency shutdowns, and ice machine operation. The ethylene glycol loops for each unit of the twin-unit plant are separate, but can be interconnected. The many air handling units in each ice condenser also have excess capacity so that the ice bed compartment temperature, and temperature gradient across the ice bed or between adjacent ice bed bays/zones are minimized and easily maintained if fans are shut down for defrosting the coils, for maintenance of the equipment, or as a result of

equipment failure. The air distribution duct system feeding the insulated duct panels is continuous around the periphery of the upper plenum in the ice condenser compartment, so that all the panels are open to available air flow.

The double-walled insulation adjacent to the ice compartment provides additional protection against heat gain if the entire refrigeration system were to shut down. Because of the air gap and second insulation layer, the time required to raise the ice and internal structures to melting temperature would be about one week, allowing more than sufficient time to repair or restart the refrigeration equipment. An additional 2 weeks would elapse before even 10 per cent of the ice would melt if the refrigeration system were not operating.

Ice Bed Loading

Prior to the initial plant startup, the ice condenser compartment is cooled to operating temperature, 10-20°F, and loaded with ice. One or more ice making machines generates the ice in flake form for ease in handling. The ice is moved into the containment through a normally closed penetration by a pneumatic conveying system. The system feeds ice to a loading head, which is positioned by the bridge crane, through removable tubes in the plenum. This system is also used to replenish sections of the ice condenser, if required.

Ice Condenser Inspection

The design of the ice condenser and its components is such that a minimum amount of surveillance is required. However, in order to ensure satisfactory performance in the event of a loss-of-coolant accident, the following inspections and monitoring are required:

- a) Total weight of ice initially installed in the condenser is determined by weighing the loads of ice being installed.
- b) Flow passages and ice beds typical of all sections of the condenser are visually inspected to check that no significant changes are occurring.

- c) Selected full-height ice columns are weighed to check that no significant changes are occurring.
- d) The door panels are visually inspected for any evidence of frost formation. Further, these door panels are manually opened momentarily and the opening force measured to confirm that the doors are functioning properly.
- e) The chemical composition in the ice is checked by chemical analysis.
- f) Temperatures in selected locations in the ice condenser are monitored continuously to ensure that the cooling system is operating satisfactorily.
- g) A monitoring system is provided to indicate if the inlet doors are open.
- h) Prior to plant heatup, the hatches in the operating deck are inspected to ensure that they are properly secured.

The following paragraph is retained for historical reasons only:

After the ice condenser has been cooled and loaded with ice, sufficient time will elapse before the reactor plant is heated to temperature for initial power operation. During this period, a number of inspections are made of the ice condenser and its systems. After power generation begins, the surveillance program continues at the frequency specified by Technical Specification. Access to the upper plenum of the ice condenser compartment is possible while the reactor is at full power; therefore, inspections b and e outlined above, as well as f and g, are possible at any time.

Technical Specifications require that a representative sample of ice baskets be weighed periodically. These baskets are selected and weighed in order to ensure that (1) there is sufficient ice in the ice condenser and (2) the ice is uniformly distributed throughout the ice condenser.

The basket ice weights are statistically analyzed to demonstrate that the minimum average weight is greater than 1220 pounds of ice per basket at the 95% confidence level.

Ice Condenser Maintenance

As stated in the Design Bases, the ice condenser internals are designed for a lifetime consistent with that of the plant. Specifically, the design of the ice support structure and insulation duct panels is intended to minimize the potential for maintenance. The ice condenser and its components have been designed to be maintainable.

Sensitivity to Small Leaks

Consideration has been given to the possibility that a small leak from the reactor coolant system could melt ice without the leak or the ice melt being detected. Temperature detectors in the ice condenser and the inlet door position indicators would provide one indication of this condition. In order for steam to enter the ice condenser, the door opening differential pressure, produced by the cold air density in the ice condenser, must first be overcome. Because of the holes in the operating deck, the reactor coolant leakage has to be high enough to generate sufficient differential pressure across the deck before the inlet doors would begin to open.

Calculations of the leakage required to generate a 1 psf differential pressure through the assumed 5 sq. ft. deck leakage area indicate that the reactor coolant leakage would be between 70 gpm and 240 gpm, depending on the mixture concentration of steam and air passing through the deck. These calculations take no credit for heat removal by structures and by the containment ventilation system. This range of leakage would quickly be detected by reactor coolant system instrumentation, and plant shutdown would be initiated. Also, this range of leakage into the containment

TABLE 5.3-1

ICE CONDENSER DESIGN PARAMETERS

Reactor Containment Volume (net free volume)	
Upper Compartment, ft ³	745,896
Ice Condenser, ft ³	126,940
Lower Compartment (active), ft ³	306,800
Total Active Volume, ft ³	1,179,636
Lower Compartment (dead-ended), ft ³	61,702
Total Containment volume, ft ³	1,241,338
Reactor Containment Air Compression Ratio*	1.41
Reactor power, MWt (design basis)	3391
Design Energy Release to Containment	
Initial blowdown mass release, lb	549,000
Initial blowdown energy release, Btu	346.7 x 10 ⁶
Allowance for undefined energy release in addition to core residual heat, Btu	50 x 10 ⁶
Ice Condenser parameters	
Weight of ice in condenser, lb (Tech. Spec.)	2.37 x 10 ⁶

*Defined in Section 14.3.

TABLE 5.3-1 (cont'd.)

Dimensions of ice condenser

O.D., ft	115
I.D., ft.	89
Average length, ft	267
Width (less insulation panels), ft	11
Ice bed height, ft	48
Inlet door flow area, ft ²	1000
Ice condenser flow area, ft ²	1326
Ice Condenser inlet door opening pressure, lb/ft ²	1/2 to 1.0
Ice boron concentration, ppm boron	2000
Refrigeration cooling capacity (current as of 1/82)	
Installed cooling capacity for compartment, Tons	75
Maximum compartment heat input, Tons (per unit)	35
Total cooling capacity for plant, Tons	
(capacity shared by two units)	250

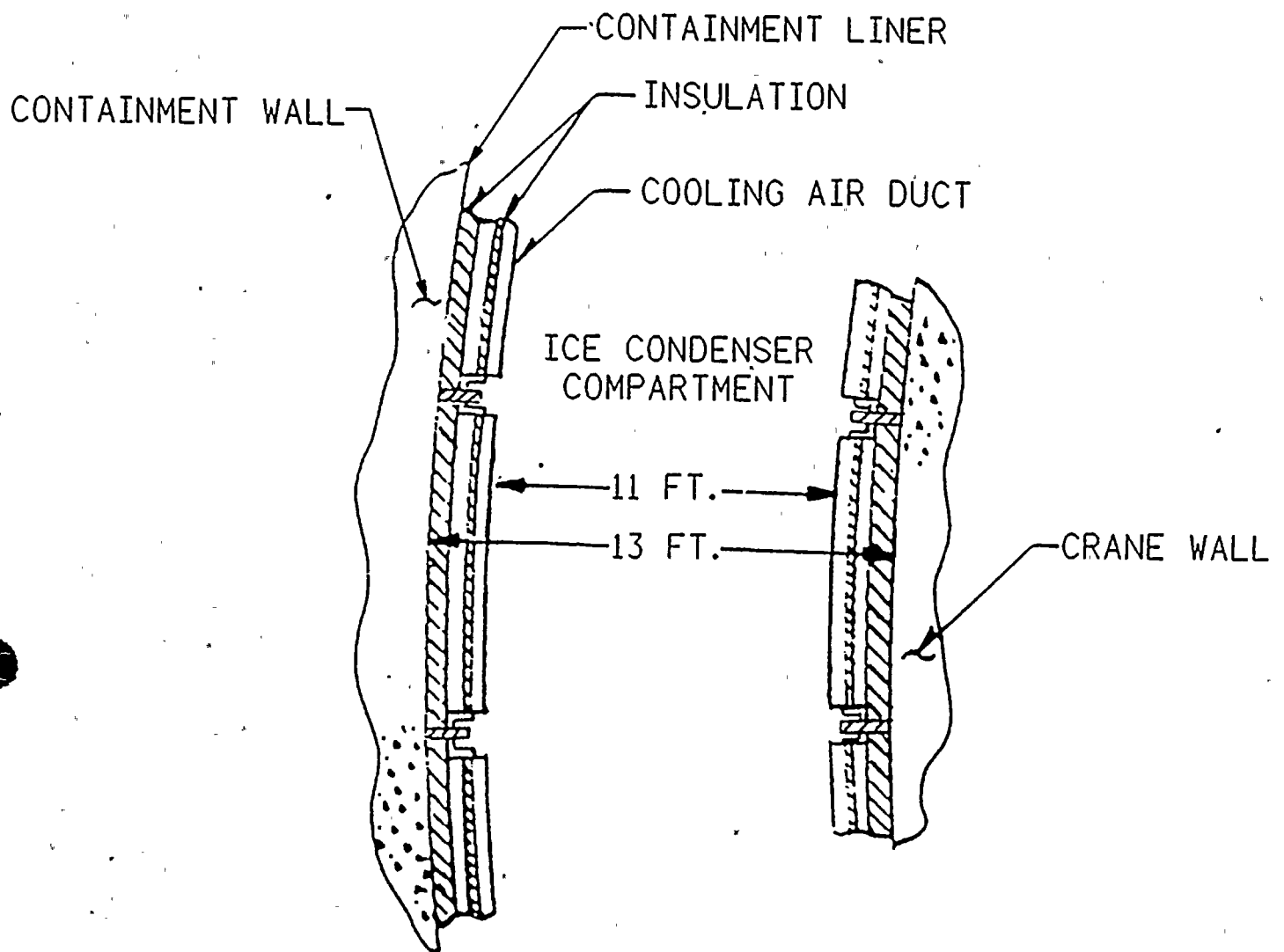


FIG. 5.3-3
ICE CONDENSER INSULATED DUCT PANELS
PLAN VIEW

JULY, 1997

Class G

The Fuel Transfer (CPN-1), Ice Loading (CPN-57) and Return (CPN-80), Containment Thimble Removal (CPN-76), and Containment Service (CPN-71) penetrations are special penetrations used only during outages. For Unit 2 only - CPN-67 also has a special penetration similar in design to the above which is considered as a spare and is not used during outages. During power operations, where containment integrity is required, these special penetrations are closed by a blind flange. These blind flanges are type B Local Leak Rate Tested and tested as part of the type A Integrated Leak Rate Test for verification of containment integrity. The inboard blind flange is considered to be the single containment pressure boundary for these penetrations. The outboard blind flange, where used, is primarily to facilitate type B testing of the inboard flange.

5.4.2 CONTAINMENT ISOLATION SYSTEM DESIGN

The general design basis covering the number and location of isolation valves required to assure reactor containment integrity is given in Section 5.4.1.

Check valves may be employed as one of the two barriers for incoming lines. Test connections and pressurizing means are provided to test each isolation valve or barrier for leak tightness. Either water or a gas is used as the pressurizing medium depending on the requirements of each case. Where it is necessary to make a quantitative leakage test, provision is made to:

- a) measure the inflow of the pressurizing medium, or
- b) collect and measure the leakage, or
- c) calculate the leakage from the rate of pressure drop.

The test connections are isolated when not in use by closed manual valves and/or caps and administratively controlled to ensure containment integrity.

All isolation valves are missile protected. Isolation valves, actuators, and control devices required inside the containment are located between the missile barrier and the containment wall. Isolation valves, actuators and control devices outside the containment are located outside the path of potential missiles or provided with missile protection. There are two levels of automatic containment isolation identified as Phase A and Phase B. Phase A isolation closes all lines penetrating the containment except essential lines such as Safety Injection and Containment Spray which are not isolated, and component cooling water to the reactor pumps and service water to the ventilation units which isolates on Phase B. (For Phase A and B initiating signals see Chapter 7 Instrumentation and Control.) All automatic isolation valves are able to be closed from the main control room. Position indicators are provided for each valve near its manual control switch in the main control room.

Specific administrative procedures govern the positioning of all isolation valves except check valves as well as any flanged closures during normal operation, shutdown and incident conditions. Check valves in incoming lines open only when the fluid pressure in the line coming from the outside is higher than the pressure on the containment side. Gravity or a spring holds the valve closed in the balanced pressure condition.

5.4.3 DESIGN EVALUATION

The containment isolation system provides two barriers to prevent leakage of radioactivity at each containment opening. Either barrier is sufficient to keep the leakage within limits.

5.4.4 TEST AND INSPECTION

All valve leak testing for Inservice Inspection (ISI) and Integrated Leak Rate Test (ILRT) program and surveillance requirements are performed in accordance with Appendix J, Option B, to 10 CFR 50 for Type A, B and C type testing. Also certain valves will be tested for operability in accordance with the applicable edition of the ASME OM Standards.

5.5.1 GENERAL DESCRIPTION

The Containment Ventilation System is designed to maintain temperatures in the various portions of the Containment within acceptable limits for operation of equipment, and for personnel access for inspection, maintenance and testing as required. It also has capability for purging the Containment atmosphere to the environment via the plant vent. The system can also cleanup airborne contamination in the containment prior to personnel entry. There is one plant vent for each unit. In the event of a loss-of-coolant accident, portions of the Containment Ventilation System aid in reducing Containment pressure after blowdown and also prevent the potential accumulation of any hydrogen in "pockets" within the Containment from reaching the flammable limit.

The Containment Ventilation System is shown in Figures 5.5-1 and 5.5-2. It consists of nine, essentially independent, sub-systems as follows:

- a. Containment Purge Supply and Exhaust System
- b. Instrumentation Room Purge Supply and Exhaust System
- c. Containment Pressure Relief System
- d. Upper Compartment Ventilation System
- e. Lower Compartment Ventilation System including Control Rod Drive Mechanism Ventilation System, Reactor Cavity Ventilation System and Pressurizers Compartment Ventilation System
- f. Containment Instrumentation Room Ventilation System

- g. Containment Auxiliary Charcoal Filter System
- h. Containment Air Recirculation/Hydrogen Skimmer System
- i. Hot Sleeve Ventilation System

Unit No. 1 and Unit No. 2 are each supplied with a separate system. These systems are essentially identical. All ventilation systems with the exception of the purge and pressure relief systems are of the recirculating type (d through i, above).

The containment and instrumentation room purge exhaust and containment pressure relief systems discharge to the unit vent where they are monitored before release.

5.5.2 DESIGN BASES

The Containment Ventilation System is designed to the following parameters:

- a. Purge the containment atmosphere to the plant vent. System capacity is sufficient to provide 1.5 changes of the containment air volume in one hour.
- b. Limit containment pressure to 0.3 psig (maximum) during normal plant operations.
- c. Maintain a maximum temperature of 100°F in the containment upper compartment during plant operation and a minimum of 60°F during plant shutdown to permit personnel access as required.
- d. Maintain a maximum temperature of 120°F in the lower compartment (135°F inside the primary concrete shield) during plant operation and a minimum of 60°F during an outage.

- e. Maintain a maximum temperature of 100°F and a minimum temperature of 60°F in the Containment Instrumentation Room.
- f. Purge the In-core Instrumentation Room atmosphere to the unit vent during periods of personnel access to this room.
- g. Ensure that a reliable supply of cooling air is provided to the Control Rod Drive Mechanisms.
- h. Reduce the concentration of airborne fission products (particulates, iodine and methyl iodine gases) which may be introduced into the containment atmosphere via leakage from the Reactor Coolant System (concurrent with 1 percent fuel cladding defects).
- i. Aid in reduction of Containment pressure in the event of an accident. (See Chapter 14.)
- j. Ensure that, in the case of a loss-of-coolant accident, any hydrogen that may be formed will not accumulate in pockets in excess of 4 percent (by volume).
- k. Maintain concrete temperature below 150°F at the crane wall sleeves serving the RHR system when that system is operating.

5.5.3 SYSTEM DESCRIPTION

Containment Purge Supply and Exhaust System

One Containment Purge Supply and Exhaust System is supplied for each Containment structure so that, prior to entry, if required, radioactivity can be reduced to safe levels.

Purge air is supplied to the containment through two 16,000 CFM fans and their associated filters and heating coils. Purged air is exhausted through two 16,000 CFM capacity fans and high efficiency particulate air filters to the unit vent where it is monitored before release to the atmosphere. The purge-air supply and exhaust fans and filters are located in the Auxiliary Building.

There are four air penetrations of the Containment associated with this system, a supply and an exhaust penetration into both the upper and lower compartment. Each penetration has two fail-closed isolation valves. (These valves are normally closed when the purge systems are not in operation.)

The Containment Purge Supply and Exhaust System has a total capacity of 32,000 CFM which affords approximately 1.5 air changes per hour.

The Containment Purge Supply and Exhaust System takes outside air through intake vents and passes it through medium-efficiency particulate filters (NBS Dust Spot Efficiency for atmospheric dust of 50%) and steam coils when necessary prior to discharge into the containment. The upper compartment purge exhaust plenum draws 11,000 CFM of air through inlets along the periphery of the refueling canal. The lower compartment purge exhaust plenum draws 21,000 CFM of air through inlets along the periphery of the reactor well cavity.

The Containment Purge Supply and Exhaust System serves to provide: 1) a means of reducing the radiation level in the containment to a safe value for containment entry, 2) a continuous airflow through the containment during refueling operations, 3) heated air to the containment necessary for comfort of personnel working in the containment, and 4) a backup means of pressure relief, in the event that the containment pressure relief system is out of service.

The Containment Purge Supply and Exhaust System is not normally operated. If, prior to containment entry, the containment radiation monitors indicate radiation levels in the containment area in excess of the appropriate Federal regulations for radiation exposure to an individual worker (per 10 CFR 20), and if it is determined that the radiation level within the containment is at a safe level for purging, then the Containment Purge Supply and Exhaust System is activated to reduce the radiation level within the containment to a safe value for containment entry.

In the unlikely event that radiation levels in the containment are too high for purging, the Containment Auxiliary Charcoal Filter System may be operated until radiation levels are low enough for purging. When the containment radiation level has been reduced to an acceptable point for purging, the Containment Purge Supply and Exhaust System isolation valves will be opened and the purge system will be actuated.

The Containment Purge Supply and Exhaust System fans are operated remotely from the Control Room. The isolation valves close automatically upon a safety injection signal or a high containment radiation level.

During purge operations, the rate of purge can be controlled by the operator who has the option of operating any desired combination of the Containment Purge Supply and/or Exhaust System fans or by repositioning as necessary volume dampers (the volume dampers are located in the Auxiliary Building). Operation in this manner will also provide a means of vacuum relief in the event of a negative containment pressure. Because containment pressures can be controlled entirely by operation of the Containment Purge Supply and/or Exhaust System during purging operations, there will be no need to use the Containment Pressure Relief System during Containment Purge.

Purge operation is permitted in all operating modes. For Modes 1 through 4, the Cook Nuclear Plant purge estimate goal is two hundred and forty (240) hours each year for each unit. This purge estimate is based on a plant capacity factor of 95%, and accounts for two purge operations per week. Each purge operation is assumed to be approximately 2 1/2 hours in duration. The annual 240-hour purge operation time limit amounts to less than 3% of the estimated plant operation time in Modes 1 through 4. In Modes 1 through 4, purge operation is limited to one supply and one exhaust flow path.

Reasons to operate the system include the need to improve containment working conditions, e.g., reduce airborne activity, to perform surveillance and/or maintenance on a safety-related system or piece of equipment, or to relieve containment pressure if the containment pressure relief system is out of service. The purge/vent system is not intended to be used to routinely control containment atmosphere temperature and humidity. It is intended that purging and venting times will be as short as possible. Allowing purge operations in Modes 1, 2, 3, and 4 is more beneficial than a cooldown to Mode 5 from the standpoint of (a) imposing unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems. The containment purge system is designed in accordance with the requirements of NRC Branch Technical Specification CSB 6-4, Rev. 1. This includes, but is not limited to, an analysis of the impact of purging on ECCS performance, an evaluation of the radiological consequences of a design basis accident while purging, and limiting purge operation to using no more than one supply path and one exhaust path at a time. The purge isolation valves have been demonstrated capable of closing against the dynamic forces associated with a loss-of-coolant accident and are assured of receiving a containment ventilation isolation signal. Reset switches have been protected against inadvertent use in a manner which facilitates the administrative controls governing their use. The purge and vent isolation valves do not use resilient seating/sealing material and are not subject to the type of environmental degradation common to resilient materials.

Instrumentation Room Purge Supply and Exhaust System

The Containment Instrumentation Room is isolated from the general Containment atmosphere and has a separate and independent purge system consisting of a 1000 CFM supply unit and a 1000 CFM exhaust unit.

The supply unit draws outdoor air through an intake louver, passes it through a medium-efficiency particulate filter and electric blast coil heaters and discharges it into the Containment Instrumentation Room. The exhaust unit draws air from the Containment Instrumentation Room, passes it through both high efficiency particulate air (HEPA) and charcoal filters and discharges it to the unit vent where it is monitored before release. This operation affords approximately 3-1/2 air changes per hour for the Containment Instrumentation Room.

Both the Containment Instrumentation Room purge supply and purge exhaust penetrations have two isolation valves similar in type and function to those provided for the Containment Purge Supply and Exhaust System.

Containment Pressure Relief System

Containment pressure relief is provided by a 1000 CFM exhaust unit composed of a fan, a HEPA filter and a charcoal filter. This system is located in the Auxiliary Building. There is a single penetration of the containment barrier for this system with two isolation valves similar in type and function to those provided for the Containment Purge Supply and Exhaust System.

A flow diagram of the Containment Pressure Relief System is shown in Figure 5.5-2. The system fan draws containment atmosphere through a register in the upper compartment where, prior to discharge to the plant vent, it is passed through a filter unit containing both HEPA and charcoal filters. Additional features of the system design include two isolation valves, an automatically operated flow regulating damper

which limits flow through the filters to 1000 cfm, a backdraft damper in the duct to the unit vent to prevent backflow from the unit vent into the containment, and a bypass path around the fan so that containment pressure relief can be provided in the event the pressure relief unit fan fails to start.

The system can be operated manually from the Control Room any time that containment pressure exceeds ambient. However, if the containment pressure should reach 0.2 psig, an alarm will sound in the control room to alert the operator to actuate the system.

The operator action required to actuate the system consists of opening the normally closed isolation valves and starting the fan motor. Such operator action will limit the containment internal pressure to less than 0.3 psig for normal atmospheric fluctuations.

Whenever operation of the Containment Pressure Relief System occurs, the containment atmosphere will always be exhausted through the charcoal and HEPA filters in the unit. This should be sufficient to prevent any adverse radioactivity from being exhausted to the environment. The Containment Pressure Relief System isolation valves automatically close on upon receipt of a containment ventilation isolation signal. This will prevent any further release of adverse radioactivity to the environment.

The containment pressure relief system is intended for use only for normal operation when it is necessary to reduce internal containment pressure. It is not intended for use when the Containment Purge Supply and Exhaust Systems are operating, since the Containment Purge Supply and Exhaust fans themselves provide the necessary means of controlling internal containment pressure. The Containment Purge Supply and Exhaust Systems provide a backup means to relieve containment pressure, in the event that the containment pressure relief is out of service. It exhausts through HEPA filters to the plant vent.

Containment Instrumentation Room Ventilation System

The in-core instrumentation room is an isolated sector of the lower compartment. The temperatures in the room are controlled by two free-standing, 9,600 cfm recirculation ventilation units (1 standby). Each unit is composed of a fan, water cooling coil and electric blast coil heaters. The water for coils is supplied by the non-essential service water system. Water flow is regulated in the same manner as for the upper compartment ventilation units. Maximum water flow per unit is 50 gpm. (Flow rate may be exceeded during ventilation unit flushing operations.) The instrumentation room is kept at a constant temperature of approximately 90°F during plant operation.

Containment Auxiliary Charcoal Filter System

This system consists of two 8000 cfm fan-filter units located in the lower containment compartment. Each unit contains both high efficiency particulate air and charcoal filters, for reduction of fission product particulate activity which may be airborne in the lower compartment.

The containment atmosphere is monitored for radioactivity during reactor power operation, and the number of auxiliary charcoal filter units in operation (none, 1, or 2) depends on the airborne activity levels observed.

Containment Air Recirculation/Hydrogen Skimmer System

The containment air recirculation/hydrogen skimmer system is the only safety related ventilation system within the containment. This system functions only in the event of a hi-hi containment pressure signal. It consists of two redundant independent systems which include fans, back draft dampers, valves, piping and ductwork.

Both containment air recirculation hydrogen skimmer system fans are located in the upper volume. The fans discharge, via the annular space

between the crane wall and the Containment liner, into the lower compartment. The fans are provided with back draft dampers on the discharge to prevent backflow during initial blowdown.

Figure 5.5-2 shows the various components of this system and Figure 5.5-3 shows the recirculation flow patterns that are created by this system. The system includes provisions for providing both 1) general recirculation of containment atmosphere between the upper and lower compartments following a loss-of-coolant accident, and 2) preventing the improbable accumulation of hydrogen in restricted areas within the containment following a loss-of-coolant accident.

The potential areas of hydrogen pocketing are the top of the containment dome, and the lower compartment enclosures which include the three rooms in the annular space between the crane wall and the liner, the steam generator enclosures, and the pressurizer enclosure. Hydrogen pocketing is prevented by continuously drawing air out of the top of each of the above areas at such a rate as to limit the potential local hydrogen concentration to less than 4% by volume.

Each of the two independent systems fan has its own intake system composed of three separate headers. These headers draw 39,000 CFM from the upper compartment in the immediate vicinity of the fan, draw 1,000 CFM from the upper compartment at the top of the dome, and draw air from the potential hydrogen pockets in the lower compartment (this is the hydrogen skimmer header). Each header has volume control dampers in the line or at the air intake to balance flow. The hydrogen skimmer header is composed of two pipe branches, one which draws 500 CFM from the top of each double steam generator enclosure and pressurizer enclosure and one which draws 100 CFM from each of three rooms in the annular space. There is a normally closed, motor-operated hydrogen skimmer valve on each main hydrogen skimmer header to prevent ice condenser bypass during initial blowdown.

D
requiring the operator to actuate the Containment Pressure Relief System when the internal containment pressure reaches 0.2 psig assures that internal containment pressure will never reach 0.3 psig during normal plant operations.

The automatic air-operated damper in the Containment Pressure Relief System provides a means of maintaining a constant air flow through the charcoal and HEPA filters in the unit. Regulation of the flow in this manner will optimize the iodine absorption capability of the impregnated activated charcoal by limiting the face velocity through the charcoal filters, thus providing a minimum residence time of airflow of 0.25 seconds in each of the six 2-inch deep charcoal beds in this unit.

The HEPA and charcoal filters in the Containment Pressure Relief System have an exceedingly high capability for removal of both airborne particulate matter and airborne radioactive iodine. The Containment Pressure Relief System has more than adequate capacity for retention of both particulates and iodine for the intended use of the system. The impregnated activated charcoal has a minimum absorption capability of 2.5 mg. of iodine for every gram of charcoal (total charcoal in this unit is a minimum of 37,100 grams). The single 24" x 24" x 12" HEPA filter is capable of holding at least 4 pounds of NBS Cottrell Precipitate Standardized Test Dust at a pressure drop of no more than 2.0 inches w.g.

5.5.5 INCIDENT CONTROL

In the event of an incident the two independent Containment Air Recirculation/Hydrogen Skimmer System fans automatically start within 9 ± 1 minutes after initiation of 2/4 hi-hi containment pressure signals. The operation of either fan ensures the reduction of the containment pressure to the limits described in Chapter 14.

At the same time the Air Recirculation/Hydrogen Skimmer fans start, the hydrogen skimmer valves in the two Containment Air Recirculation/Hydrogen Skimmer headers open, thus causing the Air Recirculation/Hydrogen Skimmer System fans to continuously purge all potential hydrogen pockets in the Containment.

All other Containment Ventilation Systems are not designed for operation during a loss of coolant accident.

The occurrence of a High Containment Radiation Signal from the upper compartment area or lower containment particulate/radiogas monitors will automatically trip the purge fans and close all ventilation system isolation control valves, thus isolating the Containment.

5.5.6 MALFUNCTION ANALYSIS

Sufficient redundancy exists in all recirculation ventilation systems to ensure a normal operation with one active component out of service.

The two filter cleanup units provide redundancy for small leakage rates. The Containment Purge Supply and Exhaust System is fitted with dual supply and exhaust fans. Simultaneous failure of a supply and an exhaust fan would result in an 80-minute purge rate.

The Containment Air Recirculation/Hydrogen Skimmer (CEQ) Systems are two redundant systems that are cooled from a common Component Cooling Water (CCW) header. The loss of either CEQ system or any component of either CEQ system will not impair system operation. In the event that flow to the CCW header is lost, procedural guidance is in place to ensure that it is expediently restored.

The Containment Purge Supply and Exhaust System is available for relieving containment pressure in the event that the containment pressure relief system is out of service.

All systems are inspected, tested and balanced upon installation. Charcoal and particulate filters are individually tested before shipment, upon installation and periodically thereafter as required. Replacement filters will be tested in the same manner.

The Containment Air Recirculation/Hydrogen Skimmer fans were tested during installation and are tested periodically to ensure proper functioning. The initial test of these fans were conducted at both no flow and full flow, verifying the fan capability to deliver the required amount of air. The periodic fan tests are conducted at no flow to assure that the fan is still operable.

5.7.2 INITIAL CONTAINMENT (PRE-OPERATIONAL) LEAKAGE RATE TESTS

Integrated Leakage Rate Tests

After completion of the containment and after loading the ice condenser, an integrated leakage rate test was carried out using a test procedure which was written using the American National Standard - ANSI N45.4-1972 and 10 CFR 50, Appendix J as guidelines.

The integrated leakage rate tests were conducted with the weld channel zones open to the containment atmosphere. The containment was pressurized to 12 psig, the containment design pressure, using air dried to a dew point below the coldest temperature in the ice condenser to eliminate the possibility of condensing water vapor during the test.

The design leakage rate under accident conditions is 0.25% of the containment free volume per 24 hours.

Sensitive Leakage Rate Tests

The sensitive leakage rate tests are performed using testing procedures written for testing liner weld channels and penetrations using 10 CFR 50 Appendix J as a guide.

Since the volumes contained in the weld channels and penetrations are significantly smaller than the containment free volume, the test sensitivity is correspondingly greater than that of an integrated leakage rate test. These tests are conducted with 12 psig in the weld channels and penetrations and with the containment at atmospheric pressure.

5.7.3 CONTAINMENT PERIODIC (POST-OPERATIONAL) LEAKAGE RATE TEST

There is a small combined volume of enclosed space in the double barrier penetration, the penetration weld seam channels and the liner weld channels installed on the inside of the liner in the containment. Since it is easy to monitor these small volumes, a sensitive and accurate means of periodically monitoring their status with respect to leakage is provided.

With this provision, there is no need to perform integrated leak rate tests of the containment vessel unless major maintenance or modifications of the containment are made. To allow for this possibility, it is permissible to pressurize the containment vessel to the design pressure.

Observations of the vessel will be made from platforms or by other means with special attention given to areas of major discontinuities.

Provisions have been made in the design of the Ice Condenser structure to permit periodic inspection of the containment liner in the area behind the ice condenser. Inspection of the liner is accomplished through "Inspection Ports" located around the ice condenser, to permit access to the liner.

Periodic leak testing of the containment is performed in accordance with the Technical Specifications and 10 CFR 50 Option B Appendix J Type A, B, and C leak tests. The leak rate test is done to determine the leak tightness of the containment vessel and containment isolation valves and not to measure the structural response of the containment. The leak rate test is performed with the ice in place and at the design pressure of 12 psig.

Since maximum $NPSH_r$ and minimum $NPSH_a$ occur at the runout flow for the pumps, this flow was assumed for calculation purposes. The minimum temperature of the refueling water storage tank is 70°F for both units.

The maximum sump temperature considered was 160°F (190°F for Unit 2) during recirculation for purposes of calculating $NPSH_a$. This exceeds the maximum expected sump temperature. Friction losses were calculated using the conservative pipe and fitting resistances given in the Crane Co. Technical Paper Number 410.

The containment spray and RHR pumps take suction during the recirculation phase from the containment sump. The water is at a higher temperature than during the injection phase, but the elevated containment pressure following the DBA somewhat offsets the higher vapor pressure of the water. However, no credit is taken for this elevated containment pressure. In addition, the piping to the pump suctions is direct, hence friction losses are small.

Engineered Safety Features Components Capability

Criterion: Engineered Safety Features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident (LOCA) to the extent of causing undue risk to the health and safety of the public.

The majority of the active components of the Emergency Core Cooling System and the Containment Spray System whose failure would affect the health and safety of the public are located outside the containment and not subject to containment accident conditions. Instrumentation, motors, cables, and penetrations located inside the containment which are required to function are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to with

stand without failure, the effects of radiation, temperature, pressure, and humidity expected during the required operational period for individual specific accident conditions.

Accident Aggravation Prevention

Criterion: Protection against any action of the Engineered Safety Features which would accentuate significantly the adverse after-effects of a LOCA shall be provided.

The reactor is maintained subcritical following a LOCA. Introduction of borated cooling water into the core results in a net negative reactivity addition. The RCCAs insert and remain inserted, although credit for this is not taken in the large break analysis (See Subchapter 14.3).

The supply of water by the Emergency Core Cooling System to cool the core cladding does not produce significant water-metal reaction (See Subchapter 14.3). The delivery of cold emergency core cooling water to the reactor vessel following a LOCA does not cause further loss of integrity of the reactor coolant system pressure boundary. Accumulator actuation, including possible nitrogen addition is evaluated in Chapter 14 and is shown not to aggravate any loss-of-coolant accident (LOCA).

Instrumentation, motors, cables and penetrations located inside the containment which are required to function are selected to meet the most adverse accident conditions to which they may be subjected (Chapter 5 and 7). These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the effects of radiation, temperature, pressure, and humidity expected during the required operational period for individual specific accident conditions.

Protection, in the form of restraints, supports and physical separation has been provided for the ECCS to assure no loss of core cooling capability.

Power sources are arranged to permit individual actuation of each active component of the Emergency Core Cooling System.

Testing of Emergency Core Cooling System

Criterion: Capability shall be provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical.

An integrated system test can be performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the Reactor Coolant System but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The accumulator tank pressure and level are continuously monitored during plant operation and discharge flowpath availability can be checked at anytime by noting the outlet isolation valve position indication on the main control board.

The accumulators and the safety injection pipe up to the final isolation valve are maintained full of borated water at refueling water concentration while the plant is in operation. The accumulators and injection lines will be refilled with borated water as required by using the safety injection.

Small fill and drain lines are provided for this purpose.

Flows in each of the centrifugal charging and safety injection pump discharge headers and in the main flow lines for the residual heat removal pumps are monitored by flow indicators. Pressure instrumentation is also provided for the main flow paths of the safety injection pump and centrifugal charging pump headers and residual heat removal pumps. Level and pressure instrumentation are provided for each accumulator tank.

Testing of Operational Sequence of Emergency Core Cooling System

Criterion: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System into action, including the transfer to alternate power sources.

The design provides for capability to test initially, to the extent practical the full operational sequence up to the design conditions for the Emergency Core Cooling System to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.2.5, Test and Inspections.

Codes and Classifications

Tables 6.2-1 tabulates the codes and standards to which the emergency core cooling system components are designed.

Service Life Under Accident Conditions

Portions of the system located within the containment are designed to operate under the most adverse accident conditions without benefit of maintenance and without loss of functional performance for the duration of time the component is required following the accident.

6.2.2 SYSTEM DESIGN AND OPERATION

System Description

The Emergency Core Cooling System is shown in Figures 6.2-1 and 6.2-1A, and 9.2-1. These figures illustrate the redundancy of components and piping systems.

The operation of the Emergency Core Cooling System, following a loss of coolant accident, can be divided into two distinct phases: 1) the injection phase in which any reactivity increase attending the accident

is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system is replenished, and 2) the recirculation phase in which long term core cooling is provided during the accident recovery period. A discussion of each phase is given below. Accidents analyzed in Chapter 14 assume a pump head degradation from vendor curves of 10% for centrifugal charging pumps and 15% for the safety injection and residual heat removal pumps.

Injection Phase

The major equipment involved in the injection phase are:

- a. Two centrifugal charging pumps (a third, positive displacement, charging pump is not involved in the injection system).
- b. Two safety injection pumps
- c. Two residual heat removal pumps
- d. Four accumulators (one for each loop)
- e. Refueling water storage tank (RWST)

The relative importance of the various pieces of injection equipment is dependent upon the size and location of the primary system break. For a large break, the accumulators represent the principle injection mechanism in the sense that they are the first piece of equipment to be effective. For further details see Chapter 14, and Figures 6.2-2 and 6.2.3.

The accumulators, utilizing a compressed nitrogen cover gas, inject borated water into the cold legs of the reactor coolant piping when the primary system pressure falls below nominal 600 psig. One accumulator is provided for each cold leg of the Reactor Coolant System. They are located inside the containment but outside the missile barrier, and are therefore protected against credible missiles. Accumulator water level can be adjusted remotely during normal power operation. Borated makeup water from the refueling water storage tank is added using a safety injection pump. Water level is reduced by draining to the reactor coolant

drain tank. Samples of the solution in the accumulator tanks are taken in the sampling station for periodic checks of boron concentration. Provisions are also included for remote nitrogen makeup. The accumulators are passive components of the injection system because they require no external source of power or signal in order to function. The remainder of the major pieces of equipment comprising the emergency core cooling system are active components which are actuated by any of the Safety Injection Signals:

- a) Low steam line pressure in 2 of 4 steam lines. (Possible steam line break).
- b) High differential pressure between any two steam generators (Possible steam line break).
- c) Low pressurizer pressure (Possible LOCA).
- d) High containment pressure (Possible LOCA or steam line break).
- e) Manual actuation (the Control Panel includes a switch for each train).

The safety injection signal initiates a reactor trip (this may have already occurred), starts the diesel generators, opens the boron injection tank isolation valves and the charging pump refueling water storage tank suction valves, and starts the centrifugal charging pumps, the safety injection pumps, and the residual heat removal pumps. In addition, isolation valves on the volume control tank discharge, charging line, and centrifugal charging pump minimum flow lines close. Finally, a safety injection signal will produce a phase A containment isolation signal which results in the closure of the majority of the automatic containment isolation valves, isolating all non-essential process lines. (See Sub-Chapter 5.4)

The containment recirculation sump is protected at entry by coarse and fine screens supported within a substantial frame. Water flowing into the sump passes through the coarse and fine screens and downwards under the crane wall. The flow is then turned upwards and enters the twin recirculation pipes connecting the sump to the RHR and containment spray pumps. The two sets of grating act as flow straighteners and mitigate vortex formation by equalizing local velocity differences.

The adjacent containment sump is also equipped with both coarse and fine screens at its entrance. This sump and the recirculation sump are connected via an 8" pipe at the bottom of the sumps.

The sump is designed with a large flow area, allowing low water velocities, such that build-up of debris against the screens is minimized. The low velocities make it unlikely that air bubbles could be carried into the pump suction area of the sump. Each recirculation line from the sump is run outside the containment to a sump isolation valve. This valve is surrounded with a leak tight steel enclosure and the section of piping joining it to the sump is run within a guard pipe welded to the valve enclosure. Any leakage from the sump piping or valve body will be contained and cannot leak into the atmosphere or cause a loss of recirculation fluid. The pressure relief for each valve enclosure is routed to the associated residual heat removal pump room sump. The relief valve set point is 35 psig which is also the design pressure for the valve enclosure. The drain lines from the enclosures to the RHR pump room sumps are normally closed. The enclosures are ASME Section III Class B vessels which require pressure relief provision.

The sump isolation valves are interlocked with the RHR pump suction supply valves from the RWST so that the supply line(s) from the sump cannot be opened until the RHR pump suction valve(s) is (are) fully closed. These interlocks are train oriented and will prevent air from getting into the RHR pump suction. Any excessive leakage or passive failure downstream of the sump valves can be controlled and isolated by closure of the sump valve in the affected train.

Within the containment, continuity of the liner is assured by welding of the

sump discharge piping to the liner plate and fitting of a weld test channel over the seal weld. The liner extends under the sump area to ensure containment integrity (see Chapter 5).

Change-Over from Injection Phase to Recirculation Phase

The general sequence, from the time of the safety injection signal, for the changeover from the injection to the recirculation phase is as follows:

- a) First, sufficient water is delivered to the containment to provide adequate net positive suction head (NPSH) for the residual heat removal pumps. This is the Refueling Water Storage Tank (RWST) low level setpoint and corresponds to a volume in the RWST of 131,980 gallons (tank height 7 feet above lo-lo level setpoint).
- b) Second, the low level alarm on the RWST sounds. At this point, the operator initiates transfer to recirculation. If all ECCS pumps are operable, the operator aligns the west RHR pump and west containment spray pump to take suction from the containment sump. Both sets of high head pumps (centrifugal charging pumps and safety injection pumps), the east RHR pump and the east containment spray pump continue to take suction from the Refueling Water Storage Tank.
- c) Third, the north and south safety injection pumps and the east and west centrifugal charging pumps are aligned for cold leg recirculation. They achieve this by taking suction from the west RHR pump after their realignment is complete.
- d) Finally, the operator completes the switch-over operation by transferring the suction of the east containment spray and RHR pumps from the RWST to the containment sump. The operator does not begin this until the RWST has reached its lo-lo level setpoint (usable water volume in the RWST of 37,250 gallons). When reaching this setpoint, the operator terminates all suction flow from the RWST to the ECCS and containment spray systems.

The emergency operating procedures provide detailed sequence for the changeover from injection to recirculation.

The operator in the control room implements the changeover from injection to recirculation via a series of manual switching operations. An automatic pump trip will occur once the refueling water storage tank (RWST) reaches lo-lo level. This protects the residual heat removal pumps aligned to the RWST from cavitation. The power supply for each pump trip is from an independent power source. The pump trip and associated circuitry are designed to be consistent with the remainder of the plant engineered safety features. Should there be a trip on lo-lo RWST level, the pump can be restarted by operator action once the RWST suction has been isolated and the recirculation sump suction opened. This automatic trip feature is a back-up to the manual switchover.

Following an accident the shortest time when the operator must take action to perform the necessary switchover results when both trains of ECCS and spray pumps are in operation at full runout conditions. This situation empties the RWST at the fastest possible rate, thus requiring the most rapid operator action to perform the switchover from injection to recirculation.

Earlier studies (Reference 1) have shown the minimum time for the operator to complete the switchover of the west containment spray pump; the west RHR pumps; and the north and south SI pumps is about four minutes (232 seconds). Subsequent to these studies, the transfer process was modified to include the transfer of the centrifugal charging pumps prior to transferring the east RHR pump and east containment spray pump. The time to complete the transfer to cold leg recirculation of the west RHR pump; the west containment spray pump; the north and south SI pumps; and the east and west centrifugal charging

pumps has increased until it can no longer be fully assured that the operator is able to complete the transfer of one entire train¹ of ECCS pumps that is assumed in the FSAR².

The valve stroke times for switchover to cold leg recirculation which are used in the Chapter 14 safety analysis are described in Unit 1, Section 14.1 of this UFSAR. Related information regarding the large break loss of coolant accident evaluation for a 3 minute SI interruption may be found in Section 14.3.1.1.2 (Unit 2) and just preceding the "Core and System Performance" section of 14.3.1 (Unit 1).

Steam Break Protection

Following a steam line break, the reactor control system, in response to the apparent load, would tend to increase reactor power. For larger breaks, a reactor trip would occur. Continued secondary steam blowdown cools the reactor coolant causing a positive reactivity insertion. Analyses described in Chapter 14 indicate that breaks large enough to produce a reactivity insertion sufficient to cause a return to criticality also produce

¹The FSAR analysis assumed the availability of pumped water from one RHR pump, one SI pump, and one centrifugal charging pump. This is referred to as a train of pumps. When transitioning to cold leg recirculation both the centrifugal charging pumps align to a common suction, making it impossible for realignment of only one pump; i.e., both the east and west centrifugal charging pumps must be realigned at the same time. The same situation exists for the SI pumps.

²Following a large break loss of coolant accident, the operator is always able to transfer one containment spray pump and one RHR pump. The operator is not always able to demonstrate the capability to complete the transfer of at least one SI pump and one centrifugal charging pump.

sufficient depressurization and shrinkage of the primary coolant to initiate safety injection. The high pressure delivery of boric acid solution by the centrifugal charging pumps from the RWST then reestablishes adequate shutdown margin even for the case where the most reactive control rod is stuck in the fully withdrawn position.

Components

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is

isolated from the Reactor Coolant System by two check valves in series.

Should the Reactor Coolant System pressure fall below the accumulator pressure, the check valves open and borated water is forced into the Reactor Coolant System. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold legs.

The accumulators are passive engineered safety features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the Reactor Coolant System.

The design capacity of the accumulators is based on the assumption that the contents of one of the accumulators spills onto the containment floor through the ruptured loop, and the contents of the remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and a portion of the core.

The accumulators are carbon steel, clad with stainless steel and designed to ASME B&PV Code Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided. The accumulator design parameters are given in Table 6.2-2.

The margin between the minimum operating pressure and design pressure provides a band of acceptable operating conditions within which the accumulator system meets its design core cooling objectives. The band is sufficiently wide to permit the operator to minimize the frequency of adjustments in the amount of contained gas or liquid to compensate for leakage. See Table 6.2-8.

Boron Injection Tank

The boron injection tank, constructed of carbon steel clad with stainless steel, is located in the auxiliary building and contains approximately 2% (by weight) boric acid solution. The tank design parameters are given in Table 6.2-3. The originally supplied tank heaters, pipe heat tracing and recirculation lines have been disconnected. These support systems to the tank are no longer required as a high concentration of boric acid solution is no longer maintained in the boron injection tank.

Refueling Water Storage Tank

The Cook Nuclear Plant is equipped with two (2) refueling water storage tanks, one for each unit.

The function of the refueling water storage tank is:

1. To provide sufficient volume of borated water to fill the refueling cavity for refueling operations.
2. To provide sufficient volume of borated water for emergency (post-accident) operations. This includes the ability to maintain the core subcritical during the long term cooling phase of a LOCA, even in the unlikely event that the control rods do not drop into the core.
3. To ensure that the ECCS pumps are provided with adequate NPSH.

The tank is maintained with a minimum of 350,000 gallons of borated water above the bottom of the discharge pipe. This ensures that an adequate amount of water will be delivered to the sump before the operators begin switching from the injection mode to the sump recirculation mode of operation. The switchover is initiated upon receipt of a low level alarm which indicates that the tank level has drained down to a level which is (nominal) 9 feet - 9 inches above the bottom of the discharge pipe.

A high level alarm is provided to alert the operator of potential overflow conditions. A minimum level alarm is provided to assure that 350,000 gallons of usable water are in the RWST.

The Unit No. 1 refueling water storage tank is heated by means of two 100% capacity heat tracing circuits with separate thermostatic controls. The tank is insulated with 2 inch thick fiberglass insulation. A temperature sensor attached to the outside of the tank will actuate a low temperature alarm in the control room in the event that the tank temperature falls below the design basis temperature requirement. The setpoint of the alarm is typically set approximately 5°F above the minimum temperature.

The Unit No. 2 refueling water storage tank is heated by means of a 15 gpm pump which recirculates tank water through two electric heaters. The RWST heating pump operates continuously, when required, with the heaters energizing automatically on a low RWST temperature signal. The system is seismic category I with respect to protection of the tank boundary and is designed to maintain RWST temperature at design basis conditions. The Unit 2 RWST is insulated with 2 inch thick fiberglass insulation, and has a temperature sensor and alarm similar to that of the Unit 1 tank.

Each tank is equipped with an 8 inch vent and a 10-inch overflow line and a 3-inch return line. The overflow lines terminate in the pipe tunnel. Should the 8-inch vent become plugged the 10 inch overflow line would maintain sufficient venting area to prevent any adverse effect on the safety function of the tank. The 3-inch return line is routed internally in the tank to enhance mixing of the tank contents.

Missile protection is not provided for the RWST since in the event of tornado or turbine-missile damage to it, the unit can be safely shut-down without the RWST and can be maintained shut-down.

Pumps

Design parameters for the emergency core cooling system pumps are included in Table 6.2-5.

The two centrifugal charging pumps are horizontal, electric motor driven multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the volume control tank or the pump suction manifold. This bypass is automatically isolated upon initiation of safety injection. The minimum flow, motor-operated valve reopens if the reactor coolant system pressure increases above 2000 psig to protect the pumps from deadheading.

The two safety injection pumps are horizontal, electric, motor-driven, multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event that the reactor coolant system pressure is above the shutoff head of the pumps.

The two residual heat removal pumps are vertical, electric, motor-driven, single-stage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or of equivalent corrosion resistant material. Pump minimum flow bypass connection is located downstream of the residual heat exchanger and the bypass flow returns to the pump suction.

The pressure containing parts of the pumps are stainless steel castings conforming to ASTM A-351 Grade CF8 or CF8M, stainless steel castings procured per ASTM A-743 Grade CA-15 or A-487 Grade CA6NM or carbon steel forgings to ASTM A-266 Class 1 and ASTM A-181 Grade 1 clad with austenitic steel or ASME SA-182 Grade F304. Parts fabricated of stainless plate are constructed to ASTM A-240 Type 304 or 316. The bolting material conforms to ASTM A-193 or ASTM A-453 Grade 660.

Materials such as weld-deposited Stellite or Colomony may be used at points of close running clearances in the pumps to prevent galling and to ensure long term performance ability in high velocity areas subject to erosion. In other cases wear points are of ASTM A-420 Grade stainless steel, heat treated to give the required anti-galling properties.

Pressure containing parts of the pumps were chemically and physically analyzed and the results are checked to assure conformance with the applicable ASTM specification. In addition, pressure containing parts of the pump are liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid penetrant test is the ASTM Pump & Valve Code.

Pump design was reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that they are adequate for the specified service.

Where welding of pressure containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. This procedure includes evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement also applied to any repair welding performed on pressure containing parts.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for thirty minutes.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shut-off head and three additional points to verify performance characteristics. Where NPSH was critical, this value was established at design flow by means of adjusting suction pressure.

Heat Exchangers

The two residual heat exchangers of the Residual Heat Removal System cool the water from the recirculation sump. These heat exchangers are sized for the cooldown of the Reactor Coolant System. Table 9.3-2 gives the design parameters of the heat exchangers.

The residual heat exchangers are designed to the ASME Boiler and Pressure Vessel Code, Sections III & VIII and conform to the requirements of TEMA (Tubular Exchanger Manufacturers Association) for Class R heat exchangers.

Additional design and inspection provisions include: confined-type gaskets, general construction and mounting brackets suitable for the plant seismic design requirements, tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Boiler and Pressure Vessel Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and all hot or cold formed parts, a hydrostatic test duration of not less than thirty minutes, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The residual heat exchangers are conventional vertical shell and U-tube type units (tube sheet down). The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has a SA-515 GR70 carbon steel shell, SA-213 TP-304 stainless steel tubes, SA-240 Type 304 stainless steel channel, SA-240 Type 304 stainless steel channel cover and a tube sheet of forged steel SA-105 GR.II with 1/4 inch minimum TP-304 weld overlay.

When the Reactor Coolant System is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is about 100 psi differential across the valve. This test confirms the seating of the disc and provides a quantitative leakage rate measurement which can be compared with the results of earlier tests. When this test is completed, the discharge line test valves are opened and the Reactor Coolant System pressure increase continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept some leakage back from the Reactor Coolant System without compromising their availability. Table 6.2-8 indicates the frequency that the accumulator level would have to be readjusted as a function of leakage rate. Tables 6.2-6 and 6.2-7 summarize the single failure analyses of recirculation phase. Table 6.2-9 summarizes the estimated leakage during recirculation.*

Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are considered in Chapter 14.

6.2.4 SAFETY LIMITS AND CONDITIONS

Limiting Conditions for Operation

The limiting conditions for operation are detailed in the Technical Specifications. These conditions apply to both active components and tanks of the Emergency Core Cooling System.

* See footnote, Table 6.2-9

Limiting Conditions for Maintenance

The Technical Specifications also establish limiting conditions governing the maintenance of Emergency Core Cooling System components during plant operation. Maintenance on a component is permitted providing the redundant component is operable and capable of being powered from an emergency power source.

The design philosophy with respect to active components in the safety injection and residual heat removal systems is to provide duplicate equipment so that maintenance is possible during operation without impairment of the safety function of the systems.

6.2.5 TESTS AND INSPECTIONS

All active and passive components of the Emergency Core Cooling System are inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pump seals, valve packing, and flanged joints during system testing.

In addition, to the extent practical, the critical parts of the injection nozzles, pipes, valves and safety injection pumps are inspected for erosion, corrosion, and vibration wear evidence.

Components Testing

Pre-operational performance tests of the components were performed in the manufacturer's shop. An initial system flow test demonstrates proper functioning of the system. Thereafter, tests are performed in accordance with the provisions of ASME B&PV Code Section XI and, beginning with the 3rd 10 year interval ISI program, pump and valve tests are in accordance with ASME O&M Standards and NUREG-1482 to demonstrate that the components are functioning properly.

System Testing

Testing is conducted during plant shutdown to demonstrate proper automatic operation of the emergency core cooling system. A test signal is applied to initiate automatic action and verification made that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

The periodic testing of pumps in the emergency core cooling and containment spray systems requires a flow of water from the refueling water storage tank. Demonstration of proper operation of these pumps will also demonstrate the operability of the line from the refueling water storage tank. Testing procedures are employed to assure that the motor operated isolation valves function normally.

The accumulator pressure and level are continuously monitored during plant operation.

The accumulators, and their injection piping up to the accumulator isolation valve are maintained full of borated water while the plant is in operation. The boron concentration is checked periodically by sampling. The accumulators and injection lines are refilled with borated water as required by using the safety injection pumps. A small test line is provided for this purpose in each injection header.

The motor-operated valves in the recirculation suction lines from the containment recirculation sump to the RHR pumps are normally closed. At or between each major refueling shutdown, the 14" locked open hand valve to the residual heat removal pump suction will be closed, the line drained and the sump valve exercised.

Flow in each of the main safety injection lines and in the main flow line for the residual heat removal pumps is monitored by flow indicators on the main control board. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps.

Operational Sequence Testing

After hot functional testing and prior to initial fuel loading, the Emergency Core Cooling System plus a portion of the Containment Spray System were operationally tested. These tests include individual pump full flow tests, accumulator operation and complete system operational flow tests, with the reactor head removed. Water was supplied from the refueling water storage tank.

Separate full flow tests were performed for a minimum of one hour to assure that all safety injection, residual heat removal and containment spray pumps are capable of sustained operation. The containment spray pump discharge flow was piped directly to the containment recirculation sump via temporary piping. Water was returned to the refueling water storage tanks by the residual heat removal pumps.

The accumulators were tested by charging the tanks to 100 psig and normal water level with the isolation valves closed. With the reactor head removed, the isolation valves were opened and proper performance verified.

A complete operational flow test was performed including the simultaneous full flow operation of all safety injection pumps, containment spray pumps, residual heat removal pumps and charging pumps. The purpose of this test was to demonstrate the proper functioning of the instrumentation and actuation circuits and to evaluate the dynamics of placing the system in operation.

To initiate the test, the Emergency Core Cooling block switch was moved to the unblock position thereby allowing the automatic actuation of the Emergency Core Cooling System relays from the pressurizer low pressure signals. A simulated high containment pressure signal initiated operation of the Containment Spray System. Special test instrumentation and data obtained provided information to confirm valve operating times, pump motor starting times, and delivery rates of injection water to the reactor coolant system.

REFERENCES

1. Appendix Q, Amendment 78 to Unit 2 FSAR, Question 212.36, October, 1978.

TABLE 6.2-1

SAFETY INJECTION SYSTEM CODE REQUIREMENTS**

<u>Component</u>	<u>Code</u>
Refueling Water Storage Tank	Not applicable
Residual Heat Exchanger	
Tube Side	ASME B&PV Code Section III Class C
Shell Side	ASME B&PV Code Section VIII
Accumulators	ASME B&PV Code Section III Class C
Valves	ANSI B16.5, MSS-SP-66, and ASME B&PV Code Section III, 1968 Edition*
Piping	USAS B31.1, 1967 Edition*
Boron Injection Tank	ASME B&PV Code Section III Class C

* Repairs and replacements are conducted in accordance with ASME Section XI

**Subsequent procurement of equipment as replacement is being done in accordance with codes and specifications equivalent to the original codes and specifications, updated as appropriate.

TABLE 6.2-2

ACCUMULATOR DESIGN PARAMETERS

Number	4 per unit
Type	Stainless steel clad/ carbon steel
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	120
Normal pressure, psig	621.5
Minimum pressure, psig	585.0
Total volume, ft ³	1350
Maximum water volume at operating conditions, ft ³	971
Minimum water volume at operating conditions, ft ³	921
Boron concentration (as boric acid), ppm	2400 to 2600
Code	ASME B&PV Code Section III Class C

TABLE 6.2-3

BORON INJECTION TANK DESIGN PARAMETERS

Number	1 per unit
Total Volume, gal (also useable volume)	900
Boron concentration, wt % (approximately)	2
Design pressure, psig	2735
Design temperature, °F	300
Operating pressure, psig (Injection Mode)	2340
Operating pressure, psig (Standby)	atmospheric
Operating temperature, °F	ambient
Material	SS Clad Carbon Steel
Code	ASME B&PV Code Section III Class C

TABLE 6.2-4

REFUELING WATER STORAGE TANK DESIGN PARAMETERS

Number	1 per unit
Tank Capacity, gal.	420,000
Required Capacity, gal.	350,000
Design pressure, psig	Static head and sloshing
Design temperature, °F	-30 to 100
Normal pressure, psig	Atmospheric
Liquid temperature, °F	70-100
Inside diameter, ft (approx.)	48
Straight side height, ft	31
Material	Stainless Steel

6.3 CONTAINMENT SPRAY SYSTEMS

6.3.1 DESIGN BASES

Containment Heat Removal Systems

Criterion: Where active heat removal systems are needed under accident conditions to prevent exceeding containment pressure, at least two systems, each with full capacity, shall be provided.

Adequate heat removal capability for the Ice Condenser Containment is provided by two separate Containment Spray Systems and two (redundant) portions of the Residual Heat Removal System. The sequential modes of operation are given in Section 6.3.2.

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident to prevent containment pressure from exceeding the design value. The design of the Containment Spray System is based on the conservative assumption that the core residual heat is released to the containment as steam. The heat removal capability of each Containment Spray System is sized to remove the reactor residual heat during cool down from operation at 3391 MWt after a loss-of-coolant accident. The residual heat (during ice melt) plus an undefined energy margin of 50×10^6 BTU is absorbed by the operation of the Containment Spray System and the Ice Condenser, respectively. The sizing of the Containment Spray Systems also provides for absorption of steam leaking through the operating deck at the maximum long term deck differential pressure (1/2 to 1 lb per square foot, the pressure required to open the Ice Condenser doors). Refer to Chapter 14.3 for Containment Integrity Analysis including Containment Spray System Modelling.

The secondary purpose of the Containment Spray System is the removal of fission products (radioactive iodine isotopes) from the containment atmosphere. The Containment Spray System is designed to deliver

sufficient sodium hydroxide solution which, when mixed with water from the Refueling Water Storage Tank which contains approximately 1.5% by weight boric acid (2000 ppm Boron), reactor coolant system water and the melted ice, gives a final spray water pH of approximately 9.3 after the spray additive (NaOH) tank is emptied. The performance of the Containment Spray System for iodine removal with a single Containment Spray Pump operating adequately fulfills the requirement of 10 CFR 100 as described in Chapter 14.

Inspection of Containment Pressure-Reducing Systems

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

Where practicable, active and passive components of the Containment Spray Systems are inspected periodically to demonstrate system readiness. The pressure containing components are inspected to detect leaks from pump seals, valve packing, flanged joints and safety valves. During operational testing of the Containment Spray Pumps, the portions of the system containing pump pressure are inspected to detect leaks. Design provisions for inspection of portions of the Emergency Core Cooling System which functions as part of the Containment Spray System are described in Section 6.2.5.

Testing of Containment Spray Systems

Criterion: A capability shall be provided, to the extent practical, to periodically test the delivery capability of the Containment Spray Systems as close to the spray nozzles as possible.

The Containment Spray Pump test lines are provided to verify flow from the RWST through the pump discharge. Water is pumped through the Containment Spray pumps and returned to the Refueling Water Storage Tank from a point upstream of the two parallel motor operated discharge valves via the

Spray Pump Test Line which includes a flow meter. The motor-operated valves in the RHR spray lines downstream of the RHR heat exchangers remain closed during testing of that portion of the RHR system which is a part of the spray system. Testing of this flow path is accomplished by a recirculation flow around the Residual Heat Removal Heat Exchanger.

Test connections are provided downstream of the block valves for checking (with air) for unobstructed flow through the spray nozzles.

Testing of Containment Pressure-Reducing Systems Components

Criterion: The containment pressure-reducing systems shall be designed, to the extent practical, so that active components can be tested periodically for operability and required functional performance.

Consideration was given in the system design for provisions to permit periodic testing of active components. Periodic tests are performed to verify proper component functioning in accordance with the requirements of the applicable edition of the ASME OM Standards. Testing of those components of Emergency Core Cooling System which are used for containment spray purposes is described in Section 6.2.5.

Testing of Operational Sequence of Containment Pressure-Reducing Systems

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

The design of the Containment Spray System provides, to the fullest practical extent, the capability to perform an initial test of the full operational sequence to demonstrate the state of readiness of those

sections of the system which do not function during normal plant operation. This testing included a full-flow test through special test connections which, for test purposes, replaced the check valves before the nozzles. Transfer to emergency power source was also demonstrated during this test. Air flow tests through each of the nozzles was used for verification of unobstructed flow. The transfer to emergency power source test and the air flow test through the nozzles are performed periodically.

6.3.2 SYSTEM DESIGN

System Description

Adequate containment pressure reduction and iodine removal are provided by the Containment Spray Systems whose components operate in sequential modes as follows:

- a) 'A' mode. Spraying a portion of the contents of the Refueling Water Storage Tank into the containment atmosphere using the Containment Spray Pumps. During this mode, the contents of the spray additive tank are mixed into the spray system to provide adequate iodine removal.
- b) 'B' mode. Recirculation of water from the containment sump by the Containment Spray Pumps through Containment Spray Heat Exchangers and back to the containment after the Refueling Water Storage Tank has been isolated, but while there is still ice in the Ice Condenser. This spray reduces the containment atmosphere temperature and prolongs the effective life of the ice.
- c) During the entire 'A' mode and continuing into the 'B' mode, NaOH is metered into the spray solution by an eductor system, using the Containment Spray Pump discharge for motive water.

- a. Low pressurizer pressure signal in 2/3 channels. May be manually blocked below P-11 and is automatically unblocked above P-11.
- b. High containment pressure in 2/3 channels.
- c. Steam line pressure in one steam line (2/3 channels) low in comparison to the other three steam lines (high steam line pressure differential).
- d. Steam line pressure low in two out of four steam lines.
- e. Manual actuation from one panel mounted switch per train.

These trips are listed in Table 7.2-1.

Turbine Generator Trip

A turbine trip is sensed by two out of three signals from low emergency trip fluid pressure. A redundant 4/4 stop valve closed signal will also indicate a turbine trip condition. A turbine trip causes a direct reactor trip above P-7 and results in a controlled short term release of steam to the condenser which removes sensible heat from the reactor coolant system and thereby avoids steam generator safety valve actuation.

The turbine control system automatically trips the turbine generator under any of the following conditions:

1. Mechanical overspeed trip

2. Backup overspeed trip (electrical on Unit 1, mechanical on Unit 2)
3. Low condenser vacuum
4. Thrust bearing failure
5. Reactor trip
6. Excessive shaft vibration
7. Moisture separator drain system level high
8. Loss of stator cooling (low flow, low pressure, or high temp.)
9. Safety injection
10. High-high water level in steam generator (1/4 loops)
11. Low bearing oil pressure
12. Manual operation of any of several trip levers
13. Loss of both feed pump turbines
14. Low shaft driven oil pump pressure (Unit 1 only)
15. EHC trip system pressure low (Unit 1 only)
16. EHC loss of speed feedback (Unit 1 only)
17. Low EHC pressure (Unit 1 only)
18. Initiation of AMSAC (ATWS Mitigation System Actuation Circuitry):
less than 25% flow to 3/4 loops and above 40% reactor power
19. Unit or overall differential
20. High exhaust hood temperature at no load (Unit 1 only)

Low Feedwater Flow Trip

This trip protects the reactor from a sudden loss of its heat sink. The trip is actuated by a steam/feedwater flow mismatch (1/2) in coincidence with low water level (1/2) in any steam generator.

Low-Low Steam Generator Water Level Trip

The purpose of this trip is to prevent a loss of the reactor's heat sink in the case of a sustained steam/feedwater flow mismatch of sufficient magnitude to cause a low feedwater flow reactor trip. The trip is actuated on two out of the three (2/3) low-low water level signals in any steam generator.

TABLE 7.2-2 (cont'd.)

<u>Designation</u>	<u>Derivation</u>	<u>Function</u>
P-12	$2/4 T_{avg}$ channels below setpoint	Permits manual block of safety injection on low steam line pressure. Permits or causes steam line isolation on high steam line flow. Blocks condenser steam dump.
P-12 Reset	$3/4 T_{avg}$ above setpoint	Prevents or defeats manual block of safety injection on low steam line pressure. Prevents or defeats steam line isolation on high steam flow. Permits condenser steam dump.

TABLE 7.2-2 (cont'd.)

<u>Designation</u>	<u>Derivation</u>	<u>Function</u>
P-13	1/2 turbine first stage pressure channel above setpoint	Inputs to P-7.
P-13 Reset	2/2 turbine first stage pressure below setpoint	Inputs to P-7.
P-14	2/3 hi-hi steam generator level (any steam generator)	Permits the initiation of: <ul style="list-style-type: none"> -Feedwater isolation -Main feedwater pump trip. -Main turbine trip.
P-14 Reset	2/3 hi-hi steam generator level (any steam generator)	Prevents or defeats initiation of: <ul style="list-style-type: none"> -Feedwater isolation. -Main feedwater pump trip. -Min turbine trip.
C-1	1/2 Intermediate range neutron flux above setpoint	Blocks automatic and manual control rod withdrawal. <p>May be manually blocked above P-10.</p> <p>Automatically unblocked below P-10.</p>
C-2	1/4 Power range neutron flux above setpoint	Blocks automatic and manual control rod withdrawal.
C-3	2/4 Overtemperature Delta T above setpoint	Blocks automatic and manual control rod withdrawal. <p>Actuates turbine runback via load reference.</p>

TABLE 7.2-5 (cont'd.)

Sheet	Function	
28	Volume Control Tank Level Control	Control
29	Boric Acid Blend Control	Control
30	Rod Insertion Limit	Control
31	ΔT /Auctioneered ΔT Deviation Alarms	Control
32	T AVG./Auctioneered T AVG. Deviation Alarms	Control
33	Steam Generator Level Control	Control
34	Steam Generator Level (Wide Range)	Control
35	ΔT & ΔT - S.P. Recording	Control

TABLE 7.2-6
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>
1. Power Range, Neutron Flux (High and Low setpoint)	Less than or equal to 0.5 seconds*
2. Overtemperature delta T	Less than or equal to 6.0 seconds*
3. Overpower delta T	Less than or equal to 6.0 seconds*
4. Pressurizer Pressure - low	Less than or equal to 2.0 seconds
5. Pressurizer Pressure - High	Less than or equal to 2.0 seconds
6. Pressurizer Water Level - High	Less than or equal to 2.0 seconds
7. Loss of Flow - Single Loop (Above P-8)	Less than or equal to 1.0 seconds
8. Loss of Flow - Two Loops (Above P-7 and below P-8)	Less than or equal to 1.0 seconds
9. Steam Generator Water Level - Low-Low	Less than or equal to 2.0 seconds
10. Undervoltage - Reactor Coolant Pumps	Less than or equal to 1.5 seconds
11. Underfrequency - Reactor Coolant Pumps	Less than or equal to 0.6 seconds

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in the channel.

TABLE 7.2-7
ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATING SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
1. <u>Containment Pressure - High</u>	
a. Safety Injection (ECCS)	Less than or equal to 27.0 ^{1,4} /27.0 ^{2,3}
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Essential Service Water System	Less than or equal to 13.0 ¹ /48.0 ²
2. <u>Pressurizer Pressure - Low</u>	
a. Safety Injection (ECCS)	Less than or equal to 27.0 ^{1,4} /27.0 ^{2,3}
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Essential Service Water	Less than or equal to 13.0 ¹ /48.0 ²
3. <u>Steam Flow in Two Steam Lines - High Coincident with Steam Line Pressure Low</u>	
a. Safety Injection (ECCS)	Less than or equal to 27.0 ^{1,4} /37.0 ^{2,4}
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Steam Line Isolation	Less than or equal to 11.0
4. <u>Containment Pressure - High-High</u>	
a. Containment Spray	Less than or equal to 45.0
b. Steam Line Isolation	Less than or equal to 10.0
c. Containment Air Recirculation Fan	Less than or equal to 600.0
5. <u>Steam Generator Water Level - High - High</u>	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0

TABLE 7.2-7 (cont'd)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATING SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
6. <u>Steam Generator Water Level - Low - Low</u>	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
7. <u>4160 volt Emergency Bus Loss of Voltage</u>	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
8. <u>Loss of Main Feedwater Pumps</u>	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
9. <u>Reactor Coolant Bus Undervoltage</u>	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

Notes:

DEFINITION: The ENGINEERED SAFETY FEATURE RESPONSE TIME (ESF) shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required value, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

1. Diesel generator starting and sequence loading delays NOT included. Offsite power available.
2. Diesel generator starting and sequence loading delays included.
3. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.
4. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

7.3 CONTROL SYSTEMS

7.3.1 DESIGN BASIS

The reactor automatic control system is designed to reduce nuclear plant transients for design load perturbations, such that reactor trips will not occur because of them.

Overall reactivity control is achieved by the combined use of chemical shim and Rod Cluster Control Assemblies (RCCA). Long-term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short-power changes is accomplished by moving RCCA's.

The function of the reactor control system is to provide automatic control of the RCC Assemblies during power operation of the reactor. The system uses input signals including neutron flux, coolant temperature, and turbine load. The Chemical and Volume Control System (Chapter 9) supplements the reactor control system by the addition and removal of varying amounts of boric acid solution.

When the reactor is critical, the best indication of the reactivity status of the core is the position of the control rod groups in relation to power and average coolant temperature. There is a direct relationship between control rod position and power and it is this relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor which is described in Sub-Chapter 7.2.

Any unexpected change in the position of the control group under automatic control, or a change in coolant temperature under manual control provides a direct and immediate indication of a change in

the reactivity status of the core. In addition, periodic samples are taken to determine the coolant boron concentration whose variation during core life provides a further check on the reactivity status of the reactor including core depletion.

The reactor control system is designed to enable the reactor to follow load changes automatically when the output is above approximately 15 percent of nominal power. Control rod positioning may be performed automatically, when plant output is above this value, and manually at any time.

The operator is able to select any single bank of rods for manual operation. This is accomplished with a multiposition switch so that he may not select more than one bank. He may also select automatic reactor control, in which case the control banks can be moved only in their normal sequence with program overlap. As one bank reaches 128 steps, the next bank begins to withdraw.

The system enables the nuclear unit to accept a step load increase of 10 percent and a ramp increase of 5 percent per minute within the load range of 15 percent to 100 percent without reactor trip subject to possible xenon limitations. Similar step and ramp load reductions are possible within the range of 100 percent to 15 percent of nominal power except for between 40% to 25% where AMSAC limits the decrease to approximately 2%/min.

The control system is capable of restoring coolant average temperature to within the programmed temperature deadband, following a scheduled or unexpected change in load.

The pressurizer water level is programmed as a function of auctioneered coolant average temperature. This minimizes the demands on the chemical and volume control and waste disposal systems resulting from coolant density changes during loading and unloading.

heaters, which are used to control small pressure variations due to heat losses, including those due to a small continuous spray in the pressurizer, and backup heaters which are turned on when the pressurizer pressure controller signal is below a given value.

A spray nozzle is located in the upper portion of the pressurizer cavity. Spray is initiated when the pressure controller signal is above a given set point, and spray rate increases proportionally with increasing pressure. Steam is condensed by the spray which will return the pressurizer pressure to its Program Value. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and to help maintain uniform water chemistry and temperature in the pressurizer.

Three pressurizer power relief valves limit system pressure for large load reduction transients.

Three spring-loaded safety valves limit system pressure should a complete loss of load occur without direct reactor trip or steam dump actuation.

Pressurizer Level Control

The water inventory in the Reactor Coolant System is maintained by the Chemical and Volume Control System. During normal plant operation, the pressurizer level is controlled by the charging-flow controller which controls the charging flow control valve or the positive displacement charging-pump speed to produce the flow demanded by the pressurizer-level controller.* The pressurizer water level is programmed as a function of coolant average temperature. The pressurizer water level decreases when load is reduced. This is the result of coolant contraction following programmed coolant temperature reduction from full power to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant

* The positive displacement charging pumps are not currently used for plant operations.

temperature changes. To permit manual control of pressurizer level during startup and shutdown operations, the charging flow can be manually regulated from the control room.

Secondary System Control

The secondary system includes the steam from the steam generators and the condensate and feedwater systems.

Steam Dump

The steam dump system is designed to relieve steam from the steam generators to the condenser thus reducing the sensible heat in the primary system in the event of net load reduction not exceeding 50 percent.

The steam dump design capacity is 40 percent of full load steam flow at full load steam pressures. All steam dump steam flows to the main condensers via the steam lines.

When a load rejection occurs, if the difference between the required temperature set point of the Reactor Coolant System and the actual average temperature exceeds a predetermined amount, a signal will actuate the steam dump to maintain the Reactor Coolant System temperature within control range until a new equilibrium condition is reached.

The steam dump flow reduces proportionally as the control rods act to reduce the average coolant temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

7.4 NUCLEAR INSTRUMENTATION

7.4.1 GENERAL DESIGN CRITERIA

Fission Process Monitors and Controls

Criterion: Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

The primary function of nuclear instrumentation is to safeguard the reactor by monitoring the neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. It also provides a secondary control function and indicates reactor status during startup and power operation. The Nuclear Instrumentation System uses information from three separate types of instrumentation channels to provide three discrete protection levels. Each range of instrumentation (source, intermediate, and power) provides the necessary overpower reactor trip protection required during operation in that range. The overlap of instrument ranges provides reliable continuous protection beginning with source level through the intermediate and low power level. As the reactor power increases, the overpower protection level is increased by administrative procedures after satisfactory higher range instrumentation operation is obtained. Automatic reset to core restrictive trip protection is provided when reducing power.

Various types of neutron detectors, with appropriate solid-state electronic circuitry, are used to monitor the leakage neutron flux from a completely shutdown condition to 120 percent of full power. Because of the wide range of neutron flux, monitoring with several ranges of instrumentation is necessary. The lowest range ("source" range) covers six decades of leakage neutron flux. The next range

("Intermediate" range) covers eight decades. Detectors and instrumentation are chosen to provide overlap between the higher portion of the source range and the lower portion of the intermediate range. The highest range of instrumentation ("power" range) covers approximately two decades of the total instrumentation range. This is a linear range that overlaps with the higher portion of the intermediate range. The power range channels are capable of recording overpower excursions up to 200 percent of full power.

The system described above provides control room indication and recording of signals proportional to reactor neutron flux during core loading, shutdown, startup and power operation, as well as during subsequent refueling. Start-up-rate indication for the source and intermediate range channels is provided at the control board. Reactor trip and rod stop control and alarm signals are transmitted to the Reactor Control and Protection System for automatic plant control.

7.4.2 NUCLEAR INSTRUMENTATION SYSTEMS DESIGN AND EVALUATION

A comprehensive discussion of the Nuclear Instrumentation System (NIS), covering design bases and a detailed description of the system, can be found in Reference 7. In addition, two neutron flux monitoring channels have been added to Units 1 and 2 for indication purposes only. Both channels have been qualified for post accident monitoring. Wide range and source range flux indication is provided by both channels in the control room and one channel also provides source range flux indication on a local shutdown indication panel. The neutron flux monitoring channels that were added perform none of the tripping or protective functions described in Section 7.2.2. Both channels can be configured to provide backup monitoring for the Source Range channels during shutdown conditions.

Engineered Safety Features and Associated System Actuation

Table 7.2-1 includes the engineered safety features and associated systems actuation signals.

Engineered Safety Features Vital Functions

The engineered safety features actuation system automatically performs the following vital functions:

- a) Starts operation of the Safety Injection System upon:
 - 1. Low pressurizer pressure
 - 2. High Containment pressure
 - 3. High differential pressure between steam lines
 - 4. Low steam line pressure
- b) The Safety Injection Signal will also:
 - 1. Initiate Phase "A" containment isolation (A) and containment ventilation isolation (CVI)
 - 2. Initiate main feedwater isolation
 - 3. Actuate the auxiliary feedwater system
 - 4. Start the diesel generators
- c) Closes the steam generator main steam stop valves on:

High-High containment pressure or low steam line pressure or high steam line flow coincident with low-low Tavg.

- d) Initiates the Containment Spray System and a Phase "B" containment isolation (B) on a hi-hi containment pressure signal.

Reset Capability

To allow for post incident recovery flexibility as well as recovery from spurious actuation, push buttons are provided to reset the following actuating signals:

- a) Safety Injection
- b) Phase "A" Containment Isolation
- c) Containment Ventilation Isolation
- d) Steam Line Isolation
- e) Phase "B" Containment Isolation
- f) Containment Spray
- g) Feedwater Isolation

Each of these reset push buttons has an alarm to indicate that it has been pushed and a sealed cover to prevent its inadvertant use.

Engineered Safety Features Calibration and Test

The engineered safety features actuation channels are designed with sufficient redundancy to provide the capability for channel calibration and test during power operation. Except for containment spray actuation, removal of one actuation channel for test is accomplished by placing that channel in a tripped mode; i.e., a two out of three matrix logic becomes a one out of two matrix logic. Testing does not trip the system unless a trip condition occurs in a redundant channel.

Feedwater Isolation

Any safety injection signal will isolate the main feedwater lines by closing the flow control and isolation valves, tripping the main feedwater pumps and closing the pumps' discharge valves.

Main Steam Isolation

Protection against a steam line break is provided by safety injection actuation, feedwater isolation - to prevent excessive cooldown of the primary side, and main steam isolation - to prevent the uncontrolled blowdown of more than one steam generator. Closure of the steam line isolation valves is initiated by the signals previously described in section 7.5.2 and included in Table 7.2-1 as part of an automatic actuation system designed to meet the requirements for protective systems as described in sections 7.2.1 and 7.5.2. Main steam isolation may also be initiated manually from the control room.

Engineered Safety Features Instrumentation

The following describes the instrumentation which ensure monitoring of the Engineered Safety Features.

Ice Condenser Instrumentation

The ice condenser instrumentation serves to monitor the operation of the equipment and the ice bed status by providing to the operator the control room information listed below. These features are informative but are not required for proper ESF action.

a) Temperature Measurements:

The monitoring of ice bed temperatures provides information to the operator about possible thermal gradients as well as the general ice bed condition. The temperature recorder is located in the control room. The recorder monitors 96 independent temperatures. The recorder is provided with alarm switches and the alarm is activated if a preselected temperature set-point is exceeded.

The thermal status of ice condenser floor cooling and wall duct panels is monitored at 32 sensing points which are recorded by an additional recorder in the control room.

b) Door Position Indications:

The 48 lower inlet doors are arranged in pairs to cover the 24 openings. Each door has two limit switches which monitor its position. One of each door's switches is wired to an individual status lamp on the CAS sub-panel. The 48 remaining switches are connected in parallel to a common annunciator on the SV panel in the control room. Thus, if any door is open there will be an alarm in the control room and the identity of the open door or doors can be determined by observing the status lamps.

The lower personnel access door also has two limit switches. One lights a status lamp on the CAS sub-panel and the other actuates its own annunciator on panel SV in the event the door is opened.

water pumps. This level control function involves remote manual positioning of auxiliary feedwater flow control valves in order to maintain proper steam generator water level. Steam generator water level indication and controls are located in the control room and at a Hot Shutdown Panel located in the other unit's control room.

Motor and Valve Control

For starting pump motors, the control relays are energized to energize the closing coil on the circuit breaker or the motor starter. When motor starters are used the starter operating coil will be supplied by power from the same source as the motor. When circuit breakers are used for motor control the circuit breaker closing and trip coils will be supplied by power from a 250-volt d-c battery bus described in Chapter 8.

For valve motor control, the control relay causes the coil of the main contactor for the actuating circuit to be energized.

Air actuated containment isolation valves are spring loaded to close upon loss of air pressure.

Environmental Capability

The engineered safety features instrumentation and equipment inside the containment is designed to operate under the credible accident environments of a steam-air mixture and radiation.

Table 7.5-2 lists the equipment both inside and outside containment exposed to harsh environments which is required for post-accident operation and indicates whether each is an initiation and/or long-term recirculation time span required component.

Failure of the equipment identified in Table 7.5-2 after the specified time will not increase the severity or consequence of the accident.

The reactor protection control and instrumentation equipment and electrical equipment for engineered safety features located in the auxiliary building will operate in a normal ambient environment following a postulated accident.

A "type" or "similar component" environmental testing program has been completed on the equipment exposed to harsh environment and used for engineered safety features. The current results of this testing are presented in response submittals to Inspection and Enforcement Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment".

Figures 7.5-2 and 7.5-3 give the Chapter 14 accident analysis envelope required for predicted in-containment post-LOCA and in-containment Main Steam Line Break (MSLB) conditions, respectively. Outside containment equipment locations and associated environments are discussed in Sub-Chapter 14.4.

Tables 3.3-11 (Unit 1) and 3.3-10 (Unit 2) in the Technical Specifications provide details of the minimum number of channels of post-accident monitoring instrumentation that are required.

Table 7.8-1 (sheet 1 of 2)
 TYPE "A" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

	<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
A-1	Centrifugal Chg. Pump Flow (CCP)	4	0-200 GPM	Control Room Panel SIS	Maintain pressurizer level during S/G tube rupture
A-2	RCS pressure (wide range)	2	0-3000 psig	Control Room Panel RHR	Manual trip of RC pumps based on RCS pressure
A-3	S/G Pressure	12	0-1200 psig	Control Room Panel SG	Determination of required core exit temperature by S/G Pressure
A-4	Containment Water Level	2	599'-3" to 614' elevation (cont. floor to max flood level)	Control Room Panel RHR	Determination of adverse containment
A-5	S/G Level (narrow range)	12	From below 1st stage separator to 2nd stage separator	Control Room Panel SG	Manual reduction of ECCS flow (secondary heat sink capability)
A-6	Pressurizer Lev.	3	0-100% (96% of total flow)	Control Room Panel PZR	Manual reduction of ECCS flow
A-7	Containment Area Radiation monitor high range	2	1 R/HR to 1X10 ⁷ R/HR	Control Room Panel RMS	Determination of adverse containment
A-8	Containment Pressure (narrow range)	4	-5 to +12 psig	Control Room Panel SPY	Manually establish or trip containment spray

Table 7.8-1 (sheet 2 of 2)
 TYPE "A" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

	<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
A-9	Auxiliary Feedwater Flow	4	0-250 x 10 ³ PPH	Control Room Panel SG	Manual reduction of ECCS flow (secondary heat sink capability).
A-10	RWST Level	2	Essentially top (bottom of over- flow) to bottom (100% of total volume)	Control Room Panel SPY	Manual transfer to cold leg recirculation on low level in RWST
A-11	Degrees sub- Cooling	NA	-50°F Superheat to +350°F Subcool	Control room Panel BA	Manual trip or reduction of pressurizer spray and ECCS flow
A-12	Core Exit T/C's	T/C 1-65	200-2300°F	Control Room Panel FI (U-1) Panel RMS (U-2)	Manual reduction of ECCS Flow
A-13	CCP Breaker Status	4	OPEN/CLOSE	Control Room Panel BA	Manual trip of RCPs
A-14	SI Pump Breaker Status	4	OPEN/CLOSE	Control Room Panel SIS	Manual trip of RCPs
A-15	Safety Injection Pump Flow	2	0-800 GPM	Control Room Panel SIS	Manual trip of RCPs

Table 7.8-3 (sheet 1 of 2)
 TYPE "C" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
C-1 Core Exit Temperature	(See Item A-12)			Fuel Cladding
C-2 Radioactive Concentration or Radiation Level in Circulating Primary Water	2	NA	NA	
C-3 Analysis of Primary Coolant (gamma spectrum)	(See Item C-2)			
C-4 RCS Pressure	(See Item A-2)			Reactor Coolant Pressure Boundary
C-5 Containment Pressure	(See Item A-8 and B-13)			
C-6 Containment Sump Water Level	(See Item B-12)			
C-7 Containment Area Radiation	(See Item A-7)			
C-8 Effluent Radioactivity-Noble Gas from Condenser Air removal System Exhaust	1	9E-07 to 9E+04 uCi/CC	Control Room CT-1 Control Terminal	
C-9 RCS Pressure	(See Item A-2)			

Table 7.8-3 (sheet 2 of 2).
 TYPE "C" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

	<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
C-10	Containment Hydrogen Concentration	9	0 - 30 Volume %	Control Room Panel IV	
C-11	Containment Pressure	(See Item A-8 and B-13)			
C-12	Containment Effluent Radioactivity- Noble gases from Identified Release Points	1	9E-07 to 9E+04 uCi/cc	Control Room CT-1 Terminal	
C-13	Effluent Radioactivity- Noble Gases (from Buildings or Areas where Penetrations and Hatches are located, eg. Secondary Containment and AUX Buildings that are in direct Contact with Primary Containment	(see Item C-12)			

Table 7.8-4 (sheet 5 of 5)
 TYPE "D" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
D-32b 4KV Safety Related Power Systems Status	4	0-150V	Control Room Panel SA	
D-32c 250VDC Battery Power System Status	2	0-300V	Control Room Panel SA	
D-32d 120VAC Safety Related Power Systems Status	4	0-150V	Control Room Panel SA	
D-32e Instrument Air Status	1 1 1 1	0-150 psig 0-100 psig 0-60psig 0 160 psig	Control Room Panel SV	

Table 7.8-5 (sheet 1 of 3)
 TYPE "E" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

	<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
E-1	Containment Area Radiation High Range	(See Item A-7)			Containment Radiation
E-2	Radiation Exposure Rate (inside buildings or where areas of access are required to service equipment important to safety)	12 8 5	.01 to 1000 R/HR .0001 to 10 R/HR .001 to 10 R/HR	Control Room CRT	Area Radiation
E-3a	Containment or Purge effluent	(See Item E-3e)			Noble Gases and Vent Flow Rate
E-3b	Reactor Shield Building Annulus	(See Item E-3e)			
E-3c	Auxiliary Building	(See Item E-3e)			
E-3d	Condenser Air Removal System Exhaust	1 1	9E-07 to 9E+04 uCi/cc 0-250 scfm	NA	
E-3e	Common Plant Vent	1 1	9E-07 to 9E+04 uCi/cc 0-200K scfm	Control room CT-1 Control Terminal	
E-3f	Vent from S/G Safety Relief valves	1	0.-1 to 100 uCi/cc	Control Room Panel RMS	

Table 7.8-5 (sheet 2 of 3)
 TYPE "E" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
E-3g Other Identified Release Points	1	9E-07 to 9E+04 uCi/cc	Control Room Panel FI	
	1	U1-0-1500 SCFM U2-0-4500 SCFM		
E-4 All Identified Release Points (except S/G safety related valves and condenser air removal system exhaust) Sampling and onsite analysis	(See Item E-3e)			Particulates and Halogens
E-5a Airborne Radioactivity and Particulates Sampling and Analysis (portable)	NA	1E-9 to 1E-3 uCi/cc (minimum)	NA	Environs Radiation and Radioactivity
E-5b Plant and environs Radiation (portable)	NA	Gamma 1.0E-3 to 1.0E4 R/HR Beta/low energy gamma 1.0E-3 to 1.0E4 Rad/hr	NA	
E-5c Plant and environs Radioactivity (portable)	NA	Isotopic Analysis	NA	
E-6 Wind Direction	4	0-360°	Control Room Panel Fix and/or CRT	Meterology

Table 7.8-5 (sheet 3 of 3)
 TYPE "E" VARIABLES PROVIDED THE OPERATOR FOR MANUAL FUNCTIONS
 DURING AND FOLLOWING AN ACCIDENT

<u>Parameter</u>	<u>Number of Channels</u>	<u>Range</u>	<u>Display Location</u>	<u>Purpose</u>
E-7 Wind Speed	4	0-100 mph	Control Room Panel Fix and/or CRT	
E-8 Estimation of Atmospheric Stability	4	-30 to 50°C	Control Room Panel Fix and/or CRT	
E-9a Gross Activity	3	1 uCi/ml to 10 Ci/ml	NA	Accident Sampling Primary Coolant and Sump
E-9b Gamma Spectrum	3	0.050 to 2.05 MeV Isotopic Analysis	NA	
E-9c Boron Content	3	375 to 2000 ppm	NA	
E-9d Chloride Content	2	0.01 to 20 ppm	NA	
Chloride Content	3	10 to 20,000 ppm	NA	
E-9f Dissolved H ₂ or Total Gas	3	0-2000 cc/kg	NA	
E-9g Dissolved O ₂	3	0-20 ppm	NA	
E-9h pH	3	1.0 to 13.0 pH	NA	
E-10a H ₂ Content	1	NA	NA	Containment Air
E-10b O ₂ Content	NA	NA	NA	
E-10c Gamma Spectrum	1	1 uCi/cc to 10 Ci/cc Isotopic Analysis	NA	

8. ELECTRICAL SYSTEMS

Section 8 describes the electrical systems and equipment required to generate power and deliver it to the high voltage system. The systems described herein consist of two identical units (Units No. 1 and No. 2) and include facilities for providing power to and controlling the operation of electrically driven plant auxiliary equipment and instrumentation. Figures 8.1-1a and 8.1-1b show the Unit 1 auxiliary electrical one line diagram. Figures 8.1-2a and 8.1-2b show the interconnections between the plant and its offsite power sources.

The main generator output of each unit is fed into the transmission network of the American Electric Power System. While generating, all auxiliary power is supplied from the generator terminals through the normal auxiliary transformers (1AB and 1CD for Unit 1 and 2AB and 2CD for Unit 2). Upon turbine-generator trip, the station auxiliaries are automatically and instantaneously transferred to the preferred offsite power source auxiliary transformers (101AB and 101CD for Unit 1 and 201AB and 201CD for Unit 2) to assure continued power to equipment when the main generator is off the line.

The preferred offsite power source auxiliary system for both units is arranged so that either the 345MVA, 34.5kV tertiary winding of transformer No. 4, or the low voltage winding of 150MVA 345/34.5kV transformer no.5 supplies four transformers (101AB and 101CD for Unit No. 1 and 201AB and 201CD for Unit No. 2). Transformer No. 5 has been installed as a full service alternate to transformer No. 4. In addition, the alternate offsite power source, a 69/4.16 Kv transformer, located at the plant site, has the necessary capacity to operate the engineered safeguard equipment in one unit while supplying safe shutdown power in the other. Essential instrumentation, including the reactor protection system and the engineered safety features instrumentation, is fed from vital instrumentation buses to provide

continuous monitoring and control. The station batteries provide circuit breaker control, control room emergency lighting, and operating power for certain electrically operated valves and vital bus inverters.

8.1 DESIGN BASES

The plant electrical systems are designed to ensure a continuous supply of electrical power to all essential plant equipment during normal operation and under abnormal conditions.

8.1.1 GENERAL DESIGN CRITERIA

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon, such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect:

- a) Appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area.
- b) An appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Applicable standards and codes as detailed in the Electrical Equipment Specifications for the D. C. Cook Nuclear Plant have been complied with in the design, manufacture, and testing of all electrical equipment vital to the operation of the engineered safety features. Accordingly, elec-

U trical equipment directly related to the operation of the engineered safety features, or to the safe shutdown of the units has been designated as Class I, which designation assures compliance with the seismic Class I criteria as defined for the Cook Nuclear Plant (Reference subchapter 2.9 of the FSAR).

The design of all cable trough (trays) and conduit systems vital to the operation of the engineered safety features has been analyzed and documented to assure compliance with the seismic Class I criteria as defined for the Cook Nuclear Plant. Power, control and instrumentation cabling, motors and other electrical equipment required for operation of the engineered safety features have been inherently designed to withstand the effects of a nuclear system accident or severe external phenomena, as required by their safety function, thus assuring a high degree of confidence in the operability of such components should their use be required.

Emergency Power

Criterion: An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

Each unit has two 3500 kW emergency diesel generators which are individually capable of supplying sufficient power to operate the engineered safety features and protection systems required to avoid undue risk to public health and safety.

The diesel generators start automatically and accept load within 10 seconds after the loss of normal and Preferred Offsite Power Sources to the buses which supply vital loads.

The diesel generator capacity is established on the basis of the operation of engineered safety features during a maximum hypothetical incident concurrent with a loss of (offsite) power and is adequate for safe and orderly shutdown of the unit.

All necessary safety features are duplicated and power supplies so arranged that failure of any one of the applicable buses to energize or failure of one diesel generator to start, does not prevent operation of a sufficient amount of equipment to ensure protection of the public.

In addition, the diesel generators may be test started and loaded to approximately fifty percent of rated load via the diesel generator load bank resistors.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failure of which would result in undue risk to health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

The applicable portions of the Missile Protection Criteria as stated in Section 1.4 apply to Class I equipment in this chapter.

8.1.2 FUNCTIONAL CRITERIA

In addition to the aforementioned criteria, the following functional criteria will be employed to achieve maximum reliability and operating efficiency of the electrical systems.

- a) The main turbine-generator for each unit, described in Section 10, feeds electrical power at 26 kV through the isolated phase bus to its main step-up transformer and the unit auxiliary transformers located adjacent to the turbine building.

- b) The primary sides of the unit auxiliary transformers 1AB and 1CD for Unit 1 and 2AB and 2CD for Unit 2 are connected to the isolated phase bus at a point between the generator terminals and the low voltage connection of the main step-up transformer. During normal operation, station auxiliary power is taken from these transformers. These transformers are each rated 18/24/30 MVA, 26/4.16 kV. The 4160 volt secondaries feed four independent 4160 volt auxiliary buses of each unit. The short circuit fault duty of each bus is limited to within the interrupting capability of the 250 MVA air circuit breakers. This functional alignment permits limited station operation when one 4160 volt bus is out of service.
- c) The preferred offsite power source for the two units is either the 345 MVA 34.5 Kv tertiary winding of transformer No. 4, a 1500 MVA 765/345 transformer or the 150 MVA 345/34.5 Kv transformer No. 5 which has been installed as a full service alternate to transformer No. 4 for purposes of supplying the plant's auxiliary loads. Transformer No. 4 or transformer No. 5 may be connected to transformers 101AB, 101CD, 201AB, and 201CD (each an 18/24/30 MVA, 34.5/4.16 Kv transformer), which supply the reserve auxiliary power for both units.
- d) A 69 kV line operating on a right-of-way off the plant property has been tapped to feed a 7500 kVA 69/4.16 kV transformer located at the plant site. The 4160 volt power is used as the alternate offsite power source to both units. The 69/4.16 kV alternate offsite power source is manually connected to the 4160 volt buses. The breakers which connect this source to the 4160 volt buses are interlocked so they will not close if any other 4160 volt bus source is closed. In addition, the availability of the 69 kV alternate offsite power source is constantly monitored and its loss annunciated.

- e) The 4160 volt system for Unit 1 is divided into eight bus sections (1A, 1B, 1C, 1D and T11A, T11B, T11C and T11D). Buses 1A and 1B are supplied either from transformer 1AB when the main generator is in operation or from Transformer 101AB when the main generator is not in operation. Buses 1C and 1D are supplied in a similar manner, from either transformer 1CD or transformer 101CD. Buses T11A, T11B, T11C and T11D are supplied from buses 1A, 1B, 1C and 1D respectively. Upon unit trip, 4160 volt bus 1A, 1B, 1C, and 1D automatically transfer from their normal auxiliary source to the preferred offsite power source. Motors 400 hp or larger are operated at 4160 volts and all emergency motors of this size are operated from buses T11A or T11D. An identical bus arrangement (2A, 2B, 2C, 2D and T21A, T21B, T21C and T21D) is provided for Unit 2.
- f) The 600 volt system for Unit 1 is divided into six bus sections, four of which (11A, 11B, 11C and 11D) contain motors up to 400 hp including emergency equipment required in the event of a power failure. These four sections are each normally fed by a 2000 kVA, 4160/600 volt transformer from 4160 volt buses T11A, T11B, T11C and T11D, respectively, which, in turn, are fed from 4160 volt buses 1A, 1B, 1C and 1D, or by diesel generators, 1AB or 1CD during a loss of power incident. T11A, T11B, T11C and T11D are also directly alignable to the alternate source of off-site power, the 69/4.16 kV transformer. There are also two 600 volt bus sections (11BMC and 11CMC), each fed by a 1500/2000 kVA, 4160/600 volt transformer, which supply power to non-essential 600 volt equipment rated 100 hp or less and which is grouped into motor control centers. An identical bus arrangement (21A, 21B, 21C, 21D and 21BMC and 21CMC) is provided for Unit 2.

The 480 volt systems for each unit are divided into two bus sections (11PHA, 11PHC in Unit 1 and 21PHA, 21PHC in Unit 2) and are used to provide power to the pressurizer heater system.

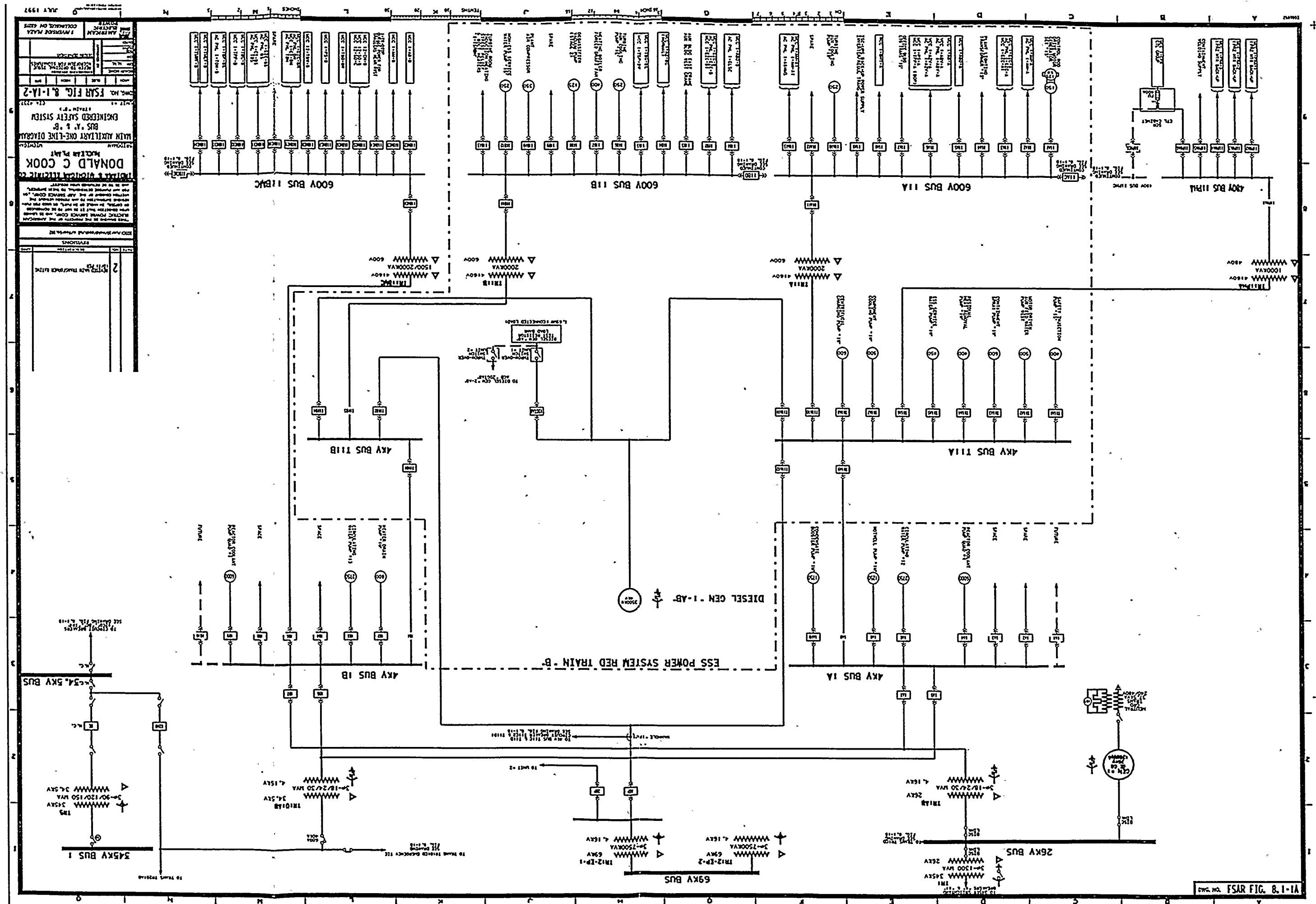
Buses 11PHA and 11PHC are fed from two 4160/480V, 1000 kVA transformers (TR11PHA and TR11PHC, respectively). These transformers are fed from 4160 volt buses T11A and T11D respectively. Buses 21PHA and 21PHC are fed from two 4160/480 volt, 1000 kVA transformers (TR21PHA and TR21PHC, respectively). These transformers are fed from 4160 volt buses T21A and T21D respectively.

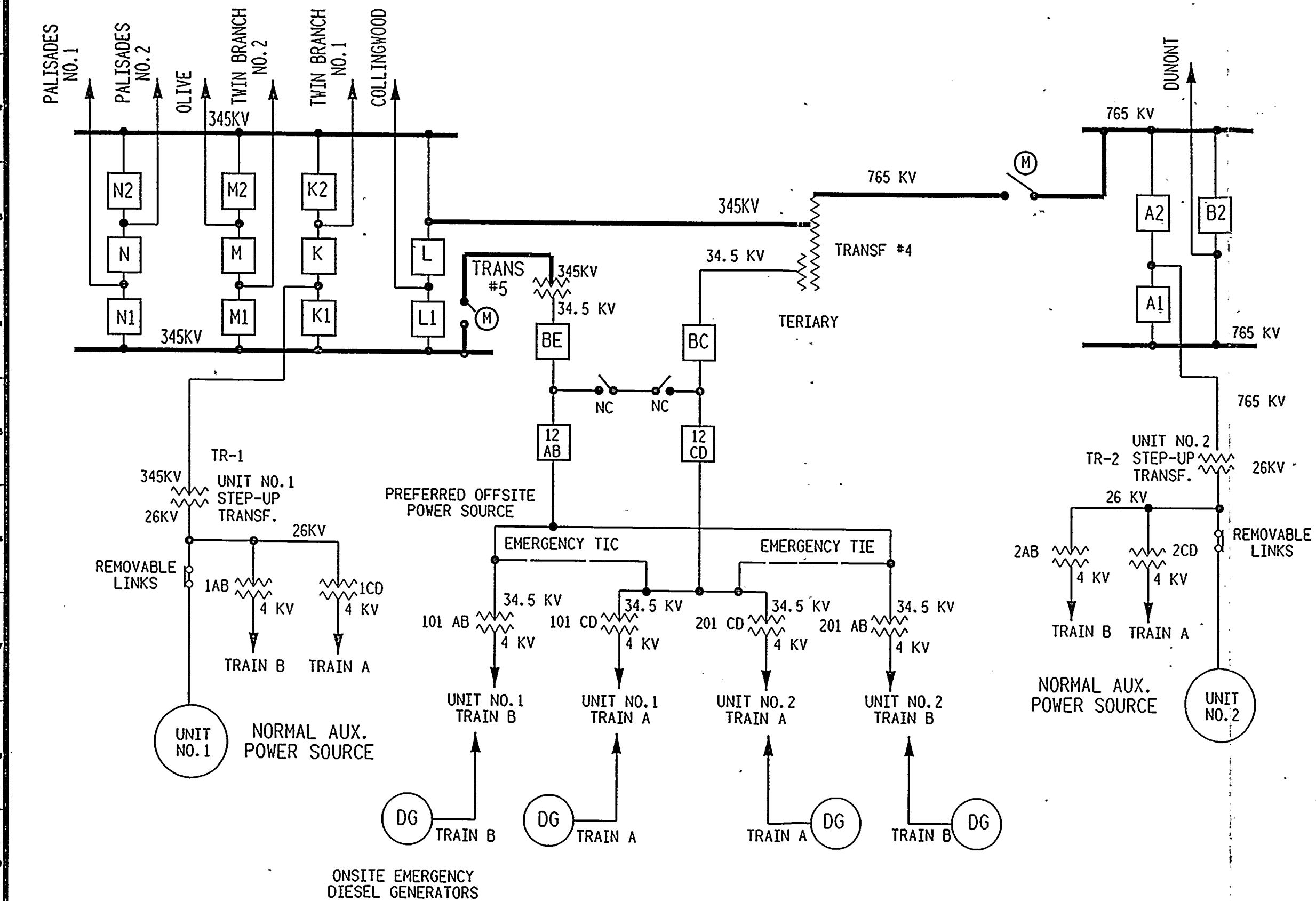
- g) The 4160 volt, 600 volt and 480 volt switchgear is of metal-clad construction with closing and tripping control power taken from the station batteries. Each breaker cubicle is isolated from the adjacent cubicle with metal barriers and each bus section is physically separated from all others, with the exception of buses 11A and 11C, and 11B and 11D, which are separated by means of bus tie breakers and the metal barriers between the adjacent end cubicles.
- h) The system has been so designed that a single failure of any electrical device to operate shall not prevent the protection and safety features from providing the required safety functions.
- i) Power cables are distributed from the switchgear by means of steel conduit, plastic conduit imbedded in concrete and cable trays. Control cables are run in steel conduit or cable trays.
- j) The feed from the generator terminals to the main step-up transformer bank is isolated phase, forced air cooled, bus duct.
- k) The main feeds and feeder motor cables in 4160 volt service are insulated cables rated at 5000 volts. The exact construction of the cables and method of support conform to the requirements of the individual service. Single conductor cables are

shielded and provided with a fire retardant jacket. Three conductor cables are triplexed. Copper conductors are used within the containment.

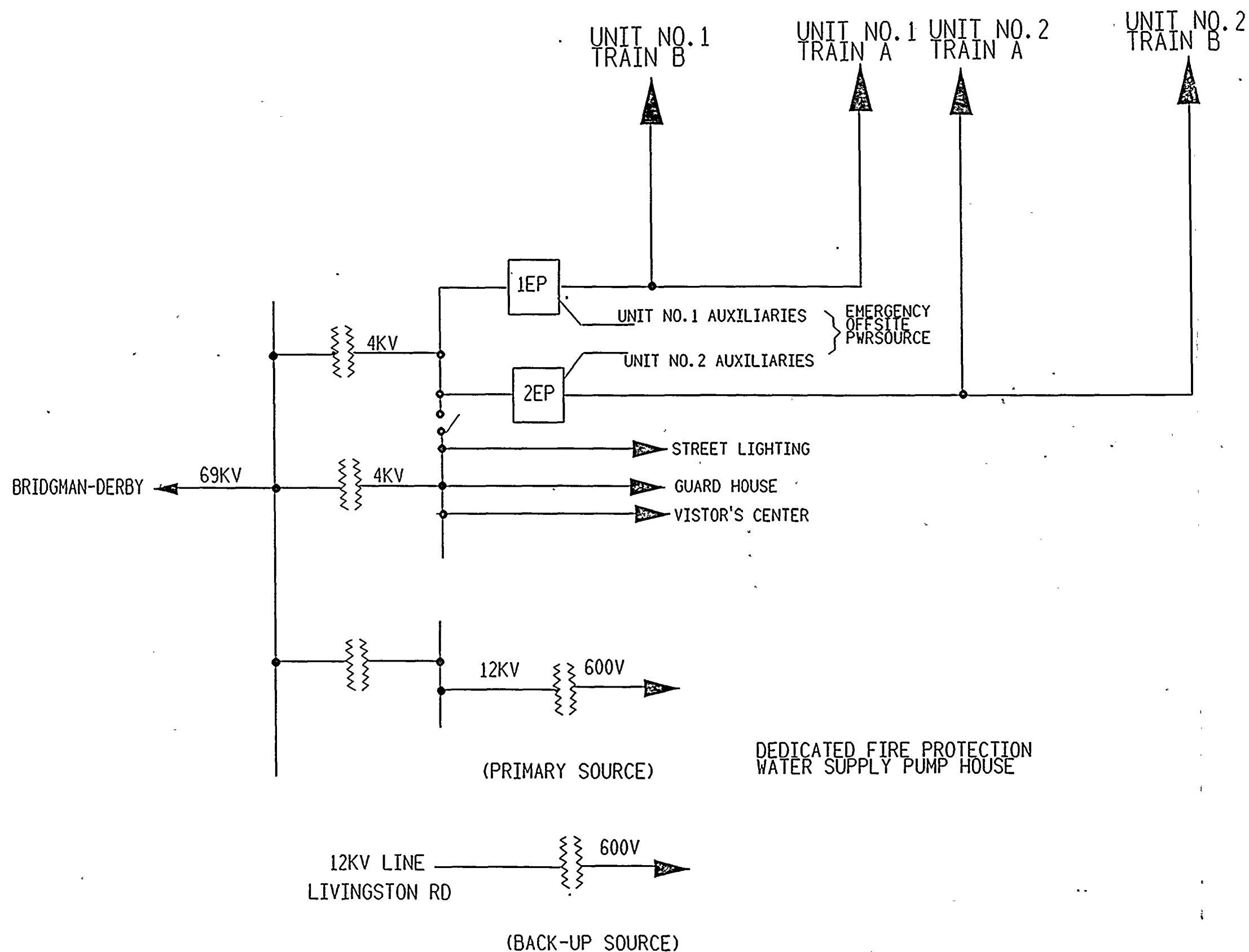
- l) Power cables for the 600 V service are insulated cables rated at 600 V in single or triplex construction as required. Copper conductors have been used within the containment.
- m) Control cables are of single or multi-conductor copper construction rated at 600 V with overall flame retardant jacket.
- n) Low voltage instrument cables are rated at 300 V. These cables have total coverage electrostatic shield and are flame retardant.
- o) The normal current rating of all insulated conductors is limited to that continuous value which does not cause excessive insulation deterioration from heating. Selection of conductor sizes is based on "Power Cable Ampacities," published by the Insulated Power Cable Engineers Association (IPCEA).
- p) Vital instrument buses are provided for essential instrumentation and reactor protection circuits. Each bus is fed from an inverter which receives its normal source of power from the 250 V DC bus. If the 250 V DC bus or the DC to AC section of the inverter fails, the vital bus is transferred to a regulated 600/120 V AC Balance of Plant Source. The 600/120 V AC Balance of Plant Source is a transformer which is regulated to provide 120 V AC \pm 3%. This transfer from the 250 V DC bus to the Balance of Plant Source is accomplished without voltage variations. The output frequency of the inverter is synchronized to the normal plant AC source. On loss of the plant AC, the inverter will maintain an output frequency of 60 Hz \pm 0.5 Hz. The alternate source of power to the vital instrument buses is the unit auxiliary power system (see Figure 8.3-1).

- q) Motor and electrical switchgear enclosures conform to the expected environmental conditions and are designed in accordance with specifications issued by the National Electrical Manufacturers Association (NEMA). The station batteries are sized to operate turbine shutdown oil pumps, instrumentation and vital nuclear channels for three hours without benefit of any station AC power.
- r) All electrical equipment and cables operate within their normal rating or temperature rise. Motor loading does not exceed its nameplate rating.





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DONALD C. COOK NUCLEAR PLANT			
SIMPLIFIED OFFSITE POWER SOURCES			
DWS NOFSAR FIG. 8.1-2A-0			
ARCH	ELEC	MECH	STR
DATE	NO.	REV.	BY
10/1/84	1	1	1
DRAWN BY: J. L. COOK			
CHECKED BY: J. L. COOK			
APPROVED BY: J. L. COOK			
AEP SERVICE COOP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43260			



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DRAFT

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DATE: []

REVISIONS

NO.	DESCRIPTION	DATE
1	ISSUED FOR CONSTRUCTION	10/1/78

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AMERICAN ELECTRIC POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT

SIMPLIFIED OFFSITE POWER SOURCES

DWG NO. FSAR FIG. 8.1-28-0

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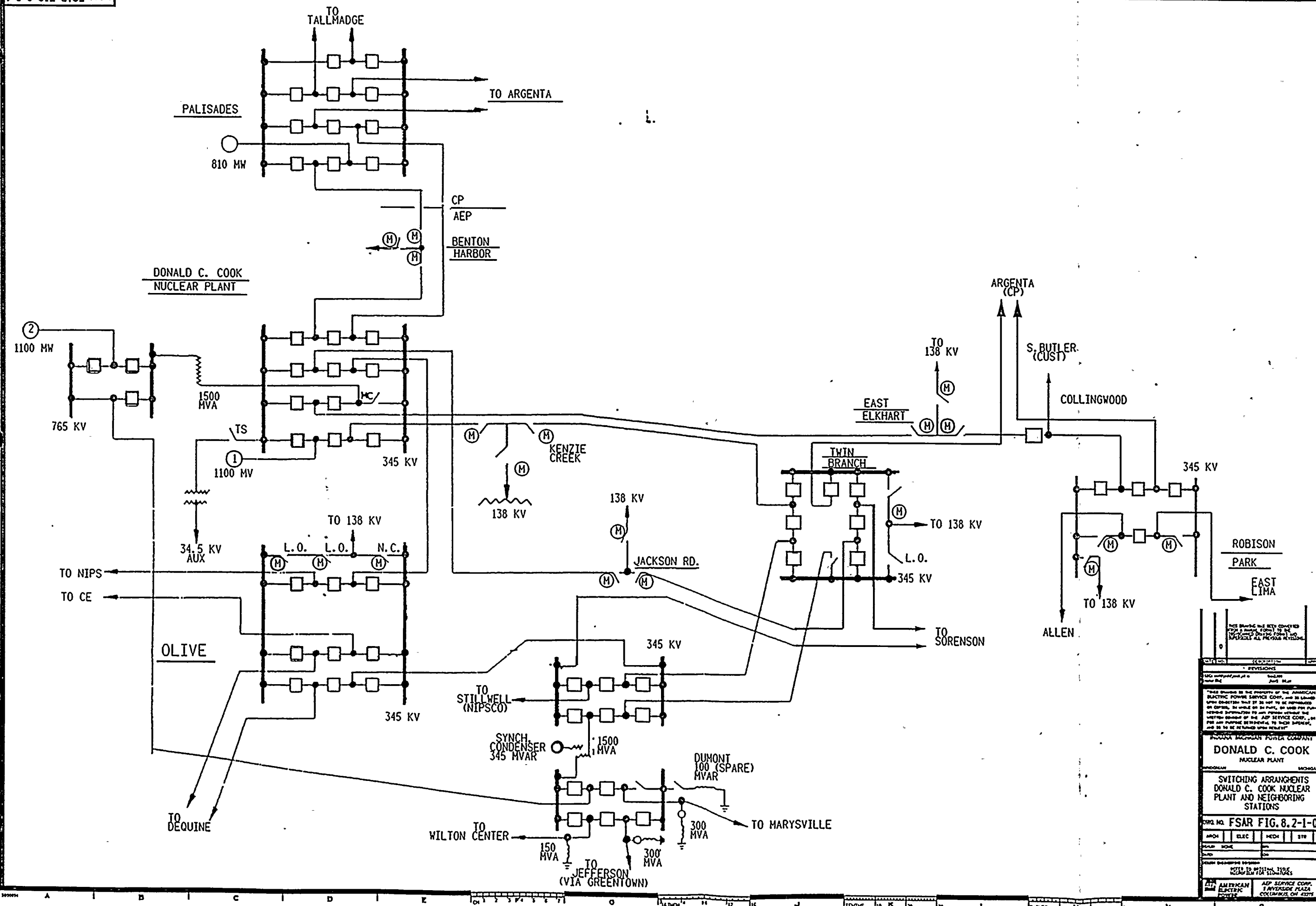
Electrical energy generated at 26 kV is stepped-up to 345 kV and 765 kV by the main power transformers of Unit No. 1 and Unit No. 2, respectively. Energy from Unit No. 1 feeds into a 345 kV switchyard consisting of eleven 3000-ampere, 25,000 MVA circuit breakers. Two 345 kV circuits connect with the Palisades Plant of Consumers Power Company. Four additional 345 kV circuits connect into the American Electric Power Company bulk power supply system; two circuits to Twin Branch and one circuit each to Olive and Collingwood Stations. Except for the Collingwood line, all 345 kV circuits (Palisades, Olive, (2) circuits each, and Twin Branch, (2) circuits each) terminate on a breaker-and-a-half bus scheme through 345 kV, 3000 ampere, 25,000 MVA circuit breakers. The Collingwood line terminates on an incomplete breaker-and-a-half bus scheme.

Energy from Unit No. 2 feeds into a 765 kV ring bus consisting of three 765 kV, 3000 ampere, 35,000 MVA circuit breakers. The 765 kV and 345 kV switchyards are connected through a 1500 MVA, 765/345 kV autotransformer bank comprised of three single-phase units. In addition, a 765 kV circuit from the plant terminates on a ring bus at Dumont Station through two 765 kV, 3000 ampere, 35,000 MVA circuit breakers.

The facility is designed such that loss of one or both units will not perturb the external grid to the extent that offsite power will be unavailable.

Figure 8.2-1 is a one-line diagram of the existing bulk power supply transmission system in the vicinity of the plant.





8.3.1 4160 VOLT SYSTEM

Unit auxiliary power is distributed from the 4160 volt switchgear which is energized from the main generator through unit auxiliary transformers 1AB and 1CD during normal operation, and from Preferred Offsite Power Source auxiliary transformers 101AB and 101CD during start-up or shutdown operations. The 4160 volt system is duplicated for Unit No. 2.

The 4160 volt switchgear is arranged in eight bus sections. Buses 1A, 1B, 1C, 1D, T11A and T11D each have a capacity of 2000 amperes. Buses T11B and T11C, which serve only transformers T11B and T11C, have a capacity of 1200 amperes. All feeder and motor circuits are protected by:

- a) Overcurrent relays which trip the associated breaker in the event of a sustained overload or fault.
- b) Instantaneous relays for ground fault and motor cable faults.

During start-up, the total unit power demand is supplied from Preferred Offsite Power Source auxiliary transformers 101AB and 101CD. Upon attaining operating conditions, and after the turbine generator has been synchronized and connected to the system, the auxiliary load is transferred to unit auxiliary transformers 1AB and 1CD. The transfer is effected without a power interruption, by momentarily feeding the 4160 volt switchgear from both the reserve and unit transformers. Once the transfer is complete, each turbine-generator supplies its own auxiliaries. A trip of the unit automatically trips the normal source breakers (unit auxiliary transformers) and transfers the auxiliary loads to the Preferred Offsite Power Source. Motors 400 hp or larger operate from the 4160 volt buses.

The 4160 volt buses (T11A, T11B, T11C and T11D) may also be fed from a 4160 volt diesel generator, to supply power to the engineered safety features and other necessary equipment in the event of a loss of offsite power. There are two diesel generators associated with each unit. Each diesel generator is connected to two 4160 volt buses, one to buses T11A and T11B and one to buses T11C and T11D. Upon loss-of-power to a 4160 volt bus, the associated diesel generator starts automatically. The circuit breaker which normally supplies power to that bus from the main 4160 bus is tripped. A 4160 volt circuit breaker in each bus is automatically closed when its diesel generator is at speed and rated voltage and re-energizes the bus. The diesel generators will then supply all equipment which must operate under emergency conditions. The diesel generator system is described in detail in Subchapter 8.4.

The alternate offsite power source has been provided (Ref. Figures 8.1-1a, 8.1-1b and 8.1-2) by tapping a 69 kV transmission line which is located adjacent to the plant property. This line is run overhead to the 69/4.16 kV transformer and the 4160 volt main bus is run underground to connect to buses T11A, T11B, T11C, T11D and T21A, T21B, T21C and T21D. This transformer has been sized to provide necessary capacity to operate the engineered safeguards equipment in one unit while supplying safe shutdown power in the other. The breakers which connect this source to the 4160 volt bus are manually operated, and are interlocked to prevent parallel operation with any other 4160 volt source.

8.3.2 LOW VOLTAGE POWER SYSTEMS

The 600 volt auxiliary system distributes power for all low voltage station service demands other than the pressurizer heaters. The normal source of power for the 600 volt system is the 4160 volt system buses via the 4160/600 volt transformers. The pressurizer heaters are fed from the 4160 volt system buses via their 4160/480 volt transformers. (Ref. Figures 8.1-1a and 8.1-1b).

The switchgear is metal-clad with 250 volt dc operated air circuit breakers. The 4160/600 volt transformers are filled with non-flammable liquid. The 600 volt system is divided into six bus sections, four of which (11A, 11B, 11C and 11D) will feed the motors up to 400 hp. Each motor over 100 hp is energized by a 600 volt circuit breaker. Motors 100 hp and less are fed from motor control centers. The power source for each of these buses is a 2000 kVA, 4160/600 volt transformer whose primary is connected to buses T11A, T11B, T11C and T11D respectively. Bus tie breakers between buses 11A and 11C and buses 11B and 11D are provided so a 2000 kVA transformer can feed two adjacent 600 volt buses should one of the transformers fail. Upon signal to start the diesel generators, the 600 volt bus tie breakers are opened automatically. They cannot be automatically closed after diesel start, thus eliminating the possibility of inadvertent parallel operation of diesels. An identical 600 volt system is provided for Unit 2.

Bus tie breakers 11AC and 11BD are interlocked to close automatically only when a hand reset auxiliary relay (HEA) operates from protective relays which indicate a fault in a 4160/600 volt transformer area. In addition, the closing circuits of the bus tie breakers are interlocked to insure that the faulted section's 600 volt feeder breaker (11 A1 etc.) is also open. An identical 600 volt system is provided for Unit 2.

Two 600 volt buses, (11BMC and 11CMC) are fed from two of the 4160 volt buses (1B and 1C) via two 1500 kVA, 4160/600 volt transformers. These buses supply power to the 100 hp and smaller non-safety related motors fed from motor control centers throughout the plant. A bus tie breaker enables one 1500 kVA transformer to feed both buses should the other transformer fail. An identical 600 volt system is provided for Unit 2.

Two (2) 480 volt buses, 11PHA and 11PHC, are fed from two of the 4.16 kV buses, T11A and T11D respectively via two 1000 kVA, 4160/480 volt transformers. These buses supply power to the pressurizer heater loads. An identical 480 volt system is provided for Unit 2.

8.3.3 120 VOLT AC VITAL INSTRUMENT BUS SYSTEM

The 120 volt ac vital instrument bus system consists of four separate vital buses which are supplied by four independent 7.5 kVA, single phase static inverters, as shown in Figure 8.3-1. Two of the inverters connect to one of the station batteries, the other two connect to a second station battery. Each inverter cabinet output may derive its input from any one of three sources:

- a) The output of a battery charger whose input is a 600 volt Engineered Safety System (ESS) source.
- b) A 250 volt station battery, should the ac powered battery charger fail.
- c) The output of a balance of plant regulating transformer whose input is a 600 volt ESS source separate from the battery charger source.

Transfers between sources are automatic and will not disturb vital bus voltage and frequency.

The inverter voltage output is regulated automatically at 118 volt ac $\pm 3\%$. The output frequency is synchronized with the frequency of the ac supply voltage when the ac source is energized. When free running, the frequency is maintained within 0.5 Hz of the rated value.

The vital buses constitute a very reliable electrical system. The four vital buses provide a continuous source of power to vital instruments and equipment, independent of any momentary interruption of the ac power system.

The output of each inverter is connected to a distribution cabinet through a normally closed circuit breaker. The distribution cabinets have 15 and 20 ampere branch circuit breakers to feed reactor pro-

The plant power system includes an on-site, independent, automatically starting emergency power source which supplies power to essential auxiliaries if normal or preferred offsite power source is unavailable.

The emergency power source for each unit consists of two 4160 volt, 3-phase, 60 cycle, 3500 kW diesel generators as shown in Figures 8.1-1a and 8.1-1b. The arrangement of the emergency diesel generators and their fuel oil system is shown in Figure 8.4-1. Each diesel engine is equipped with its own auxiliaries. These include starting air, fuel oil, lube oil, cooling water, intake and exhaust system, voltage regulator and controls. Cooling water is provided from the Essential Service Water System while electric power for each engine's auxiliaries is provided by its own generator.

Cranking power for each diesel is supplied from its respective high pressure starting air system. Energy for starting a diesel is derived from two (2) air receivers each containing enough high pressure compressed air to provide for two starting sequences.

There are two diesel fuel oil storage tanks on site, physically separated from each other. The piping is arranged so that each storage tank supplies fuel to one emergency diesel generator in each unit. Each storage tank contains enough fuel oil to run one emergency diesel generator at full load continuously for greater than seven days.

The emergency power sources for the two units are identical and are electrically and physically isolated from one another, as are the diesel generator sets for each unit. Each diesel generator is full capacity with one supplying power to buses T11A and T11B, (T21A and T21B for Unit 2) and the other supplying power to T11C and T11D (T21C and T21D for Unit 2).

Loss of voltage to the 4160 volt buses above is sensed by undervoltage relays. Upon sensing, master relays automatically start the emergency generators, trip the normal feed circuit breakers for the 4160 volt buses and trips all motor feeder breakers and 480 volt bus transformer feeder breakers on the buses, the 600 volt bus tie breaker, all non-essential 600 volt motor feeder breakers and 480 volt bus breakers. The emergency generator circuit breaker which connects the diesel generator output to the 4160/600 volt bus system is automatically closed when rated voltage and speed are obtained. The diesel generators supply power to 600 volt buses, 11A, 11B, 11C, and 11D through the 4160 volt buses T11A, T11B, T11C, and T11D respectively. The 600 volt bus tie breakers cannot close automatically after diesel start, thus eliminating the possibility of parallel operation of diesels.

Each emergency generator comes up to speed and is capable of accepting load within 10 seconds. If either diesel fails to start, the remaining one is capable of supplying the required engineered safeguard load. A safety injection signal will also start the diesels.

The diesel generators are sized at 3500 kW each to assure available power to operate the following equipment assuming a loss-of-power concurrent with a loss-of-coolant accident:

<u>Number</u>	<u>Component</u>	<u>Rating (Horsepower)</u>	Nominal Start Time After Safety Injection Signal and Blackout
			<u>(sec)</u>
1	Centrifugal Charging Pump	600	13
1	Safety Injection Pump	400	17
1	Residual Heat Removal Pump	400	21
1	Component Cooling Water Pump	500	25
1	Essential Service Water Pump	450	30
1	Motor Driven Auxiliary Feedwater Pump	500	35
1	Containment Spray Pump	600	41
1	Non-essential Service Water Pump	250	47

The motors listed previously start automatically in sequence as determined by the initiating event after the diesel generator has energized the appropriate buses. In addition, other plant electrical loads fed from these buses may be energized manually provided the operating diesel generators capacity is not exceeded.

All safety equipment is duplicated with one connected to an A or B emergency bus and the other connected to a C or D emergency bus. Should one bus section fail to energize, or one diesel generator fail to start, safety is maintained by the continued integrity of the duplicate system. All switching flexibility of the 600 volt system as previously described, is also maintained. If any safety feature fails to operate automatically, manual operation is possible from the control room.

The emergency power system and the diesel generators are equipped with monitors and annunciators to insure adequate information on system status. Suitable protective devices are provided to initiate prompt automatic detection and isolation of defective or faulted equipment. All annunciators and protective devices are in service as applicable during diesel generator testing. Only the diesel generator differential protection and overspeed trips are operative during actual or simulated emergency conditions.

Diesel generator testing is facilitated by load banks, test circuit breakers and switching equipment that make it possible to load the diesel generators without the need of paralleling the diesel generator to the energized safety buses.

The diesel generators can be started, stopped and their voltage and speed controlled locally via subpanels in each diesel generator room. In this mode of operation, diesel generator control is independent of the control room.



All plant electrical systems are designed to ensure maximum operating efficiency and reliability under all conditions. The plant is connected to seven independent external circuits, (six via the 345 kV switchyard and one via the 765 kV switchyard). The switchyards are interconnected and all switchyard equipment is protected from lightning.

Transformer ratios and tap settings have been chosen to insure that all safety system electrical equipment connected to or powered from the auxiliary system is operated within voltage rating.

During normal operation, auxiliary bus voltages are controlled by the main generator automatic voltage regulator. The main generator may be switched to manual voltage regulation and manually regulated by the control room operator.

All 4160 volt and 600 volt safety buses serving motor loads have been equipped with undervoltage relays to alarm low bus voltage to the operator in the main control room.

While operating from the preferred offsite power source the bus voltages are dependent on the system power grid. In order to prevent a degradation of the offsite power grid from reducing bus voltage beyond equipment ratings, special relaying has been installed to disconnect the ESS buses and automatically transfer them to the on site emergency generators.

Plant auxiliary electrical systems are designed so each bus may be fed from several sources. Components which perform duplicate functions receive their power from different buses to ensure functional reliability. Inherent in system design is the ability to accept a single component failure or fault without jeopardizing plant safety or causing undue risk to public health and safety.

Redundancy in the Emergency Power System ensures the availability of adequate power needed to effect an orderly shutdown under a loss-of-power condition or a concurrent loss-of-power, loss-of-coolant accident. Both emergency diesel generators associated with each unit are protected from natural phenomena, are capable of supplying required power should either generator fail to start, and can be operated locally independent from the control room should that become necessary.

System Description

During plant operation, reactor coolant flows through the letdown line from one of the reactor coolant loop cold legs on the suction side of the reactor coolant pump and is returned through the charging line on the discharge side of the reactor coolant pump of the same loop. An alternate charging connection is provided on the cold leg of a different loop. Current operating practice includes simultaneous use of both the normal and alternate charging connections. This practice has been adopted to address thermal stress concerns in piping connected to the reactor coolant system. An excess letdown line is also provided as an alternate in case the normal letdown circuit is inoperative.

Each of the CVCS connections to the Reactor Coolant System has an isolating valve. In addition, a check valve is located downstream of each charging line isolating valve. Reactor coolant entering the Chemical and Volume Control System flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through a letdown orifice which reduces the coolant pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the letdown heat exchanger followed by a second pressure reduction by the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank which is predominantly hydrogen and water vapor. The hydrogen within the tank is, in turn, the supply source to the reactor coolant. Fission gases are removed from the system by venting the volume control tank to the Waste Disposal System prior to a cold or refueling shutdown.

To enter the Reactor Coolant System the coolant flows from the volume control tank to the charging pumps which raise the pressure above that in the Reactor Coolant System. The coolant then enters the containment, passes through the tube side of the regenerative heat exchangers, and returns to the Reactor Coolant System. A portion of the high pressure charging flow is filtered and injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to particulate matter in the reactor coolant. Part of the flow cools the lower radial bearing and enters the Reactor Coolant System through a labyrinth seal on the pumps shaft. The remainder, which is the shaft seal leakage flow, is filtered, cooled in the seal water heat exchanger and returned to the suction of the charging pumps.

Coolant injected through the reactor coolant pump labyrinth seals returns to the volume control tank by the normal letdown flow path through the regenerative heat exchanger. When the normal letdown route is not in service, labyrinth seal injection flow returns to the suction of the charging pumps through the excess letdown and seal water heat exchangers.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium which is formed from the $B^{10} (n, \alpha) Li^7$ reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately twelve weight percent. The batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. One of four boric acid transfer pumps is used to transfer the batch to the boric acid tanks. The batching tank and the boric acid tanks are shared by Units 1 and 2. Small quantities of boric

suction of the charging pumps is automatically aligned to take suction from the refueling water storage tank.

The maximum rate of boration of the primary system with the 75 gpm discharge of a boric acid transfer pump directed to the charging pump suction is 24.0 ppm/minute, which compensates for a cooldown rate of 6°F/minute at the end of core life when the moderator temperature coefficient is most negative.

The maximum rate of boration with the two centrifugal charging pumps delivering water from the refueling water storage tank at a concentration of 2400 ppm boron is 11 ppm/minute. This compensates for a cooldown rate of 2°F/minute at the end of core life when the moderator temperature coefficient is most negative. By comparison, normal cooldown rates are about 0.8°F/minute.

Alarm Functions

The reactor makeup control is provided with alarm functions to call the operator's attention to the following conditions:

- a. Deviation of reactor primary water makeup flow rate from the control set point.
- b. Deviation of concentrated boric acid flow rate from control set point.
- c. Low level (makeup initiation point) in the volume control tank when the primary water makeup control selector is not set for the automatic makeup control mode.
- d. Low level (between makeup initiation point and automatic alignment charging pump suction to refueling water storage tank) in the volume control tank to allow the operator to manually initiate makeup prior to refueling water automatic alignment.

Charging Pump Control

Positive Displacement Charging Pump *

The positive displacement charging pump has a variable speed drive and supplies charging flow to the Reactor Coolant System. The speed of this pump can be controlled manually, or automatically by pressurizer level. During load changes the pressurizer level set point varies automatically with T_{avg} , compensating partially for the expansion or contraction of reactor coolant associated with T_{avg} changes. Charging pump speed will not change rapidly with pressurizer level control. If the pressurizer level increases, the speed of the pump decreases; conversely, if the level decreases, the speed increases. If the positive displacement charging pump reaches the high speed limit, it becomes necessary to place a centrifugal pump in operation to provide the higher flow capacity and to remove the positive displacement pump from service.

To ensure that the charging pump flow is always sufficient to meet both the seal water and minimum charging flow requirements, the pump has a variable control stop which prevents pump flow lower than the specified minimum. The control stop is variable to permit higher minimum flow limits to be set if mechanical seal leakage increases during plant life.

Centrifugal Charging Pumps

The centrifugal pumps are constant speed pumps with flow control accomplished by a modulating valve in the pump discharge line. When the positive displacement pump is in operation, this control valve is in the wide open position.

* The positive displacement charging pumps are not currently used for plant operations.

Each demineralizer is sized to accommodate the maximum letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of Li^7 which builds up in the coolant from the $\text{B}^{10} (n, \alpha) \text{Li}^7$ reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below $1.0 \mu\text{Ci/cc}$ with 1% defective fuel. The demineralizer is used intermittently to control cesium.

The demineralizer vessel is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with resin retention screens. The cation bed demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

Reactor Coolant Filter

The filter collects resin fines and particulates from the letdown stream. The vessel is provided with connections for draining and venting. The nominal flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements are used.

Volume Control Tank

The volume control tank is an operating surge volume compensating in part for reactor coolant releases from the Reactor Coolant System as a result of level changes. The volume control tank also acts as a head tank for the charging pumps and reservoir for the leakage from the reactor coolant pump controlled leakage seal. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 35 cc per kg of water (STP).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in the tank.

Charging Pumps

Three charging pumps are provided for injecting coolant into the Reactor Coolant System. Two are centrifugal pumps and the third is a positive displacement pump equipped with variable speed drive. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. The centrifugal pump packing glands and positive displacement pump stuffing box are provided with leakoffs to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the drain header disposal. The pump design prevents lubricating oil from contaminating the charging flow. The integral discharge valves on the positive displacement pump act as check valves.

The positive displacement pump is designed to provide the full charging flow and the reactor coolant pump seal water supply during normal seal leakage and normal letdown.* The centrifugal pumps have a higher flow capacity and are currently used in normal plant operation. Each pump was designed to provide charging and seal injection flows with normal letdown flow (75 gpm) or maximum letdown flow (120 gpm), provided that the RCS cold leg backpressure is at normal operating conditions, and provided that the charging pump minimflow path is isolated during maximum letdown flow.

The positive displacement charging pump is designed to be used to hydrotest the Reactor Coolant System.

Either the positive displacement charging pump or a centrifugal charging pump can take suction from the volume control tank and discharge to the normal charging and reactor coolant pump seal water injection paths. When the positive displacement pump is not used, one of the centrifugal charging pumps is operated.* The flow paths remain the same but flow control is

*The positive displacement charging pumps are not currently used for plant operations.

accomplished by a modulating valve on the discharge side of the centrifugal pumps. For periods when maximum letdown or purification flow is required, a centrifugal pump is operated to provide the necessary flow. The centrifugal charging pumps also serve as high head safety injection pumps in the Emergency Core Cooling System (Chapter 6).

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of caustic solutions for pH control and hydrazine for oxygen scavenging.

The capacity of the chemical mixing tank is determined by the quantity of 35% hydrazine solution necessary to increase the hydrazine concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to permit the preparation of the appropriate quantity of pH control chemical solution for the Reactor Coolant System.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger is designed to cool the amount of reactor coolant letdown equal to the nominal injection rate through the reactor coolant pump labyrinth seal, when the normal letdown path is not usable. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

Seal Water Heat Exchanger

The seal water heat exchanger removes heat from several sources; the reactor coolant pump seal water returning to the volume control tank, the reactor coolant discharge from the excess letdown heat exchanger and the centrifugal charging pump by-pass flow. Reactor coolant flows

through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tube sheet to prevent leakage in either direction and undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow, the pump seal water flow and the centrifugal charging pump by-pass flow to the temperature normally maintained in the volume control tank.

Seal Water Filter

This filter collects particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump seals. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

Seal Water Injection Filters

The filter collects particulates from the reactor coolant pump seal water inlet. Two filters are provided in parallel, each sized for the maximum design pump seal flow rate. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the charging pump suction line or boric acid blender. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The filter elements are disposable

cartridges. Provisions are included for venting and draining the filter.

Boric Acid Tanks

Three boric acid tanks are shared by Units 1 and 2. The total boric acid tankage capacity is sized to store sufficient boric acid solution, recovered from the recycle processing train or mixed in the batching tank, for simultaneous refueling plus enough boric acid solution for a cold shutdown shortly after full power operation is achieved. One tank provides sufficient boric acid solution for cold shutdown even if the most reactive RCC assembly is not inserted. One tank supplies boric acid for each reactor coolant makeup system during normal operating, while the third tank serves as a spare.

The concentration of boric acid solution in storage is maintained between 11.5 and 12.5% by weight. Periodic manual sampling and corrective action, if necessary, insures that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution.

Batching Tank

The batching tank (shared by both units) is sized to hold one week's makeup supply, per unit, of boric acid solution for transfer to the boric acid tanks. The basis for makeup is an arbitrary reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage.

Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of a pair of holdup tanks for sampling or to transfer the contents to another pair of holdup tanks. The pump may also be used to fill the spent fuel pit transfer canal from the holdup tanks. The wetted surface of this pump is constructed of austenitic stainless steel.

Boric Acid Evaporator Feed Pumps

The three feed pumps (shared by both units) supply feed to the boric acid evaporator trains from the holdup tanks. The capacity of each pump is equal to the boric acid evaporator capacity. The non-operating pump is a standby and is available for operation in the event the operating pump malfunctions. These canned centrifugal pumps are constructed of austenitic stainless steel.

Evaporator Feed Ion Exchangers

Four flushable evaporator feed ion exchangers (shared by both units) remove cations (primarily cesium and lithium) and anions from the holdup tank effluent. Two of the demineralizers are of the mixed bed type and the other two are of the cation bed type. One of each type are in series in each processing train.

The design flow rate is equal to the boric acid evaporator processing rate. The demineralizer vessels are constructed of austenitic stainless steel and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with resin retention screens.

Ion Exchanger Filters

These filters collect resin fines and particulates from the evaporator feed ion exchangers. The vessels are made of austenitic stainless steel, and are provided with connections for draining and venting. Disposable filter elements are used. The maximum design flow capacity is equal to the boric acid evaporator flow rate.

Boric Acid Evaporators

A boric acid evaporator is provided which will process 30 gpm of dilute radioactive boric acid and produce distillate and approximately 12 weight percent of concentrated boric acid stripped of the radioactive gases. The other boric acid evaporator and associated equipment has been converted to a radioactive waste evaporator as described in Chapter 11. Radioactive gas stripping is achieved by passing heated feed through packed towers employing stripping steam which removes nitrogen, hydrogen and fission gases from the feed and is designed to reduce the influent gas concentration by a factor of 10^5 .

Evaporator Condensate Demineralizers

An anion demineralizer removes any boric acid contained in the evaporator condensate. The other anion demineralizer has been converted to a radioactive waste disposal function as described in Chapter 11. Hydroxyl based ion-exchange resin is used to produce evaporator condensate of high purity by releasing a hydroxyl ion when a borate ion is absorbed. Facilities are provided for regeneration of the resin. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank. Each demineralizer is sized for a flow rate equal to the evaporator flow rate. The demineralizer vessel is made of all-welded austenitic stainless steel, and is equipped with a resin retention screen.

Condensate Filter

The filter collects resin fines and particulates from the boric acid evaporator condensate stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Disposable filter elements are used. The design flow capacity of the filter is equal to the total installed boric acid evaporator flow rate.

Monitor Tanks

Two shared monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and either reprocessed, discharged to the Waste Disposal System or pumped to the primary water storage tank. The other two monitor tanks have been converted to a radioactive waste disposal function as described in Chapter 11.

Each of the tanks has sufficient capacity to hold the condensate produced during 12 hours of operation from an evaporator at full output with only two lab analyses per day.

The tanks are fitted with a nylon, rubber-coated membrane to prevent absorption of oxygen by the water stored in the tank. The portion of the tank above the membrane is vented to the auxiliary building atmosphere.

Monitor Tank Pumps

Two shared monitor tank pumps discharge water from the monitor tanks. Each pump is sized to empty a monitor tank in approximately 3 hours. The pumps are constructed of austenitic stainless steel.

Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the Reactor Coolant System fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time when boron concentration is low. Hydroxyl based ion-exchange resin is used to reduce Reactor Coolant System boron concentration by releasing a hydroxyl ion when a borate ion is absorbed. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

Each demineralizer is sized to remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintain full power operation near the end of core life.

Concentrates Filter

The filter removes particulates from the evaporator concentrates. Design flow capacity of the filter can accommodate the total installed boric acid evaporator capacity. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

Concentrates Holding Tank

The shared concentrates holding tank is sized to hold approximately the production of concentrates from one batch from both evaporators. The tank is supplied with an electrical heater which prevents boric acid precipitation.

TABLE 9.2-1

CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III*, Class C
Letdown heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Excess letdown heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Cation bed demineralizer	ASME III, Class C
Seal water injection filters	ASME III, Class C
Boric acid filter	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Concentrates filter	ASME III, Class C
Evaporator feed ion exchangers	ASME III, Class C
Ion exchanger filter	ASME III, Class C
Condensate filter	ASME III, Class C
Piping and valves	USAS B31.1**

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 - Code for Pressure Piping, USA Standards, and special nuclear cases where applicable. Repairs and replacements for piping are conducted in accordance with ASME Section XI.

TABLE 9.2-2

CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERSGeneral

Plant design life, years	40
Seal water supply flow rate:	
Normal, gpm	32
Maximum, gpm	113
Seal water return flow rate:	
Normal, gpm	12
Maximum gpm	93
Letdown flow:	
Normal, gpm	75
Minimum, gpm	45
Maximum, gpm	120
Charging flow:	
Normal, gpm	55
Minimum, gpm	25
Maximum, gpm	100
Temperature of letdown reactor coolant entering system, °F	Unit 1: 518.9 to 543.5 Unit 2: 511.4 to 547.6
Centrifugal pump miniflow, gpm	60 (each)
Temperature of charging flow directed to Reactor Coolant System, °F	495
Temperature of effluent directed to holdup tanks, °F	127

(volumetric flow rates in gpm are based upon 130°F and 2350 psig)

TABLE 9.2-3

PRINCIPAL COMPONENT DATA SUMMARYRegenerative Heat Exchanger

Number 1 (per unit)
 Heat transfer rate at design conditions, Btu/hr 10.3×10^6

Shell Side

Design pressure, psig 2485
 Design temperature, °F 650
 Fluid Borated reactor coolant
 Material of construction Austenitic stainless steel

	Normal (Design)	Maximum Purification	Heatup
Flow, lb/hr	37,050	59,280	59,280
Inlet temperature, °F	545	545	547
Outlet temperature, °F	290	237	366

Tube Side

Design pressure, psig 2735
 Design temperature, °F 650
 Fluid Borated reactor coolant
 Material of construction Austenitic stainless steel

	Normal (Design)	Maximum Purification	Heatup
Flow, lb/hr	27,170	49,400	29,640
Inlet temperature, °F	130	130	130
Outlet temperature, °F	495	461	521

TABLE 9.2-3 (cont'd.)

Letdown Orifice

Design pressure, psig	2485
Design temperature, °F	650
Normal operating inlet pressure, psig	2085 (U1) 2235 (U2)
Normal operating temperature, °F	290
Material of construction	Austenitic stainless steel

	<u>45 gpm</u>	<u>75 gpm</u>
Number	1 (per unit)	2 (per unit)
Design flow, lb/hr	22,230	37,050
Differential pressure at design flow, psig	1900	1900

Letdown Heat Exchanger

Number	1 (per unit)
Heat transfer rate at design conditions (heatup), Btu/hr	14.8×10^6

Shell Side

Design pressure, psig	150
Design temperature, °F	250
Fluid	Component cooling water
Material of construction	Carbon steel

	<u>Normal</u>	<u>Heatup</u> <u>(Design)</u>	<u>Maximum</u> <u>Purification</u>
Flow, lb/hr	203,000	492,000	496,000
Inlet temperature, °F	95	95	95
Outlet temperature, °F	125	125	125

TABLE 9.2-3 (cont'd.)

Tube Side

Design pressure, psig	600
Design temperature, °F	400
Fluid	Borated reactor coolant
Material of construction	Austenitic stainless steel

	<u>Normal</u>	<u>Heatup</u> (Design)	<u>Maximum</u> Purification
Flow, lb/hr	37,050	59,280	59,280
Inlet temperature, °F	290	380 (max.)	380 (max.)
Outlet temperature, °F	127	127	127

Mixed-Bed Demineralizers

Number	2 per unit)
Type	Flushable
Vessel design pressure:	
Internal, psig	200
External, psig	15
Vessel design temperature, °F	250
Resin volume, each, ft ³	30
Vessel volume, each, ft ³	43
Design flow rate, gpm	120
Minimum decontamination factor as	
measured by I-131 removal	10
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Resin type	Cation and anion
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Reactor Coolant FilterGeneral:

Number	1 (per unit)
Type	Disposable Cartridge
Flow Rates,	
Nominal, gpm	120
Maximum, gpm	150

Vessel:

Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design ΔPressure, psi	75
Design Temperature, °F	180
Absolute Retention Size, micron	≤ 6

Volume Control Tank

Number	1 (per unit)
Internal volume, ft ³	400
Design pressure:	
Internal, psig	75
External, psig	15
Design temperature, °F	250
Operating pressure range, psig	0 - 60
Spray nozzle flow (maximum), gpm	120
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Batching Tank and Batching Tank Heater Jacket (Continued)

Initial ambient temperature	32
Final fluid temperature, °F	165
Heatup time, hrs	3 (approximately)
Tank material of construction	Austenitic stainless steel
Jacket material of construction	Carbon steel

Batching Tank Agitator

Number	1 (shared)
Fluid handled, boric acid, wt%	12
Service	Continuous
Operating temperature, °F	165
Operating pressure	Atmospheric
Material of construction	Austenitic stainless steel

Excess Letdown Heat Exchanger

Number	1 (per unit)
Heat transfer rate at design conditions, Btu/hr	4.61×10^6

	<u>Shell Side</u>	<u>Tube Side</u>
Design pressure, psig	150	2485
Design temperature, °F	250	650
Design flow rate, lb/hr	115,000	12,380
Inlet temperature, °F	95	545
Outlet temperature, °F	135	195
Fluid	Component cooling water	Borated reactor coolant
Material of construction	Carbon steel	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Seal Water Heat Exchanger

Number	1 (per unit)
Heat transfer rate at design conditions, Btu/hr	2.49×10^6

	<u>Shell Side</u>	<u>Tube Side</u>
Design pressure, psig	150	150
Design temperature, °F	250	250
Design flow, lb/hr	99,500	160,600
Normal operating flow, lb/hr (includes miniflow)	99,500	36,000
Design operating inlet temperature, °F	95	143
Design operating outlet temperature, °F	120	127
Fluid	Component cooling water	Borated reactor coolant
Material of construction	Carbon steel	Austenitic stainless steel

Seal Water FilterGeneral:

Number	1 (per unit)
Type	Disposal Cartridge
Flow Rates,	
Nominal, gpm	12
Maximum, gpm	325

Vessel:

Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

TABLE 9.2-3 (cont'd.)

Boric Acid FilterGeneral:

Number	1 (per unit)
Type	Disposable Cartridge
Design Flow Rate, gpm	150

Vessel:

Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design, ΔPressure, psi	150
Design Temperature, °F	250
Nominal Retention Size, micron	20

Boric Acid Transfer Pump

Number	4 (shared)
Type	Two-speed horizontal centrifugal
Design flow rate, each, gpm	75 at high speed
Design pressure, psig	150
Design discharge head, ft.	235
Design temperature, °F	250
Temperature of pumped fluid, °F	170
Available NPSH at 170°F, ft.	15
Material of construction	Austenitic stainless steel

Boric Acid Blender

Number	1 (per unit)
Design pressure, psig	150
Design temperature, °F	250
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Cation Bed Demineralizer

Number	1 (per unit)
Type	Flushable
Vessel design pressure:	
Internal, psig	200
External, psig	15
Vessel design temperature, °F	250
Resin volume, ft ³	20
Vessel volume, ft ³	30
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Design flow, gpm	72
Resin type	Cation
Material of construction	Austenitic stainless steel

Chemical Mixing Tank Orifice

Number	1 (per unit)
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	2
Material of construction	Austenitic stainless steel

Boric Acid Tank Orifice

Number	3 (shared)
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	3
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Deborating Demineralizers

Number	2 (per unit)
Type	Fixed bed
Vessel design pressure, psig	
Internal	200
External	15
Vessel design temperature, °F	250
Resin Volume, ft ³	43
Vessel volume, ft ³	56
Normal flow, gpm	120
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Resin type	Anion
Material of construction	Austenitic stainless steel

Seal Injection FiltersGeneral:

Number	1 (per unit)
Type	Disposal Cartridge
Flow Rates,	
Nominal, gpm	32
Maximum, gpm	80

Vessel:

Design pressure, psig	2735
Design temperature, °F	200
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design ΔPressure, psi	75
Design Temperature, °F	180
Absolute Retention Size, micron	≤ 6

No. 1 Seal By-Pass Orifice

Number	4 (per unit)
Design pressure, psig	2485
Design temperature, °F	250
Design flow, gpm	1.0
Differential pressure at design flow, psi	300

TABLE 9.2-3 (cont'd.)

Holdup Tanks

Number	6 (shared)*
Type	Horizontal, cylindrical
Capacity, each pair, gal.	128,000
Design pressure, psig	15
Normal operating pressure, psig	3
Design temperature, °F	200
Normal operating Temperature, °F	130
Material of construction	Austenitic stainless steel

Recirculation Pump

Number	1 (shared)
Type	Centrifugal
Design flow, gpm	500
Available NPSH at 130°F, ft.	15
Design head, ft.	100
Design pressure, psig	150
Design temperature, °F	200
Normal operating temperature, °F	150
Material of construction	Austenitic stainless steel

Boric Acid Evaporator Feed Pumps

Number	3 (shared)
Type	Canned
Design flow, gpm	30
Design head (TDH), ft.	320
Design pressure, psig	150
Design temperature, °F	200
Normal fluid temperature, °F	115
Material of construction	Austenitic stainless steel
NPSH at 115°F, ft.	15

* Three pairs of tanks

TABLE 9.2-3 (cont'd.)

Evaporator Feed Ion Exchangers (Continued)

Normal flow, gpm	30
Normal operating temperature, °F	130
Normal operating pressure, Psig	75
Resin type	Cation (2 of 4 units) Mixed Bed (2 of 4 units)
Material of construction	Austenitic stainless steel

Concentrates FilterGeneral:

Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	40

Vessel:

Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design, ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

Concentrates Holding Tank

Number	1 (shared)
Type	Cylindrical, heated
Volume, gal.	2,000
Design Pressure	Atmospheric
Design temperature, °F	250
Normal operating temperature, °F	150
Material of construction	Austenitic stainless steel

Concentrates Holding Tank Electric Heater

Number	1 (shared)
Heat transfer rate, KW	6.0
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

Concentrates Holding Tank Transfer Pump

Number	2 (shared)
Type	Centrifugal can
Design flow rate, gpm	40
Design head, ft.	150
Design temperature, °F	250
Design pressure, psig	150
Available NPSH at 180°F, ft.	10
Material of construction	Austenitic stainless steel

Ion Exchanger FilterGeneral:

Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	35

Vessel:

Design pressure, psig	200
Design temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

Condensates FilterGeneral:

Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	35

Vessel:

Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel

Cartridge:

Maximum Design ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

The Residual Heat Removal System is designed to remove residual (sensible) heat from the core and reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System (Chapter 10).

The Residual Heat Removal System is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the Reactor Coolant System are approximately 400 psig and less than 350°F, respectively. Under normal operating conditions, the Residual Heat Removal System will reduce the temperature of the reactor coolant to 140°F within 20 hours following reactor shutdown. The design residual heat load is based on the residual heat fraction of full core MW (thermal) power level that exists at 20 hours following reactor shutdown from an extended power run near full power. These cooldown rates can be achieved with 15% RHR pump head degradation. The design parameters of the system are shown in Table 9.3-2.

As a secondary function, the Residual Heat Removal System is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations.

In addition, portions of the system are utilized as parts of the Emergency Core Cooling System and the Containment Spray Systems. These functions and the associated analyses are discussed in Chapters 6 and 14.

The Residual Heat Removal System provides sufficient capability in the emergency operational mode to accommodate any single active or passive failure and still function in a manner to avoid risk to the health and safety of the public.

The system design precludes any significant reduction in the overall design reactor shutdown margin when cooling water is introduced into the core for decay heat removal or during the emergency core cooling recirculation mode of operation.

System components whose design pressure and temperature are less than the Reactor Coolant System design limits are provided with redundant isolation means and overpressure protective devices.

All system active components which are relied upon to perform the system functions are redundant and the system design includes provision for hydrostatic testing of system components to applicable code test pressures.

Codes and Classifications

All piping and components of the Residual Heat Removal System are designed to the applicable codes and standards listed in Table 9.3-1. Since the loop contains reactor coolant when it is in operation, austenitic stainless steel piping is employed.

9.3.2 SYSTEM DESIGN AND OPERATION

System Description and Operation

The Residual Heat Removal System (shown in Figure 9.3-1) consists of two residual heat exchangers, two residual heat removal pumps and associated piping, valves, and instrumentation. The instrumentation is discussed in Chapter 7.

During system operation, coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The inlet line to the Residual Heat Removal System loop begins at the hot leg of

returned to the RCS (and thus the refueling canal) by using one train of the RHR/ECCS operating in the recirculation mode. Heat removal from the system would be via the RHR heat exchanger. If the RCS can be pressurized and heat removal is via the steam generators, this cooling path could be established in a few minutes. If the RCS can not be pressurized the short term cooling path could also be established in a few minutes. The follow-up, long-term cooling path could be established in another one to two hours.

System and equipment actuated could include (depending on the method of core cooling employed) the centrifugal charging pumps, the reactor coolant pumps, the steam dump valves (either to atmosphere or the main condenser), the auxiliary feedwater system, the safety injection pumps, the portable pumps, the refueling canal drains, and one train of RHR/ECCS operating in the recirculation mode. The centrifugal charging pumps, the auxiliary feedwater system, the safety injection pumps, and the ECCS are of safety grade design.

9.3.4 MALFUNCTION ANALYSIS

A failure analysis of residual heat removal pumps, heat exchangers and valves is presented in Table 9.3-3.

9.3.5 TESTS AND INSPECTIONS

The residual heat removal pump flow instrumentation is calibrated on a periodic basis. Periodic visual inspections and preventative maintenance are also conducted. The system components are tested in accordance with the requirements of the applicable edition of the ASME B&PV Code Section XI and beginning with the 3rd 10 year interval ISI program, pump and valve tests are in accordance with ASME O&M Standards and NUREG-1482 (Refer to Chapter 6).

9.3.6 SAFETY LIMITS AND CONDITIONS

9.3.6.1 Limiting Conditions for Maintenance

- a. Administrative controls at the Plant have been established to permit the removal of RHR system equipment from service only to perform absolutely required maintenance when the RHR system is operating in the decay heat removal mode. If the equipment has to be removed from service, consideration must be given to alternate decay heat removal methods.
- b. Administrative controls at the plant have been established requiring that during the condition when the reactor coolant system is depressurized and vented with air in the steam generator tubes, and the reactor vessel head in place (with or without bolting), both RHR trains must be available with either both emergency diesel generators or one diesel generator and the alternate reserve source available.

9.3.6.2 Limiting Conditions for Operation

- a. A requirement to have only one RHR pump in operation whenever the reactor coolant system is drained to half-loop and vented, has been incorporated into applicable operating procedures. The second pump will be in manual standby. This requirement will reduce total system flow which in turn reduces the possibility of vortex formation and air entrainment at the suction line.
- b. Only one RHR pump will be operated when the RCS is open to the atmosphere to prevent damaging both pumps in the unlikely event that the suction valve from the RCS should close.
- c. The motor operated valves in the RHR bypass line are normally closed during power operation. Closing these RHR cross-tie valves makes the miniflow circuits for each RHR pump independent thereby removing the potential for deadheading the weaker pump.

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 DESIGN BASES

The Spent Fuel Pool Cooling System shown in Figure 9.4-1 is designed to remove from the spent fuel pool the heat generated by stored spent fuel elements. The system serves the spent fuel pool which is shared between the two units.

The system design allows for the need to totally unload a reactor vessel (193 fuel assemblies) for maintenance or inspection at a time when as many as 3420 spent fuel elements are already residing in the spent fuel storage pool.

The system design incorporates two separate cooling trains. System piping is arranged so that failure of any pipeline does not drain the spent fuel pool below the top of the stored fuel elements.

Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which results in undue risk to the health and safety of the public.

The Spent Fuel Pool Cooling System has two cooling trains capable of handling the heat load generated by 3420 spent fuel assemblies plus an additional 80 assembly offload, maintaining the pool temperature below 132°F. The system, with both cooling trains operating, is also capable of maintaining pool temperature below 144°F when one complete core is unloaded and stored in the pool in addition to 3420 spent fuel assemblies already stored.

The system design will keep the maximum bulk pool water temperature below 160°F assuming an 80 assembly offload with one cooling train operational. The minimum time to boil in the event that both loops of the cooling system

become inoperable is 5.74 hours, assuming a worst case maximum heat load and a bulk pool temperature of 144°F prior to the loss of cooling. Any spent fuel pool loading scenario which meets the 160°F peak bulk pool temperature and 5.74 hours to boil criteria is acceptable.

Codes and Classifications

All piping and components of the system are designed to the applicable codes and standards listed in Table 9.4-1.

9.4.2 SYSTEM DESIGN AND OPERATION

System Description

Each of the two cooling loops in the Spent Fuel Pool Cooling System (see Figure 9.4-1) consists of a pump, heat exchanger, strainer, piping, associated valves and instrumentation. The pump draws water from the pool, circulates it through the heat exchanger and returns it to the pool. Component cooling water cools the heat exchanger.

The clarity and purity of the spent fuel pool water is maintained by passing up to 150 gpm of the cooling flow through a filter and demineralizer. Skimmers are provided to prevent dust and debris from accumulating on the surface of the water.

The refueling water purification pump and filter can be used separately or in conjunction with the spent fuel pool demineralizer to regain refueling water clarity after a crud burst in either unit. This can prevent loss of time during refueling due to poor visibility. The system is also used to maintain water quality in the Refueling Water Storage Tanks of both units.

The spent fuel pool pump suction lines penetrate the spent fuel pool wall above the fuel assemblies stored in the pool to prevent loss of water as a result of a suction line rupture. The pool is initially filled with water at the same boron concentration as in the refueling water storage tank.

There is sufficient capacity in the spent fuel pool to store up to 3420 spent fuel assemblies above and beyond the space required for the complete unloading of one unit (193 fuel assemblies). If any of this extra storage capacity is being utilized, it is by "cold" spent fuel assemblies. These are assemblies that have been removed from the reactor (e.g. during previous refuelings) and have been stored sufficiently long to reduce decay heat production to a relatively low level.

During normal operation, with two cooling trains operating and with up to 3420 spent fuel assemblies stored in the pool, the cooling system will maintain the pool temperature below 132°F . In the same scenario, with only one cooling train in operation, the pool temperature is analyzed to remain below 160°F . Under the maximum anticipated heat loading - 3420 spent fuel assemblies plus one complete core, and only one cooling train available, the temperature is analyzed to remain below 180°F . This scenario is not part of the design basis of the system and results in an unacceptable bulk pool temperature. With the maximum heat loading 3420 spent fuel assemblies plus one complete core and two cooling trains operating, the temperature is analyzed to remain below 144°F .

If all cooling is lost and 3420 spent fuel assemblies are stored in the pool, the time required for the spent fuel pool to boil (approximately 242°F) with one complete core added, is approximately 5.74 hours assuming an initial steady state bulk pool temperature of 144°F .

A failure consideration applicable to both units is a remote occurrence. However, should both cores require removal when up to 3420 fuel assemblies are already in the spent fuel pool, one of the cores is placed in the spent fuel pool and the other is left in its reactor vessel. The core added to the spent fuel pool brings the inventory up to 3613 assemblies, which can be safely handled. The other core is left in place in its reactor vessel, with the residual heat removal system in service, until there is space available for it in the spent fuel pool.

The spent fuel pool is located outside the reactor containment. During refueling the water in the pool can be isolated from that in the refueling canal by a gate valve so that there is only a very small amount of interchange of water as fuel assemblies are transferred.

Components

Spent Fuel Pool Cooling System component design data are listed in Table 9.4-2.

Spent Fuel Pool Heat Exchangers

The two spent fuel pool heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pool water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

Spent Fuel Pool Pumps

The two spent fuel pool pumps circulate water in the spent fuel pool cooling loops. All wetted surfaces of the pump are austenitic stainless steel, or equivalent corrosion resistant material. The pumps are operated manually from a local station.

Spent Fuel Pool Filter

The spent fuel pool filter removes particulate matter larger than 5 microns from the spent fuel pool water. The filter element is disposable. The vessel shell is austenitic stainless steel.

Spent Fuel Pool Strainer

A stainless steel strainer is located at the inlet of each fuel pool cooling suction line for removal of relatively large particles which might otherwise clog the spent fuel pool demineralizer or damage other components in the system.

Spent Fuel Pool Demineralizer

The demineralizer is sized to pass up to 150 gpm of the cooling flow to provide adequate purification of the fuel pool water for unrestricted access to the working area and to maintain water clarity.

Spent Fuel Pool Skimmer

A spent fuel pool skimmer pump, strainer, filter, and two skimmers are provided for surface skimming of the spent fuel pool water. This subsystem maintains the needed clarity for visual observations of the pool water.

Refueling Water Purification Pump

The shared refueling water purification pump provides for circulation of refueling water from either the refueling canal or the refueling water storage tank for purification. Its wetted surfaces are austenitic stainless steel.

Refueling Water Purification Filter

The refueling water purification filter removes particulate matter larger than 5 microns from the refueling water. The filter element is disposable.

Spent Fuel Pool Cooling System Valves

Manual stop valves are used to isolate equipment and manual throttle valves provide flow control. Valves in contact with spent fuel pool water are austenitic stainless steel or equivalent corrosion resistant material.

Spent Fuel Pool Cooling System Piping

All piping in contact with spent fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pumps, heat exchangers, and filters to facilitate maintenance.

9.4.3 DESIGN EVALUATION

Availability and Reliability

The availability of two-cooling trains allows prolonged outages of either cooling loop.

Leakage Provisions

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A purification loop is provided for removing these fission products and other contaminants from the water.

Incident Control

The most serious failure of this system would be complete loss of water in the storage pool. To protect against this possibility, the spent fuel pool cooling connections enter near the water level so that the pool cannot be gravity-drained.

Malfunction Analysis

Failure analyses of system pumps, heat exchangers and valves are presented in Table 9.4-3.

9.4.4 TESTS AND INSPECTIONS

The active components of the system are in continuous use during normal plant operation. The spend fuel pit pumps are periodically tested in accordance with the requirements of the applicable edition of the ASME OM Standards. Additionally, periodic visual inspections and preventive maintenance are conducted following normal industry practice.

TABLE 9.4-2 (cont'd.)

Spent fuel pool skimmer pump

Number	1 (Shared)
Design pressure, psig	50
Design temperature, °F	200
Design flow rate, gpm	100
Minimum developed head, ft.	50
Temperature of pumped fluid, °F	75 - 150
Fluid	Spent fuel pool water
NPSH, ft. (available/required)	30/2
Material	Austenitic Stainless Steel

Refueling water purification pump

Number	1
Design pressure, psig	600
Design temperature, °F	200
Design flow rate, gpm	Nom. 100, Max 150
Minimum developed head, ft.	130
Fluid	Refueling water
NPSH, Ft. (available/required)	@ 100 gpm 30/5, @ 150 gpm 43/7
Material	Austenitic stainless steel

Spent fuel pool demineralizer

Number	1 (Shared)
Type	Flushable
Vessel design pressure, psig Internal -	200
External -	15
Vessel design temperature, °F	250
Design flow rate, gpm Maximum	100
Minimum D/F	10
Normal flow, gpm	100, Max 150
Normal operating temperature, °F	120
Normal operating pressure, psig	250
Resin type	anion and cation

TABLE 9.4-2 (Cont'd)

Spent fuel pool filter

Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass/resin)
Internal design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	Nom. 100, Max. 150
Filtration requirement	98% retention of particles above 5 micron

Spent fuel pool skimmer filter

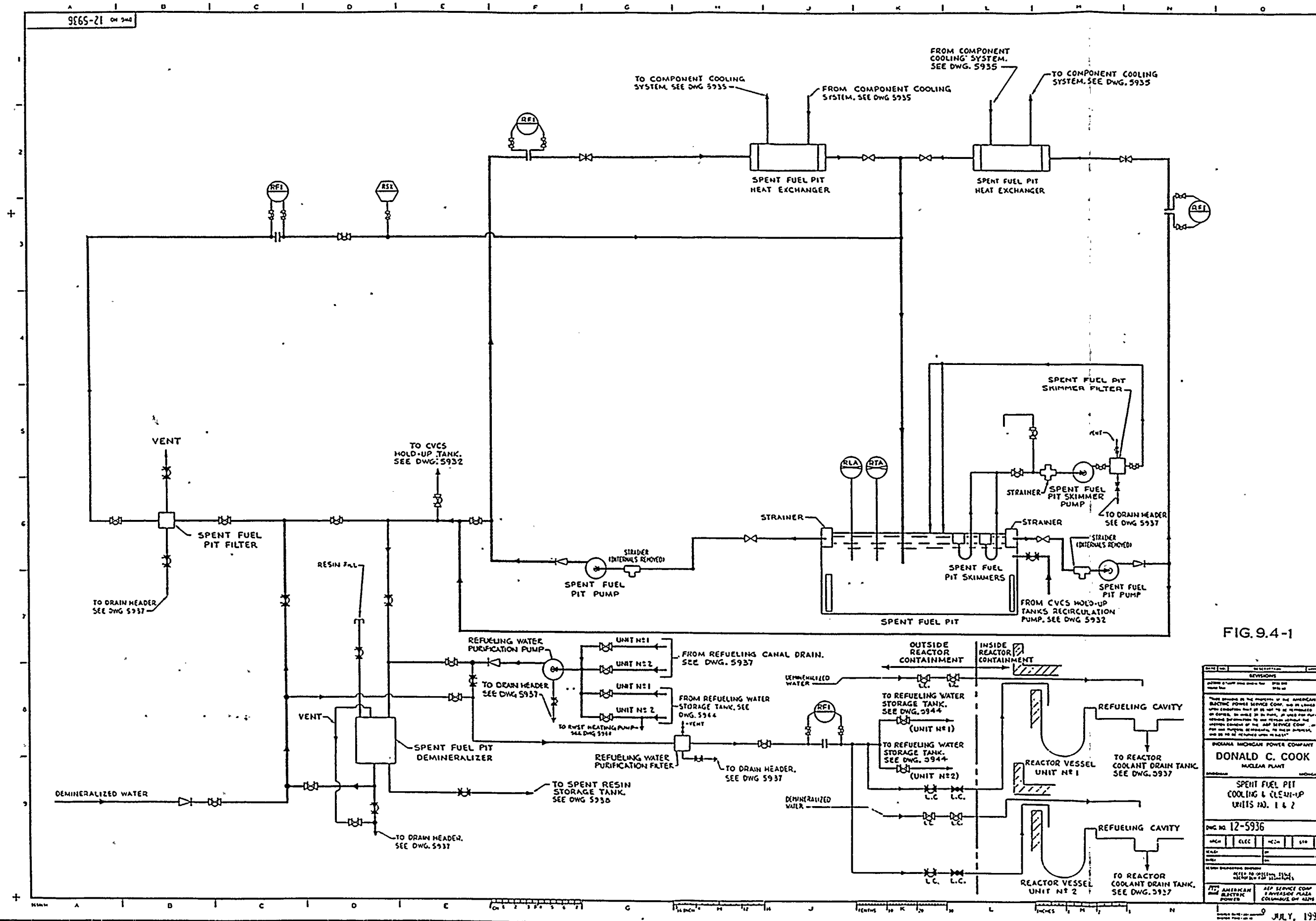
Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass/resin)
Internal design pressure, psig	200
Design Temperature, °F	250
Design flow rate, gpm	150
Filtration requirement	98% retention of particles above 5 micron

Refueling water purification filter

Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass/resin)
Internal design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	Nom. 100, Max. 150
Particle size retained, minimum, micron	5

Spent fuel pool strainer

Design flow rate, gpm	2300
Fluid	Borated demineralized water

[illegible]

The Component Cooling System, shown in Figure 9.5-1, is duplicated for each unit. The only shared piece of equipment is the maintenance spare Component Cooling pump installed in the Unit 1 area. The miscellaneous service train can be fed from either safeguards train.

9.5.1 DESIGN BASES

The system is designed to: a) remove residual and sensible heat from the Reactor Coolant System, via the Residual Heat Removal System, during plant shutdown; b) cool the spent fuel pool water and the letdown flow to the Chemical and Volume Control System during power operation; c) provide cooling to dissipate waste heat from various primary plant components, and d) provide cooling for safeguards equipment.

The system design provides radiation monitors for the detection of radioactivity entering the system from the Reactor Coolant System and its associated auxiliary systems, and includes provisions for isolation of system components.

All piping and components of the Component Cooling System have been designed to the applicable codes and standards listed in Table 9.5-1. Component cooling water contains a corrosion inhibitor to protect the carbon steel piping and equipment.

9.5.2 SYSTEM DESIGN AND OPERATION

The Component Cooling Water (CCW) System provides cooling for the following heat sources:

Safeguards Train

- a. Residual Heat Removal Heat Exchanger
- b. Centrifugal Charging Pump Gear, Lube Oil, and Seal Heat Exchangers
- c. Safety Injection Pump Seal and Lube Oil Heat Exchangers
- d. Residual Heat Removal Pump Seal Heat Exchangers
- e. Containment Spray Pump Seal Heat Exchangers

Miscellaneous Services Train

- a. Sample Heat Exchangers
- b. Reciprocating Charging Pump Bearing and Fluid Drive Heat Exchangers
- c. Spent Fuel Pit Heat Exchanger
- d. Waste Gas Compressor and Seal Water Heat Exchangers
- e. Reactor Coolant Pump Seal Water Heat Exchanger
- f. Letdown Heat Exchanger
- g. Boric Acid Evaporator Heat Exchangers
- h. Steam & Feedwater Containment Penetration Heat Exchangers
- i. Excess Letdown Heat Exchanger
- j. Reactor Support Coolers
- k. Reactor Coolant Pump Thermal Barrier Heat Exchanger
- l. Reactor Coolant Pump Motor Upper Bearing Oil Cooler
- m. Reactor Coolant Pump Motor Lower Bearing Oil Cooler
- n. 15 GPM Waste Evaporator Heat Exchangers
- o. Containment Air Recirculation Fan Motor Coolers

CCW minimum flow requirements for normal operation, LOCA injection and recirculation phases and cool down are tabulated in Table 9.5-2.

The CCW system is arranged in three flow circuits, two parallel safeguards equipment trains, and one miscellaneous services train which can be served by either of the safeguards trains.

Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system between the reactor coolant and the service water system and insures that any leakage of radioactive fluid from the components being cooled is contained within the plant. The surge tank accommodates expansion and contraction, and ensures a continuous component cooling water supply. Because this tank is normally vented to the auxiliary building atmosphere, a radiation monitor is provided in the supply piping to each component cooling heat exchanger. These monitors actuate an alarm and close the surge tank vent valve when the radiation level reaches a preset level above the normal background.

The Component Cooling System consists of two component cooling pumps, two component cooling heat exchangers, one surge tank and associated piping and valves to serve each unit. One pump and heat exchanger, with associated equipment, forms a 100% train. An additional pump is provided as an installed maintenance spare for either unit and is located in a cross tie header between the Unit 1 and 2 systems. The piping and valve arrangement is such that the maintenance spare can supply water to any one of the four trains, after the electrical controls have been transferred to it from the affected train.

One pump and one heat exchanger are required for the removal of residual and sensible heat from the reactor coolant system via the residual heat removal system during the cooldown of one unit. Full power operation of one unit, including cooling of a spent fuel pit heat exchanger, likewise requires one pump and one heat exchanger. Therefore, the remaining train serves as a standby and can be placed in service, if required, to increase system capability. Provision is made to add makeup to the system through lines connected to the surge tank.

The operation of the system is monitored with the following instrumentation:

- a) Temperature recorder and alarm in the outlet lines for each of the component cooling heat exchangers
- b) A pressure and flow indicator in the supply line to each of the component cooling heat exchangers
- c) A radiation monitor in the supply lines to the component cooling heat exchangers
- d) Flow indicators and/or alarms located in the discharge lines of the major heat exchangers served by the system
- e) Temperature indicators and/or temperature test points located in the discharge lines of the major heat exchangers served by the system.

In the event of a loss of coolant accident, one pump and one heat exchanger are capable of fulfilling system requirements. Following a LOCA, both trains receive an automatic start signal. Cooling water for the component cooling heat exchangers is supplied from the Essential Service Water System (Chapter 9) insuring a continuous source of cooling medium.

9.5.3 COMPONENTS

Component Cooling System component design data are listed in Table 9.5-3.

Component Cooling Heat Exchangers

The component cooling heat exchangers are of the shell and tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. The shell side is of carbon steel and the tubes are of arsenical copper.

Component Cooling Pumps

The component cooling water pumps which circulate water through the component cooling water loops are horizontal, centrifugal units and motor driven. The motors receive electric power from normal and emergency sources.

Component Cooling Surge Tank

The component cooling water surge tank accommodates changes in component cooling water volume and is constructed of carbon steel. In addition to piping connections at each pump's suction, the tank is provided with a means of adding a chemical corrosion inhibitor to the component cooling loop. The tank is internally divided (baffled) to form, in effect, two compartments. This arrangement provides redundancy for a passive failure during recirculation phase following a LOCA.

Valves

The valves used in the component cooling loop are constructed of carbon steel with the internals upgraded to stainless steel as needed during repairs. Since the component cooling water is normally not radioactive, special provisions to prevent leakage to the atmosphere are not provided. Relief valves are provided for lines and components that could be pressurized beyond their design pressure by improper operation or malfunction.

The relief valves on the component cooling water lines downstream from each reactor coolant pump thermal barrier are designed to relieve excessive pressure that may be caused by over heating. The relief valve set pressure equals the design pressure of the particular segment of piping between the upstream check valve and downstream motor-operated discharge valves.

The relief valves on the cooling water lines downstream of the sample, excess letdown, seal water, spent fuel pit and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated and high temperature liquid flows through the tube side. The set pressure is less than or equal to the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water that would enter the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The set pressure assures that the design pressure of the component cooling system is not exceeded. The discharge of this valve is directed to the waste holdup tank.

The component cooling water surge tank vent-overflow line, which is open to the auxiliary building atmosphere, is equipped with an air-operated valve that will close automatically if radiation is detected in the system. A vacuum breaker valve is also provided to prevent collapsing this tank in the event of a large loss of water in the system.

Piping

The component cooling loop piping is carbon steel with flanged joints and connections at components which might require removal for maintenance. All other joints are welded. One exception to the carbon steel is that portion of the piping between the double check valves and the motor-operated discharge isolation valves for the reactor coolant pump thermal barrier cooling which is stainless steel.

9.5.4 SYSTEM EVALUATION

Availability and Reliability

The component cooling pumps, heat exchangers, and associated valves, piping and instrumentation are located outside of the containment and are therefore available for maintenance and inspection during power operation. Replacement of a pump, or maintenance on a heat exchanger is practical while redundant units are in service. Sufficient cooling capability is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safeguards equipment.

Incident Control

If outleakage occurs anywhere in the Component Cooling System, including a non-seismic I component served by the Miscellaneous Service Train, detection is accomplished by falling level in the surge tank. The surge tank is equipped with a low level alarm that annunciates in the control room. Level alarms from the sumps to which this water will drain, also serve as leak indicators.

The leaking portion of the system is then shut down and isolated and the backup train is put in operation. To minimize the possibility of leakage from piping, valves, and equipment, welded construction is used wherever possible.

For leakage into the Component Cooling System, a high level alarm is provided at the surge tank.

The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the chemical and volume control, residual heat removal, sampling or the spent fuel pool cooling system or from a leak in a cooling coil for the thermal barrier cooler on a reactor coolant pump.

The detection of this contamination is by a radiation monitor located in the component cooling water supply to each of the component cooling water heat exchangers.

Component cooling water flow at a reduced rate is automatically established to the residual heat removal heat exchanger at the safety injection signal. Since the thermal demand on this heat exchanger is minimal at this time, full design component cooling water flow is not required. When it has been established that both component cooling water pumps have been started, full design flow will be established to the residual heat removal heat exchangers.

The component cooling water lines to and from the reactor support coolers and the excess letdown heat exchanger have valves outside the containment wall which are automatically closed on the Phase A isolation signal.

If normal seal water supply is unavailable to the reactor coolant pumps, the cooling water to the RCP thermal barriers should be available to assure that there will be no mechanical damage to the pump. Therefore, isolation valves for the component cooling water for this service are not automatically closed until a Phase B (containment spray) containment isolation signal is received. The cooling water supply line to the reactor coolant pumps contains two remote-operated valves in series outside the containment wall. The return lines from the thermal barriers and RCP motor bearings each have two remote-operated valves in series outside the containment wall. These redundant valves assure the ability to isolate this circuit if a leak

is detected. Leak detection is accomplished by flow alarms and indicators in the supply and return lines of this circuit.

Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the component cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Malfunctions Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 9.5-4.

9.5.5 MINIMUM OPERATING CONDITIONS

Minimum operating conditions are given in the technical specifications.

9.5.6 TESTS AND INSPECTIONS

Components of the Component Cooling System are tested in accordance with the requirements of ASME B&PV Code Section XI and, beginning with the 3rd 10 year interval ISI program, pump and valve tests are in accordance with ASME O&M Standards and NUREG-1482. Containment isolation valves will be tested periodically in accordance with procedures established in Chapter 5. Periodic visual inspection and preventative maintenance are conducted following normal industry practice.



TABLE 9.5-1

COMPONENT COOLING SYSTEM CODE REQUIREMENTS

Component cooling heat exchangers

ASME B&PV Code
Section VIII
1968 Edition

Component cooling surge tank

ASME B&PV Code
Section VIII
1968 Edition

Component cooling loop piping and valves

USAS B31.1
1967 Edition*

Pressure containing components (or compartments of components) through which reactor coolant circulates at pressures and temperatures significantly less than the reactor operating conditions at rated power, will comply with the following codes:

- a. System Pressure Vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C.
- b. System Valves, Fittings and Piping - USAS B31.1. 1967 Edition*.

*Repairs and replacements for piping are conducted in accordance with ASME Section XI.

TABLE 9.5-2
COMPONENT COOLING WATER SYSTEM
MINIMUM FLOW REQUIREMENTS PER TRAIN (GPM)

<u>Service</u>	<u>NORMAL OPERATION</u>	<u>LOCA INJECTION</u>	<u>LOCA RECIRCULATION</u>	<u>COOLDOWN</u>
<u>Safeguard Train¹</u>				
RHR Heat Exchanger	-	-	4950	4950
CCP PP Hx	31	31	31	31
SI PP Hx	-	20	20	20
RHR PP Hx	-	5	5	5
CTS PP Hx	-	3	3	3
Subtotal	31	59	5009	5009
<u>Miscellaneous Train</u>				
BA Evaporator	1442 ¹	-	-	-
SFP Hx ⁴	2980	-	-	-
Waste Gas Compressors	42.5	-	-	42.5
Sample Coolers (U1/U2)	139/169 ¹	-	-	139/169 ¹
Post Accident Sampling System ¹	-	-	44	-
Letdown Hx ⁴	984 ¹	-	-	984 ¹
Seal Water Heat Exchanger	199	-	-	199
Ctmt. Pen. Cooling	300	-	-	300
CEQ Fan Motors ¹	-	-	15	-
RCP Motors	404	-	-	404
RCP Thermal Barrier Hxs	140	-	-	140
Reactor Support Ctrs	40	-	-	40
Subtotal (U1/U2)	6670.5/6700.5	59/59	59/59	2248.5/2278.5
Totals (U1/U2)	6701.5/6731.5	59/59	5068/5068	7257.5/7287.5

- Notes: 1. The flows shown reflect the use of one safeguard's train. The second safeguard train may be placed in service provided the necessary equipment is operable. Single train operation results in minimum safeguard's requirements and a minimum cooldown.
2. For LOCA Recirculation only one CEQ fan is required. An analysis was performed which determined acceptable performance at a reduced flow of 15 gpm.
3. The 44 gpm flow is based on the use of 3 model QC-563 (10 gpm ea.) and 1 model QC-501 (14 gpm) sample coolers.
4. SFP Hx is assumed to be on the non-accident unit.
5. These flows represent the maximum flows; they may be significantly reduced as necessary to control process temperatures.
6. The Letdown Hx is assumed to be inservice. The excess letdown Hx is placed inservice if the letdown Hx is unavailable. The excess letdown Hx's design flow rate is 230 gpm.

Typical of the analyses performed on samples are boron concentration, fission product radioactivity level, dissolved gas content, and corrosion product concentration. In addition, local sample points are provided at various locations outside the reactor containment for occasional sampling of other systems. These are not considered part of the sampling system. Analytical results are used to regulate boron concentration, evaluate fuel element integrity, evaluate mixed bed demineralizer performance, regulate additions of corrosion controlling chemicals and monitor primary and secondary water purity. Except for the steam generator blowdown sampling, the NSS is designed to be operated manually and intermittently for conditions from full power operation to cold shutdown.

Gamma spectrometric analyses of the liquid primary coolant samples are performed, where possible, without further preparation of the sample. In instances involving separation techniques, the time delay is dependent upon the particular component of interest. For normal routine analyses of liquid samples, including non-radioactive species, completion can usually be accomplished within 4 hours.

For gaseous components of primary coolant, liquid samples are collected in pressure sampling vessels and degassed in the laboratory according to detailed plant procedures. Gas samples are then counted utilizing gamma spectrometry. In most cases, this can be accomplished within 1-2 hours after sampling.

The NSS incorporates means of purging a sample line for a sufficient period of time to ensure collection of a representative sample. Local flow, temperature and pressure measuring devices have been included in the nuclear sampling room to monitor these parameters.

Liquid samples are cooled and depressurized. Temperatures are maintained high enough after cooling to prevent solids from precipitating out. In addition, sample runs are kept to the minimum practicable and all sample lines and coils are constructed of materials compatible with coolant chemistry.

The reactor coolant sample points which are normally inaccessible and which require frequent sampling are permanently piped to a sampling room. The sample lines originating inside the reactor containment have remotely-operated isolation valves outside the containment. A delay coil located inside the containment provides for decay of short-lived radioactive isotopes present in the reactor coolant system samples. With the delay coil, it takes 2 1/2 to 3 1/2 minutes for a sample increment to reach the sampling room. The samples are cooled as they flow through the sample heat exchangers and the pressure is reduced by pressure-reducing needle valves. The sample flow is directed to the volume control tank through a purge line until sufficient volume has passed to obtain a representative sample. A portion of the flow is then diverted to the sample sink where the sample is collected. Reactor coolant gas samples and pressurizer steam samples are collected in sample vessels.

Liquid samples originating upstream and downstream of the Chemical and Volume Control System mixed bed demineralizer pass through a common sample line at the sample sink. The sample from the volume control tank gas space of the Chemical and Volume Control System is also collected in a sample vessel.

The Reactor Components and Fuel Handling System provides a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. Each unit has its own fuel handling equipment within its containment and an independent fuel transfer mechanism. Other fuel handling equipment used in and around the spent fuel pool is shared.

The system is designed to minimize the possibility of mishandling or of maloperations that could cause fuel damage and potential fission product release.

The Reactor Components and Fuel Handling Systems consist basically of:

- a) The reactor and refueling cavities.
- b) The transfer canal and the spent fuel pool, which are accessible to operating personnel.
- c) The Fuel Transfer System, which consists of an underwater conveyor, RCC changing fixture, new and spent fuel handling crane, manipulator crane, transfer tube, and new fuel elevator.
- d) Fuel racks.
- e) Polar crane.

Prevention of Fuel Storage Criticality

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

During reactor vessel head removal, and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core to a $K_{eff} = 0.95$. Refueling water boron concentration is verified in accordance with technical specification surveillance requirements to ensure the proper shutdown margin.

The new fuel storage racks are designed so that it is impossible to insert assemblies in other than the storage cells in the racks, thereby maintaining separation. The poisoned high density spent fuel storage racks are designed such that no assembly can be placed any closer to another assembly than that required by the critical analysis to maintain the required K_{eff} of ≤ 0.95 . The new fuel storage rack accommodates 144 fuel assemblies, over two-thirds of a core, and a spent fuel storage pit accommodates 3613 fuel assemblies, slightly more than eighteen and one-half cores, plus the required spent fuel shipping cask area. Borated water is used to fill the spent fuel storage pool and maintain it at a concentration to match that used in the refueling cavity and refueling canal during refueling operations. (The fuel is stored in a vertical array with sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.95$ [even if unborated water is used to fill the pool.])

The new fuel storage vault (NFSV) rack analysis is based on maintaining $K_{eff} \leq 0.95$ under full water density conditions and ≤ 0.98 under low water density (Optimum moderation) conditions.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective neutron multiplication factor, K_{eff} , of the NFSV when flooded with full density water will be less than 0.95 as recommended by ANSI 57.3-1983 and NRC guidance. Furthermore, the effective neutron multiplication factor, K_{eff} , of the NFSV under optimum moderation (aqueous foam) conditions will be less than 0.98 as recommended by NUREG-0800.

Fuel assemblies and enrichments up to 4.55 w/o ^{235}U can be safely stored in the NFSV. The maximum 95/95 K_{eff} determined for full water density flooding is 0.9495 and the maximum 95/95 K_{eff} determined for optimum moderation flooding is 0.8974. Based on these previously calculated K_{eff} values, the acceptance criteria are met for both full and optimum water density flooding of the new fuel storage racks. A maximum nominal enrichment of 4.95 weight percent U-235 for Westinghouse fuel types is acceptable provided that sufficient integral fuel burnable absorber is present in each fuel assembly stored in the new fuel storage racks such that the maximum reference fuel assembly k_{∞} is less than or equal to 1.4857 at 68°F.

An exemption from the requirements of 10 CFR 70.24, which requires a criticality monitoring system and emergency procedures for the handling and storage of unirradiated fuel, has been granted. The basis for the exemption is that inadvertent or accidental criticality will be precluded through compliance with the Cook Technical Specifications, the geometric spacing of fuel assemblies in the new fuel storage facility and spent fuel storage pool, and administrative controls imposed on fuel handling procedures.

Detailed information is available for use by refueling personnel. These instructions, safety limits and conditions and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incidents can occur during the refueling operations that could result in a hazard to public health and safety.



fuel provides an effective, economic and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Boric acid is added to the water to further ensure subcritical conditions during refueling.

In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pool, fuel is removed from the transfer system and placed in the poisoned high density storage racks with a long manual tool suspended from an overhead hoist. After a sufficient decay period, the fuel is expected to be removed from storage and loaded into a cask for removal from the site or continued on-site dry storage, unless it is desired to retain them in the spent fuel pool. Up to 3420 fuel assemblies may be stored and still retain capacity to store up to an additional 193 fuel assemblies which corresponds to a complete unloading of one unit. New fuel assemblies are received and eventually transferred to the spent fuel pool or new fuel storage vault for temporary storage or to the reactor core. The new fuel storage vault is sized for storage of the fuel assemblies and other nuclear fuel components normally associated with the replacement of up to 144 assemblies for either or both units. New fuel is loaded into the reactor by either lowering it into the refueling canal from the new fuel storage vault and taking it through the transfer system, by transferring it from the spent fuel pool via the transfer system or by transferring it directly from the receipt canister via the transfer system.

The refueling cavity, refueling canal and spent fuel storage pool are reinforced concrete structures with seam-welded stainless steel plate liners. These Class I structures are designed to withstand the anticipated earthquake loadings and to prevent liner leakage even in the event the reinforced concrete develops cracks.

Refueling Operation

The refueling operation follows a detailed procedure which provides a safe, efficient refueling operation. The following significant points are assured by the refueling procedure:

- 1) The refueling water and the reactor coolant contain approximately 2,400 ppm boron, or a boron concentration sufficient to ensure that the $K_{eff} \leq 0.95$, whichever provides more margin to criticality.
- 2) The water level in the refueling canal is maintained high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core. This water also provides adequate cooling for the fuel assemblies during transfer operations.
- 3) The handling of heavy loads is controlled to reduce the possibility of damage to nuclear fuel and/or equipment that may be required to achieve safe shutdown and continued decay heat removal. This is more fully described in Section 12.2.

While one unit is being refueled, there are no restrictions on the operation of the other unit. Refueling of one unit does not affect the safety aspects of the other unit.

Refueling Procedure

Preparation

The following general tasks are required prior to refueling:

- The reactor has been subcritical for at least 168 hrs and cooled to ambient conditions. The basis for 168 hours of subcriticality is the maximum decay head load to ensure adequate heat removal capability in the spent fuel pool.
- A radiation survey is made. A procedure is followed to checkout the functioning and operability of radiation monitors important to refueling operations. This includes radiation monitors, both in the containment and in the auxiliary building spent fuel ventilation system.

- The reactor missile shields and the control rod drive mechanism (CRDM) seismic restraint are removed. |
- The bulkhead sections between the reactor cavity and the refueling cavity are removed. |
- CRDM cables and cooling air ducts are disconnected and removed. |
- Reactor vessel head insulation and instrument leads are removed. |
- The reactor vessel head nuts are loosened with the hydraulic tensioner. |
- The reactor vessel head studs are removed. |
- The canal drain holes are plugged and the fuel transfer tube flange is removed. |
- Checkout of the fuel transfer device and manipulator crane is started. |
- Guide studs are installed in three stud holes and the remainder of the stud holes are plugged. |
- Install the reactor vessel to cavity seal. |
- Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed. |
- The reactor vessel head is unseated and raised. |
- The lift of the reactor vessel head is stopped at several specified heights to check that: |
 - the reactor head is level
 - the head is not binding on the guide studs

- the protective sleeves for the instrument port seal assemblies are not being lifted.
- At the appropriate reactor vessel head lift height, a check is made that the RCCA drive shafts are clear of the CRDM housings, and are not being lifted with the head. The reactor vessel head is lifted to clear and is taken to its storage pedestal.
- The reactor cavity and refueling canal are flooded with water to the level required for unlatching the RCCA drive shafts.
- The control rod drive shafts are unlatched.
- The reactor vessel internals lifting rig is lowered into position and latched to the support plate.
- The reactor cavity and refueling canal are flooded with water to the level required for refueling.
- The reactor vessel upper internals are lifted out of the vessel and placed in the underwater storage rack.
- The core is now ready for refueling.

Refueling

Refueling is performed with the manipulator crane, following the general tasks listed below.

- Spent fuel, which is to be discharged, is removed from the core and placed on the fuel transfer conveyor for removal to the spent fuel pool.
- Partially spent fuel is relocated within the core or moved to the spent fuel pool.
- New fuel assemblies and the removed, partially spent fuel assemblies to be used in the upcoming cycle of operation are transferred from the new fuel storage area, new fuel receipt canister or the spent fuel pool into the refueling canal and are brought through the transfer system and loaded into the core.

- Whenever fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

If a transfer of the rod cluster control (RCC) elements between fuel assemblies is required and the reactor core is not completely offloaded to the spent fuel pool, the assemblies can be taken to the RCC change fixture to exchange the RCC elements from one assembly to another. Should a full core offload be performed during the refueling, the RCC exchange can be performed in the spent fuel pool with a long handled tool. Such an exchange is required whenever a spent fuel assembly containing RCC elements is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangements. If the previous core design contained burnable poison rod (BPR) elements, then fuel assemblies with BPR elements are moved to the spent fuel pool where the BPR element is removed using the burnable poison handling tool, and a thimble plugging device is inserted to restrict the flow through the guide thimbles. Such an operation is necessary whenever a fuel assembly containing a BPR element is to be reinserted into the core.

Reactor Reassembly

The following general tasks are required following refueling:

- The fuel transfer car is parked and the fuel transfer tube isolation valve is closed.
- The reactor vessel internals package is replaced in the vessel. The reactor vessel internals' lifting rig is removed to storage.
- The control rod drive shafts are relatched to RCC elements.
- The manipulator crane is parked.
- The old seal rings are removed from the reactor vessel head, the grooves cleaned and new rings installed.
- The reactor vessel head is picked up and positioned over the reactor vessel.
- The water level is lowered and the reactor vessel head is lowered.
- The refueling cavity and refueling canal are completely drained and the flange surface is manually cleaned.
- The reactor vessel head is seated.
- The guide studs and the stud hole plugs are removed.
- The head studs are replaced and the head nuts are retorqued.
- The canal drain holes are unplugged and the fuel transfer tube flange is replaced.

- Electrical leads and cooling air ducts are reconnected to the CRDM's.
- Vessel head insulation, CRDM seismic restraints, and instrumentation leads are replaced.
- Remove the reactor vessel to cavity seal.
- Control rod drives are checked.
- The reactor missile shield is picked up with the polar crane and replaced.
- Pre-operational tests are performed.

Major Structures Required for Refueling

Refueling Cavity

The refueling cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling.

The cavity is filled so that at least 23 feet of water is maintained over the reactor pressure vessel flange. The radiation at the surface of the water is limited to a level as low as reasonably achievable during those periods when a fuel assembly is transferred over the reactor vessel flange.

The reactor vessel flange is sealed to the reactor cavity by a Preferred Engineering mechanical seal which prevents leakage of refueling water from the refueling cavity. This seal is installed after reactor cooldown but prior to flooding the refueling cavity for refueling operations.

The floor and sides of the refueling cavity are lined with stainless steel. The refueling cavity has been designed to be within the stress and strain limitations of the ACI Code 318-63, using working stress design criteria for operating conditions, and ultimate strength design criteria for accident conditions. Analysis of the refueling cavity has been made using the AEP FRAME Program. The heat generation rates due to radiation in the primary concrete were calculated by using a point kernel analysis technique. In addition to the reactor core sources, the code considers the capture gamma and inelastic neutron scattering contributions outside the core, and within the concrete.

Refueling Canal

The refueling canal is a passageway extending from the refueling cavity to the inside surface of the reactor containment. The canal is formed by two concrete shielding walls which extend upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity to provide the greater depth required for the fuel transfer tipping device and the control cluster changing fixture located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. The canal is a stainless steel lined reinforced concrete structure.

The refueling cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools.

The refueling cavity and refueling canal are modeled as one unit; as a grid of beams and columns. The static and thermal loads are introduced as input at the node points of the gridwork. Seismic loading is entered using the acceleration responses determined from previous analyses. All stresses in a loading combination are combined algebraically. The seismic stresses are considered to be reversible in sign, so as to give maximum calculated combined stresses. The refueling cavity/refueling canal area is further checked for seismic condition by means of the FRAME Program dynamic routines.

Cask Drop Protection System

The proposed, but not installed*, cask drop protection system (CDPS) consists of a circular base plate attached to the bottom of the spent fuel shipping cask and a combination guide structure - dashpot assembly. The guide structure guides and restrains the falling cask in the event it is dropped, and the dashpot decelerates the cask to a low velocity to reduce the impact load on the floor of the pool to an acceptable value. The function of the base plate is to act as a piston within the dashpot.

To lower the cask into the pool, the cask with its base plate attached is moved from the base plate attachment area to a position over the center of the CDPS. The cask follows a particular path so that in the event that the cask is dropped at any point along this path, the cask will not tip into spent fuel pool.

Cask Decontamination Facilities

Once the spent fuel shipping cask has been loaded, it would be removed from the spent fuel pool and placed on a pad just beyond the pool for decontamination prior to shipment.

The pad has a stainless steel lined base and a curb is provided around it to prevent the water and solvents used during decontamination from spreading over the auxiliary building floor. Drains in the floor of the pad remove the decontaminants to the waste disposal system for processing.

New Fuel Storage

New fuel assemblies and new control rod clusters may be stored in an area adjacent to the spent fuel pool, whose location facilitates the unloading of new fuel assemblies from delivery trucks. This storage vault is designed to hold new assemblies in specially constructed racks. A total of 144 storage positions are provided. Prior

*System will be installed when the need arises.

to initial core loading for Unit 1 assemblies in excess of the number which could be accommodated in the new fuel storage area were stored in the dry spent fuel pool. For Unit 2, temporary storage facilities were established adjacent to the new fuel storage area.

Use of the new fuel storage vault is not required. New fuel may be loaded directly into the new fuel elevator from the new fuel shipping canister(s) if desired, for temporary storage in the spent fuel pool or for direct transfer to the appropriate refueling cavity for insertion into the reactor.

Major Equipment Required for Refueling

Reactor Vessel Stud Tensioner

Stud tensioners are used to make up the head closure joint.

The stud tensioner is a hydraulically operated (oil is the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs 120° apart. However, procedures exist that allow use of only two tensioners 180° apart, if necessary. One hydraulic pumping unit operates the tensioners which are hydraulically connected in parallel. The studs are tensioned to their operational load in two or three steps to prevent high stresses in the flange region and unequal loadings in the studs. An overstroke alarm is provided on each tensioner to alert the operator that a tensioner is about to reach maximum stroke. Charts indicating the stud elongation and load for a given oil pressure are included in the transient operating instructions. In addition, measurements of the elongation of the studs are performed after tensioning.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations.

All charcoal filter equipped air handling units in the auxiliary building and for the control rooms are provided with manual water spray deluge systems to extinguish the charcoal filter fire. Continuous strip thermistors provide detection and a high temperature alarm in the associated control room. A detection alarm also sends a signal to open the isolating valves in the auxiliary building supply header and automatically opens the charcoal filter system valve. The control valve to the affected charcoal filter water spray system is then manually opened to fight the fire.

Hydrogen tubes outside the auxiliary building are equipped with a water spray dry pilot deluge system similar to that provided at the office/service building hydrogen tubes.

Ionization fire detection is provided on each floor of the auxiliary building for general alarm of fire as follows:

- Elev. 573' a. Containment Spray and Residual Heat Removal Pump Cubicles
(Units 1 and 2)
b. Normally accessible common areas of the Auxiliary Building

- Elev. 587' a. Transformer Rooms (Units 1 and 2)
b. Sampling Room (common to both units)
c. Spray Additive Tank Room (common to both units)
d. Charging and Safety Injection Pump Cubicles (Units 1 and 2)
e. Drumming/Drum Storage (common to both units)
f. Normally accessible common areas of the Auxiliary Building

- Elev. 609' a. Access Control (common to both units)
and 612' b. AB and CD (EL 625'-10") Battery Rooms (Units 1 and 2)
c. El. 617' Valve Gallery (common to both units)
d. NESW Valve Gallery (Units 1 and 2)
e. Normally accessible common areas of the Auxiliary Building

- Elev. 633' a. New Fuel Storage Room (common to both units)
b. N-Train Battery Rooms (Units 1 and 2)
c. Normally accessible common areas of the Auxiliary Building

- Elev. 650' a. Control Room Equipment Rooms (Units 1 and 2)
b. Normally accessible common areas of the Auxiliary Building
c. Computer Rooms (Units 1 and 2) (High Voltage Detectors)

A combination of thermal and infrared detectors is provided in the Main Steam Valve Enclosures East, and a combination of ionization and infrared detectors is provided in the Main Steam Line Area of Units 1 and 2 at elevation 612'.

Reactor Containments

Containment cable trays, reactor coolant pumps and HVAC charcoal filters are equipped with continuous strip thermistor fire detection which will annunciate in the control rooms.

The HVAC charcoal filters have water spray deluge fire suppression systems and are actuated by the thermistor detection.

Reactor coolant pumps are equipped with preaction water spray systems, manually operated from the control rooms in the event of a lubricating oil fire. Additionally, the RCP motors are provided with an oil spillage control and retention system to preclude spreading oil from a pressure or gravity type leak.

Water supply to containment fire protection is from the non-essential service water system.

Low-Pressure Carbon Dioxide System

A 17-ton capacity low-pressure carbon dioxide system, located in the auxiliary building, is provided for automatic and/or manual protection of various areas as listed below. The amount of CO₂ in the system is sufficient to protect the largest single hazard in the plant. The CO₂ is stored in an insulated pressure vessel having an automatically operated refrigeration system. Operation of the CO₂ systems is annunciated and activates the control room alarm system.

The areas protected by the low-pressure CO₂ system and the type of fire detection are as follows:

1. Turbine Building

- a) Lubricating oil storage rooms Units No. 1 and No. 2. Manual (backup to Automatic Sprinkler System).
- b) Main turbine oil tank rooms Units No. 1 and No. 2. Manual (backup to Automatic Sprinkler System).

2. Auxiliary Building

- a) AB and CD emergency diesel generator rooms Units No. 1 and No. 2. Continuous-strip thermistor detection. (2 zones for each room)
- b) Diesel oil pump and valve station rooms Units No. 1 and No. 2. Continuous-strip thermistor detection.
- c) Electrical switchgear rooms Units No. 1 and No. 2.
 - 1. 4.16 kV switchgear rooms. Infrared and ionization detection.
 - 2. 4.16 kV/600 V transformers and engineered safety equipment rooms. Infrared and ionization detection.
 - 3. 4.16 kV/600 V transformers, control rod drive and inverter rooms. Infrared and ionization detection.
- d) Electrical switchgear room cable vaults Units No. 1 and No. 2. Infrared and ionization detection.
- e) Auxiliary cable vaults Units No. 1 and No. 2. Ionization detection.
- f) Control room cable vaults Units No. 1 and No. 2. Manual (backup to Halon 1301 systems).

g) Electrical penetration area cable tunnels Units No. 1 and No. 2.

1. Quadrant 1. Infrared and ionization detection.
2. Quadrant 2. Infrared and ionization detection.
3. Quadrant 3 north. Infrared and ionization detection.
4. Quadrant 3 middle. Infrared and ionization detection.
5. Quadrant 3 south. Infrared and ionization detection.
6. Quadrant 4. Infrared and ionization detection.

3. Carbon dioxide hose reel stations are provided for manual fire fighting in the auxiliary building, switchgear rooms, and at the entrances to the control rooms, diesel generator rooms, and electrical penetration area cable tunnels.

Halon 1301 Systems

Halon 1301 systems are provided for automatic fire protection in various areas of the plant. Locations of these systems include the control room cable vaults, the computer rooms and underfloor, control points for the plant security system, and as previously mentioned, the TSC computer room, TSC console room, and TSC UPS inverter room. Actuation is by two zones of ionization detection for each system.

Control Room Fire Protection

The control rooms are equipped with portable fire extinguishers. Detection systems of the ionization type are installed. The control rooms are occupied at all times by operators who have been trained in fire extinguishing procedures. All areas of the control rooms are accessible for fire fighting.

The Control Air System includes sufficient capacity to supply the control and instrument air requirements with the equivalent of approximately 5 minutes of control air output after a loss of power incident. Additionally, certain vital control valves within the containment are each equipped with a local receiver tank with capacity to activate the valve. Also, the control air compressors can be supplied with electric power from both normal and emergency sources so that a supply of compressed air can be made available in any foreseeable circumstance.

The Compressed Air System includes normal accessory equipment such as dryers, filters, storage receivers, after-coolers, and safety valves in addition to the compressors. A descriptive summary of the major pieces of equipment in the system is included in Table 9.8-2.

9.8.2.3 Design Evaluation

The Compressed Air System is designed to provide a reliable source of compressed air for all plant uses.

During normal operation, either one of the two plant air compressors is capable of supplying the entire demand of both plant and control-instrument air requirements for both units.

Low plant air header pressure will automatically start the second plant air compressor. A lower control air header pressure in either unit will automatically start that unit's control air compressor. A further degradation in the plant air header pressure will cause the four air-operated isolation valves located in the plant air ring header to close, thus completely isolating the control air systems of the two units.

This system arrangement allows either unit's plant air system to be removed from service should that become necessary while allowing the remainder of the plant air system as well as both unit's control air system to continue in operation. This isolation can be achieved by closing the two air-operated isolation valves which serve the effected unit.

In this manner, each unit still retains a backup supply of compressed air from its own control air compressor.

A failure in the control air system of one unit will not affect the control air system of the other unit because check valves in the control air off-takes from the plant air header prevent back flow.

9.8.2.4 Tests and Inspections

The "Compressed Air System" pre-operational test procedure verified the system's automatic start sequences, the isolation of the crosstie headers between each unit and the interlocks which assure proper operation of the equipment. Performance tests are performed on both the plant air and control air compressors in which the capacity, pressure and temperature of the compressed air are measured. The surge point for the plant air compressors is determined as is the load and unload pressure of the control air compressors. Individual components such as after-coolers, pre- and after-filters, and air dryers are also tested to assure proper operation of the system.

9.8.3 SERVICE WATER SYSTEMS

The Service Water Systems are shared by both units.

9.8.3.1 Design Basis

The Service Water Systems supply cooling water to various heat exchangers in both the primary and secondary systems of each unit. Provisions are made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety both during normal operation or under accident conditions. Sufficient redundancy of piping and components is provided to insure that cooling is maintained to vital services at all times.

9.8.3.2 Description

Service water is provided by two independent systems, the Non-Essential Service Water System shown in Figures 9.8-4, 9.8-5 and 9.8-6 and the Essential Service Water System shown in Figure 9.8-7. Each system consists of four operational pumps, each with a duplex automatic backwashing strainer in its discharge line, and associated piping and valves. The design parameters of these components are listed in Table 9.8-3.

Non-Essential Service Water System

The Non-Essential Service Water System supplies cooling water to the following components. Turbine oil coolers, air compressors, the upper and lower containment ventilation units, reactor coolant pump motor air coolers, and miscellaneous services, none of which are required for plant safety related functions. Cooling requirements are given in Table 9.8-4. Three of the four pumps are normally operated to provide service water to the two units with one pump held in standby. All pumps are able to take suction from either the Unit 1 or Unit 2 Circulating Water Intake Tunnels or discharge tunnels. The system discharges into either the Unit 1 or Unit 2 Circulating Water Discharge Tunnels. Thus, Non-Essential Service Water supply to both units is assured, even if the tunnels of one unit are out of service.

Following a loss of all off-site power, the non-essential service water pumps are automatically started as soon as the emergency diesel generator power becomes available. Under those conditions the pumps are primarily used to supply cooling water to the control air compressors in order to restore control air service. All motor-operated valves on the non-essential water systems are operated from the station battery system. Cross-ties between the pumps permits any one pump to supply the initial blackout requirements for both units.

The discharge strainers of the pumps are of duplex construction, with automatic backwashing. Each strainer is effectively two strainers in one casing with flow directed through one half, while slide gates block off the other half. When the strainer is in service and if it becomes dirty or clogged, a high differential pressure signal initiates a shift of the slide gates blocking the flow to the dirty basket and directing it through the clean basket. The dirty basket is then backwashed and is ready for re-use.

Essential Service Water System

The Essential Service Water (ESW) System supplies cooling water to the following components:

- a. Component Cooling Heat Exchangers
- b. Containment Spray Heat Exchangers
- c. Emergency Diesel Generators
- d. Auxiliary Feedwater System
- e. Control Room Air Conditioners

During normal operations essential service water is supplied continuously to the Component Cooling Heat Exchangers and the Control Room Air Conditioners while the Containment Spray Heat Exchangers and the Emergency Diesel Generators are supplied only when these systems are in operation. In addition, the essential service water system serves as back-up water sources to the auxiliary feedwater pumps for use when the condensate storage tank, the normal supply for the auxiliary feed-water system, is either empty or otherwise lost as a source of supply.

The system consists of four essential service water pumps, four duplex strainers and associated piping and valves. System piping is arranged in two independent headers, each serving certain components in each unit as follows:

For the detection of large leaks, the Essential Service Water System is equipped with flow and pressure alarms and/or indicators which will signify losses from the supply headers. In addition, flow indicators are located in the Essential Service Water lines for each Component Cooling and Containment Spray Heat Exchanger as well as each Diesel Generator. The header supply valves are remotely operated, facilitating isolation of the supply header or pump which has failed.

9.8.3.3 Design Evaluation

Non-Essential Service Water System

The Non-Essential Service Water System is not required for the maintenance of plant safety related functions in the event of an accident. During normal operation, the system remains functional even if one Unit is out of service and its circulating water tunnels are dewatered.

Essential Service Water System

The Essential Service Water System is designed to prevent any failure in its system from curtailing normal plant operation or limiting the ability of the engineered safeguards to perform their functions in the event of an accident. Since the Essential Service Water System is required for long term heat removal, it is designed to withstand a passive failure on a long term basis. Although it is not a design requirement, the Essential Service Water System has sufficient capacity to handle a LOCA on one unit and hot shutdown in the other considering the single failure criterion. Sufficient pump capacity is included to provide design service water flow under all postulated conditions. The headers are arranged such that even loss of a complete header does not jeopardize plant safety related functions. Table 9.8-6 gives a malfunction analysis of a pump, valve and strainer.

9.8.3.4 Tests and Inspections

System components were hydrostatically tested prior to station startup and are accessible for periodic inspections or tests during operation.

Electrical components, switchovers, and starting controls are tested periodically.

The essential service water pumps, valves and components are periodically tested in accordance with the applicable edition of the ASME OM Standards and NUREG-1482. Periodic testing of the non-essential service water pumps is conducted in accordance with normal industry practice.

9.9.1

GENERAL DESCRIPTION

The auxiliary building ventilation systems, shown in Figures 9.9-1 and 9.9-2, consist of:

- a. Engineered Safety Features Ventilation System (one per plant unit).
- b. Fuel Handling Area Ventilation System (one shared system).
- c. General Ventilation Systems (one per plant unit with crosstie).
- d. General Supply System (one per plant unit).

The auxiliary building is basically a five-level compartmented structure containing the auxiliary nuclear equipment for both units. All equipment handling radioactive fluids is located on the lower four levels of the auxiliary building. The fourth level also houses the two control rooms and the ventilation equipment.

The auxiliary building ventilation systems are designed to maintain temperatures in the various portions of the building within design limits for operation of equipment and for personnel access for inspection, maintenance and testing as required.

9.9.2

DESIGN BASES

Outside ambient conditions used for design purposes are 91°F summer dry bulb, 75°F summer wet bulb and -7°F winter dry bulb. Ventilation is based on limiting temperatures in all area to a predetermined maximum, generally 110°F, and heating is provided to maintain a 60°F minimum temperature.

All ventilation systems serving the auxiliary building are once-through systems. Supply air is introduced to the areas least likely to be contaminated, and exhausted directly from those with the greatest contamination potential. Additionally, the exhaust systems are of greater capacity than the supply systems, thus maintaining the area within the auxiliary building pressure boundary at a slightly negative pressure. The auxiliary building pressure boundary is the area within the auxiliary building which is maintained at a negative pressure by the HVAC system, as required for radiological control.

All exhaust air from the auxiliary building is directed to the unit vents. There is a vent for each unit. Each vent has radiation detectors for continuous monitoring of the exhaust air during release to atmosphere.

High efficiency particulate air filter cells are designed to remove as much as 99.97 percent of solid particulates of 0.3 micron mean diameter in size. Performance characteristics of the charcoal adsorbent provide for removal of as much as 99.9 percent of any entrained methyl iodide or iodine vapor. Supply and exhaust unit roughing filters have a NBS duct spot efficiency (Cottrell Precipitate) of 75%.

9.9.3 SYSTEM DESCRIPTIONS

9.9.3.1 Engineered Safety Features Ventilation

The enclosures for the engineered safety features equipment for both units are located in the lower three levels of the auxiliary building. (The containment spray heat exchanger and residual heat exchanger enclosures extend up into the fourth level with access into the enclosures from the third level only.) The enclosures for each unit's safety feature equipment are ventilated by two separate ventilation systems. The areas serviced by this system are: the containment spray pump enclosures, the residual heat removal pump enclosures, the safety injection pump enclosures, the residual

heat exchanger enclosures, the containment spray heat exchanger enclosures and the reciprocating and centrifugal charging pump enclosures. Figure 9.9-2 shows a flow diagram of the engineered safety features ventilation system and is typical for the system serving either unit.

The exhaust ventilation system is composed of two 25,000 cfm fan/ filter exhaust units (1 standby) which draw air from the auxiliary building through the equipment enclosures via a common vent shaft and discharge it to the unit vent. Each fan/filter unit is composed of a 100% capacity bank of roll media roughing filters, high efficiency particulate air filters, charcoal filters, and a 100% capacity exhaust fan. (There is a bypass on the charcoal filter bank.) This is a Class I ventilation system, therefore each fan/filter unit receives power from a separate engineered safeguards system bus which can be fed from the diesel bus and all components up to the connection to the unit vent are of Class I design.

Normally, one fan/filter unit operates continuously, directing the exhaust air through the roughing filter and high efficiency particulate air filter, bypassing the charcoal filter, and discharging it to the unit vent. This operation aids in the air distribution within the auxiliary building, isolates the atmosphere in the enclosures by inducing a draft through the entering portals and removes any heat generated within the enclosures.

In the event of a Phase B Isolation signal the standby fan/filter unit is energized and the charcoal filter bypasses are automatically closed and the air is directed through the charcoal filters in addition to the roughing and high efficiency particulate air filters. There are two independent air operated, fail-closed, dampers in the charcoal filter bypass. The charcoal filters can be placed in service when gaseous contamination warrants their operation.

Make-up air for the Engineered Safety Features Ventilation System is normally provided by the Auxiliary Building general supply. Partial make-up air can be provided during a loss of off-site power by three 15,000 cfm fans blowing outdoor air into the component cooling pump area of the Auxiliary Building (third level). The fans are Class I design and are provided primarily for use in emergency conditions.

Power for two of the 15,000 cfm fans can be provided by the Unit No. 1 diesel-generators, and for the third by Unit No. 2 diesel-generators.

These fans are dual purpose during an emergency, aiding in providing safe ambient temperature for the component cooling pump motors and providing partial make-up air for the engineered safety features ventilation system. The capacity of these fans is less than the engineered safety features ventilation system exhaust fans, thus ensuring a negative pressure within the auxiliary building pressure boundary during an emergency.

In addition to the engineered safety features ventilation system described above, the emergency diesel-generator rooms, the auxiliary feed pump enclosures, essential service water pump enclosures, safety related battery rooms, and the electric relay rooms are ventilated by systems powered by the emergency diesels. These systems include supply and/or exhaust fans sized to maintain design ambient temperatures within the various rooms and enclosures.

9.9.3.2 Fuel Handling Area Ventilation System

The fuel handling area is a shared facility and its ventilation system is therefore a shared facility consisting of an exhaust system and a supply system.

The fuel handling area exhaust system is composed of two 30,000 cfm fans (1 standby) which draw air through a common slot exhaust plenum along the north side of the spent fuel pool to direct it through a filter housing and discharges it to the unit No. 1 vent. The filter assembly is composed of roll media roughing filters, high efficiency particulate air filters and charcoal filters. There is a normally open bypass on the charcoal filters.

The Fuel Handling Area Supply Air System is made up of four supply units composed of fans, filters and steam coils. Two 11,000 cfm supply units are located in the western section of the Fuel Handling Area and two 2,500 cfm supply units are located in the eastern section of the Fuel Handling Area. Normally, all four supply units operate, drawing outside air through the steam coils and filters and discharging it into the fuel handling area. The air is drawn through the Fuel Handling Area into the exhaust plenum, and passed through the roughing and high efficiency particulate air filters by a continuously operating exhaust fan and discharged into the unit no. 1 vent. The combined capacity of the four supply units is less than that of a single exhaust fan, thus the Fuel Handling Area, as well as the entire space within the auxiliary building pressure boundary, are maintained at a slightly negative pressure.

In the event that the area radiation monitors in the Fuel Handling Area give a high radiation signal the charcoal filter bypass dampers are tripped closed thus passing the exhaust air through the charcoal filters prior to discharge to the vent. The Fuel Handling Area Supply Units are also tripped on the high radiation signal, thus ensuring a negative pressure within the space.

Operation of this system is the same for both summer and winter conditions. During winter operation the heating capacity of the supply units is supplemented by steam unit heaters located throughout the Fuel Handling Area.

9.9.3.3 General Ventilation System

All areas except the fuel handling area and the safeguard equipment areas are exhausted in each unit by a ventilation system consisting of two 50% capacity fans with roughing and high efficiency particulate air filters. There is no standby capacity in these systems, however there is a normally closed tie-line between the Unit No. 1 and Unit No. 2 exhaust units.

Normally, all fans operate at their design speed and direct their air flow through the filters and then to the unit vent. This operation induces a draft of 50 to 150 fpm through the entrance portals of the various enclosures thus removing any heat, vapors or particulate matter generated within the enclosures.

The hot laboratory chemical hood and cabinet exhaust fans, sample room sink hood and sample rack exhaust fans also discharge into this system. In the event of a high radiation signal from the vent monitor, the gas decay tank discharge is automatically closed.

The hot laboratory is located in the access control area of the Auxiliary Building. The access control area includes a radiation control office, a radiation protection supervisor's office, a chemical foreman's office, and other miscellaneous rooms which have no internal contamination potential, and a hot laboratory, chemical counting room, and R. P. counting room and decontamination area which are in a potential contamination area. The clean, or non-contaminated rooms are air-conditioned by a conventional, partial recirculation system which also pressurizes these areas. The potentially contaminated areas are air-conditioned by a once-through system with 100% fresh air supply of conditioned air which is exhausted to the auxiliary building general exhaust system.

The spray additive tank room houses the post-accident sampling system panel and is normally ventilated by the auxiliary building general exhaust system. When necessary, the spray additive tank room can be isolated from the auxiliary building general exhaust system and ventilated by the spray additive tank room filter unit and the spray additive tank room sample filter unit. The spray additive tank room filter unit consists of a roughing filter, HEPA filter, charcoal absorber, a second HEPA filter, and fan. This unit combines makeup air from outdoors with recirculated air to both pressurize the room and remove radiation contamination in order to maintain

the room habitable for plant personnel. The spray additive tank room sample filter unit exhausts air from the post-accident sampling system panel and discharges into the auxiliary building general exhaust system to prevent contamination from the panel being discharged to the room. The spray additive tank room sample filter unit consists of a canister HEPA filter, canister charcoal filter, and fan.

9.9.3.4 General Supply System

Normal make-up air from the outdoors for the engineered safety features ventilation system and the auxiliary building exhaust system is provided by the auxiliary building general supply system. This system consists of four 35,000 cfm capacity fans, 2 in each unit, with steam heating coils and air filters. There is no standby capacity in this supply air system.

Normally all fans operate at their design speed and direct outdoor air through the air filters and steam coils and into the building. The air is distributed throughout the building by the suction of the various exhaust ventilating systems.

The steam coils are activated during cold weather to temper the incoming air. Sufficient heat is added to the air flow to maintain the general ambient temperature of the building at or above the 60°F design minimum. Steam and/or electric heaters located in various areas of the building are used to ensure a satisfactory minimum temperature.

All ventilation system equipment is located within the building. During operation of either unit, all of its auxiliary building ventilation systems will be activated to "normal" operation. During shutdown of either unit, its auxiliary building ventilation systems may operate in part or in total to suit maintenance, inspection, testing, refueling, etc. conditions.

Continuous local monitoring of temperature and radiation is provided at appropriate areas throughout the auxiliary building to alert operating personnel of any abnormality in these parameters.

9.9.4 DESIGN EVALUATION

The Auxiliary Building Ventilation and Heating Systems capacity is adequate for the maintenance of proper temperatures in the building under operating or shutdown conditions in all types of weather.

Sufficient redundancy is included in the Engineered Safety Features Ventilation System to insure proper operation of these systems with one active component out of service. Even if the three 15,000 cfm fans and the four auxiliary building supply fans are not available, or if the 15,000 cfm fan's intake dampers are closed, sufficient ventilation is available for the Component Cooling Water pumps, and sufficient air flow exists for proper operation of the Emergency Safeguards Ventilation System.

The Fuel Handling Area Ventilation System has sufficient redundancy to ensure proper operation of this system with one exhaust fan out of service. Charcoal, roughing and high efficiency particulate air filters on the Fuel Handling Area Exhaust System provide protection against release of radioactivity from this area to the atmosphere.

The General Ventilation System and the General Supply System each consist of two 50% capacity segments per unit with a crosstie between the Unit No. 1 and Unit No. 2 exhaust systems, thus minimizing the possibility of losing the total system of the plant unit.

Under normal operating conditions, the total exhaust flow exceeds the total fan supply. Therefore, all areas within the auxiliary building pressure boundary are at a negative pressure with respect to atmosphere. All exhaust flows from within the boundary are directed to the vent of the respective unit and monitored before release to the atmosphere. All supply air is pre-filtered. The fuel handling area exhaust system is directed to the Unit 1 vent.

All systems are located within the building and generally grouped for ease of access, control and monitoring.

9.9.4.1 Test and Inspections

The systems are inspected, tested and balanced upon installation. Particulate and charcoal filters were individually tested by the manufacturer after fabrication and again after installation. The engineered safeguard ventilation system and the fuel handling area ventilation system are tested on a regularly scheduled basis over the life of the plant. Replacement filters will be tested in the same manner. Filter banks can be tested for leakage and dioctylphthalate smoke test efficiency while in place, and defective cells identified for removal and replacement.

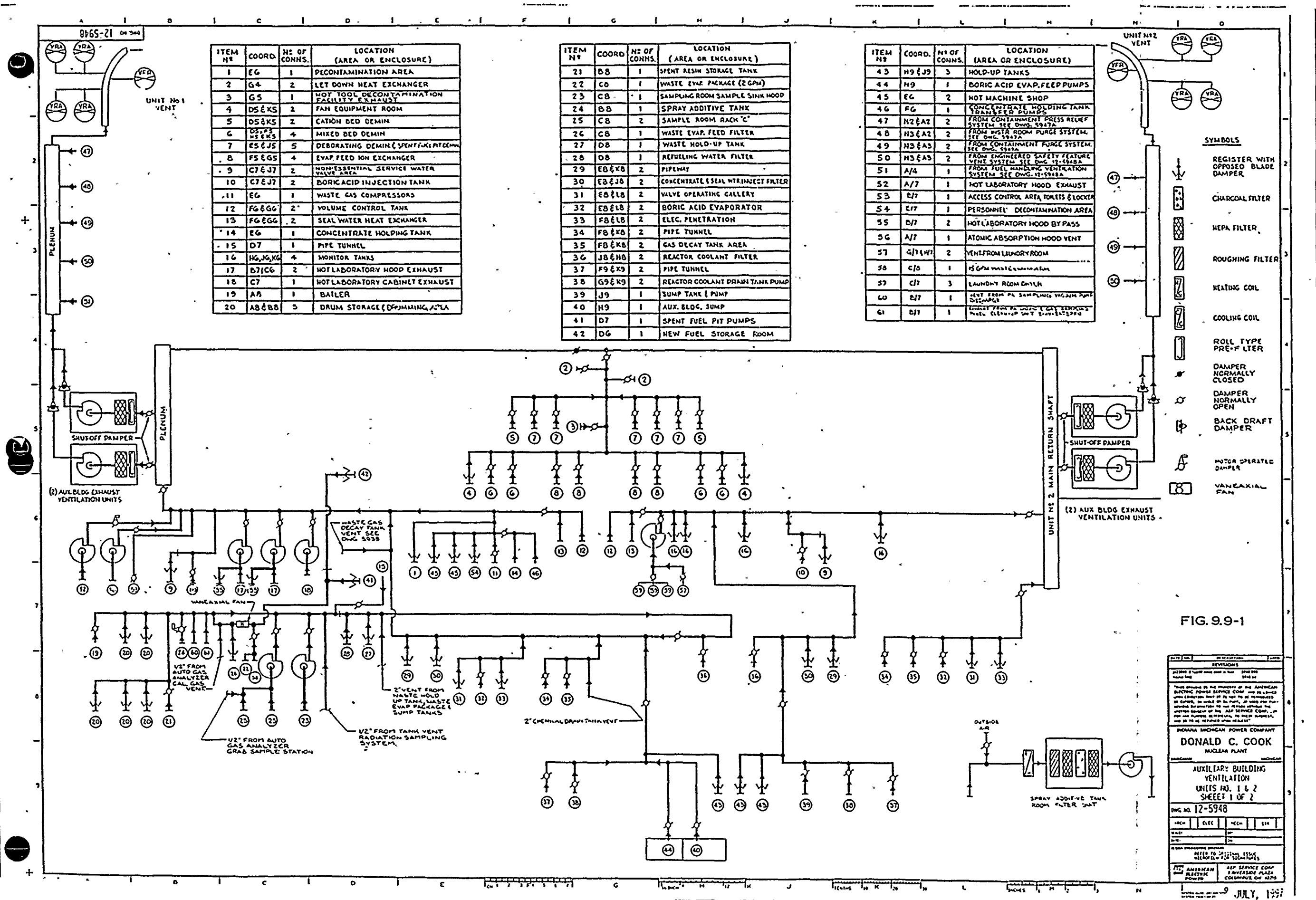


FIG. 9.9-1

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1 JULY 1959

9.10 CONTROL ROOM VENTILATION SYSTEM

9.10.1 GENERAL DESCRIPTION

The control rooms for Unit No. 1 and Unit No. 2 are both physically located on El 633' 0" of the auxiliary building with normal access from the turbine building. Control room air conditioning equipment is in an equipment room directly above the control room. Both control rooms are enclosed in a missile and tornado proof structure. The control room ventilation system is shown in Figure 9.10-1.

9.10.2 DESIGN BASES

The control room air conditioning system is designed to maintain room temperature within limits required for operation, maintenance and testing of plant controls and uninterrupted safe occupancy during post-accident shutdown.

The control room air conditioning system is designed to maintain a temperature of 75°F dry bulb and 25-80 percent relative humidity under normal operating conditions. The design is based on outside temperatures ranging from -7°F winter dry bulb to 91°F summer dry bulb and 75°F summer wet bulb. The system operates during normal or emergency conditions as required.

Conditioned air is supplied to the control room by either of two full-capacity 15,000 CFM air-handling units (one standby). Each unit includes a roughing filter, medium efficiency filter, chilled-water coil, and a fan. Downstream of each air handler in the duct system is an electric blast coil heater and an electric humidifier. Each unit is provided with chilled water from an associated 30-ton liquid-chiller. Each air-handler/liquid-chiller combination is independently capable of fulfilling design objectives. Condenser water for each liquid chiller is taken from a different header of the Essential

service water system. The air conditioning liquid chiller package was not designed to seismic Class I standards. For emergency cooling the essential service water can be manually diverted directly through the seismic Class I air handling coil, thus bypassing the liquid chillers.

Continuous pressurization of the control room is normally provided by the air conditioning system to prevent the entry of dust and dirt. Emergency filtration and pressurization are provided by a separate 6,000 CFM air-handler with roughing filters, high efficiency particulate air filters and charcoal adsorbers. This unit can also be used in the recirculation mode as a cleanup system. The performance characteristics of the high efficiency particulate air filter cells provide for removal of as much as 99.97 percent of solid particulates of 0.3 micron mean diameter. Performance characteristics of the charcoal adsorbers provide for removal of as much as 99.9 percent of entrained methyl iodide or iodine vapor*. All air conditioning equipment, pressurization fans and auxiliary equipment can be powered from emergency buses.

9.10.3 SYSTEM OPERATION

Two fresh-air intakes are provided for each control room. Both air conditioning units share one intake. A separate intake is provided for the pressurizer/cleanup filter unit. Both fresh-air intakes are fitted with a motor-operated isolation damper for control room isolation. Normally, a fixed proportion of room air and outside air is supplied to the control room through one of the air-handling units. Temperature is controlled by thermostats located in the control room. Each liquid chiller has an independent control system. Outdoor air supplied to the control room through the air-handling unit maintains a positive pressure within the room with respect to the surrounding environs to prevent entry of dust, etc.

A toilet facility is located in the Unit No. 2 control room. A small exhaust fan continuously purges this room. The exhaust vent is fitted with an isolation damper.

*For accident analysis, the high efficiency particulate air filter is assumed to remove 99% of all radioactive particulates with the adsorber removing 95% of methyl iodine.

The Control Room Pressurizer/Cleanup Filter Unit does not normally operate. In the event of a fire signal from the cable enclosure below the Control Room, the air conditioner fresh-air intake isolation damper is closed, the Control Room Pressurizer/Cleanup Filter Unit started. These operations are all performed automatically.

The Air Conditioning System then functions as a 100 percent recirculation system and pressurization air is supplied separately through the high efficiency particulate air and the charcoal filters of the Control Room Pressurization/Cleanup Filter Unit before discharging into the Control Room. The controls for isolating the normal fresh-air intake and starting the Emergency Pressurizer/Cleanup Filter Unit are located in both the Control Room and the air conditioning equipment room and can be manually actuated from either room.

A high radiation alarm from the Control Room radiation monitor or a Safety Injection signal automatically initiates closure of the isolation dampers in the Air Conditioning System and the toilet exhaust discharge. The Air Conditioning System then functions in the 100 percent recirculation mode. Upon receipt of these same signals, the isolation damper in the pressurizer/cleanup system intake goes to a minimum position to allow sufficient outdoor air into the system to pressurize the Control Room. The Control Room Pressurizer/Cleanup Filter Unit automatically starts in the partial recirculation mode to remove radioactive particulates and iodines from within the room and from the outdoor ventilation air used for pressurization.

A manually actuated override control can be used to supply additional variable amounts of outside air (over and above the minimum makeup air required for pressurization in the cleanup mode) through the Emergency Pressurizer/Cleanup Filter Unit to purge the Control Room atmosphere (outdoor conditions permitting).

9.10.4 DESIGN EVALUATION

The control room and the ventilation equipment room are both enclosed in a missile- and tornado-proof concrete structure. The ventilation equipment room is directly accessible from the control room. All other areas in the vicinity of the control room such as cable spaces, auxiliary building, turbine building, etc. are ventilated by systems which are completely independent of the control room ventilation system, thus fire or smoke generated in such other areas would not impair the integrity or accessibility of the control room. Two independent, full capacity air conditioning systems serve each control room. Two full capacity fans are provided for the control room pressurizer/cleanup filter unit of each control room. This redundancy ensures proper room conditions with one active component out of service.

9.10.5 INCIDENT CONTROL

A safety injection signal automatically closes the normal control room air intake, thus preventing possibly contaminated air from entering the room. The control room pressurizer/cleanup filter unit is automatically operated to remove any particulates or iodine which may leak into the room. In the event of gross failure of both the seismic Class III control room liquid chillers, essential service water can be diverted directly through the seismic Class I air handler cooling coils for emergency cooling.

9.10.6 TESTS AND INSPECTION

The systems were inspected, tested and balanced upon installation. Periodic testing is performed to insure system operability.

High efficiency particulate air filters and charcoal filters are tested after fabrication by the manufacturer, again after installation and periodically over the life of the plant.

10.2

MAIN STEAM SYSTEM

The Main Steam System for Units No. 1 and No. 2 are shown in Figures 10.2-1, 10.2-1A, 10.2-1B and 10.2-1C.

10.2.1 DESIGN BASES

The design bases of the Main Steam System are largely derived from past design experience with fossil fuel stations and have evolved over a long period. They are modified in order to meet special requirements associated with nuclear application and include provisions for specific earthquake, tornado, missile and reactor protection as further described in other sections.

Design codes applicable to the main steam system include, but are not limited to:

- a. ASME Boiler and Pressure Vessel Code, Sections III, VIII, and IX.
- b. ANSI Power Piping Code B31.1
- c. AEP Specifications

10.2.2 DESCRIPTION

The Main Steam System is designed to deliver steam from the steam generators to the turbine and to other equipment or systems requiring main steam, including:

- 1) Motive steam to the turbine driver of an auxiliary feedwater pump. Steam to this turbine is supplied by 4-inch branch connections upstream of the Steam Generator Stop Valves on two of the four steam lines. Either line is sufficient to

supply steam for the turbine but two are provided for redundancy. These two 4-inch lines are tied together with a motor operated shut-off valve and a check valve in each line before the tie.

- 2) Motive steam for the main feed pump turbines during start-up and up to approximately 55% load (Unit 1) or 70% load (Unit 2).
- 3) Heating steam for the reheaters.
- 4) Turbine by-pass system (Steam Dump)
- 5) Auxiliary steam system.
- 6) Turbine steam seals (Unit 2 only).

The system is best described by following the flow path from the steam generator to the turbine. Refer to Figures 10.2-1 and 10.2-1B.

Steam from the four steam generators flows through A-155, Grade KC-70 carbon steel pipes designed for 1085 psig, 600°F, through the containment penetrations.

A steam flow measuring device located in each lead within the containment provides a signal for steam generator level control and initiation of reactor safeguards system in the event of a main steam line rupture.

Following penetration of the containment a power relief valve and bank of five safety valves are installed on each steam lead. The five identical safety valves provide a combined relieving capacity of 4,288,450 lb/hr per lead (17,153,800 lb/hr for 4 leads at 1172 psi). This capacity is sufficient for the steam generation rate at maximum calculated conditions. The capacity of the power relief valve is

approximately 10% of full load flow. It opens automatically if steam pressure exceeds a pre-set value.

Downstream of the safety valves a parallel slide gate valve is installed in each line as close to the containment wall as possible. This valve, known as the Steam Generator Stop Valve, is capable of closing rapidly in the event of a main steam line rupture occurring anywhere in the piping between the steam generator and turbine. An analysis of the steam break accident is given in Chapter 14, and a safety evaluation of the steam system is given in Section 10.2.3.

The Steam Generator Stop Valves are designed to close against flow in either the normal or reverse direction to limit the effect of a steam line rupture to the blowdown of the one affected steam generator; assuming, conservatively, the failure of one of the four valves to close.

The Steam Generator Stop Valve design incorporates a piston which is attached to the valve stem. The steam above and below the piston is normally at line pressure. The cylinder volume above the piston is piped through a three-way valve into a pair of redundant, air-operated dump valves. Upon receipt of a signal to close, the dump valves open and vent the steam from the cylinder. The steam pressure in the valve body below the piston forces the piston to move rapidly and close the valve. The valve therefore is not dependent on an external power source for emergency closure. Each valve closes within 5 seconds after receipt of the requisite safety signal. Speed of closing is controlled by the setting of a needle restrictor within the hydraulic opening and closing system.

For emergency operation, reactor protection logic is supplied to isolate all steam generators by rapid closure of the four stop valves for any of the following conditions:

- a) Containment spray actuation signal initiation (Hi-Hi pressure).
- b) High steam flow coincident with Lo/Lo T_{avg} .
- c) Steam line pressure low.
- d) In addition, emergency closure can be initiated by operator actuation of the dump valves in the steam generator stop valve control system.

In the event of a steam generator tube rupture occurring, the recovery procedure involves closure of the steam generator stop valve associated with the affected steam generator. However, for this accident, rapid closure of the valve is not essential and the operator may close the valve using the hydraulic actuator.

Normal opening and closing of the valve is achieved by use of the hydraulic actuator, which is bypassed in case of an emergency closing requirement. The operating switch, in the control room, actuates the reversing solenoid and starts the electrically driven hydraulic fluid pump supplying hydraulic fluid to the valve actuator. Limit switches are fitted to the valve and wired up to display position indication in the control room.

All four main steam lines are connected to a common header, which equalizes the pressure before the steam flows through the turbine admission valves. This header is also connected to the turbine by-pass system (steam dump system).

The capacity of the turbine by-pass system is 40% of full load steam flow. All or several of the steam dump valves open under the following conditions provided a condenser vacuum permissive interlock is satisfied:

A steady-state hydraulic analysis was performed for each case assuming the most limiting single failure, steam generators pressurized to the safety valve setting (plus 3% accumulation) and non-safety related control systems failing to operate. The results of these hydraulic analyses are used as inputs in the appropriate Chapter 14 safety analysis.

10.5.2.4 Tests and Inspections

The auxiliary feedwater system, including pumps, valves and drivers, is tested in accordance with requirements of the applicable edition of the ASME Boiler and Pressure Vessel Code Section XI and, beginning with the third 10 year interval ISI program, pump and valve tests are in accordance with ASME OM Standards and NUREG-1482. During the tests, the pumps are operated with flow back to the Condensate Storage Tank through a test/recirculation line. Performance is verified by monitoring flow meters in the test lines and pressure gauges on the suction and discharge of the pumps.

The availability of the Essential Service Water supply to the Auxiliary Feedwater System must also be determined, but without contaminating the condensate tank with lake water. Two normally closed valves, one motor operated, connect the Essential Service Water supply to each of the auxiliary feed pumps. To test, the tell-tale valve between each set of the two aforementioned valves is opened to drain that portion of the line, and the two valves are independently stroked. A visual check at the tell-tale will verify normal direction of flow through the manual valve and backflow through the motor operated valve, after which all valving is restored to its normal setting.

10.7 TURBINE AUXILIARY COOLING SYSTEM

10.7.1 DESIGN BASIS

The Turbine Auxiliary Cooling System utilizes water from the main condensate system as a coolant for:

- a. The gland steam condenser
- b. The generator hydrogen coolers
- c. The generator stator coolers
- d. The exciter cooler
- e. The bus duct enclosure

10.7.2 DESCRIPTION

The system is shown in Figure 10.5-2A and 10.5-3A. The system can operate in either of two modes as described below. The mode of operation is determined by the temperature of the condensate leaving the hotwell. In cold weather, when the temperature of the condensate is sufficiently low to effect adequate cooling of the services listed in Section 10.7.1, the system is operated in an open cycle. In warm weather, when the condensate temperature is too high to meet cooling requirements, a closed cycle is used. The systems for Units No. 1 and No. 2 are similar except for the changeover point from closed to open cycle. For Unit No. 1, when the temperature of the condensate leaving the hotwell is above 95°F, changeover from an open to a closed cycle is made, for Unit No. 2, this point is 104°F. In general, the changeover is made on a seasonal basis, by manual operation of valves.

During open cycle operation, condensate is taken from the hotwell pump discharge header and is pumped by one of two full-capacity turbine auxiliary cooling pumps through the various heat exchangers. The condensate from the heat exchangers then returns to the condensate booster pump suction header. This mode reclaims heat and improves thermal efficiency of the unit.

In closed cycle operation, the condensate is pumped by one of the two turbine auxiliary cooling pumps through the turbine auxiliary cooler, where it is cooled by circulating water in the tube circuit of the cooler. Flow through the tube circuit is in parallel with the circulating water flow through the main condenser. The pressure differential across the main condenser maintains the flow in the turbine auxiliary cooler. Cooled condensate from the shell side then flows through the heat exchangers and returns to the turbine auxiliary cooling pump suction. Makeup for the system is supplied by a small by-pass around the condensate isolation valve.

Operation of the system is monitored in the control room by flow indicators, pressure indicators and temperature recorders.

10.7.3 DESIGN EVALUATION

When the temperature of the main condensate is below 95°F, Unit No. 1, or below 104°F, Unit No. 2, the heat from the turbine auxiliary coolers is reclaimed by utilizing the open cycle. When it is above 95°F, Unit No. 1, or above 104°F, Unit No. 2, the turbine auxiliary cooler is placed in service and the heat is lost to the circulating water system.

10.7.4 TESTS AND INSPECTION

The active components of the system are in continuous use during normal plant operation. Periodic visual inspections and preventive maintenance are conducted following normal industry practice.

The Demineralized Water Make-Up System produces the high purity, degassified water required for make-up to the reactor coolant and condensate-feedwater systems for both units. Lake water from the non-essential service water system is filtered, chlorinated, and held in a retention tank to effect complete sterilization. There is an alternate source of supply from the Lake Township public water system.

The water is pumped from the retention tank through carbon filters to remove organics and residual chlorine. The water is then processed by a reverse osmosis unit and passed through cation exchangers, a vacuum degassifier, anion exchangers and mixed-bed "polishing" demineralizers. Following treatment, the demineralized, degassified water is distributed to the various points of usage.

The Primary Water System supplies water for miscellaneous purposes in the auxiliary building, primarily for reactor coolant make-up. The primary water is a mixture of demineralized, degassified make-up water and condensate recovered from processing reactor-coolant letdown fluid.

Chemical feed systems are provided for adding chemical solutions to the condensate and feedwater to scavenge dissolved oxygen, control pH and minimize corrosion. The chemicals used are hydrazine, carbohydrazide, ammonia or other amines, and boric acid. The solutions are mixed using appropriate dilutions of chemicals from bulk storage, and stored in covered stainless steel feed tanks. When needed, the solutions are pumped from these tanks by motor driven, positive displacement pumps to the points of injection. The pumps have adjustable strokes and only have manual control.

The Steam and Power Conversion System vents and drains are arranged in a similar manner to those in a fossil-fueled power station, since the system is normally non-radioactive. However, because the steam generator blowdown (SGBD) and the air ejectors discharge can become contaminated, these subsystems are monitored and discharged under controlled conditions as explained below.

10.11.1 DESIGN BASIS

The Steam Generator Blowdown System is designed to maintain the proper water chemistry within the steam generators. The secondary side water is blown down to maintain the total dissolved solids within established limits.

The Steam Jet Air Ejector unit removes non-condensable gases from the condenser shells. These exhaust gases are vented to the atmosphere. A small representative sample passes through a radiation monitor. Each of the condenser steam jet air ejector elements is designed to remove 15.0 cfm of non-condensable gases. Separate non-condensing start-up jets are used to reduce condenser back pressure to 5 in Hg abs during start-up.

10.11.2 DESCRIPTION

The steam generator blowdown and blowdown treatment systems are shown on Figures 10.2-1, 10.2-1B, and 11.5-1. The steam jet air ejector vent systems are shown on Figures 10.5-4A and 10.5-5A.

The SGBD is routed to the start-up blowdown flash tank during start-up or under abnormal operating conditions, for example, during high condenser inleakage. The steam produced in the start-up blowdown flash tank is vented to the atmosphere through a moisture separator. The water is routed to the screenhouse forebay. The start-up flash tank is equipped with a NESW supply line for quenching.

When the plant reaches normal full power operation, the start-up blowdown flash tank is taken out of service and the blowdown is routed to the normal blowdown flash tank. The blowdown flashes into a mixture of approximately 40 percent steam and 60 percent water. The steam is returned to the Condensate System through the condensers and the water is routed to the screenhouse forebay either directly or through mixed-bed demineralizers. The normal blowdown flash tank is equipped with a NESW supply line for quenching.

The blowdown rate from each steam generator is controlled by two parallel, fail closed, control valves, which are located downstream of a blowdown isolation valve. The SGBD is monitored for radioactivity before it reaches either blowdown tank. The SGBD treatment system also has a radiation monitor between the second and third treatment demineralizers (see Sections 9.6 and 11.5, respectively). These radiation monitors close the SGBD isolation valves upon detection of high radiation.

During normal operation, both elements of each of the four two-stage twin element steam jet air ejector (SJAE) units removes non-condensable gases from each of the three main condenser shells and both feedpump turbine condensers. For added flexibility, the individual air off-takes are joined to a common header with cross tie-valves. The motive steam is condensed in the SJAE inter- and after-condensers. Inter-condenser drains are returned to the condensate system via the main condenser drip leg and the miscellaneous drain tank, respectively.

Gases removed from the condensers by the steam jet air ejectors during normal operation are discharged into a common header. These non-condensable gases are then exhausted at a slightly positive pressure to the atmosphere through a vent stack. The SJAE vent stack has an air flow meter to measure the quantity of non-condensables removed from the condensers. Since the introduction of radioactivity in the main steam system by a steam generator tube leak would probably first

escape to the reactor coolant by diffusion through defects in the cladding of one percent of the fuel rods.

The waste disposal system collects and processes all potentially radioactive reactor plant wastes for removal from the plant site within limitations established by applicable governmental regulations. In addition, the system is capable of liquid waste segregation and reuse.

All planned releases may be either batch or continuous. Before a batch may be released, the tank is sampled and the sample analyzed in the laboratory. A gas release is made only if the release can be made without exceeding federal standards and lack of reserve holdup capacity requires such a release.

Radiation monitors are provided to maintain surveillance over the release operation, and a permanent record of activity released is provided by radiochemical analysis of known quantities of waste.

At least two valves must be manually opened to permit discharge of liquid or gaseous waste from the Waste Disposal System. One of these valves is normally locked or sealed closed. The other is a control valve which will trip closed on a high effluent radioactivity level signal.

As secondary functions, system components supply hydrogen and nitrogen to primary system components as required during normal operation, and provide facilities to transfer fluids from the containment to other systems outside the containment.

The system is controlled primarily from a central panel in the auxiliary building. Malfunction of the system is alarmed in the auxiliary building, and annunciated in the control room. All system equipment is located in or near the auxiliary building, except for the reactor coolant drain tanks which are located in the reactor containments.

System Description

Liquid Processing

During normal plant operation the Waste Disposal System processes liquids from the following sources:

- a) Equipment drains and leaks
- b) Radioactive chemical laboratory drains
- c) Radioactive laundry and hot shower drains
- d) Decontamination area drains
- e) CVCS demineralizer regeneration
- f) Sampling System

The system also collects and transfers liquids from the following sources in the containment for processing:

- a) Reactor coolant loops
- b) Pressurizer relief tank
- c) Reactor coolant pump secondary seals
- d) Excess letdown (during startup)
- e) Accumulators
- f) Valve and reactor vessel flange leakoffs
- g) Refueling cavity drains

- b) Displacement of cover gases as liquids accumulate in various tanks
- c) Miscellaneous equipment vents and relief valves
- d) Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases

The waste disposal system includes nitrogen and hydrogen systems which supply these gases to primary plant components. The pressure regulator in the nitrogen system header is set at 75 psig. When the nitrogen header pressure drops below a preset pressure, an alarm alerts the operator.

Most of the gas received by the waste disposal system during normal operation is nitrogen cover gas displaced from the CVCS holdup tanks as they are filled with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or no aerated liquids and the vent header itself is designed to operate at a slight positive pressure (but not high enough to cause overpressurization) including allowances for instrument uncertainties, process induced pressure changes, and any other special concerns that may be necessary to prevent oxygen in-leakage. Out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. From the compressors, gas flows

to one of eight gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select another tank for backup. When the tank in service becomes pressurized to 100 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator so he may select a new backup tank. Pressure indicators are provided to aid the operator in selecting the backup tank. The individual tank pressures are continuously recorded on the control panel in the auxiliary building.

Gas held in the decay tanks can either be returned to the CVCS holdup tanks or, if it has decayed sufficiently for release, discharged to the atmosphere. Generally, the last tank to receive gas will be the first tank recycled to the CVCS holdup tanks. This permits the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator the option to fill, reuse, and discharge gas simultaneously. During degassing of the reactor coolant prior to a cold shutdown, for example, it may be desirable to pump the gas purged from the volume control tank into a particular gas decay tank and isolate that tank for decay rather than reuse the gas in it. This is done by opening the inlet valve to the desired tank and closing the outlet valve to the reuse header. Simultaneously, one of the other tanks can be opened to the reuse header if desired, while another is discharged to atmosphere.

Before a tank is discharged to the environment, it is sampled and analyzed to determine and record the activity to be released, and then is discharged to the plant vent at a controlled rate through a radiation monitor which enables the operator to monitor the radioactivity in the gas release. Samples of the gas to be released are taken in gas sampling vessels. During release a trip valve in the discharge line is closed automatically by a high radioactivity level indication in the plant vent.

The refueling cavity and refueling canal, flooded with borated water to an elevation of approximately 645' during refueling operations, provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24 ft. above the reactor vessel flange. This height ensures that there will be sufficient water depth above the active fuel of a withdrawn fuel assembly to maintain exposures as low as reasonably achievable (ALARA).

The spent fuel assemblies and control rod clusters are remotely removed from the reactor containment through the horizontal fuel transfer tube and placed in the spent fuel pit. The transfer tube is shielded with a minimum of 5'-2" of concrete in all areas except a small piping area located under the transfer tube in the containment. The piping area is shielded with 2'-5" of concrete. It is posted as radiological conditions dictate and is protected with locked gates to ensure that personnel cannot enter this area while spent fuel is being transferred.

Fuel is stored in the spent fuel pool portion of the Auxiliary Building. Shielding for the spent fuel storage pool is provided by 6 feet thick concrete walls, and the pool is flooded to a level such that the water height is approximately 25 feet above the stored spent assemblies. During spent fuel handling, sufficient water depth is maintained above the fuel assembly being handled to maintain exposures ALARA.

The original refueling shield design parameters are listed in Table 11.2-5.

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so a compartment may be entered without having to shutdown and, possibly, to decontaminate equipment in an adjacent compartment. The shield material provided throughout the auxiliary building is normal density concrete ($\rho = 2.3$ g/cc). The principal auxiliary shielding provided and the design parameters are tabulated in Table 11.2-6.

Shielding Design Evaluation

The whole body gamma dose in the control room under accident conditions is calculated assuming the release of the following sources to the reactor containment (Per TID-14844):

- a) 100% of the noble gases
- b) 50% of the halogens
- c) 1% of the remaining fission product inventory.

These sources, tabulated in Table 11.2-7, are assumed to be homogeneously distributed within the free volume of the reactor containment. The source intensity as a function of time after the accident is conservatively determined by considering decay only; no credit is taken for washdown or spray and ice condenser removal of iodine.

TABLE 11.2-5

ORIGINAL REFUELING SHIELD DESIGN PARAMETERS *

Total number of fuel assemblies	193
Minimum full power exposure	1000 days
Minimum time between shutdown and fuel handling	100 hours
Maximum exposure rate adjacent to spent fuel pit	1.0 mrem/hr
Maximum exposure rate at water surface	2.5 mrem/hr

* These parameters are kept for historical reasons. The dose rates are no longer applicable since the design of the spent fuel pit has been changed.

TABLE 11.2-6

PRINCIPAL AUXILIARY SHIELDING

Design parameters for the auxiliary shielding include:

Core thermal power	3391 MWt
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume	12600 ft ³
Letdown flow (normal purification)	75 gpm
Cesium purification flow (intermittent)	75 gpm
Cut-in concentration deborating demineralizer	100 ppm
Dose rate outside auxiliary building	<1 mrem/hr
Dose rate in the building outside shield walls	<2.5 mrem/hr

<u>Component</u>	<u>Concrete Shield Thickness, Ft. - In.</u>
Mixed Bed Demineralizers	4 - 0
Charging pumps	2 - 6
Liquid holdup tanks	2 - 8
Volume control tank	3 - 9
Reactor Coolant filter	2 - 6
Boric Acid Evaporator	2 - 4
Gas decay tanks	3 - 3
Waste Gas Compressors	2 - 8
Waste Evaporator	2 - 0
Liquid Waste Holdup Tank	2 - 0
Spent Resin Storage Tank	4 - 0

11.3 RADIATION MONITORING SYSTEM

11.3.1 GENERAL DESIGN CRITERIA

Monitoring Radiation Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

The containment atmosphere, the unit vent, SJAE vent, turbine gland seal exhaust, steam generator blowdown, and the waste disposal system liquid effluent are monitored for radioactivity concentration during operation. The design objective is for annual average releases of radioactivity (gases and liquids) for both dose and dose rates at the critical site boundary will be to meet the requirements of 10 CFR Part 50.

Liquid release pathways are monitored by radiation detection instruments. Planned liquid effluents of the plant are released to the circulating water system.

Gaseous releases are monitored by the unit vent monitors. The gaseous effluent from the steam generator blowdown tank vent is normally routed to the main condenser, except during startup and other periods of short duration when it may be vented to the atmosphere. In addition, any time there are non-condensable radioactive gases which may be released from the blowdown flash tank, such gases would also be present in the condensate system where they would be removed by the steam jet air ejectors and be detected by the applicable radiation monitor.

Accidental spills of radioactive liquids are maintained within the auxiliary building and collected in a drain tank. Any contaminated liquid effluent discharged to the condenser circulating water is monitored. Gaseous effluent from possible sources of accidental releases of radioactivity external to the reactor containment (e.g., the spent fuel pool and waste handling equipment) is exhausted from the unit vent and monitored by a radiation monitor. Gaseous batch releases shall be made only if the release can be made without exceeding federal standards and lack of reserve hold-up capacity requires such a release.

Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

A controlled ventilation system removes gaseous radioactivity from the fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the unit vent. Radiation area monitors are in continuous service in these areas to actuate high-activity alarms on the control room board annunciator.

Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provision shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

Waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will result in doses below the limits of 10 CFR 100, as discussed in Section 11.1.1 and Chapter 14.

11.3.2 DESIGN BASIS

The Radiation Monitoring System is designed to perform two basic functions:

- a. Warn of any radiation hazard which might develop, and
- b. Give early warning which might lead to a radiation hazard or plant damage.

Instruments are located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the control room. The Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

The components of the Radiation Monitoring System are designed to operate during all expected environmental conditions for normal operation. Specific components are designed to operate during adverse or accident plant conditions. In addition, process and area radiation monitors are of a nonsaturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full scale indication.

Gaseous Release Pathways

1. Containment and Instrument Room Exhaust - Releases are through the Unit Vent. Noble gas activity and release rates are monitored and recorded. Releases are on an intermittent basis as the containment is purged only periodically. The containment atmosphere is sampled prior to release. The containment purge and exhaust isolation valves close on a containment high radiation signal. Monitors ERS 1300, 1400, 2300 and 2400 and VRS 1101, 1201, 2101 and 2201 cause the ESF actuation. The Unit Vent monitor systems also sample iodine and particulate activity. Operation of the containment purge and exhaust system is controlled by Plant Technical Specifications.
2. Auxiliary Building Ventilation - The activity in the exhaust depends on leakage into the Auxiliary Building atmosphere from the primary systems, and it is expected to be very low.

Releases are through the unit vent. Activity is measured prior to the release point of the unit vent. High radioactivity will be alarmed in the control room.

3. Steam Jet Air Ejector - A continuous release of activity exists only during periods of steam generator primary to secondary leakage. The steam jet air ejector exhaust is continuously monitored. The steam jet air ejector monitor is sensitive to total beta and gamma activity.
4. Gland Seal Condenser Exhaust - A continuous release of activity exists only during periods of steam generator primary to secondary leakage. The gland seal condenser exhaust is continuously monitored.
5. Steam Generator Blowdown Exhaust - The releases are through the main condenser while utilizing the normal flash tank. During off normal chemistry conditions, unit start-up, or unit shutdown the release is to the atmosphere via the S/G blowdown flash tank vent. The steam generator blowdown is continuously monitored.
6. Main Steam PORV/Safety Release Valves - The main steam power operated relief valves and safety valves provide pressure relief on each steam lead if steam pressure exceeds normal operating values. They also allow plant cooldown by steam discharge to the atmosphere if the turbine by-pass system is not available. The PORV discharge lines are continuously monitored.
7. Waste Gas Decay Tanks - These tanks are batch released through the unit vent. Their total activity and release rates are monitored by a radiation monitor and a flow meter and both are recorded. Isolation valves on the discharge header from the tanks close on a high radiation signal. The contents of the tanks are sampled prior to release and analyzed to determine isotopic concentrations.
8. Miscellaneous Ventilation - Releases are through the unit vent from ventilation systems such as SF pool, nuclear sampling room, etc. Noble gas activity and release rates are monitored and recorded. High radioactivity will be alarmed in the control room.

Meteorological conditions during periods of release from the above systems will be obtained from the meteorological program.

The unit vent monitors for noble gases and samples for particulates and iodine.

The reactor coolant system isotopic inventory is determined by sampling and analysis to predict any change in isotopic spectrum that would lead to measurable quantities of iodine release.

The unit vent is provided with integrating type air samplers. A sample from the unit vent is drawn continuously through a particulate filter and an iodine sampling device.

The methods and formulas for computation of doses associated with the liquid and gaseous releases are given in the Cook Nuclear Plant's Off-site Dose Calculations Manual (ODCM).

Liquid Release Pathways

Radioactive liquids are released through the waste disposal system monitor tanks, steam generator blowdown and turbine room sump. Activity is monitored and recorded on the liquid effluent monitor via the applicable pathway. Before a batch may be released, the tank is sampled and the sample analyzed. If the radioactivity level of the sample is found to be within acceptable limits, the liquid wastes will be released, monitored, and recorded. At the same time, the rate of the liquid release is measured by a flow meter. By using the rate of liquid waste releases, the rate of flow of the condenser cooling water, the activity of the liquid waste released, the rate of activity release and the concentration of activity in the condenser cooling water can be determined.

Liquid effluent and dilution volumes released are recorded. Gamma isotopic analysis is performed on the liquid effluent prior to each batch release.

The Radiation Monitoring System is divided into the following sub-systems:

- a. The Process Radiation Monitoring System monitors various fluid streams for indication of increasing radiation levels.
- b. The Area Radiation Monitoring System monitors radiation in certain areas of the plant.
- c. Environmental radiation monitoring program monitors radiation in the area surrounding the plant as described in Sub-Chapter 2.7.

11.3.3 GENERAL DESCRIPTION AND OPERATION

The original radiation monitoring channel equipment, including chassis with signal conditioning equipment, controls, power supplies, indicators and alarms is centralized in cabinets located in the control rooms for convenient operator access. Strip chart recorders are provided in these cabinets to sequentially record each monitoring channel.

This equipment has been supplemented and partially replaced by a system of distributed, multi-channel field data acquisition units. Each field unit services one or more detector channels. It measures and records the channel readings, performs alarm and other status checks and initiates trip functions (if applicable). Each field unit is connected via isolation devices to two data communication lines. Each line terminates at its associated system control terminal (CT). A CT is located in each control room and provides the control room operator with current channel status. A printer provides a record of the channels on a regular basis. Channel status changes are reported and recorded as they occur.

Typical sensitivity ranges of the various radiation monitor channels are given in Table 11.3-1 and are based on the first isotope listed in the last column of the table.

The monitor channels are response checked using a radioactive source, tested electronically, and calibrated by pulse injection methods. The detectors are calibrated using appropriate calibrated sources at a frequency listed in the Technical Specifications and/or the ODCM. Effluent monitor setpoints are determined in accordance with the ODCM and are designed to aid in maintaining ODCM limits.

11.3.3.1 Process Radiation Monitoring System

This system consists of (original and newer) channels which monitor radiation levels in various plant operating systems. High radiation level alarms are annunciated and identified in the control room.

The radiation monitoring channels employ instrument failure alarms at the radiation monitoring cabinets, control board annunciator, and at local indicators (where provided). Control interlocks fail in the 'high radiation' position upon instrument failure and must be manually reset. Instrument failure alarms are initiated upon failure of the radiation monitor, loss of detector signal or loss of power.

Gaseous effluents are scanned, alarmed, and recorded thereby providing a complete history of abnormal occurrences for evaluation.

These radiation monitoring channels are shown in Table 11.3-3.

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11.3.3.2 Area Radiation Monitoring System

This system consists of channels which monitor radiation levels in various plant areas. Certain of these monitors have been upgraded. Both the original and newer monitors are shown in Table 11.3-1.

Each original monitor consists of a fixed position Geiger-Mueller detector with local indicator and check source. An associated readout drawer in the control room provides high radiation and failure alarms, and initiates trips (if required). Channel readings are logged on a multi-point recorder.

Newer monitors consist of either Geiger-Mueller or ion chamber detectors, with check sources, connected to multi-channel field mounted data acquisition units. Each field unit reads its detectors, performs status checks, initiates trips (if required), records the readings, provides local readout and reports to the system control terminals. The control terminals poll the field units and provide channel readings and status information to the control room operators through displays, annunciators and printers. Selected channels are provided with individual indicator/alarm units near the detector.

Two high range ion chamber detectors monitor each containment. One is located in the upper containment while the second is in the lower containment about 180° apart from the first. These accident monitors are separate from the other area channels. Each has a dedicated readout module in the control room with a multi-range indicator, status lights, and test circuits. An isolated output from each module is sent to an associated field unit to provide for recording and supplemental data access via the system control terminals.

11.3.4 Reactor Coolant Activity Monitoring

Refueling shutdown programs at operating Westinghouse PWRs indicate that, during cooldown and depressurization of the Reactor Coolant System (RCS), a release of activated corrosion products and fission products from defective fuel has been found to increase the coolant activity level above that experienced during steady state operation. However, high coolant activity is avoided by implementing established shutdown procedures. These procedures

include purification of the RCS through the cation and mixed bed demineralizers and system degassification.

Table 11.3-2 illustrates the calculated coolant activity increases of several isotopes for the Donald C. Cook Plant. This table lists the calculated activities during steady state operation before refueling shutdown outage and calculated peak activities during plant cooldown operations. These data are based on measurements from an operating PWR which is similar in design to the Cook Nuclear Plant and has operated with fuel defects. The measured activity levels are also included in Table 11.3-2.

The dominant non-gaseous fission product released to the coolant during system depressurization is found to be Iodine-131. The activity level in the coolant was observed to be higher than the normal operating level for nearly a week following initial plant shutdown. Although lesser in magnitude, the other fission product particulates (e.g., cesium isotopes) exhibited a similar pattern of release and removal by purification. It is reasonable to project this data to the Cook Nuclear Plant since the purification constants are similar and as it is standard operating procedure to purify the coolant through the demineralizers during plant cooldown. Fission gas data from operating plants indicate a maximum increase of approximately 1.5 over the normal coolant gas activity concentration. However, the system degassification procedures are implemented prior to and during shutdown, and have proven to be an effective means for reducing the gaseous activity concentration and controlling the activity to levels lower than the steady state value during the entire cooldown and depressurization procedure.

The corrosion product activity releases have been determined to be predominantly dissolved Cobalt-58. From Table 11.3-2, it is noted that this contribution is less than 1% of the total expected coolant activity and is hence considered to be a minor contribution.

Since continued operation of the purification system is standard operating procedure during plant cooldown and since means for system degassification are

available for fission gas removal, the total activity concentration in the coolant can be maintained within Technical Specification limits throughout the plant shutdown, while considering the additional activity inventory released during system cooldown and depressurization. The coolant activity concentrations and inventories during the shutdown and prior to plant startup are established by chemical analysis of samples for the Reactor Coolant System.

11.3.5 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Mobile continuous air monitors are available for use in emergencies involving airborne radioactivity concerns. This equipment includes particulate, radioiodine and noble gas monitors. Regulated air samples are also available which require sample collection and laboratory analysis.

Emergency response equipment is located in the basement assembly area for counting radioactive samples.

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TABLE 11.3-1

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES, AND

DETECTING MEDIUM

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Containment-Air Particulate	ERS-1301, 1401 2301, 2401	Air	1×10^{-4} to $10 \mu\text{Ci}$	Cs^{137} , Radioactive Particulates
Containment-Air Iodines	ERS-1303, 1403 2303, 2403	Air	2×10^{-4} to $3 \mu\text{Ci}$	I^{131} , Radioiodine
Containment Radio-Gas	ERS-1305, 1405 2305, 2405	Air	9×10^{-7} to $5 \times 10^{-2} \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
	ERS-1307, 1407 2307, 2407	Air	2×10^{-3} to $2 \times 10^3 \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
	ERA-1309, 1409 2309, 2409	Air	1×10^{-1} to $9 \times 10^4 \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
Steam Jet Air Ejector Gas	SRA-1905, 2905	Air	9×10^{-7} to $5 \times 10^{-2} \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
	SRA-1907, 2907	Air	2×10^{-3} to $2 \times 10^3 \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
	SRA-1909, 2909	Air	1×10^{-1} to $9 \times 10^4 \mu\text{Ci/cc}$	Xe^{133} , Noble Gases
Component Cooling Loop Liquid	R-17A & B	Water	1×10^{-5} to $1 \times 10^{-2} \mu\text{Ci/cc}$	Co^{60} , Mixed Fission Products
Waste Disposal System Liquid Effluent	RRS-1001	Water	5 to 500,000 cpm	Co^{60} , Mixed Fission Products
Steam Generator Blowdown Liquid	R-19	Water	1×10^{-5} to $2 \mu\text{Ci/cc}$	Cs^{137} , Mixed Fission Products
Essential Service Water Liquid	R-20	Water	3×10^{-5} to $4 \times 10^{-1} \mu\text{Ci/cc}$	Cs^{137} , Mixed Fission Products

TABLE 11.3-1 (Cont'd.)

<u>Monitor Name</u>	<u>Channel Number</u>	<u>Medium</u>	<u>Typical Range</u>	<u>Detected Isotopes</u>
Steam Generator Blowdown Treatment System Liquid	R-24	Water	1×10^{-6} to 2×10^{-1} $\mu\text{Ci/cc}$	Co^{60} , Mixed Fission Products
Unit Vent Air Particulate	VRS-1501, 2501	Air	1×10^{-4} to 10 μCi	Cs^{137} , Radioactive Particulates
Unit Vent Radioiodine	VRS-1503, 2503	Air	2×10^{-4} to 3 μCi	I^{131} , Radioiodine
Unit Vent Radio Gas	VRS-1505, 2505	Air	9×10^{-7} to 5×10^{-2} $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
	VRS-1507, 2507	Air	2×10^{-3} to 2×10^3 $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
Unit Vent Hi-Level Radio Gas	VRS-1509, 2509	Air	1×10^{-1} to 9×10^4 $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
Gland Seal Condenser Exhaust Monitor	SRA-1805, 2805	Air	9×10^{-7} to 5×10^{-2} $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
	SRA-1807, 2807	Air	2×10^{-3} to 2×10^3 $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
	SRA-1809, 2809	Air	1×10^{-1} to 9×10^4 $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
Essential Service Water Liquid	R-28	Water	1×10^{-5} to 1×10^{-2} $\mu\text{Ci/cc}$	Co^{60} , Mixed Fission Products
Containment Area at Personnel Lock	VRS-1101, 2101	Air	1×10^{-1} to 1×10^4 mrem/hr	
Upper Containment Area Monitor	VRS-1201, 2201	Air	1×10^{-1} to 1×10^4 mrem/hr	
Steam Generator Power Operated Relief Valve Monitor	MRA-1600, 2600 1700, 2700	Vapor	1×10^{-2} to $1 \times 10^{+2}$ $\mu\text{Ci/cc}$	Xe^{133} , Noble Gas
Spent Fuel Area	R-5	Air	1×10^{-1} to 1×10^4 mrem/hr	

TABLE 11.3-1 (Cont'd.)

<u>Monitor Name</u>	<u>Channel Number</u>	<u>Medium</u>	<u>Typical Range</u>	<u>Detected Isotopes</u>
Sampling Room Area	R-6	Air	1×10^{-1} to 1×10^4 mrem/hr	
In-Core Instrumentation Room Area	ERA-7402 (Unit 1)	Air	1×10^{-1} to 1×10^7 mrem/hr	
	R-7 (Unit 2)	Air	1×10^{-1} to 1×10^4 mrem/hr	
Drumming Station Area	R-8	Air	1×10^{-1} to 1×10^4 mrem/hr	
High Range Containment Area Monitor	VRS-1310, -2310 -1410, -2410	Air	1 to 1×10^7 R/HR	
Vestibule Elevation 591'	ERS-1306, -2306	Air	1×10^{-3} to 1×10^2 mrem/hr	
Outside Containment Spray Pump Rooms Elevation 573'	ERS-1406, -2406	Air	1×10^{-3} to 1×10^2 mrem/hr	
West of Equipment Hatch, Elevation 650'	VRS-1506, -2506	Air	1×10^{-3} to 1×10^2 mrem/hr	
Turbine Building, Elevation 609'	SRA-1806, -1906, -2906	Air	1×10^{-3} to 1×10^2 mrem/hr	
Turbine Building, Elevation 591'	SRA-2806	Air	1×10^{-3} to 1×10^2 mrem/hr	
North of Boric Acid Tanks, Elevation 609'	RRS-1003	Air	1 X 5 to 500,000 cpm	
Unit 1 E CCP Room	ERA-7303	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 W CCP Room	ERA-7304	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 E RHR Pump Room	ERA-7305	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 W RHR Pump Room	ERA-7306	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 N SIS Pump Room	ERA-7307	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 S SIS Pump Room	ERA-7308	Air	1×10^{-2} to 1×10^3 R/hr	

TABLE 11.3-1 (Cont'd.)

<u>Monitor Name</u>	<u>Channel Number</u>	<u>Medium</u>	<u>Typical Range</u>	<u>Detected Isotopes</u>
Unit 1 Reactor Coolant Filter Cubicle	ERA-7309	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 E CCP Room	ERA-8303	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 W CCP Room	ERA-8304	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 E RHR Pump Room	ERA-8305	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 W RHR Pump Room	ERA-8306	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 N SIS Pump Room	ERA-8307	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 2 S SIS Pump Room	ERA-8308	Air	1×10^{-2} to 1×10^3 R/hr	
Unit 1 Reactor Coolant Filter Cubicle	ERA-8309	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 1 Control Room	ERS-7401	Air	1×10^{-4} to 10 R/hr	
Access Control Facility	ERA-7403	Air	1×10^{-4} to 10 R/hr	
Radio Chemistry Lab	ERA-7404	Air	1×10^{-4} to 10 R/hr	
Unit 1 N Seal Water Injection Filter Cubicle	ERA-7407	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 1 S Seal Water Injection Filter Cubicle	ERA-7408	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 1 Seal Water Filter Cubicle	ERA-7409	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 2 Control Room	ERS-8401	Air	1×10^{-4} to 10 R/hr	
609' Elevation Passageway	ERA-8403	Air	1×10^{-4} to 10 R/hr	

TABLE 11.3-1 (Cont'd.)

<u>Monitor Name</u>	<u>Channel Number</u>	<u>Medium</u>	<u>Typical Range</u>	<u>Detected Isotopes</u>
Unit 2 N Seal Water Injection Filter Cubicle	ERA-8407	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 2 S Seal Water Injection Filter Cubicle	ERA-8408	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 2 Seal Water Injection Filter Filter Cubicle	ERA-8409	Air	1×10^{-3} to 1×10^3 R/hr	
587' Elevation Passageway	ERA-7504	Air	1×10^{-3} to 1×10^2 R/hr	
Emergency Sampling Location	ERA-7507	Air	1×10^{-4} to 10 R/hr	
573' Elevation Passageway	ERA-7508	Air	1×10^{-3} to 1×10^2 R/hr	
Refueling Water Purification Filter Cubicle	ERA-7509	Air	1×10^{-3} to 1×10^3 R/hr	
Unit 1 Vent Sampling Area	ERA-7601	Air	1×10^{-4} to 10 R/hr	
Unit 1 Vent Sampling Flow Adjacent Area	ERA-7602	Air	1×10^{-3} to 1×10^2 R/hr	
Unit 2 Vent Sampling Area	ERA-7603	Air	1×10^{-4} to 10 R/hr	
Unit 2 Vent Sampling Flow Adjacent Area	ERA-7604	Air	1×10^{-3} to 1×10^2 R/hr	
633' Elevation Passageway	ERA-7605	Air	1×10^{-4} to 10 R/hr	

TABLE 11.3-2

REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES
DURING STEADY STATE OPERATION AND PLANT SHUTDOWN OPERATION

<u>Isotope</u>	<u>Operating PWR Plant</u>		<u>Donald C. Cook Plant - 1% Fuel Defects</u>	
	Measured Activity Before Shutdown	Measured Peak Shutdown Activity	Calculated Activity Before Shutdown	Expected Peak Shutdown Activity
	<u>μCi/gm</u>	<u>μCi/gp</u>	<u>μCi/gm</u>	<u>μCi/gm</u>
I-131	0.83	14.9	2.4	43.0
Xe-133	127.0	65.0*	254.0	130.0*
Cs-134	1.29	1.7	0.19	0.25
Cs-137	1.67	2.14	1.1	1.4
Cs-144	0.00068	0.0058	0.00051	0.0044
Sr-89	0.0033	0.40	0.0042	0.51
Sr-90	0.00057	0.013	0.0001	0.0023
Co-58	---	0.95	0.025	1.0

*Activity reduced from steady state level by approximately one day of system degassification prior to plant shutdown.

Table 11.3-3
Radiation Monitoring System Channels

Channel	Purpose	Associated Trip Function Overview
ERS-1301, 1401, 2301, 2401	Containment Airborne Particulates - Detection	Containment ventilation isolation, prevent further release
ERS-1303, 1403, 2303, 2403	Containment Radioiodine - Detection	None
ERS-1305, 1405, 2305, 2405	Containment Low Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-1307, 1407, 2307, 2407	Containment Mid Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-1309, 1409, 2309, 2409	Containment High Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-7401, 8401	Control Room Area Monitor	Isolate Control Room Ventilation
SRA-1905, 2905	Steam Jet Air Ejector Low Range Noble Gas - Detect primary to secondary leakage	None
SRA-1907, 2907	Steam Jet Air Ejector Mid Range Noble Gas - Detect primary to secondary leakage	None
SRA-1909, 2909	Steam Jet Air Ejector High Range Noble Gas - Detect primary to secondary leakage	None
R-17 A, R-17 B	Component Cooling Water Loop Liquid Monitor - Detect leaks from RCS or RHR into the CCW system	Isolate CCW surge tank vent
RRS-1001	Waste Disposal System Liquid Effluent Monitor	Automatic valve closure to prevent further release
R-5	SFP Area Monitor	Place SFP ventilation into service
R-19	Steam Generator Blowdown Liquid Monitor - Detect primary to secondary leakage via common blowdown header	Close containment isolation valves in the blowdown lines, the sample lines, and the blowdown tank condensate drain line.
R-20, R-28	Essential Service Water Liquid Monitor - Detect leakage in the containment spray heat exchangers, (post LOCA).	None
R-24	Steam Generator Blowdown Treatment System Liquid Monitor - measure activity in the blowdown liquid after it passes the treatment demineralizer.	Isolate steam generator blowdown system.
VRA-1501, 2501	Unit Vent Airborne Particulates - Detection	None
VRA-1503, 2503	Unit Vent Radiodines - Detection	None
VRS-1505, 2505	Unit Vent Low Range Noble Gas - Detection	Gas decay tank isolation valves *
VRS-1507, 2507	Unit Vent Mid Range Noble Gas - Detection	Gas decay tank isolation valves *
VRS-1509, 2509	Unit Vent High Range Noble Gas - Detection of accidental release	Sample pathway bypass of channels 1, 3, 5, 7 to sample pallet
SRA 1805, 2805	Gland Seal Condenser Exhaust - Low range detection	None
SRA 1807, 2807	Gland Seal Condenser Exhaust - Mid range detection	None
SRA 1809, 2809	Gland Seal Condenser Exhaust - High range detection	None
Unit Vent Continuous air flow sampler	Tritium sampling	None

* Available setpoint is used to accomodate 1) normal operation, and 2) gas decay tank release.

11.4 PLANT HEALTH PHYSICS PROGRAM

An extensive health physics program under the supervision of trained professional and technical personnel is established for the plant.. Appropriate administrative controls are developed to ensure that procedures and other requirements involving radiological safety considerations are strictly adhered to.

11.4.1 Facilities

Facilities for radiation protection, personnel and equipment decontamination, and chemical and radiochemical analysis are located in the auxiliary and turbine buildings.

a) Access Control Facilities

Access control facilities are located at the entrances to the radiologically restricted areas of the plant. These facilities are used to control normal access of personnel into posted radiologically controlled areas where the use of protective clothing and/or special radiation monitoring equipment might be necessary. Protective clothing and calibrated radiation monitoring instrumentation are available for personnel to use as needed. A locker room where personnel change into protective clothing is also available.

Radiation protection offices are located at the access control facilities.

Personnel decontamination facilities are located at the access control facilities where appropriate measures may be taken to decontaminate personnel if needed. The personnel decontamination facilities contain a wash basin, shower (east facility only), and a radiation survey instrument for monitoring personnel for residual levels of contamination following decontamination efforts.

b) Decontamination Facilities

Personnel decontamination facilities are located at the access control facilities as described above. The liquid wastes collected are normally sent to the waste disposal system prior to release.

Tools and equipment are normally decontaminated in the hot tool decontamination facility located on the 633' elevation.

Protective clothing is processed by an off-site vendor. Soiled protective clothing is periodically collected, packaged, and transported to the vendor for cleaning. Protective clothing meeting the release limits specified is received from the vendor and placed in the change-out facilities for personnel use. Sufficient protective clothing is maintained on-site.

c) Chemistry Facilities

- A sampling room where radioactive and potentially radioactive samples are collected is located in the auxiliary building.
- A chemical laboratory where radioactive samples are analyzed is located in the radiologically restricted area of the auxiliary building near the auxiliary building access control facility.
- A chemical laboratory, also used for analyzing non-radioactive samples, is located in the turbine building.
- A chemistry counting room where samples are analyzed for radioactivity is located in the radiologically restricted area near the auxiliary building access control facility.
- Chemical Supervisor's offices are located near the turbine side (east) access control facility.
- Eyewash stations and, when appropriate, safety showers are located near all chemical handling and analysis areas.

Hot Laboratory

This laboratory can be used for radiochemical work, such as chemical separations, etc., in addition to routine water chemistry. The hot laboratory includes three enclosed ventilated hoods and a ventilated storage cabinet. The flow from all these vents, as well as the rest of the ventilation flow from this area, is filtered and monitored by the auxiliary building ventilation system as described in Subchapter 9.9. Liquid wastes from the sinks in this laboratory are collected and analyzed for radioactivity before treatment or release as described in Subchapter 11.1. The waste liquid from the deluge shower, the face-eye wash and the floor drains are also treated as contaminated liquid waste as described in Section 11.1 of the FSAR.

Counting Room

This room is provided for the measurement of radioactivity in contained liquid, solid, gaseous, or particulate collection samples. No liquid wastes are disposed of in this room. Solid wastes are handled as described in Subchapter 11.1.

The ventilation for this room is part of the auxiliary building ventilation system.

d) Radiation Protection Calibration Facility

The calibration facility is located off the 609' Auxiliary Building Crane Bay. This provides an area for storage and calibration of instruments and storage of higher activity radioactive calibration sources. The room is also used as a repair facility for radiation protection equipment.

Sources available for calibrations purposes are:

- One Cs-137 Shepard model 89 calibrator and one Shepard model 1425 calibrator capable of low level to very high level radiation intensities.
- One PuBe neutron source for calibration of neutron instruments.

Access to the facility is normally controlled using the computer controlled keycard system. Methods are available for locking the facility should the computer system fail.

11.4.2 Radiation Control

Personnel exposure to radiation is maintained as low as reasonably achievable (ALARA) by controlling access, through radiation work permits and by the use of shielding when appropriate.

11.4.3 Access Control

Access is controlled to the radiologically restricted area on the basis of radiation levels and/or the presence of radioactive materials or contamination.

Any area in which radioactive material is stored, handled or processed, or in which radiation dose rates are ≥ 0.2 m R/hr is designated a radiologically restricted area. These areas are designated by signs such as RESTRICTED AREA, RADIOACTIVE MATERIALS, etc.

Within the radiologically restricted area access is further controlled based on radiation and contamination levels. The entrances to all areas within the restricted area are posted with signs stating CAUTION, DANGER, OR GRAVE DANGER and the appropriate area designation: radiation area, high radiation area, extreme high radiation area, very high radiation area, controlled surface contamination area, high controlled surface contamination area, and airborne radioactivity area.

Radiation protection personnel make routine surveys of accessible areas of the plant to establish current status of the radiation levels in these areas. Radiological information is posted showing radiation levels (and significant radiation sources) in the area.

11.4.4 Contamination Control:

The spread of contamination from one area to another is minimized by the use of step-off pads. Bags are used to carry contaminated tools and equipment.

Personnel monitoring devices such as, count rate meters with Geiger-Mueller detectors, hand and foot monitors and whole body contamination monitors are located throughout the radiological restricted area for use by personnel to monitor themselves for contamination. Personnel are also monitored for contamination in the access control facility prior to leaving the auxiliary building or a posted controlled surface contamination area.

All personnel are monitored for contamination when leaving the Protected Area through the Security access control area.

Radiation protection personnel conduct routine contamination surveys of accessible areas of the plant. Any area contaminated above set procedural levels is posted appropriately and decontaminated as soon as, and if, practical. Radiological information is available showing the contamination and radiation levels. Appropriate protective clothing to be worn when entering the radiologically restricted area is specified on radiation work permits or by radiation protection personnel.

11.4.5 Personnel Contamination Control

The potential for personnel contamination is minimized by the use of several types of protective clothing. The type and number of each specific piece of protective clothing to be worn is specified by the radiation work permit or by radiation protection personnel based on known or suspected contamination levels.

Normally, most of the plant is accessible to personnel in street or conventional clothing.

11.4.6 Airborne Contamination Control

Airborne contamination is minimized by maintaining loose surface contamination at a low level. Efforts are made to install temporary process ventilation control devices to avoid approaching or exceeding 10 CFR 20 levels. If the use of this equipment is not feasible or calculations show that the total dose would be lower, a provision may be made for personnel to use respiratory protection equipment to minimize personnel exposure to airborne radioactivity. Allowances are made for determining if personnel in radiologically restricted areas are subjected to concentrations in excess of Appendix B, Table 1, Column 3 of 10 CFR 20.

Several types of respiratory protection are available for personnel use:

- a) Full-Face Cartridge Mask
- b) Supplied - Air
 - Full-Face Mask, Constant Airflow
 - Polyethylene Hood
- c) Self-Contained Breathing Apparatus

11.4.7 External Radiation Dose Determination

Personnel expected to receive occupational dose while on site are issued self-reading dosimeters and/or thermoluminescent dosimeters prior to entering the radiologically restricted areas.

Thermoluminescent dosimeters are normally used to measure personnel radiation doses and are normally the primary basis for determining the dose of record. Self-reading dosimeters are normally used to provide a continuous readout of occupational dose accumulated between thermoluminescent dosimeter processing and also provide a backup to the thermoluminescent dosimeter data and may be used either concurrently with or in lieu of TLDs in some circumstances. Dose records are maintained on personnel that receive dose while on site.

Extremity dosimeters are issued as required. Neutron exposure monitoring is accomplished through dosimeters or the use of measured neutron dose equivalent rates and stay times.

Provisions are in place for the determination of skin dose in the event of significant skin contamination or upon repeated entry into areas having a significant airborne noble gas concentration.

An annual tabulation of the number of plant, utility, and other personnel receiving dose greater than 100 mrem in a calendar year and the associated collective dose according to work and job function is included in the annual operating report. In addition, all doses are reported as required by 10 CFR 20.

Plant supervisors are kept informed of plant personnel doses as results are received from the computerized radiation protection system.

11.4.8 Internal Radiation Dose Determination

Passive internal monitoring is routinely performed for all workers in order to detect external and/or internal deposition of radioactive materials. Additionally, investigational whole body counts are also conducted when a potential intake of radioactive material may have occurred. Any committed effective dose equivalent determination results are maintained with other dose records consistent with records retention guidelines. The determination of dose will be based upon approved radiological protection methodology.

11.4.9 Radiation Protection/Radiochemistry Instrumentation

a) Counting Room Instrumentation

Counting room instrumentation includes gamma spectroscopy equipment, tritium analysis equipment and alpha and beta counting equipment (low background).

b) Portable Radiation Detection Instrumentation

Portable instrumentation includes devices for measuring thermal and fast neutrons; alpha contamination; low, mid, and high range gamma exposure rates; and personnel contamination (e.g., friskers). A transfer standard ionization chamber is available.

c) Air Sampling Instrumentation

Air sampling instrumentation includes low and high volume air samplers and continuous air samplers which measure for particulate, radioiodines and noble gases.

d) Personnel Monitoring Instrumentation

Personnel monitoring instrumentation includes devices appropriate for monitoring deep, shallow, and lens dose equivalents are used.

e) Emergency Instrumentation

Emergency instrumentation located throughout the plant and offsite include low and high range exposure rate meters, air samplers and personnel monitoring devices.

- Self-reading dosimeters of appropriate range.

11.4.10 Tests and Inspections

a) Shielding

To assure shielding integrity is maintained, radiation surveys of plant areas are performed routinely.

b) Area and Process Radiation Monitors

Each technical specification area and process monitor is regularly tested to assure that the:

- Calibration of the monitor is correct.
- Alarm and trip points function properly.

In addition, each non technical specification radiation monitor is regularly tested to ensure the calibration of the monitor is correct.

c) Portable and Semi-Portable Radiation Monitors

This equipment is regularly tested to assure correct calibration and function.

Methods, Frequency, and Standards Used in Calibrating Instruments

1) Method

Beta-gamma portable survey instruments and portable count rate instruments are calibrated as described in Subchapter 11.4, using written procedures. Neutron survey instruments are typically calibrated offsite. Counting and measuring instruments are calibrated using low level calibration sources.

Small check sources are available for checking the operation and response of survey instruments, portal monitors, and contamination monitoring instruments.

2) Frequency

Radiation protection instruments are periodically checked, repaired, and/or calibrated by qualified personnel.

3) Standards

The calibration sources in the calibration facility are themselves calibrated using an instrument for which the standardization is traceable to the National Institute of Standards and Technology.

FIGURES 11.4-1 THROUGH 11.4-4 DELETED

11.4-12

July 1995

TABLE 11.5-1

DESIGN AND MEASURED EQUILIBRIUM REACTOR COOLANT FISSION PRODUCT
ACTIVITIES FOR OPERATING PWR'S AND CALCULATED VALUES FOR THE D. C. COOK STATIONS

	<u>**Ginna Station</u>			<u>**Beznau Station</u>			<u>Cook Station</u>
	Design* Value uc/cc	Measured Value uc/cc	Ratio <u>Measured</u> (<u>Design</u>)	Design* Value uc/cc	Measured Value uc/cc	Ratio <u>Measured</u> (<u>Design</u>)	Design* Value uc/cc
Total Activity	216	71	0.33	299	168	0.73	207
Isotopic Activity (Key Isotopes)							
I-131	1.53	0.56	0.37	0.96	0.75	0.78	1.7
I-133	2.55	1.7	0.67	1.74	2.0	1.16	2.6
Xe-133	184	45	0.24	200	119	0.60	178
Cs-134	0.19	0.06	0.32	0.22	0.075	0.35	0.13
Cs-137	0.94	0.37	0.40	1.53	0.22	0.15	0.8

*Based on an assumed 1% defect level.

**Amendment 20 to original FSAR (Mar, 1972)

TABLE 11.5-2
BLOWDOWN TREATMENT SYSTEM COMPONENTS

Pump

Number	1 per unit
Fluid	Steam generator blowdown
Pressure, Suction	Atmospheric
Temperature	200°F
Head	125 ft.
Flow	60 gpm
Type	Horizontal centrifugal
Material, Casing	Steel
Impeller	Bronze
NPSH, minimum Ft. H ₂ O	2.5

Heat Exchanger

Number	1 per unit
Shell Side (blowdown liquid)	
Inlet Temperature	200°F
Outlet Temperature	120°F
Max. pressure	70 psi
Operating pressure	50 psi
Flow	60 gpm
Material	304 Stainless Steel
Pressure drop, normal	4 psi
maximum allowable	15 psi
Tube Side (non-essential service water)	
Inlet Temperature	76°F
Outlet Temperature	106°F
Max. pressure	150 psig
Operating pressure	75 psig
Flow	160 gpm
Material	304 Stainless Steel
Pressure drop, normal	5 psi
maximum allowable	9 psi

11.6 RADIOACTIVE MATERIALS SAFETY

11.6.1 MATERIALS SAFETY PROGRAM

Licensed material is used, handled by, or under direction of one of those designated as an individual user in the license. Each individual using such radioactive material is familiar with the restrictions and limitations placed upon that particular source.

An inventory of licensed material on site is maintained and periodically updated. The inventory record contains, as a minimum, the use and locations of licensed material, and the receipt date and final disposition of material no longer in use.

11.6.1.1 Security

Sealed sources, with the exception of those installed in or on equipment and small check or calibration sources, are kept under lock and key when not in use.

11.6.1.2 Source Handling

Whenever radioactive sources are handled or used, care is taken to avoid unnecessary exposure, the spread of contamination, or damage to the source. In addition, a Radiation Work Permit is required for the following situations dealing with licensed sources:

1. Any time a sealed source capable of giving an exposure greater than 5 mrem/hr at 30 centimeters is used or handled.
2. Any time a sealed source is installed in, or removed from, equipment on which it is normally installed.

11.6.1.3 Material Handling of Special Nuclear Material (SNM)

A Nuclear Materials Management Group has the overall responsibility for SNM and the associated inventory records.

There are four Item Control Areas (ICA) designated for the plant. The custodian for each of these areas is responsible for all SNM entering, leaving, or being stored in this area. All material transfer documents for this ICA are signed by the ICA Custodian or alternate. The chain of responsibility between the Custodian and the Nuclear Materials Management Group is shown in Figure 11.6-1.

Figure 11.6-2 presents a diagram of the flow of SNM through the plant. Responsibility for the control and accountability of each physical unit of SNM begins with the on-site receipt. The responsibility for this control and accountability terminates for each physical unit when the physical unit of SNM is shipped off-site.

Each time fuel is transferred into or out of the ICA, the transfer is documented by appropriate entries on an ICA Transfer Form.

Inventory of all SNM must be taken on a periodic basis.

11.6.1.4 SNM Transfer Procedure

Fuel assembly, fission chamber detector, and moveable miniature neutron flux detector (MMNFD) movement from one ICA to another ICA is not to be initiated until an ICA Transfer Form has been approved by the Reactor Engineering Manager or alternate*. The ICA Transfer Form shows the fuel assembly and/or fission chamber detectors and/or MMNFD involved, their origin, destination, and approval by the Reactor Engineering Manager or alternate; and permits transfer of the fuel assembly, fission chamber detectors or MMNFD across the boundaries of the ICAs involved.

* During refueling, new fuel receipt, fuel exams, and fuel shuffles, the approved fuel shuffle sequence may be substituted for the ICA transfer and internal transfer forms.

Additionally, SNM movement within an ICA is administratively controlled by use of the ICA Internal Transfer Form. Each Internal Transfer Form must be signed by the Reactor Engineering Manager or alternate before it is considered approved. Each ICA Internal Transfer Form and ICA Transfer Form has a limited lifetime of ten calendar days as delineated on the form with the exception of new fuel receipt.

The detailed procedures for material handling and transfer of all Special Nuclear Material are described in the SNM Accountability Manual of the Donald C. Cook Nuclear Plant.

11.6.2 PERSONNEL AND PROCEDURES

The key personnel responsible for handling and monitoring the Special Nuclear Material are identified by title in the Special Nuclear Materials Accountability Manual.

Radiation Safety Instructions

The Radiation Protection Plan presents the philosophy and guidelines to be used to control the exposure to radiation and radioactive materials, and to effectively restrict the exposure of personnel within the plant and members of the general public to ionizing radiation resulting from the operation of the plant.

Safe handling and usage of radioactive materials are also described in Radiation Protection Procedures, Laboratory Procedures, Fuel Handling Procedures, and the Special Nuclear Material Accountability Manual.

11.6.3 REQUIRED MATERIALS

The isotope, quantity, form and use for all required byproduct, source and special nuclear material for the Donald C. Cook Nuclear Plant are identified in the Facility Operating License and subject to the conditions identified in the Technical Specifications.

11.6.4 RADIOACTIVE WASTE STORAGE

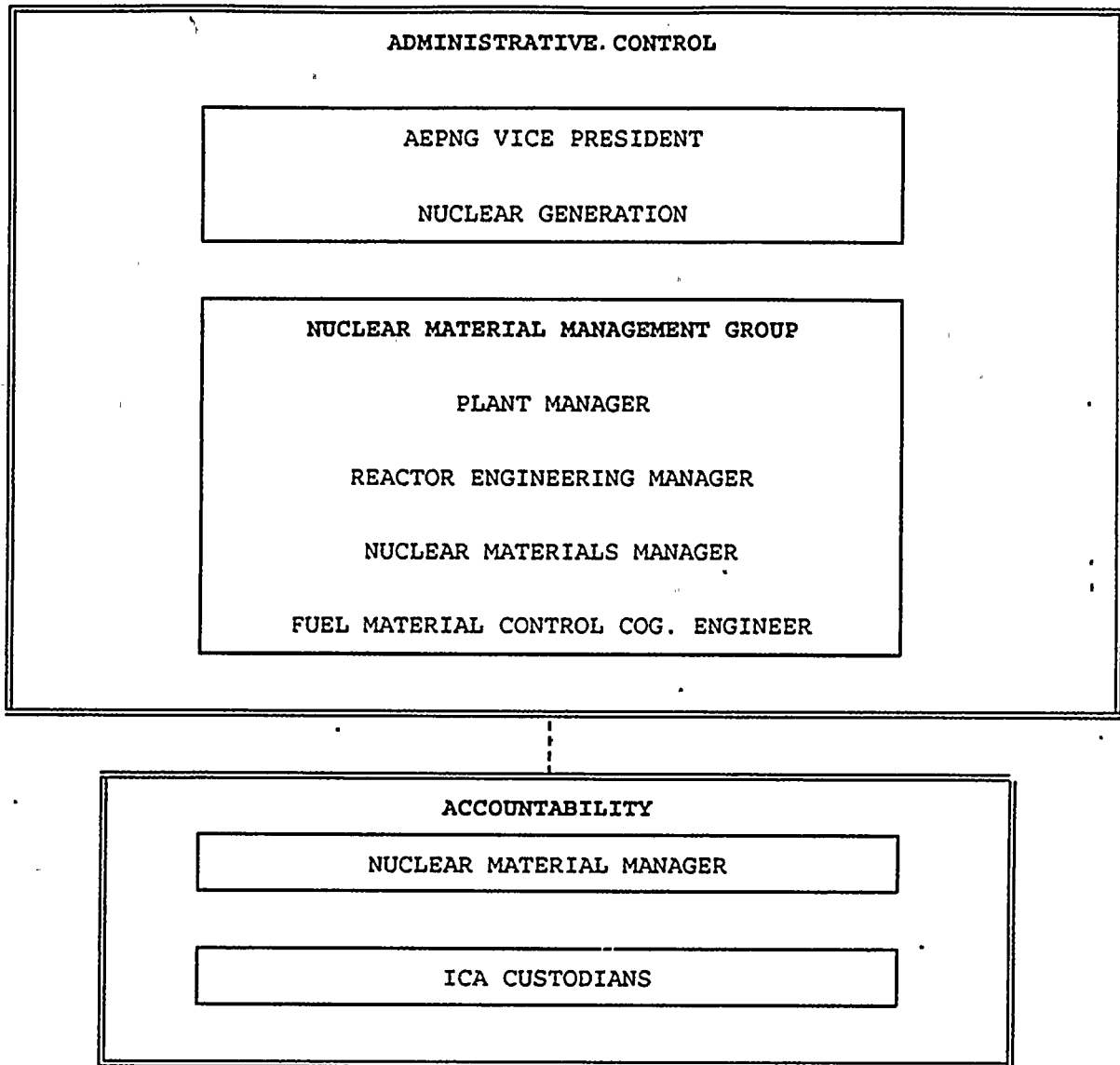
The Donald C. Cook Nuclear Plant is located in the State of Michigan. Michigan waste generators were unable to dispose of their low level radioactive waste from November, 1990 until July 1995, when the Barnwell disposal site re-opened. The Cook Nuclear Plant is storing some waste in the Radioactive Material Building.

The Radioactive Material Building was designed and constructed with the primary purpose of storing low level radioactive waste. It is located behind the Training Building about a half mile from the auxiliary and containment buildings. It has four different areas: the cell area, the dry active waste (DAW) area, the truck bay, and the service area.

The cell area will be used to store the more radioactive of the low level radioactive waste. There are twelve cells each with a pair of two-foot thick covers. Typically, waste in seven foot high, seven foot diameter, cylindrical, high integrity containers are put in the cells. They would contain filters and resin. The materials are placed into the cells with an overhead crane.

The DAW area is used to store dry active waste in boxes and drums. This waste is less radioactive and is handled using a forklift.

The truck bay was designed to accommodate a tractor and trailer. The waste is unloaded in this area with a forklift or the overhead crane. The truck bay is adjacent to the cell area.



ORGANIZATION AND FUNCTIONAL STRUCTURE

FIGURE 11.6-1

JULY 1997

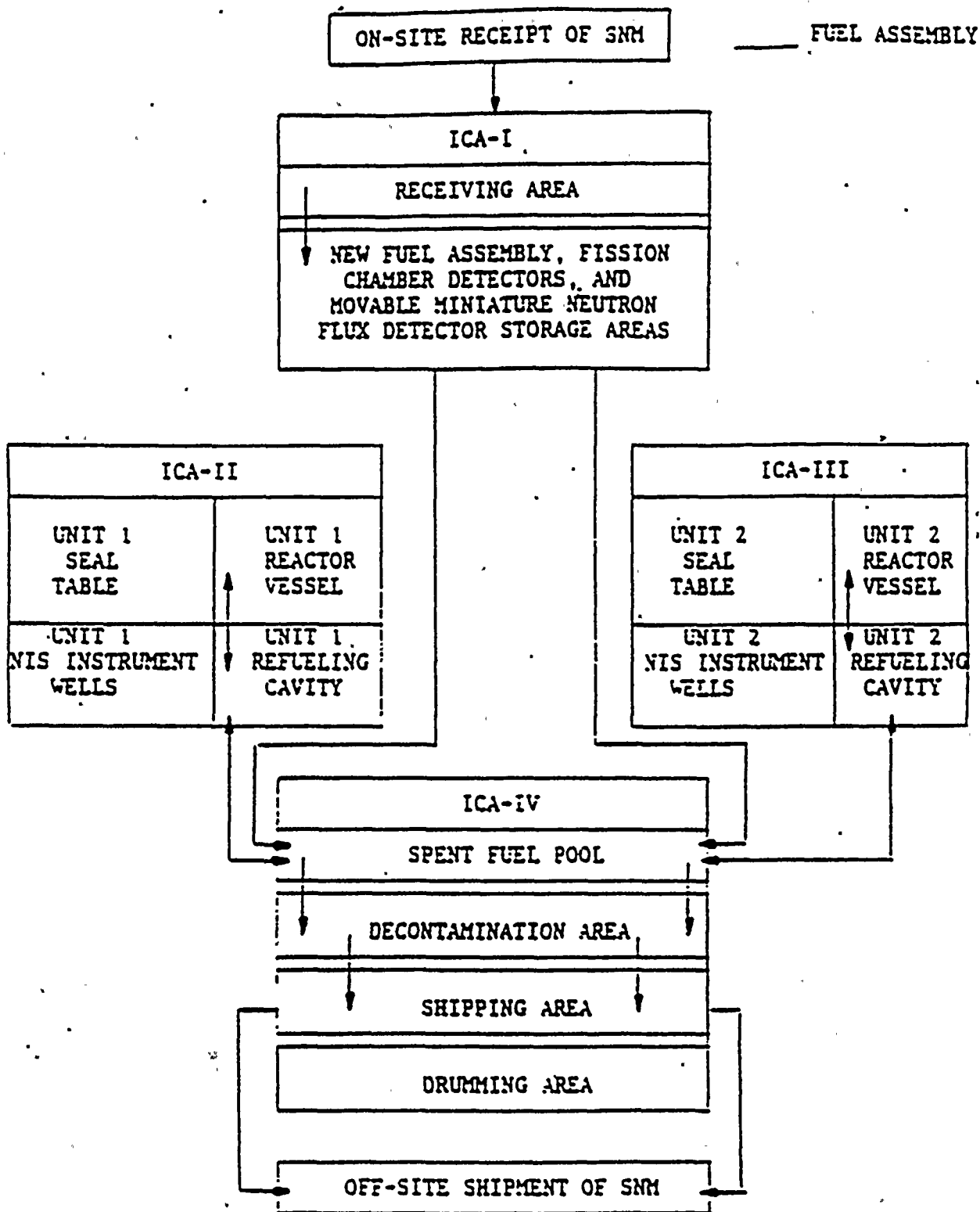


Figure 11.6-2a

FUEL ASSEMBLY FLOW CHART

July, 1986

12. CONDUCT OF OPERATIONS

12.1 ORGANIZATION AND RESPONSIBILITY

Overall responsibility for all plant operations is vested in the American Electric Power Company. The Westinghouse Electric Corporation provided technical assistance during the period of pre-operational testing, core loading, initial startup and pre-commercial operation.

The approach to operating the plant was compatible with the organizational concepts and operational philosophy that have been successfully employed for many years in the Company's conventional thermal plants. Many of the plant personnel were initially drawn primarily from the existing American Electric Power System conventional plant staff, and most had significant conventional power plant experience, plus varying degrees of nuclear experience.

The plant organization is shown in the QAPD. This organization is in accordance with the organizational practices of the Company for conventional generating plants, with increased emphasis on the technical functions required for the operation of a nuclear plant. The Site Vice President through the Plant Manager and appropriate department superintendents, provides supervision for the plant personnel and maintains direct responsibility for all plant activities. The plant organization is under the functional direction of and receives technical support from the American Electric Power Service Nuclear Generation Group, located on-site and in Buchanan, Michigan.

The individuals selected for Site Vice President position and each of the professional staff positions in the operating organization meet or exceed the minimum qualifications of ANSI N18.1-1971/ANS 3.1-71 for comparable positions. In addition: (1) the plant radiation protection manager meets or exceeds the qualifications of Regulatory Guide 1.8, September 1975; (2) the shift technical advisors have a Bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents; and (3) the operations superintendent holds or has held a senior operator license, or has been certified with senior operator equivalent knowledge.

To ensure safety and efficiency of operation of the plant, administrative procedures have been established to review the following:

1. All Plant Manager Instructions and revisions.
2. All plant operating procedures that may involve an unreviewed safety question as defined in 10 CFR 50.59.
3. Changes in plant operating procedures that may involve an unreviewed safety question as defined in 10 CFR 50.59.
4. Design changes that may involve an unreviewed safety question as defined in 10 CFR 50.59.
5. Proposed tests and experiments that may involve an unreviewed safety question as defined in 10 CFR 50.59.
6. Proposed changes to the Technical Specifications
7. Violations of the Technical Specifications
8. All reportable events.

Two committees have been established for this purpose: the Plant Nuclear Safety Review Committee (PNSRC) and the Nuclear Safety and Design Review Committee (NSDRC). The members of these committees, their responsibilities and their authority, have been noted in the Administrative Control sections of the Technical Specifications.

Audits of facility operations are conducted as previously described in Section 1.7.

12.6 NUCLEAR DESIGN AND SUPPORT CAPABILITY

The Cook Nuclear Plant organization is under the functional direction of and receives technical support from the AEP Nuclear Generation Group (NGG) which is headquartered at 500 Circle Drive, Buchanan, Michigan.

The American Electric Power Service Corporation (AEPSC), with offices currently at 1 Riverside Plaza, Columbus, Ohio, 43215, provides engineering operational support, design, legal, accounting and related services to the AEP System. Consequently, AEPSC employs engineers, designers, and drafters who are experienced in the design and construction of electric generating stations. AEPSC acts as the architect-engineer for the AEP system and as such has designed and built nearly all of the System's present generating capacity. AEPSC was responsible for the design of the Donald C. Cook Nuclear Plant and for construction of the entire plant. Design and fabrication of the nuclear steam supply system components and the initial fuel load were performed by the Westinghouse Electric Corporation and its subcontractors.

AEPSC began training employees in nuclear power in 1952 with the assignment of several engineers, designers and maintenance specialists to Oak Ridge National Laboratory, Bettis Atomic Power Laboratory, Knolls Atomic Power Laboratory, and various projects at the National Reactor Testing Station. Since that time, a large number of additional AEP personnel have completed assignments at various national laboratories or pursued graduate level work in nuclear engineering at leading universities, while others have attended shorter courses and seminars in various aspects of the nuclear power industry.

In 1953, AEP became one of the co-founders of the Nuclear Power Group, Inc., and in the ensuing years participated, technically and financially, in the development of the Dresden Nuclear Power Station. This group was then dissolved. It evolved into the East Central Nuclear Group (ECNG); and AEP was instrumental in the new group's formation. ECNG was comprised of 10 utility companies. Its goal was to research and develop

nuclear power. The AEP Service Corporation acted as architect-engineer administrator and research and development manager for the group.

ECNG's major undertakings were the development with the General Nuclear Engineering Corp. of the Florida West Coast Nuclear Group gas-cooled, heavy water moderated reactor from 1957-61, the joint development with Babcock & Wilcox of a Supercritical Pressure Steam Cooled Fast Breeder Reactor from 1963-65, the development of a Gas Cooled Fast Breeder Reactor in cooperation with Gulf General Atomic from 1965-67, the development with General Electric of a Steam Cooled Fast Breeder Reactor in 1967-1968, and from 1968 through 1982, a further project with General Atomic for the Gas Cooled Fast Breeder Reactor, first through an informal group of utilities and then through Helium Breeder Associates.

In addition, ECNG, with the aid of AEPSC staff and S. M. Stoller Associates, made a thorough study of the "The Outlook for Uranium", a survey of the likely demand and availability of nuclear fuel; and with the Massachusetts Institute of Technology produced a study of the "Effects of Changing Economic Conditions of Fuel Cycle Costs". This program investigated the ten-projected effects of private ownership of nuclear power economics. ECNG is now dissolved.

At the present time, the AEP Nuclear Generation Group consists of professional personnel who devote their professional energies to Cook Nuclear Plant and nuclear power industry issues. In addition, there are other individuals at AEPSC with substantial nuclear training or specific nuclear experience in key engineering, design and operating positions.

The Reactor Control and Protection System is relied upon to protect the core and reactor coolant boundary against the following fault conditions:

1. Uncontrolled RCCA bank withdrawal from a subcritical condition
2. Uncontrolled RCCA bank withdrawal at power
3. RCCA misalignment (this encompasses RCCA drop)
4. Uncontrolled Boron dilution
5. Loss of reactor coolant flow (including locked rotor)
6. Start-up of an inactive reactor coolant loop
7. Loss of external electrical load and/or turbine trip
8. Loss of normal feedwater
9. Excessive heat removal due to feedwater system malfunctions
10. Excessive load increase
11. Loss of offsite power (LOOP) to the station auxiliaries
12. Turbine-generator overspeed

RCS Reduced Temperature and Pressure (RTP) Operation

The safety analyses presented in this chapter include where necessary the effects of reduced RCS temperature (reactor vessel average temperature) and pressure (pressurizer pressure) operation for Cook Nuclear Plant Unit 1. Operation with a core power of 3250 MWt is supported in the range of full power primary vessel average temperatures between 553°F and 576.3°F at RCS pressure values of 2100 psia or 2250 psia. In addition, the evaluation performed supports a maximum steam generator tube plugging level of 30%. Table 14.1-1 (Cases 1 and 2) presents the range of conditions for the reduced RCS temperature and pressure operation. Steamline break mass and energy releases were evaluated to account for full power primary vessel average temperatures between 547°F and 581.3°F, thus bounding the licensed RCS temperatures of Unit 2.

The effort to support an increased steam generator tube plugging (SGTP) level of 30% consisted of re-analyses and evaluations. For the non-LOCA transients that were evaluated, the "analysis of record" continues to be the previously performed analysis which supports the future rerating of Unit 1. As such, the range of conditions presented as Cases 3 and 4 of Table 14.1-1 continue to apply. Thus, these particular transients have been shown to satisfy the applicable acceptance criteria via the rerating analysis and the subsequent 30% SGTP evaluation.

The results of the non-LOCA occurrences safety analyses presented in the following sections show that the reduced RCS temperature and pressure operation for Unit 1, can satisfy the applicable FSAR safety limits. The safety analyses support a maximum average steam generator tube plugging level of 30%, provided the minimum measured RCS flow of 84,775 gpm/loop is met and the RCS temperature and pressure presented in Table 14.1-1 (Cases 1 and 2) are not exceeded. A 5% RCS flow asymmetry is also supported by the safety analyses. Specifically, a total RCS flow rate of 339,100 gpm with a reduction of RCS flow in one loop of 5% below the average loop flow rate was evaluated in the safety analyses. As long as the total minimum measured RCS flow is equal to or greater than 339,100 gpm, the flow rate in one loop may be below 84,775 gpm by as much as 5%. Should the flow rate in more than one loop be below 84,775 gpm, a total loop flow short

fall less than or equal to 5% of 84,775 gpm is supported by the safety analyses, provided the total minimum measured RCS flow is equal to or greater than 339,100 gpm.

Reactor Protection System (RPS) and Engineered Safety Features (ESF)
Setpoints Assumed in Analysis

To enhance operating flexibility for the RCS reduced temperature and pressure operation with a maximum average steam generator tube plugging level of 30%, certain Reactor Protection System (RPS) setpoints and emergency diesel generation (EDG) requirements were revised. The revised RPS setpoints include the overtemperature ΔT (OT ΔT) and the overpower ΔT (OP ΔT) reactor trips. The revised EDG requirement is relaxed, such that the total EDG start-up delay time supported by the safety analyses is now 30 seconds.

A reactor trip is defined for analytical purposes as the insertion of all RCCAs except the most reactive one which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCCA condition existing at a time when shutdown is required. The response times of the reactor trip system instrumentation is listed in Table 7.2-6.

Instrumentation is provided for continuously monitoring all individual RCCAs together with their respective bank position. This is done in the form of a deviation alarm system. Procedures are established to correct deviations. In the worst case, the plant will be shutdown in an orderly manner and the condition corrected.

In summary, reactor protection is designed to prevent cladding damage in all fault conditions listed previously. The most probable modes of failure in each RPS channel result in a signal calling for the protective reactor trip. Coincidence of two out of three (or two out of four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria. The reactor protection system is designed in accordance with the IEEE 279 "Standard for Nuclear Plant Protection Systems," August, 1968.

Reactor Trip Setpoints

Revised OTAT and OPAT setpoints were calculated based on the new core thermal safety limits using the methodology described in Reference 1. Figure 14.1-1 presents the allowable RCS loop average temperature and ΔT for the minimum measured flow and power distribution as a function of RCS pressure. Figure 14.1-1 represents the most limiting operating configuration (nominal $T_{avg} = 576.3^{\circ}\text{F}$, nominal RCS pressure = 2100 psia) of the range of conditions described in Table 14.1-1 for the calculation of the OTAT and OPAT trip setpoints. The boundaries of operation defined by the OPAT and OTAT trip setpoints are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a reactor trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given RCS pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNBR limit line at any point for a given RCS pressure.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

The safety limit value, which was used as the DNBR limit for all accidents analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 2), is conservative compared to the actual design DNBR value required to meet the DNB design basis.

Table 14.1-2 presents the limiting reactor trip setpoints assumed in the safety analyses and the time delay assumed for each trip function. The differences between the limiting reactor trip point assumed for the safety analyses and the normal reactor trip point represent an allowance for instrumentation channel error and setpoint error. Nominal reactor trip setpoints are specified in the plant Technical Specifications. Time response testing demonstrates that actual instrument time delays are equal to or less than the assumed values. Additionally, reactor protection system channels are calibrated and instrument response times determined periodically in accordance with the Technical Specifications.

The safety analyses presented in the following sections assume that the reference average temperatures (T' and T'') used in the OTAT and OPAT setpoint equations are rescaled to the full power average temperature each time the cycle average temperature is changed. It is also assumed that the reference pressure (P') in the OTAT equation is set equal to the appropriate nominal RCS pressure (2250 psia or 2100 psia). Figures 14.1-1 through 14.1-4 illustrate the OTAT and OPAT setpoints for the endpoints of the range of average temperatures for the 30% SGTP conditions of Unit 1 at either 2100 psia or 2250 psia. The safety analyses also assume recalibration of the NIS excore detectors to compensate for the changes in coolant density each time the cycle operating conditions are changed.

Methodology

The Unit 1 non-LOCA safety analyses for the RCS reduced temperature and pressure operation with 30% steam generator tube plugging (SGTP) were performed using current Westinghouse methodology and computer codes. For the safety analyses presented in the following sections, the results show that the RCS reduced temperature and pressure operation with a maximum SGTP level of 30% for Unit 1, satisfy the applicable FSAR acceptance criteria.

Initial Conditions

All transients have been analyzed or evaluated to demonstrate that RCS reduced temperature and pressure operation with a maximum average steam generator tube plugging level of 30% can be supported. Several of the transients reflect initial condition values consistent with the previously analyzed transients that were performed to support the rerating (i.e., 3411 MWt core power) of Unit 1.

For each of the transients reanalyzed to support RCS reduced temperature and pressure operation with 30% SGTP, conservative nominal values for initial reactor thermal power and RCS temperature and pressure are assumed to bound the RCS reduced temperature and pressure operation. The initial conditions for each safety analysis are presented in Table 14.1-3.

For most transients which are DNB limited, nominal values of initial conditions and the minimum measured flow (339,100 gpm) are assumed. The allowances on reactor thermal power and RCS temperature and pressure are determined on a statistical basis and are included in the limit DNBR as described in WCAP-11397 (Reference 2). This procedure is known as the "Revised Thermal Design Procedure" (RTDP).

For occurrences that are not DNB limited or in which RTDP is not employed, the initial conditions are obtained by adding the maximum steady state errors to nominal values. The following steady state errors are considered:

- | | |
|----------------------------------|--|
| A. Core Thermal Power | + 2% calorimetric error allowance |
| B. RCS Average Temperature | $\pm 4.1^{\circ}\text{F}$ controller deadband and
measurement error allowance; plus a $+1.0^{\circ}\text{F}$
bias for cold-leg streaming |
| C. RCS (Pressurizer)
Pressure | ± 67 psi - steady state fluctuations and
measurement error allowance |
| D. Reactor Flow | Thermal Design Flow (332,800 gpm) |

Table 14.1-3 summarizes initial conditions and computer codes used in the safety analysis of occurrences in Sections 14.1.1 through 14.1.12 and Sections 14.2.5, 14.2.6 and 14.2.8, and shows which transients employed a DNB analysis using the RTDP.

Reactor Core Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of RCCAs and operation instructions. The power distribution may be characterized by the radial peaking factor, $F_{\Delta H}$, and the total peaking factor, F_Q . The peaking factor limits are given in the Technical Specifications.

For occurrences which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to RCCA insertion. This increase in $F_{\Delta H}$ is included in the core thermal safety limits. All occurrences that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial RCS thermal power level defined in the Technical Specifications.

The radial and axial power distributions are input to the THINC Code as described in Chapter 3.

For occurrences which may be overpower limited, the total peaking factor, F_Q , is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

For overpower occurrences which are slow with respect to the fuel rod thermal time constant, for example the uncontrolled boron dilution incident which lasts many minutes, and the excessive load increase incident which reaches equilibrium without causing a reactor trip, fuel temperature limits are discussed in Chapter 3. For overpower occurrences which are fast with respect to the fuel rod thermal time constant, for example the uncontrolled RCCA bank withdrawal from a subcritical condition and RCCA ejection occurrences which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation is performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and fuel rod power, a typical value at beginning-of-life for high power fuel rods is approximately 7 seconds.

Reactor Trip

A reactor trip signal acts to open the two trip breakers connected in series feeding power to the RCCA drive mechanism coils. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each reactor trip function, including delays in signal actuation, in opening the reactor trip breakers, and in the release of the RCCAs by the mechanisms. The total delay to reactor trip is defined as the time delay from the time that reactor trip conditions are reached to the time the RCCAs are free and begin to fall. The time delay assumed for each reactor trip function is given in Table 14.1-2.

The difference between the limiting reactor trip setpoint assumed for the safety analysis and the nominal reactor trip setpoint represents an allowance for instrumentation channel error and setpoint error.

The instrumentation drift and calorimetric errors used in establishing the maximum power range high neutron flux setpoint are presented in Table 14.1-4.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in RCCA worth as a function of RCCA position. RCCA positions after the reactor trip have been determined experimentally as a function of time using a prototype RCCA under simulated flow conditions. The resulting RCCA positions were combined with the RCCA worths to define the negative reactivity insertion as a function of time, according to Figure 14.1-5.

Other Assumptions

Those analyses that model the mitigation effects of Protection and/or Engineered Safety Features have used the response times provided in Table 7.2-6 and 7.2-7.

Some input assumptions differ somewhat from values that may be found elsewhere in the UFSAR. In particular, Tables 14.1-4, 14.1-5, and 14.1-6 display RCS volumes, steam generator mass, RCS pressure drops used in the current analyses. These tables can be found in reference 7. Table 14.1-7 lists the RCS pressure drops at Best Estimate flow calculated at 0% steam generator tube plugging and at 30% SGTP (Reference 7).

The time to draindown the RWST and the time to switchover to recirculation cooling affects the LOCA containment integrity analysis. For peak pressure considerations, it is conservative to switchover to recirculation sooner because of the decreased cooling effect during the recirculation phase of operation of the safety injection, the upper compartment spray, and the lower compartment spray. Because the fluid enthalpy increases during the recirculation mode, the safety injection and spray efficiency is diminished. Also, once the steam generators have equilibrated, the mass and energy releases are determined based upon a boiloff calculation that is related to the safety injection water enthalpy. The higher the enthalpy, the larger the releases. Therefore, utilizing a conservatively early switchover time sequence also results in higher and more conservative mass and energy releases to containment. Also included in the containment pressure calculation is the early part of the switchover sequence, when the containment sprays are initially drawing water

from the RWST. During this period the sprays are shut off. It is also assumed that the spray switchover sequence is started, and is completed over a 4 minute period. This results in a spray interruption (i.e., no containment spray flow) during this period. A 4 minute spray interruption is also assumed in the offsite dose analysis.

For the draindown calculation the maximum pump and spray flows are assumed during injection, and combined conservatively to shorten the time to start the switchover sequence. If necessary, valve closing time is neglected. The draindown and switchover sequence information is determined in a conservative manner to support the analytical basis for the peak pressure calculation. A table providing details of the various elements that were developed to support these assumptions is presented in Table 14.1-8. The detailed information in this table is not intended to serve as a requirement that the operators must meet while demonstrating the capability to perform emergency operating procedures related to the transfer to cold leg recirculation.

Computer Codes Utilized

Summaries of the principal computer codes used in the safety analyses are given below. The codes used in the safety analysis of each occurrence have been listed in Table 14.1-3.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.
- B. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

- C. The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 3.

LOFTRAN

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature ΔT , overpower ΔT , and high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 4.

TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes presently used for reactor core design.

The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three-dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 5.

THINC

The THINC-IV computer program, as approved by the NRC, is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core under all expected operating conditions. The THINC-IV code is described in detail in Reference 6.

Section 14.1 REFERENCES

1. Ellenberger, S. L., et. al., "Design Bases for the Thermal Overpower ΔT and Thermal Overttemperature ΔT Trip Functions," WCAP-8746, March 1977.
2. Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.
3. Hargrove, H. G., "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.
4. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.
5. Risher, D. H., Jr., and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Code," WCAP-8028-A, January 1975.
6. Friedland, A. J. and Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.
7. McFetridge, R. H., American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Engineering Report, WCAP-14286, December 1995.
8. V. VanderBurg to K. R. Worthington, Safety Review of a Proposal to Use the Valve Travel Times from the Unit 1 Steam Generator Tube Plugging and Unit 2 Uprate Programs for the Sump Recirculation Model in Lieu of Those Found in Section 6.2.2 of the UFSAR, November 13, 1996

TABLE 14.1-1

UNIT 1 DESIGN POWER CAPABILITY PARAMETERS
USED IN NON-LOCA SAFETY ANALYSES

<u>Parameter</u>	(Reduced Temperature and Pressure)		(Rerating)*	
	<u>Case 1</u>	<u>Case 2</u>	<u>Case 3</u>	<u>Case 4</u>
NSSS Power, MWt	3262	3262	3425	3425
Core Power, MWt	3250	3250	3413	3413
RCS Flow, gpm/loop	83200	83200	88500	88500
Minimum Measured Flow, gpm/loop	84775	84775	91600	91600
<u>RCS Temperatures, °F</u>				
Core Outlet	589.7	611.9	583.6	614.0
Vessel Outlet	586.8	609.1	580.7	611.2
Core Average	555.8	579.4	549.7	581.8
Vessel Average	553.0	576.3	547.0	578.7
Vessel/Core Inlet	519.2	543.5	513.3	546.2
Steam Generator Outlet	518.9	543.2	513.1	546.0
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250	2250	2250	2250
	or 2100	or 2100	or 2100	or 2100
Steam Pressure, psia	595	749	603	820
Steam Flow (10 ⁶ lb/hr total)	14.12	14.17	14.98	15.07
Feedwater Temp., °F	434.8	434.8	442.	442.
SG Tube Plugging, %	30	30	10	10

* Cook Unit 1 is not licensed to operate at the rerated conditions specified by Cases 3 and 4 with 30% steam generator tube plugging (SGTP) levels. However, several events that were previously performed using these conditions were subsequently evaluated to support the 30% SGTP program. Hence, the rerated conditions are also specified in this table for completeness.

TABLE 14.1-2
REACTOR TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN SAFETY ANALYSES^c

<u>Reactor Trip Function</u>	<u>Limiting Reactor Trip Point Assumed In Analysis</u>	<u>Time Delay (Seconds)</u>
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, see Figure 3.3-1 through 3.3-4	8.0 ^a
Overpower ΔT	Variable, see Figure 3.3-1 through 3.3-4	NA ^d
High pressurizer pressure	2420 psig	2.0
Low pressurizer pressure	1825 psig ^e	2.0
High pressurizer water level	100% NRS	2.0
Low reactor coolant flow (From loop flow detectors)	87 percent loop flow	1.0
Undervoltage trip	b	1.5
Low-low steam generator level	0.0 percent of narrow range level span	2.0
High steam generator level	82 percent of narrow range level span	2.5
Turbine Trip		11.0
Feedwater Isolation		

^a Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall. The time delay assumed in the analysis supports the 6 second response time of the RTD time response, trip circuit delays, and channel electronics delay presented in the Technical Specifications.

^b No explicit value assumed in the analysis. Undervoltage trip setpoint assumed reached at initiation of analysis.

^c The control rod scram time to dashpot is 2.4 seconds.

^d Overpower (OP) ΔT reactor trip not explicitly assumed in analysis.

^e A value of 1845 psig is used in LOCA analyses.

TABLE 14.1-3
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed			DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler						
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	Refer to Section 14.1.1			W-3/WRB-1 See Section 14.1.1	No	0	146,432	547	2033
Uncontrolled RCCA Bank Withdrawal at Power (2)	LOPTRAN	+5	.54	Min and Max (3)	WRB-1	Yes	3270 1962 327	339,100	576.3 564.58 549.93	2100
RCCA Misalignment	LOPTRAN THINC	NA*	NA	NA	WRB-1	Yes	3270	339,100	576.3	2100
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA	NA	3425 0	NA	NA	NA
Loss of Forced Reactor Coolant Flow	LOPTRAN FACTRAN THINC	+5	NA	Max	WRB-1	Yes	3270	339,100	576.3	2100
Locked Rotor (Peak Pressure)	LOPTRAN	+5	NA	Max	NA	NA	3335	332,800	581.4	2317
Locked Rotor (Peak Clad Temp)	LOPTRAN FACTRAN	+5	NA	Max	NA	NA	3335	332,800	581.4	2033

*NA - Not Applicable

(1) Minimum Doppler power defect (pcm/%power) = $-9.55 + 0.035Q$ where Q is in % power

(2) Multiple power levels, Tavg, and reactivity feedback cases were examined

(3) Maximum Doppler power defect (pcm/%power) = $-19.4 + 0.065Q$

(4) Minimum and Maximum reactivity feedback cases were examined

TABLE 14.1-1 (Continued)
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed			DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density (ADK/gm/cc)	Doppler						
Locked Rotor (Roda-in-DNB)	LOPTRAN FACTRAN THINC	+5	NA	Max	WRB-1	Yes	3270	339,100	576.3	2100
Loss of Electrical Load and/or Turbine Trip (4)	LOPTRAN	+5	.54	Max and Min	WRB-1	Yes	3262	339,100	576.3	2100
Loss of Normal Feedwater(5)	LOPTRAN	+5	NA	Max	NA	NA	3494	354,000	551.5	2285
Excessive Heat Removal(5) Due to Feedwater System Malfunction	LOPTRAN	NA	.54	Min	WRB-1	Yes	3425 0	366,400 (6)	578.7 547	2100
Excess Load Increase(5) Incident	LOPTRAN	NA	0 and .54	Max and Min	WRB-1	Yes	3425	366,400 (6)	578.7	2100
Loss of Offsite Power(LOOP) (5) to the Station Auxiliaries	LOPTRAN	+5	NA	Max	NA	NA	3494	354,000	542.5	2285
Rupture of a Steam Pipe	LOPTRAN THINC	See Figure 14.2.5-1	NA	See Figure 14.2.5-2	W-3	NA	0	332,800	547	2100
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 14.2.6	NA	Min	NA	NA	3335 0	332,800 146,432	581.4 547	2033

*NA - Not Applicable

(1) Minimum Doppler power defect (pcm/%power) = $-9.55 + 0.035Q$ where Q is in % power

(2) Multiple power levels, Tavg, and reactivity feedback cases were examined

(3) Maximum Doppler power defect (pcm/%power) = $-19.4 + 0.065Q$

(4) Minimum and Maximum reactivity feedback cases were examined

(5) Values presented were used in the rerating analysis. Subsequent evaluations support the 30% SGTP parameters presented as Cases 1 and 2 of Table 14.1-1.

TABLE 14.1-4

INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS
POWER RANGE NEUTRON FLUX

	Setpoint and Error Allowances: <u>(% of rated power)</u>	Estimated Instrumentation Errors: <u>(% of rated power)</u>
Nominal Setpoint	109	-
Calorimetric Error	2	1.55
Axial power distribution effects on total ion chamber current	5	3
Instrumentation channel drift and setpoint reproducibility	2	1.0
Maximum overpower reactor trip point assuming all individual errors are simultaneously in the most adverse direction	119	-

TABLE 14.1-5
DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM
INPUT ASSUMPTIONS FOR RCS VOLUMES

<u>Input Assumptions</u>	<u>Initial Conditions</u>	
	0% SGTP	30% SGTP
Reactor Vessel (ft ³)	4826	4826
Steam Generators (ft ³ - Total)	4308 ⁽¹⁾	3393 ⁽¹⁾
Reactor Coolant Pumps (ft ³ - Total)	314	314
Loop Piping (ft ³ - Total)	1175	1175
Surge Line Piping (ft ³)	43	43
Pressurizer (ft ³)	<u>1800</u>	<u>1800</u>
Total RCS Volume (ft ³) (Ambient Conditions)	12,466	11,551
Total RCS Volume (ft ³) (Hot Conditions includes 3% for thermal expansion)	12,840	11,898

Notes:

- (1) The SG tube volume is assumed to be 762 ft³/SG (3048 ft³ total). The increase in SG tube plugging from 0% to 30% results in a total reduction in SG tube volume of approximately 915 ft³. The reduction between the SG tube volume and SG tube plugging is assumed to be a linear relationship; e.g. at 15% SGTP, total volume reduction is $0.15 \times (3048 \text{ ft}^3) = 457.2 \text{ ft}^3$.

TABLE 14.1-6
DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM
INPUT ASSUMPTIONS FOR STEAM GENERATOR SECONDARY MASS

<u>Input Assumptions</u>	<u>Initial Conditions⁽¹⁾</u>		
	0% SGTP	30% SGTP Cases	
	Original Design Case	Low Temp Case A1	High Temp Case A2
Steam generator secondary side mass (Total lbs/SG)	106,506	106,799 ⁽²⁾	112,192 ⁽³⁾

Notes:

- (1) Initial conditions are presented for SGTP levels of 0% (Original Design) and 30% to bound the range of SGTP levels.
- (2) For Tavg of 553°F
- (3) For Tavg of 576°F

TABLE 14.1-7

DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM
INPUT ASSUMPTIONS FOR REACTOR COOLANT SYSTEM PRESSURE DROP⁽¹⁾

<u>Input Assumptions</u>	<u>Initial Conditions</u>	
	<u>0% SGTP Pressure Drop, psi</u>	<u>30% SGTP Pressure Drop, psi</u>
Reactor Vessel, including nozzles (psi)	47.21	44.26
Loop Piping (psi)	5.14	4.55
Steam Generator (psi)	<u>43.23</u>	<u>53.58</u>
Total (psi)	95.58 ⁽¹⁾	102.39 ⁽¹⁾

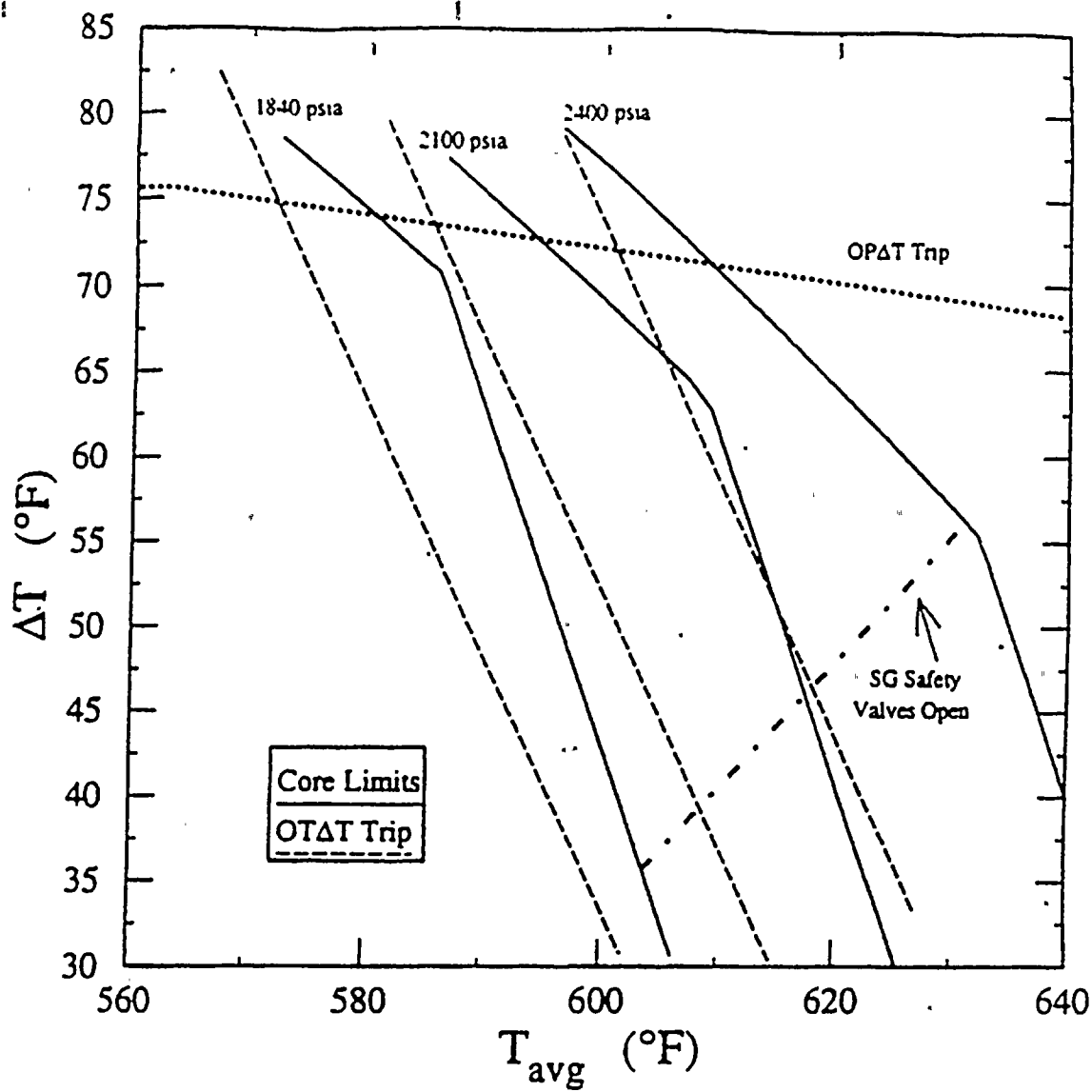
Notes:

(1) Pressure drops calculated at Best Estimate Flow.

TABLE 14.1-8

ECCS Injection to Recirculation Switchover Model for the Containment Response
Analysis Time After RWST Low Level Alarm

Event	Step Time (sec)	Cumulative Time (sec)
RWST Low Level Alarm	-	0
W RHR and W CTS pump stop	42	42
Close W RHR and W CTS pump suction valves from RWST (IMO-320, IMO-225)	155	197
Open recirculation sump to W RHR/CTS pump valve (ICM-306)	59	256
Start W RHR pump & W CTS Pump	26	282
Close SI pump recirculation line to RWST valves (IMO-262, IMO-263)	4	286
Open W RHR heat exchanger discharge tie valve to SI pumps (IMO-350)	8	294
Open SI pump suction crosstie valves to CC pumps (IMO-360, IMO-361 or -362)	3	297
Close SI pump suction from RWST (IMO-261)	0	297

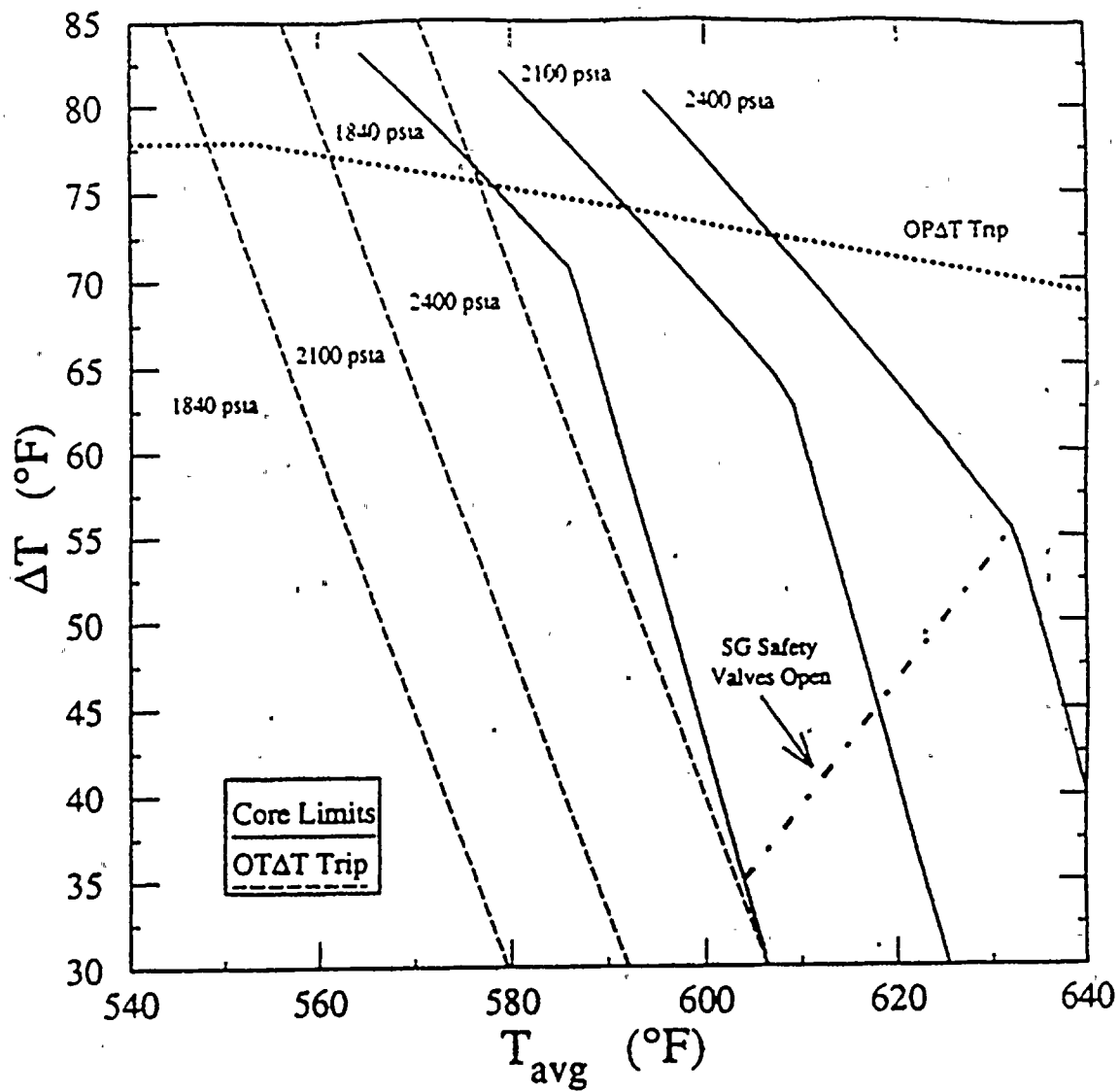


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NUCLEAR PLANT
UNIT 1

FIGURE 14.1-1

Illustration of Overtemperature and Overpower ΔT Protection
Nominal T_{avg} = 576.3°F
Nominal Pressure = 2100 psia

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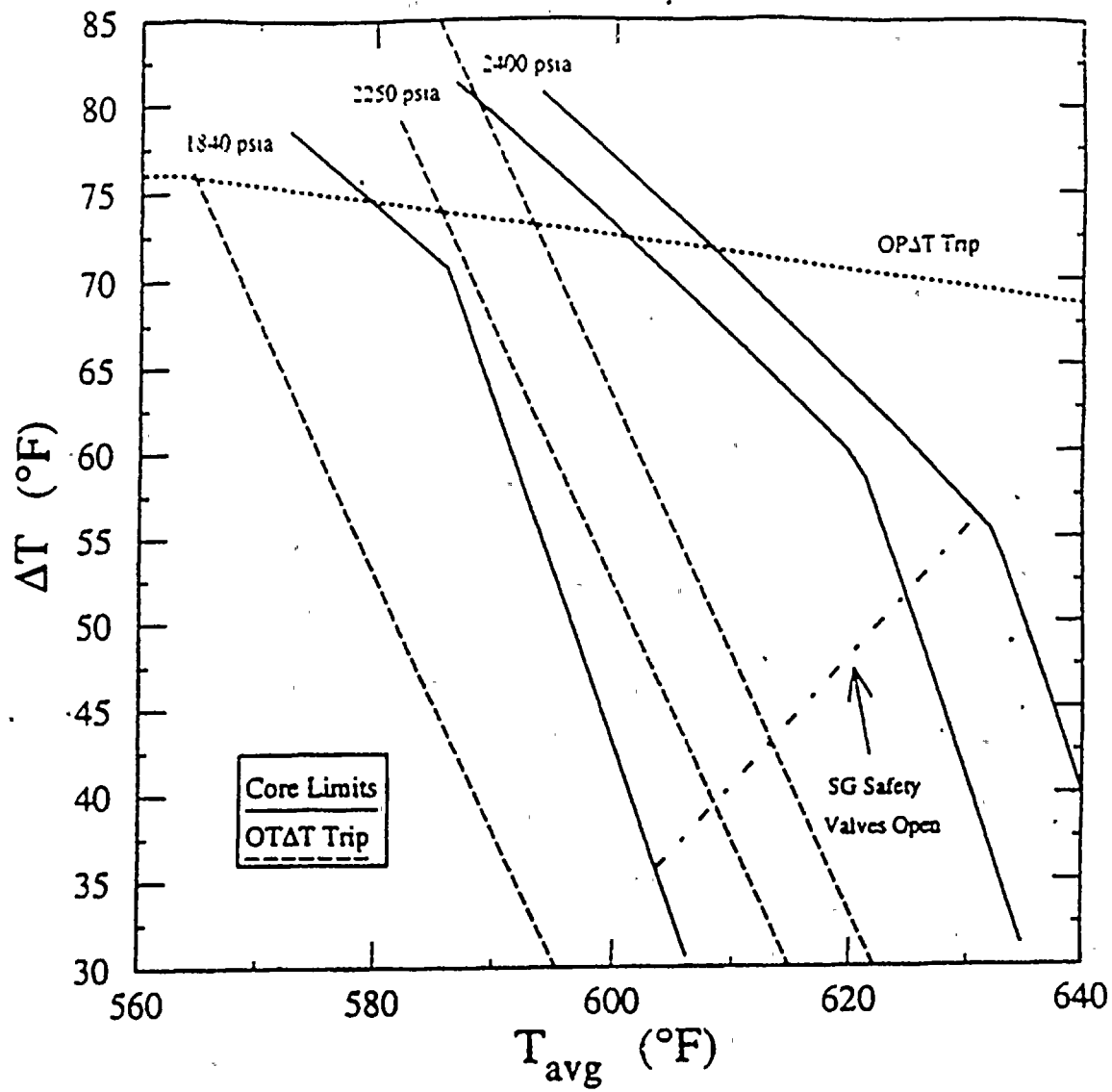


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UNIT 1**

FIGURE 14.1-2

Illustration of Overtemperature and Overpower ΔT Protection
 Nominal T_{avg} = 553.0°F
 Nominal Pressure = 2100 psia

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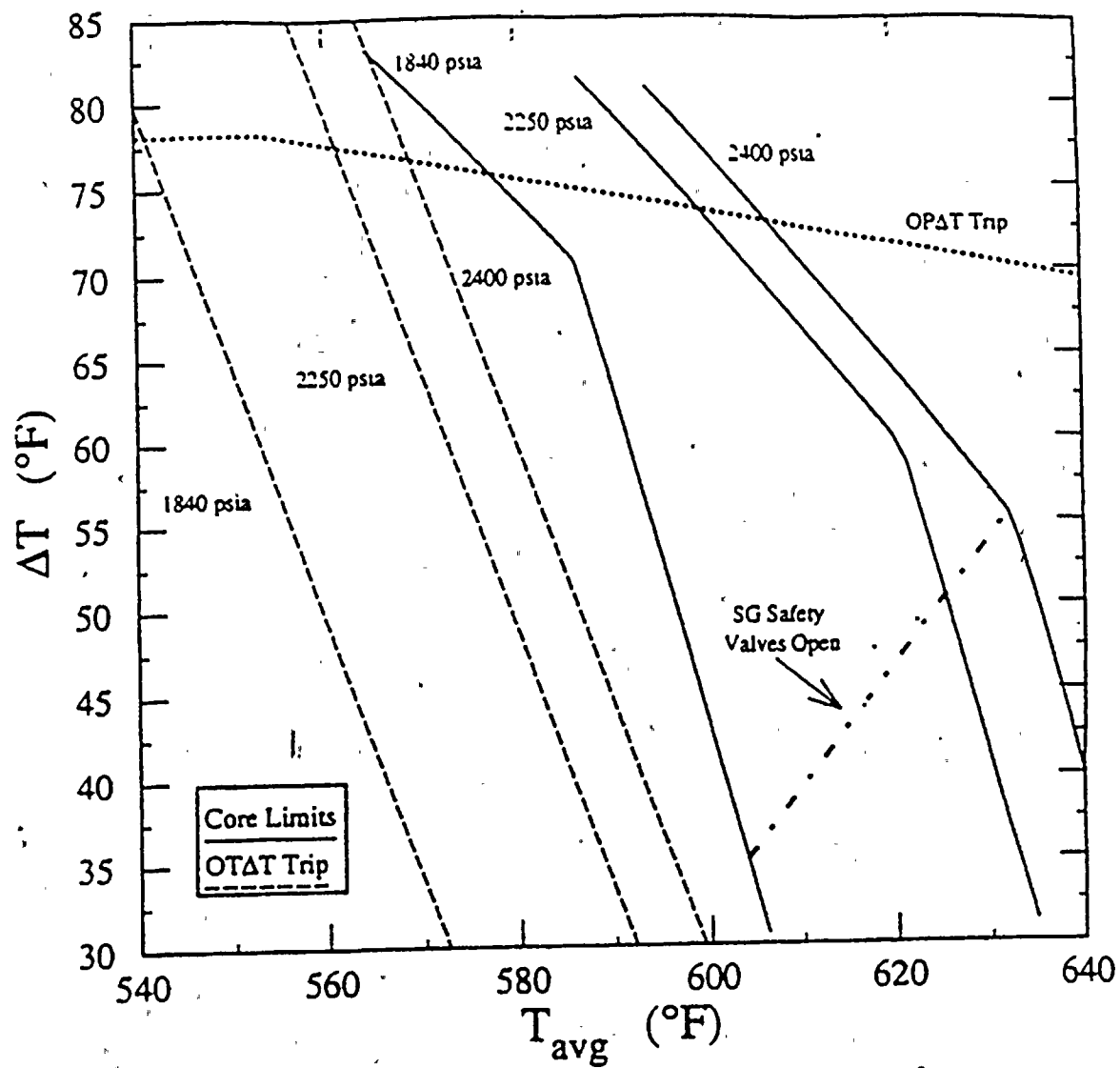


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FIGURE 14.1-3

Illustration of Overtemperature and Overpower ΔT Protection
 Nominal T_{avg} = 576.3°F
 Nominal Pressure = 2250 psia

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FIGURE 14.1-4

Illustration of Overtemperature and Overpower ΔT Protection
 Nominal T_{avg} = 553.0°F
 Nominal Pressure = 2250 psia

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Cook Nuclear Plant Unit 1 Normalized Negative Reactivity Insertion as
a Function of Time Used for the Reactor Trip In Transient Safety Analysis

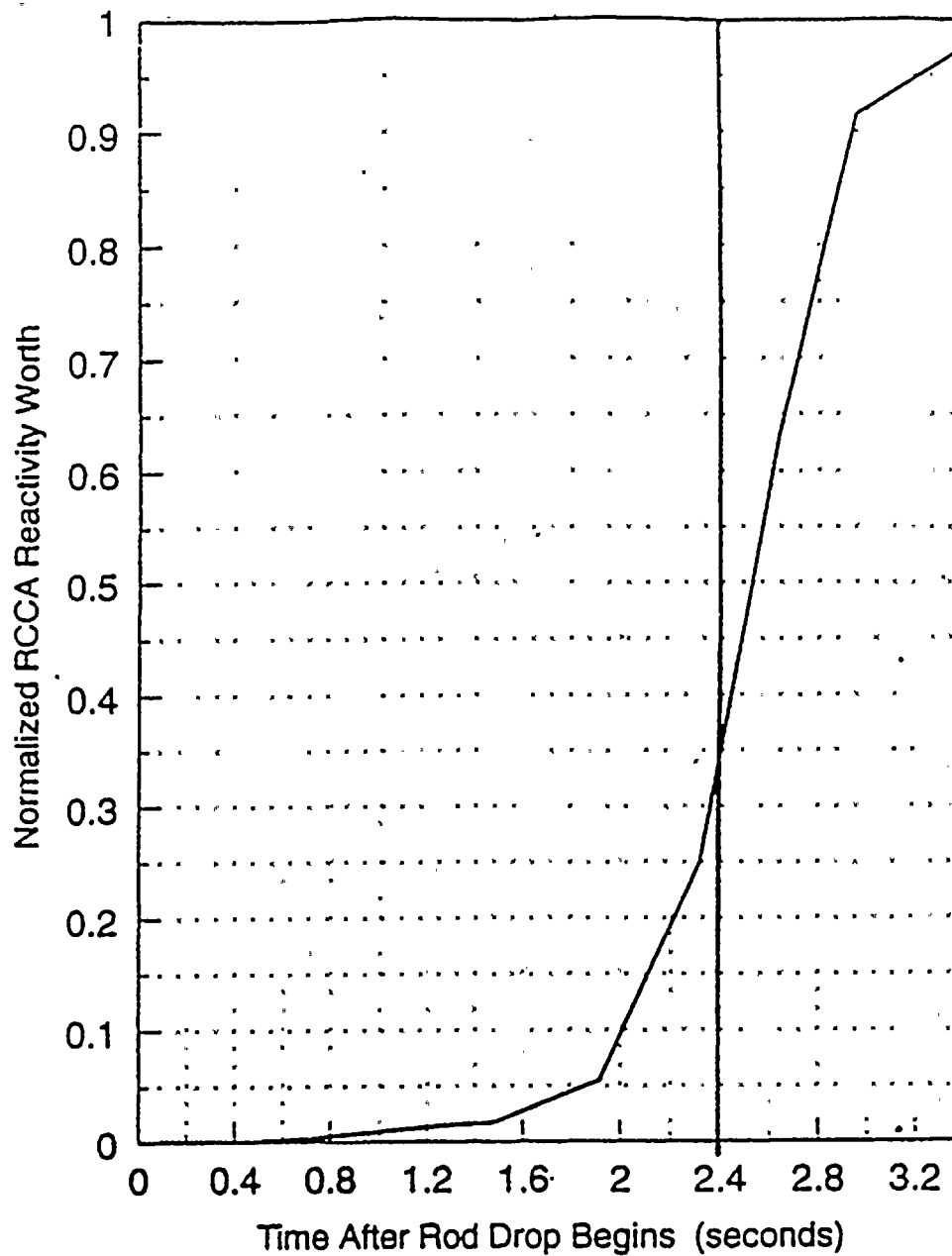


Figure 14.1-5 Trip Reactivity vs. Rod Drop Time

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After the initial power burst, the neutron flux is momentarily reduced and then, if the incident is not terminated by a reactor trip, the neutron flux increases again, but at a much slower rate.

Termination of the startup incident by the above protection channels prevents core damage. In addition, the reactor trip from pressurizer high pressure serves as a backup to terminate the incident before an overpressure condition could occur.

Method of Analysis

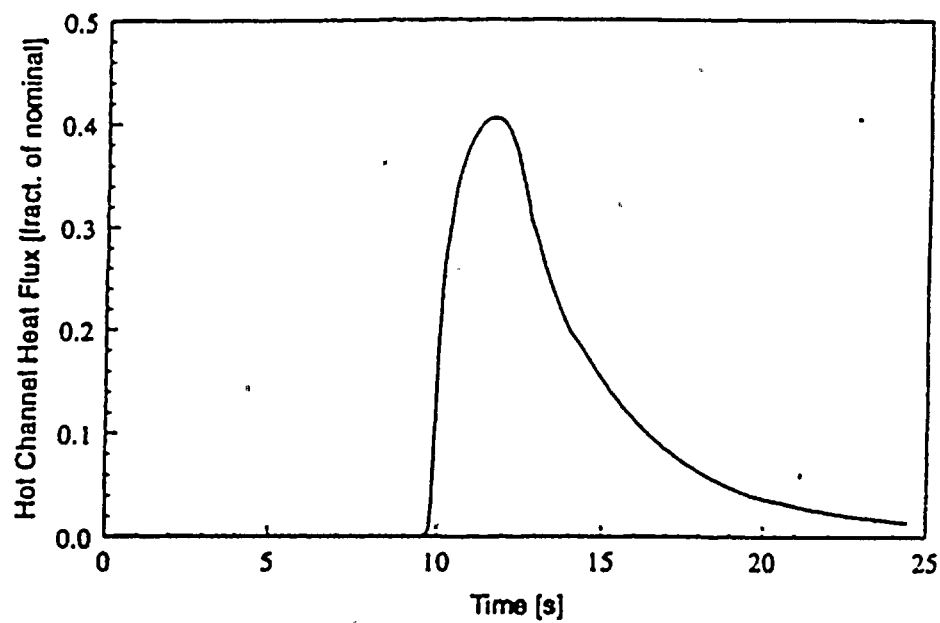
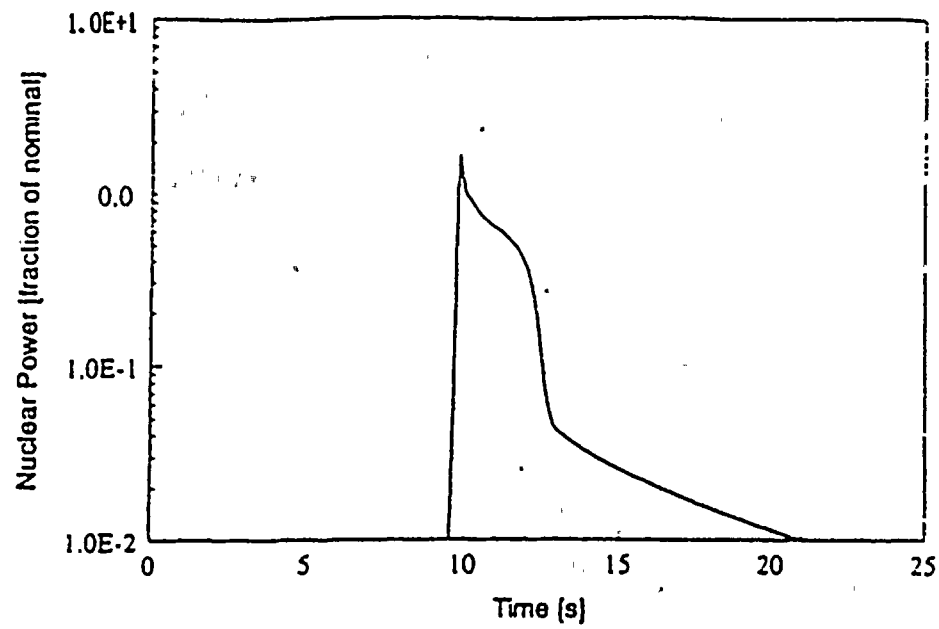
The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three states: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods (TWINKLE) to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN. The average heat flux is next used in THINC for transient DNBR calculation.

Analysis of this transient incorporates the neutron kinetics, including six delayed neutron groups and the core thermal and hydraulic equations. In addition to the neutron flux response, the average fuel, clad and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup incident, the following additional assumptions are made concerning the initial reactor conditions:

1. Since the magnitude of the neutron flux peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity defect, a conservatively low value is used for the startup incident -955 pcm.

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator temperature reactivity coefficient. The analysis is based on a moderator coefficient which was at least +5 pcm/ $^{\circ}$ F at the zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code used in the analysis is a diffusion theory code rather than a point kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature.
3. The reactor is assumed to be at hot zero power (547 $^{\circ}$ F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel to water heat transfer, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel thermal capacity and larger thermal conductivity yields a larger peak heat flux. Initial multiplication factor (k_0) is assumed to be closely approaching 1.0 since this results in the maximum neutron flux peak.
4. The most adverse combination of instrumentation and setpoint errors, as well as delays for trip signal actuation and control rod assembly release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint raising it from the nominal value of 25% to a value of 35% in addition to taking no credit for the source and intermediate range protection. Reference to Figure 14.1.1-1, however, shows that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition to the above, the rate of negative reactivity

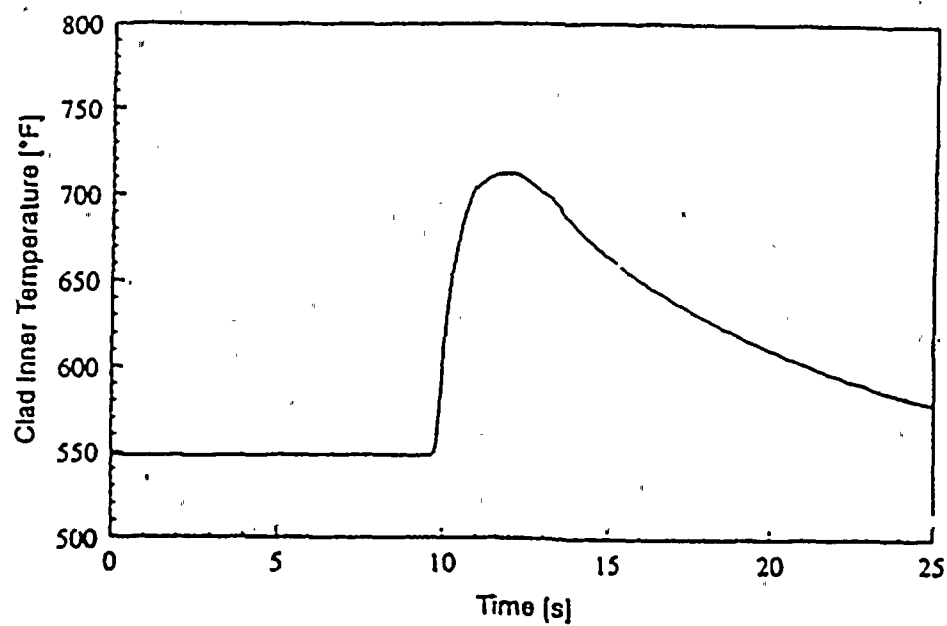
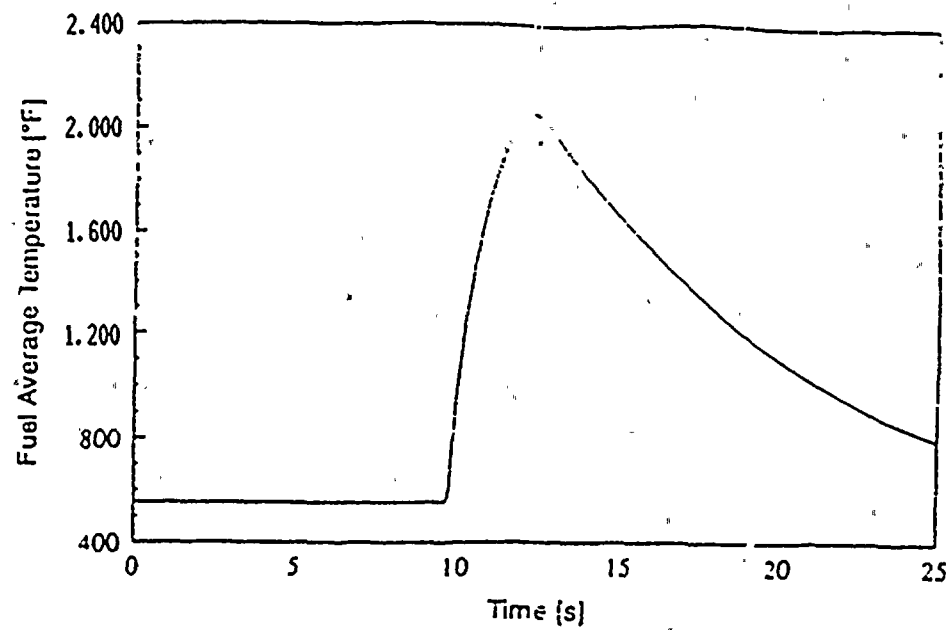


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FIGURE 14.1.1-1

Nuclear Power and Hot Channel Heat Flux vs. Time For
The Rod Withdrawal From Subcritical Event

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FIGURE 14.1.1-2

Fuel Average and Clad Temperature vs. Time For
The Rod Withdrawal From Subcritical Event

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Method of Analysis

This transient is analyzed by the LOFTRAN code. The core limits as illustrated in Figures 14.1-1 through 14.1-4 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the revised thermal design procedure described in Reference 1. Plant characteristics and initial conditions are listed in Table 14.1-3. For an uncontrolled rod withdrawal at power accident, the following conservative assumptions are made:

- A. Initial reactor power, pressure, and RCS temperatures are assumed to be at their conservative nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 1.
- B. Reactivity coefficients - two cases are analyzed:
 - 1. Minimum Reactivity Feedback. A +5 pcm/ $^{\circ}$ F moderator temperature coefficient of reactivity and a least negative Doppler only power coefficient (see Table 14.1-3) are assumed.
 - 2. Maximum Reactivity Feedback. A conservatively large negative moderator temperature coefficient and a most negative Doppler only power coefficient (See Table 14.1-3) are assumed.
- C. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.
- D. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

- E. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed.

Results

Figures 14.1.2-1 through 14.1.2-3 show the transient response for a rapid RCCA bank withdrawal incident starting from full power (case A). Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA bank withdrawal from full power (case B) is shown in Figures 14.1.2-4 through 14.1.2-6. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 14.1.2-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT functions. The minimum DNBR is always greater than the limit value.

Figures 14.1.2-8 and 14.1.2-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the limit value.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

The results of cases, which examined a conservative pressurizer water volume transient due to the uncontrolled RCCA bank withdrawal at power accident, showed that the pressurizer does not fill.

The time sequence of events for the RCCA bank withdrawal transient is shown in Table 14.1.2-1.

Conclusions

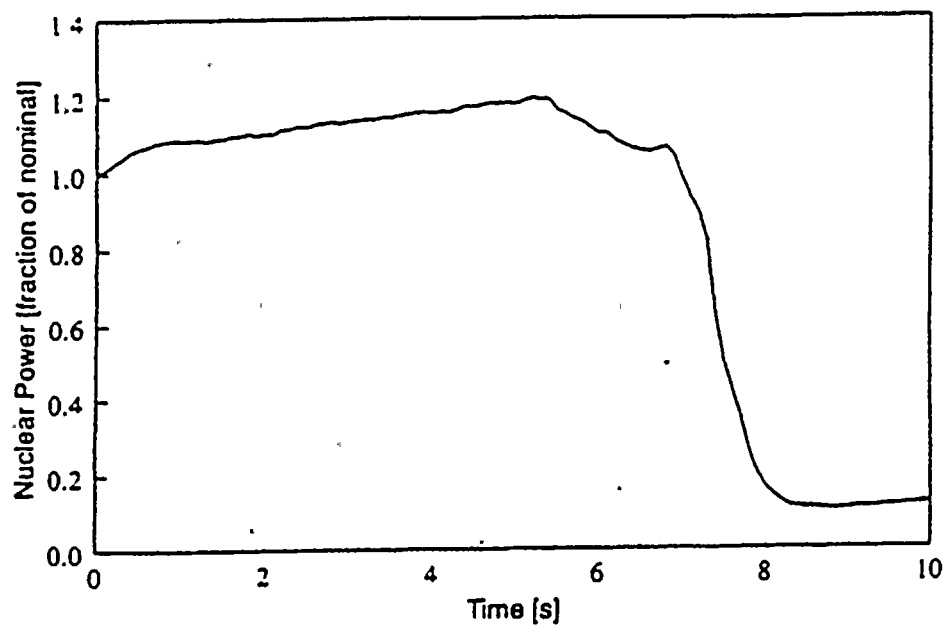
The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value for all fuel types. Also, the pressurizer does not fill.

14.1.2 References

1. Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989

TABLE 14.1.2-1
TIME SEQUENCE OF EVENTS

<u>Accident</u>	<u>Event</u>	<u>Time (sec)</u>
Uncontrolled RCCA Bank Withdrawal At Full Power		
Case A (high insertion rate max feedback)	Initiation of uncontrolled RCCA bank withdrawal at a high reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high trip signal initiated	4.8
	Rods begin to fall into core	5.3
	Minimum DNBR occurs	5.7
Case B (small insertion rate, max feedback)	Initiation of uncontrolled RCCA bank withdrawal at a small reactivity insertion rate (4 pcm/sec)	0
	Overttemperature ΔT reactor trip signal initiated	322.7
	Rods begin to fall into core	324.7
	Minimum DNBR occurs	325.2

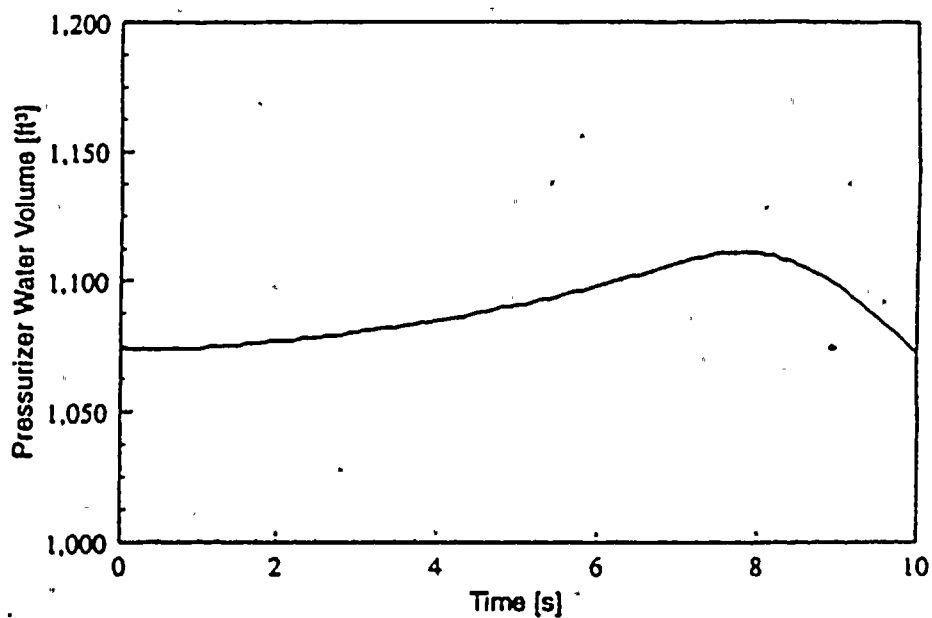
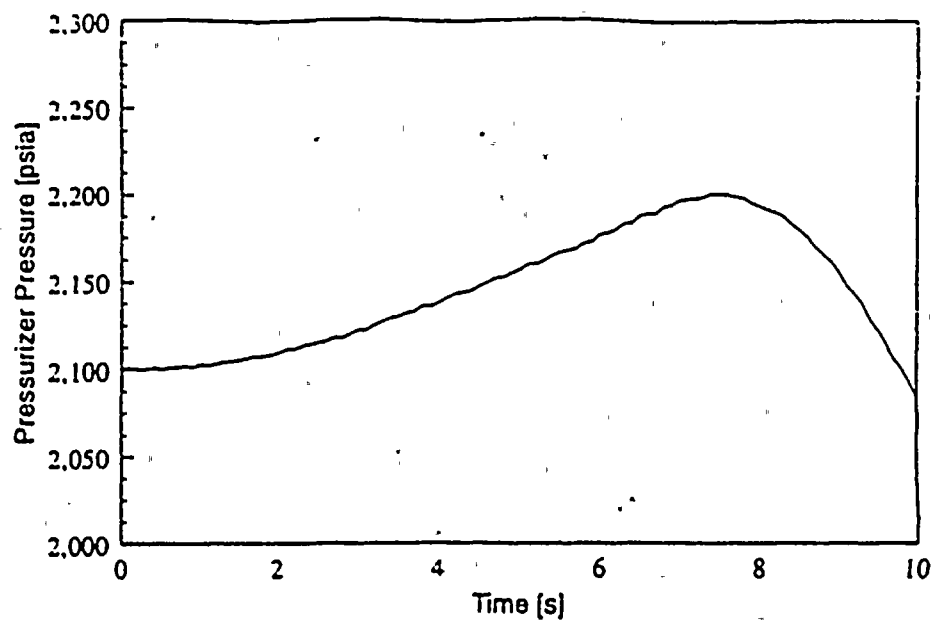


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FIGURE 14.1.2-1

Nuclear Power vs. Time For The RCCA Withdrawal
At Power Event, Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

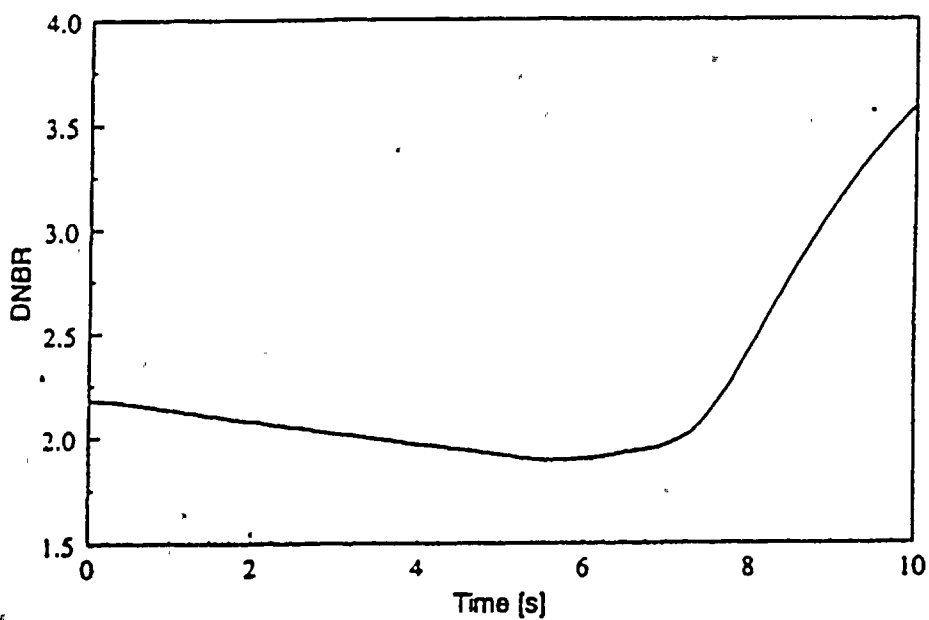
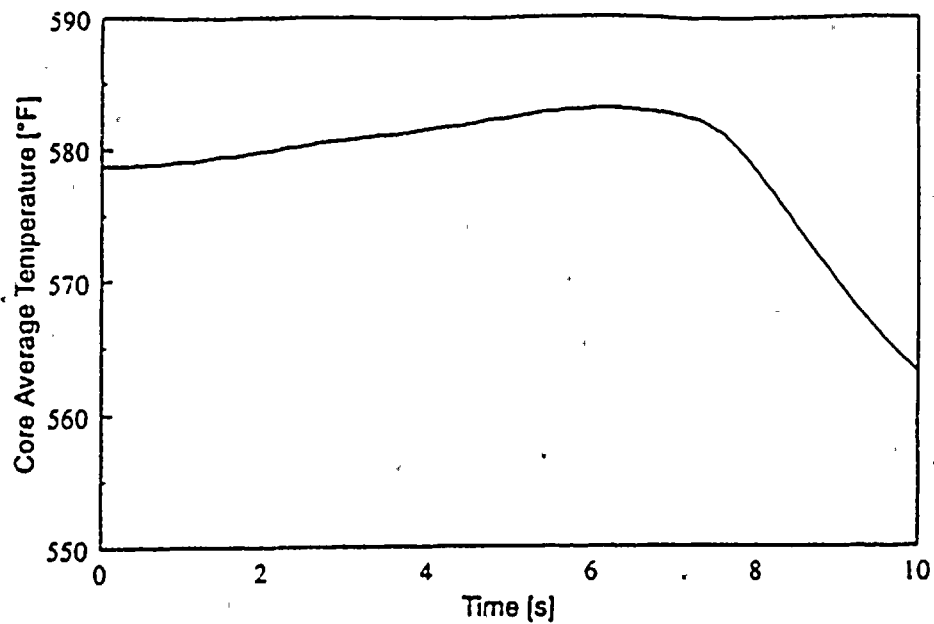
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FIGURE 14.1.2-2
Pressurizer Pressure and Pressurizer Water Volume vs. Time
For The RCCA Withdrawal At Power Event,
Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

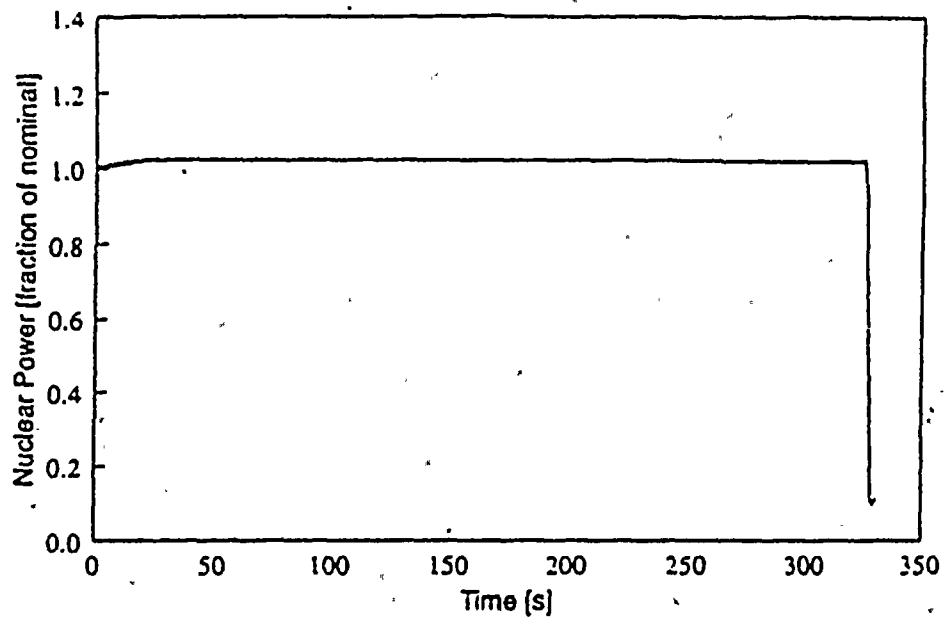
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FIGURE 14.1.2-3
Core Average Temperature and DNBR vs. Time
For The RCCA Withdrawal At Power Event,
Full Power, 80 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

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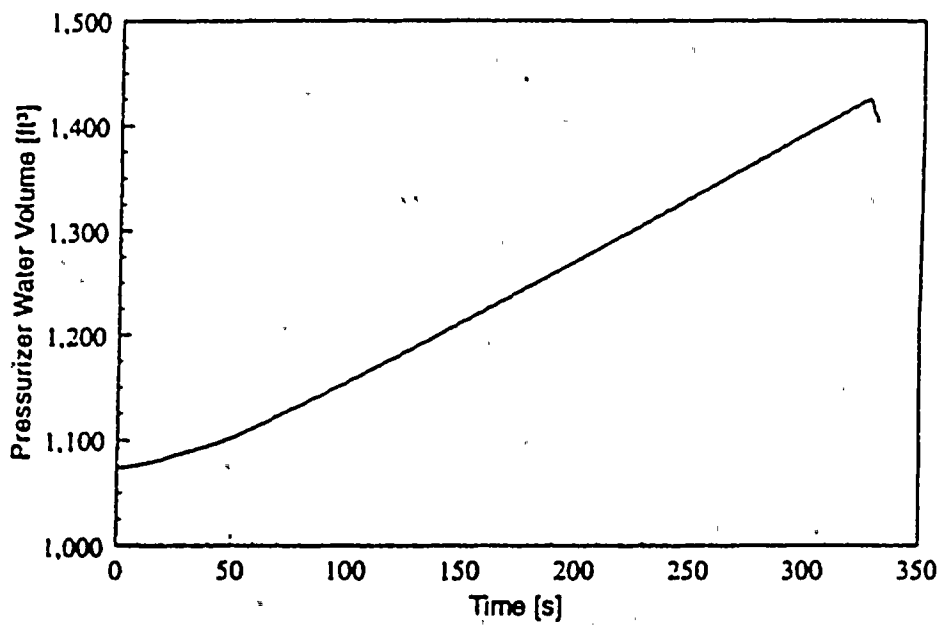
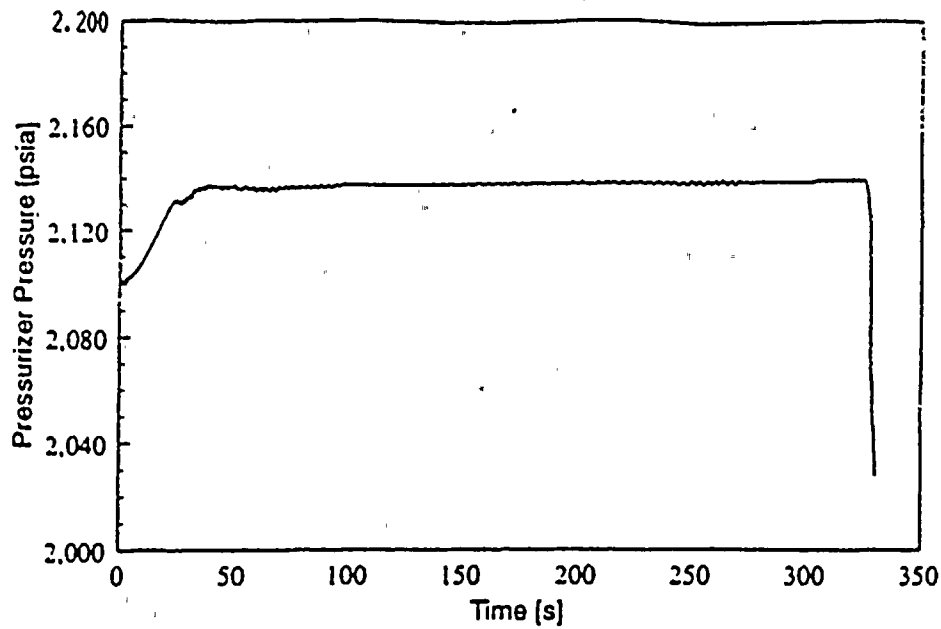


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FIGURE 14.1.2-4

**Nuclear Power vs. Time For The RCCA Withdrawal
At Power Event, Full Power, 4 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback**

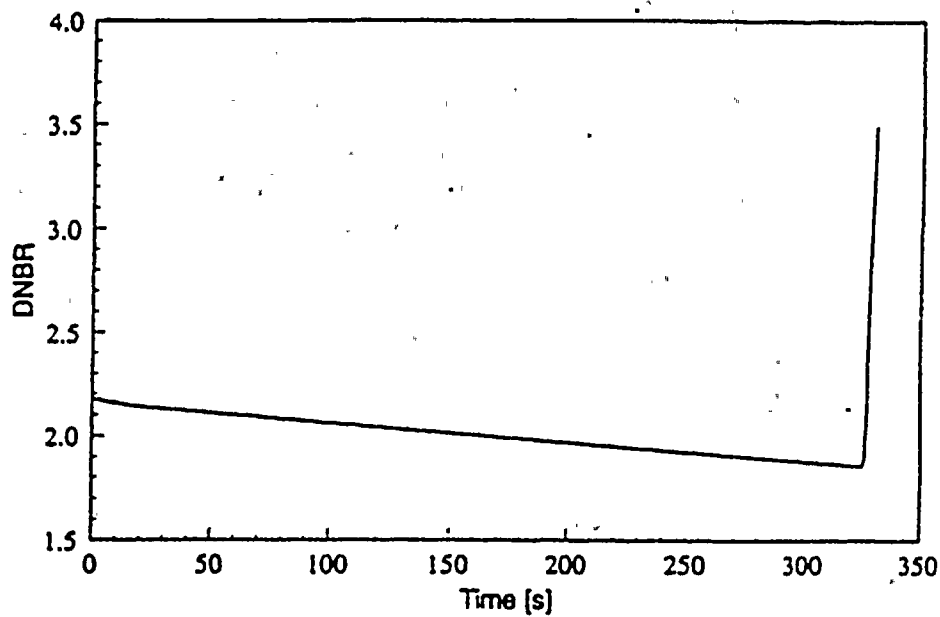
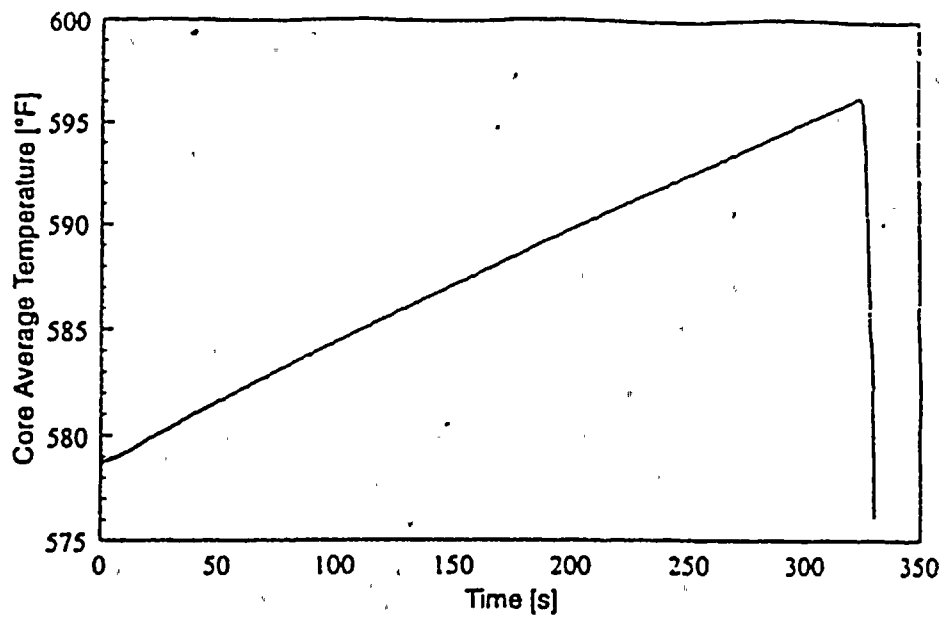
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FIGURE 14.1.2-5
Pressurizer Pressure and Pressurizer Water Volume vs. Time
For The RCCA Withdrawal At Power Event,
Full Power, 4 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

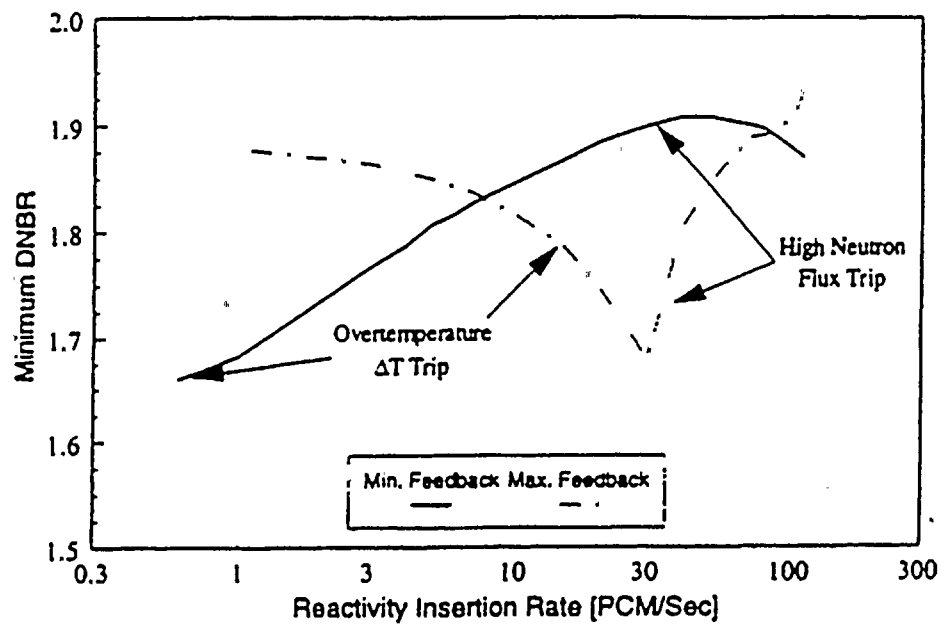
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FIGURE 14.1.2-6
Core Average Temperature and DNBR vs. Time
For The RCCA Withdrawal At Power Event,
Full Power, 4 PCM/Sec. Insertion Rate
Maximum Reactivity Feedback

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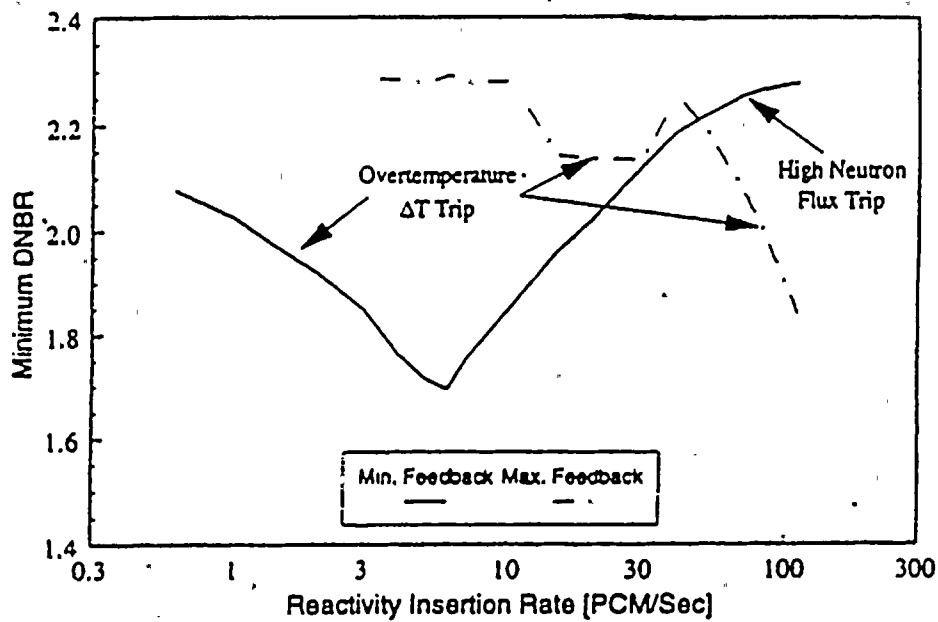


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FIGURE 14.1.2-7

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 100% Power

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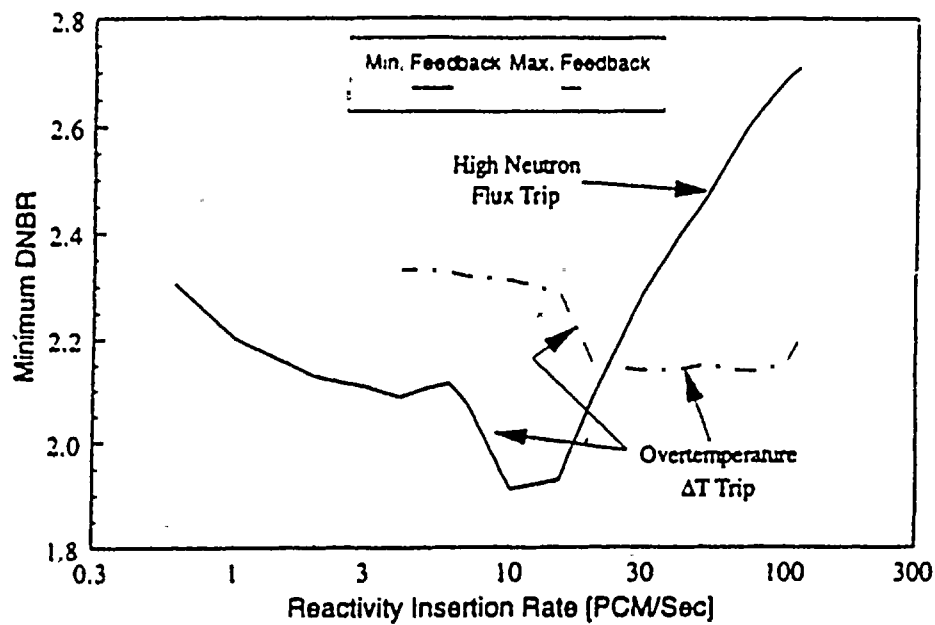


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FIGURE 14.1.2-8

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 60% Power

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FIGURE 14.1.2-9

Minimum DNBR vs. Reactivity Insertion Rate
For The RCCA Withdrawal At Power Event, 10% Power

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14.1.6 LOSS OF REACTOR COOLANT FLOW (INCLUDING LOCKED ROTOR ANALYSIS)

A loss of reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature which is magnified by the positive MTC. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly. The following trip circuits provide the necessary protection against a loss of coolant flow incident:

1. Undervoltage or underfrequency on pump power supply buses
2. Pump circuit breaker opening
3. Low reactor coolant flow

These trip circuits and their redundancy are further described in Chapter 7 (Protective Systems).

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe credible loss of flow condition. For this condition, reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent RCS overpressurization and the DNB ratio from exceeding the limit values.

Method of Analysis

The following loss of flow cases are analyzed:

1. Loss of four pumps from nominal full power conditions with four loops operating.
2. Loss of one pump from nominal full power conditions with four loops operating.

The normal power supplies for the pumps are four buses connected to the generator. Each bus supplies power to one pump. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps will continue to supply coolant flow to the core. The simultaneous loss of power to all reactor coolant pumps is a highly unlikely event. Since each pump is on a separate bus, a single bus fault would not result in the loss of more than one pump.

A full plant simulation is used in the analysis to compute the core average and hot spot heat flux transient responses, including flow coastdown, temperature, reactivity and control rod insertion effects.

These data are then used in a detailed thermal-hydraulic computation to compute the margin to DNB using RTDP. This computation solves the continuity, momentum and energy equations of fluid flow together with the WRB-1 DNB correlation.

Uncertainties in initial conditions are included in the limit DNBR as described in Reference 1. The initial conditions used are listed in Table 14.1-3.

This transient is analyzed by three digital computer codes. First the LOFTRAN code, described in Section 14.1, is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code, described in Section 14.1, is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN.

Finally, the THINC-IV code, also described in Section 14.1, is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for each type of fuel.

Results

Figures 14.1.6-1 through 14.1.6-3 show the transient response for the loss of power to all RCPs with four loops in operation. The reactor is assumed to be tripped on undervoltage signal. Figure 14.1.6-3 shows the DNBR to be always greater than the limit value for the most limiting fuel assembly cell.

Figures 14.1.6-4 through 14.1.6-6 show the transient response for the loss of one RCP with four loop operation. The reactor is assumed to be tripped on low flow signal. Figure 14.1.6-6 shows the DNBR to be always greater than the limit value for the most limiting fuel assembly cell. The sequence of events following each of these transients is included in Table 14.1.6-1.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly above their respective initial values.

Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

Locked Rotor Accident

A transient analysis has been performed for the instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid

expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in a sequence dependent on the rate of insurge and pressure increase. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray are not included in this analysis.

The locked rotor accident analysis was performed for four loop operation. The locked rotor event is examined to determine the DNB transient and to demonstrate that the peak RCS pressure and peak clad temperature remain below the limit value.

Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN code is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak RCS pressure, an additional degree of conservatism is provided by ignoring their effect. Table 14.1-3 presents the initial conditions assumed for the peak pressure transient.

The analysis assumed that the pressurizer safety valves initially open at 2575 psia and achieve rated flow at 2580 psia.

Evaluation of DNB in the Core During the Accident

For this accident, two DNB-related evaluations are made. The first evaluation has the assumption of rods going into DNB as a conservative initial condition in order to determine the clad temperature and zirconium water reaction. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (i.e., $F_Q = 2.5$) at the initial core power level. Table 14.1-3 presents the initial conditions assumed for the peak clad temperature evaluation.

A second evaluation made for this transient is to determine what percentage, if any, of rods are expected to be in DNB during the transient. For evaluation of this part of the transient, predicted core conditions are used as input to a THINC4 calculation of the minimum DNBR during the transient. Results of the THINC4 evaluation are then used to determine the percentage of fuel rods which experience DNB. Table 14.1-3 presents the initial conditions assumed for the percentage of rods in DNB analysis.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperatures (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations of the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft²-°F at the initiation of the transient. Thus the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following correlation, which defines the rate of the zirconium-steam reaction, was introduced into the model: ⁽²⁾

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(\frac{-45,500}{1.986T}\right)$$

where: w = amount reacted, mg/cm²

t = time, sec

T = temperature, °K

The reaction heat is 1510 cal/gm

Results

The transient results for the locked rotor accident are shown in Figures 14.1.6-7 through 14.1.6-9. The peak RCS pressure (2641 psia) reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. The pressure response shown in Figure 14.1.6-8 is the response at the point in the reactor coolant system having the maximum pressure. Also, the peak clad surface temperature (1934°F) is considerably less than 2700°F. The sequence of events is included in Table 14.1.6-1.

For the most limiting fuel assembly, less than 7% of the rods reach a DNBR value less than the limit value.

Conclusions

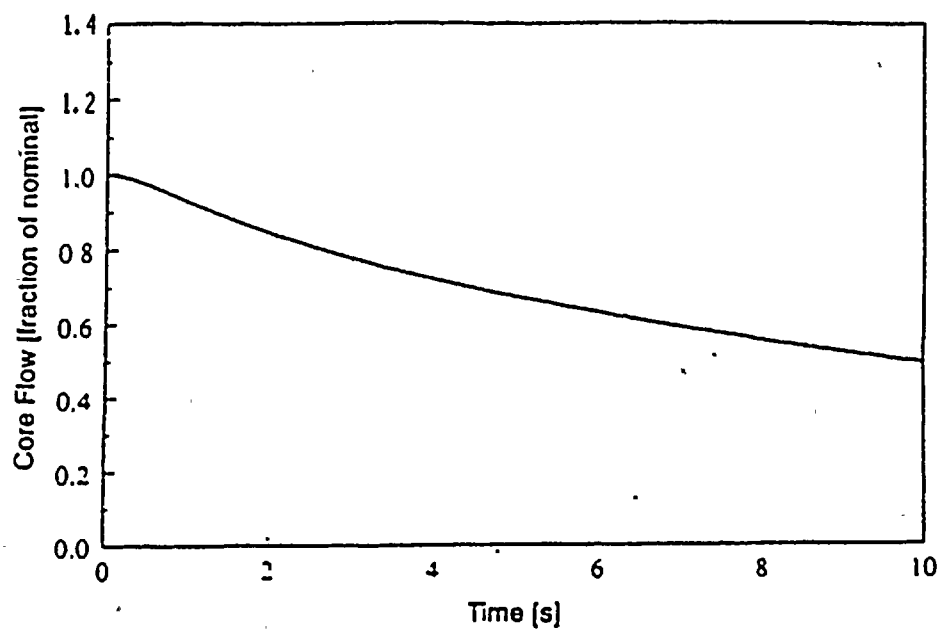
- A. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted conditions stress limits, the integrity of the primary coolant system is not endangered.
- B. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F (the temperature at which clad embrittlement may be expected), the core will remain in place and intact with no loss of core cooling capability.

14.1.6 References

1. Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April, 1989
2. Baker, L., and L. C. Just, "Studies of Metal Water Reactions of High Temperatures, III. Experimental and Theoretical Studies of Zirconium-Water Reaction," ANL-6548, May 1962

Table 14.1.6-1
Sequence of Events for Loss of
Flow and Locked Rotor Accidents

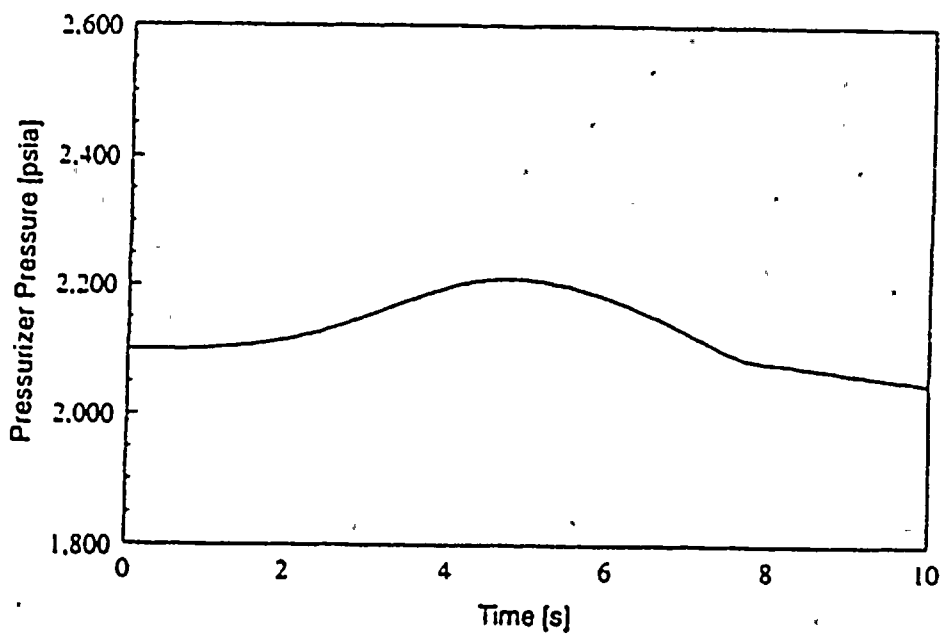
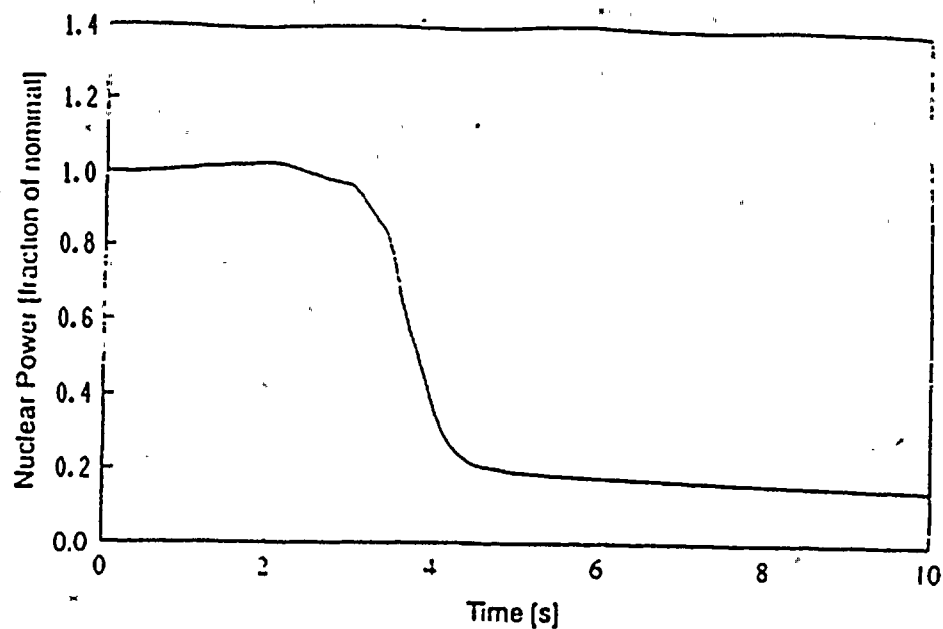
<u>Accident</u>	<u>Event</u>	<u>Time (sec)</u>
Complete Loss of Flow	All pumps lose power and begin coasting down, undervoltage trip signal generated	0.0
	Rods begin to drop	1.50
	Minimum DNBR occurs	3.40
Partial Loss of Flow	One operating pump loses power and begins coasting down	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	1.74
	Rods begin to drop	2.74
	Minimum DNBR occurs	3.90
Locked Rotor	One pump rotor seizes	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	0.04
	Rods begin to drop	1.04
	Maximum percentage of rods in DNB predicted	2.6
	Maximum RCS pressure occurs	3.20
	Maximum clad temperature occurs	3.49



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FIGURE 14.1.6-1
Total Core Flow vs. Time for
The Complete Loss Of Flow Event

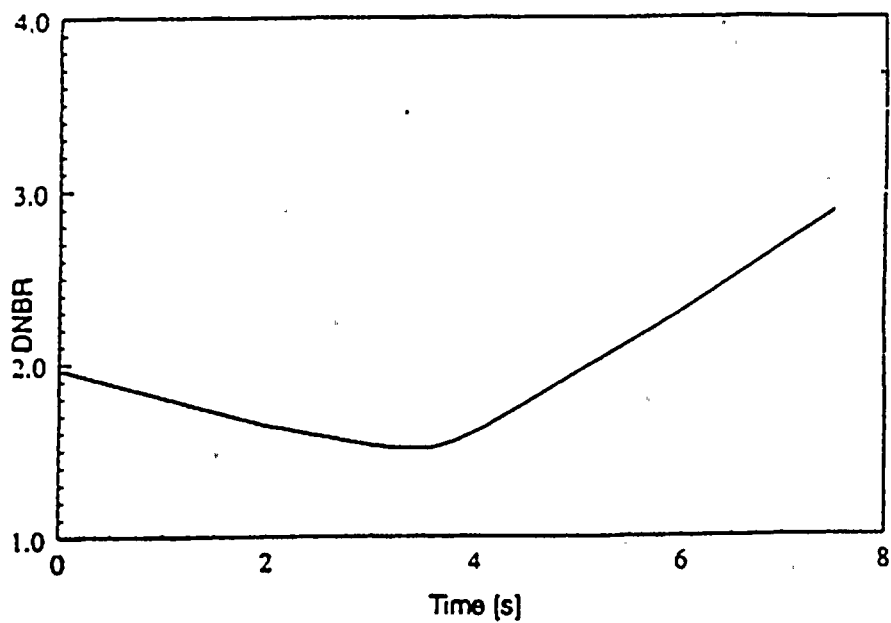
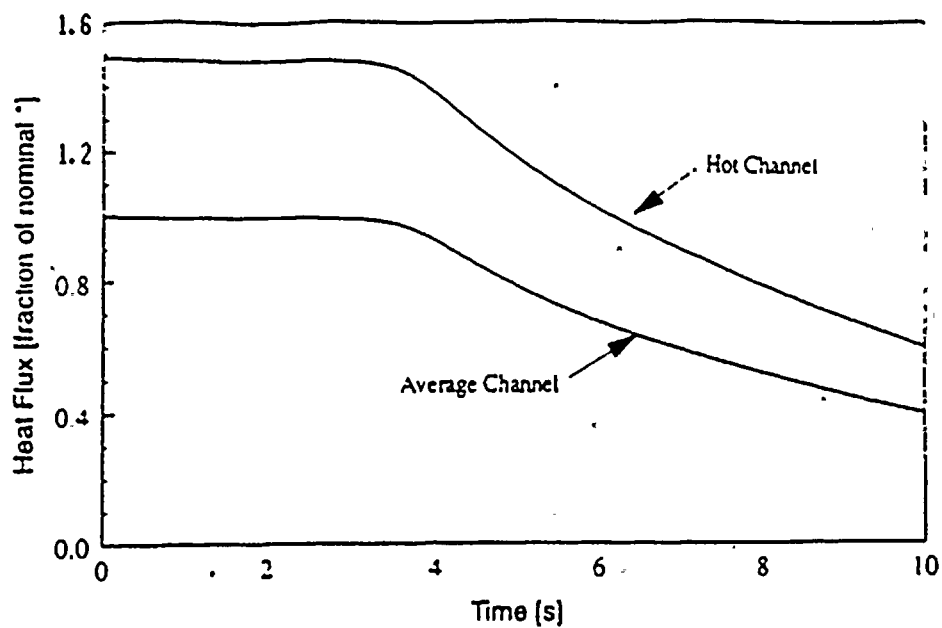
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FIGURE 14.1.6-2
Nuclear Power and Pressurizer Pressure vs. Time for
The Complete Loss Of Flow Event

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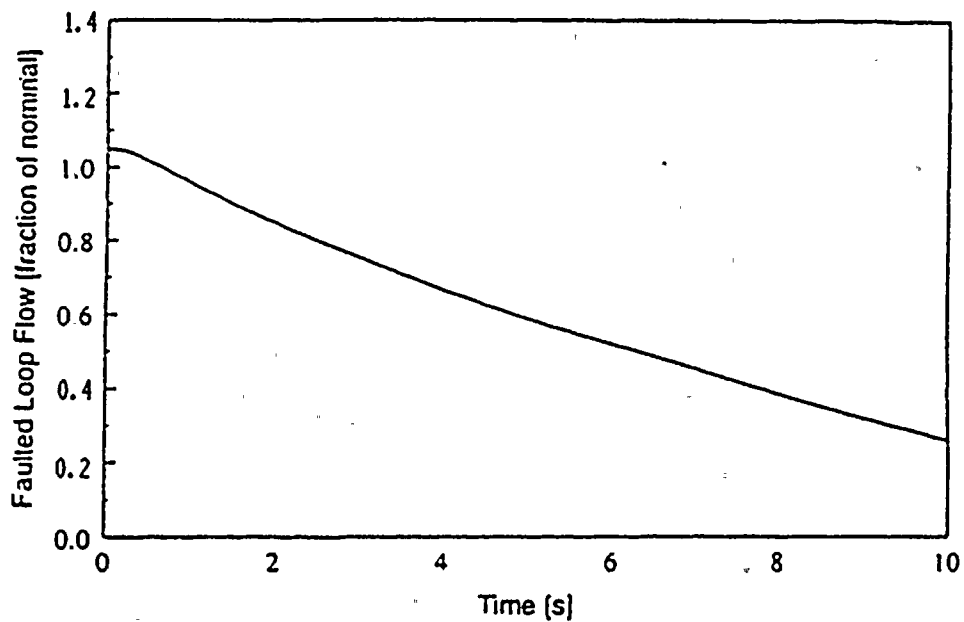
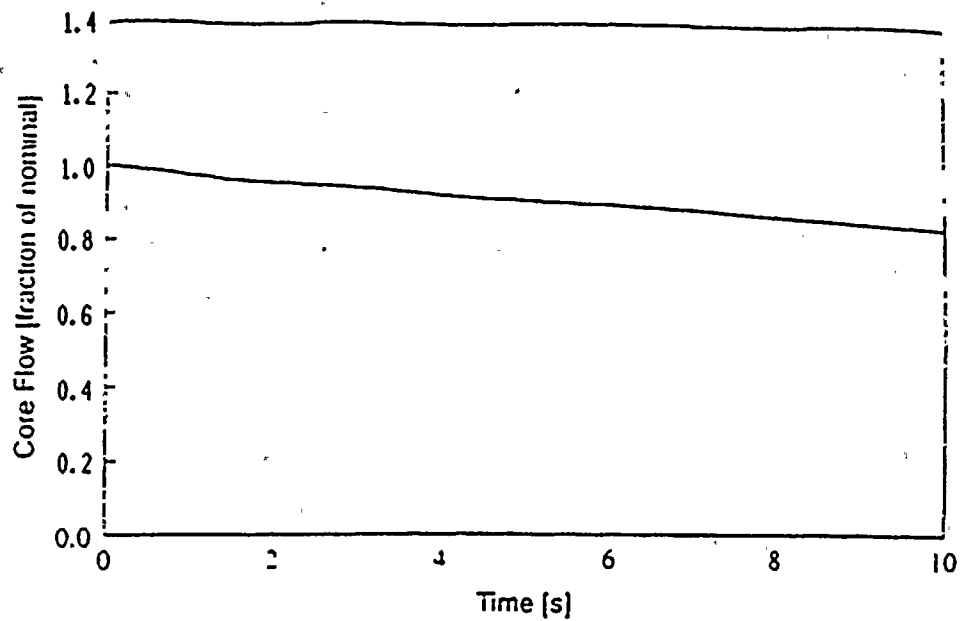
• Heat fluxes are shown as a fraction of the nominal average channel heat flux

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FIGURE 14.1.6-3

Average and Hot Channel Heat Fluxes and DNBR
vs. Time for the Complete Loss Of Flow Event

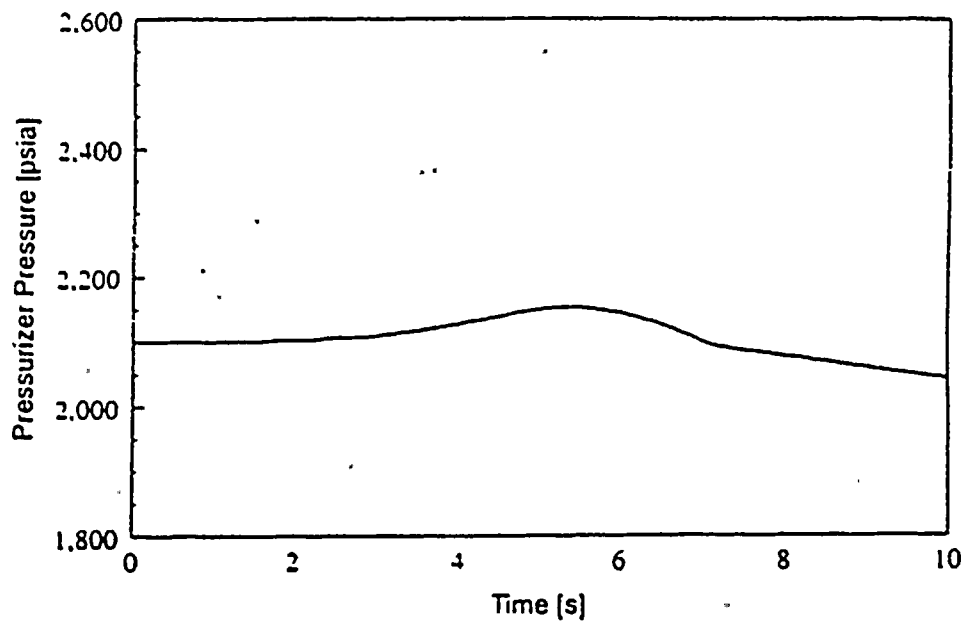
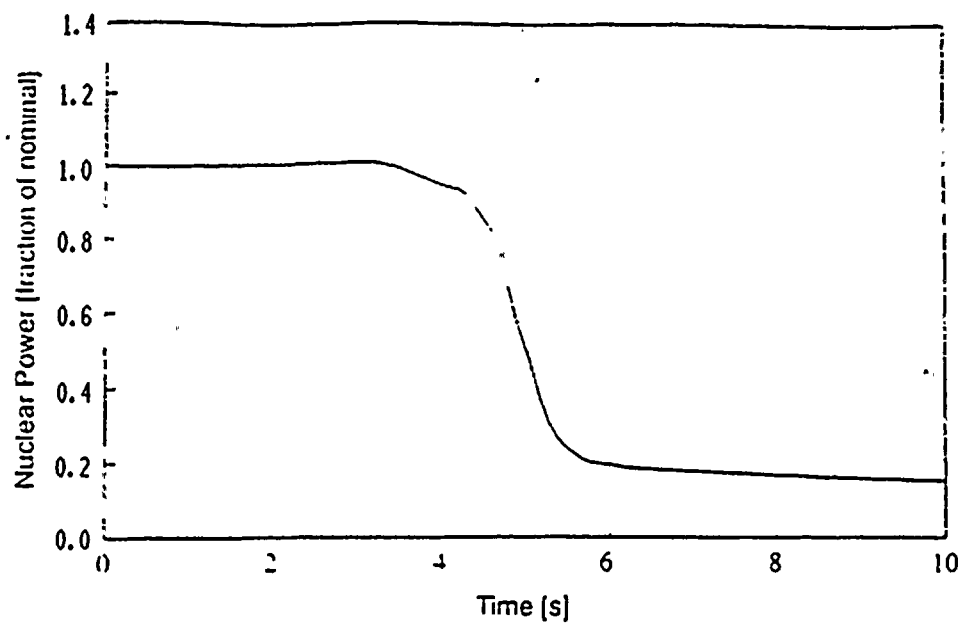
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FIGURE 14.1.6-4
Total Core Flow and Faulted Loop Flow vs. Time for
The Partial Loss Of Flow Event

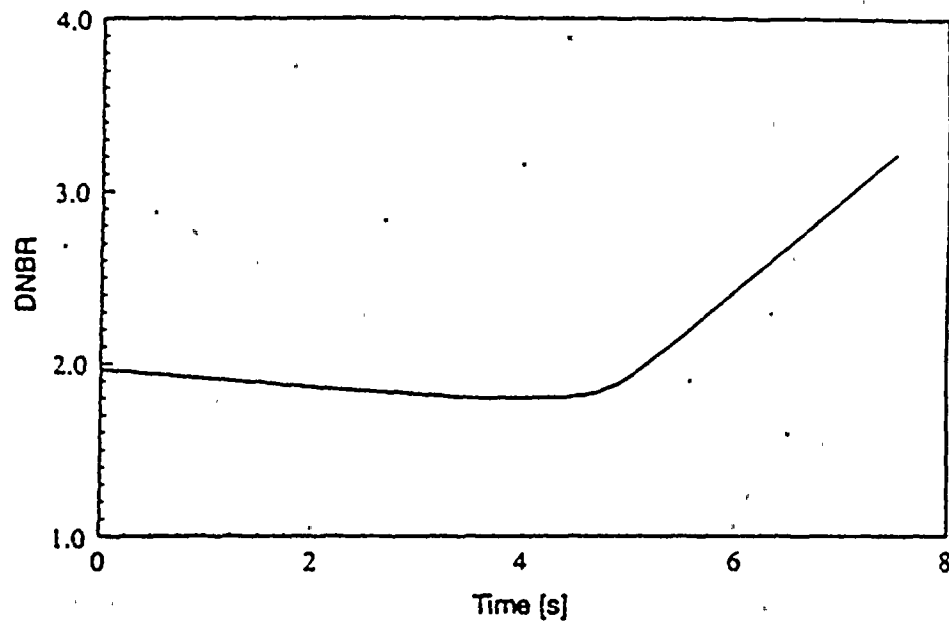
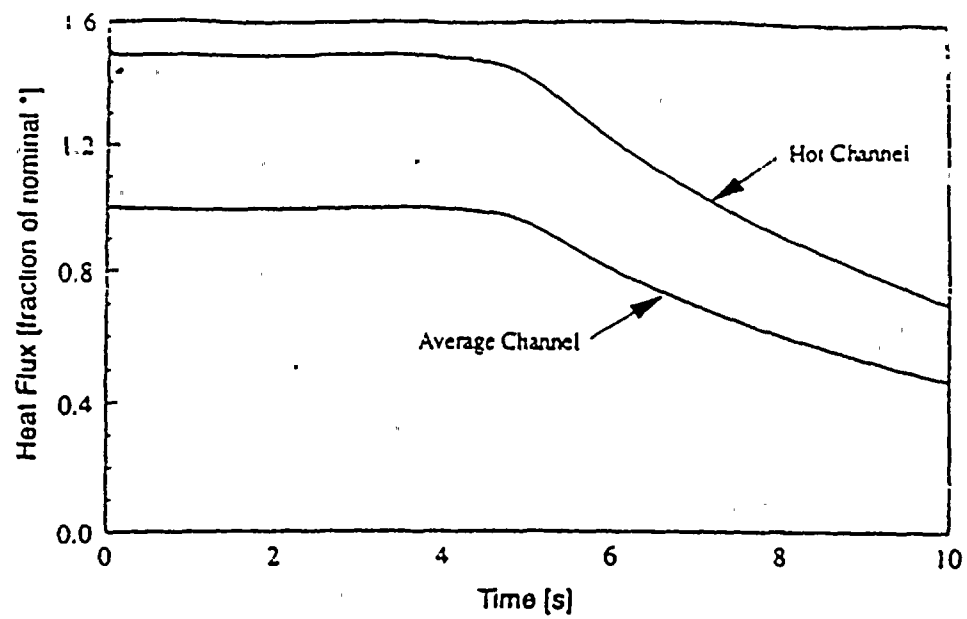
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FIGURE 14.1.6-5
Nuclear Power and Pressurizer Pressure vs. Time for
The Partial Loss Of Flow Event

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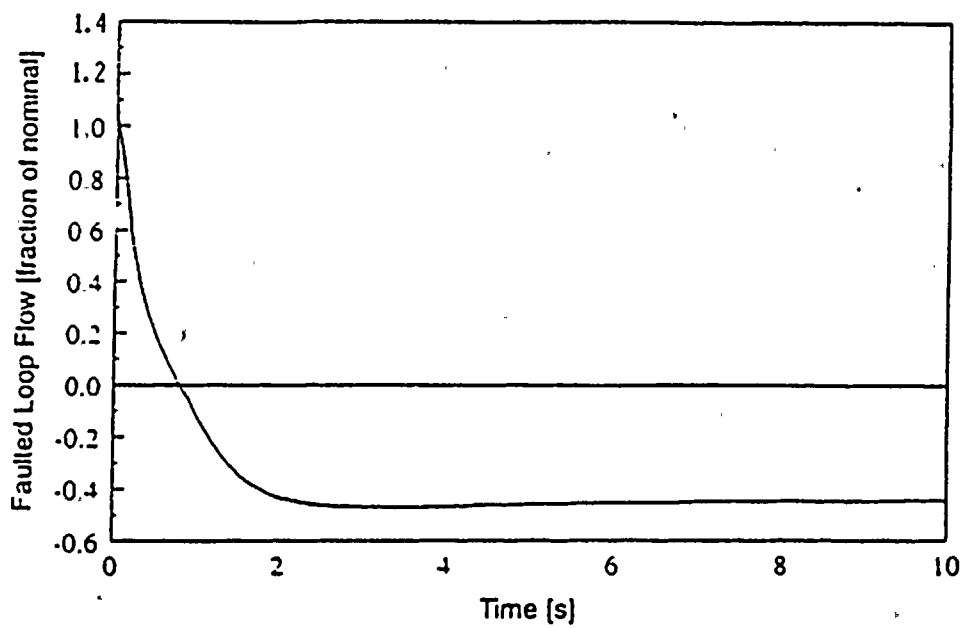
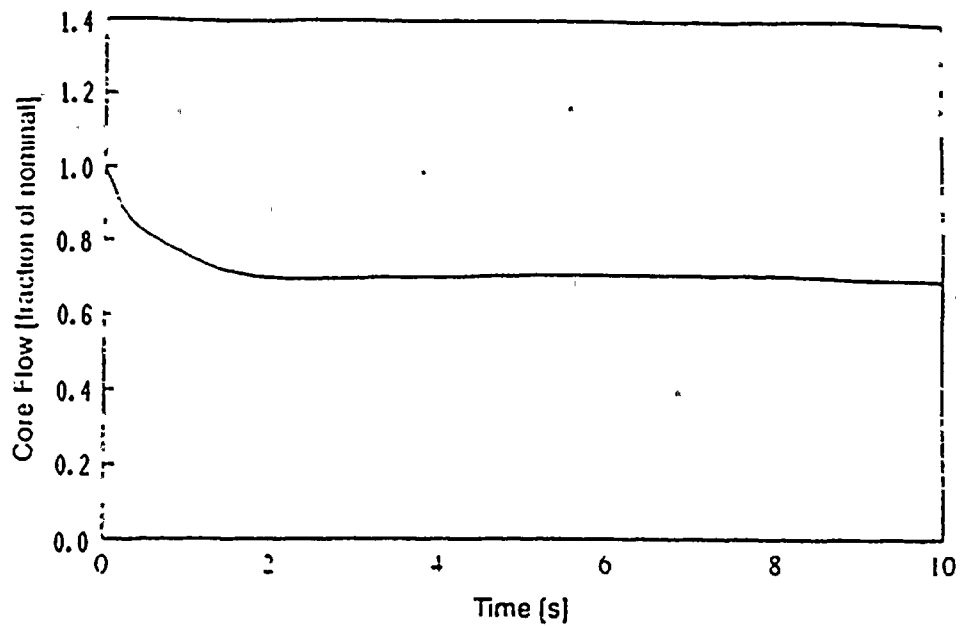
• Heat fluxes are shown as a fraction of the nominal average channel heat flux

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FIGURE 14.1.6-6

Average and Hot Channel Heat Fluxes and DNBR
vs. Time for the Partial Loss Of Flow Event

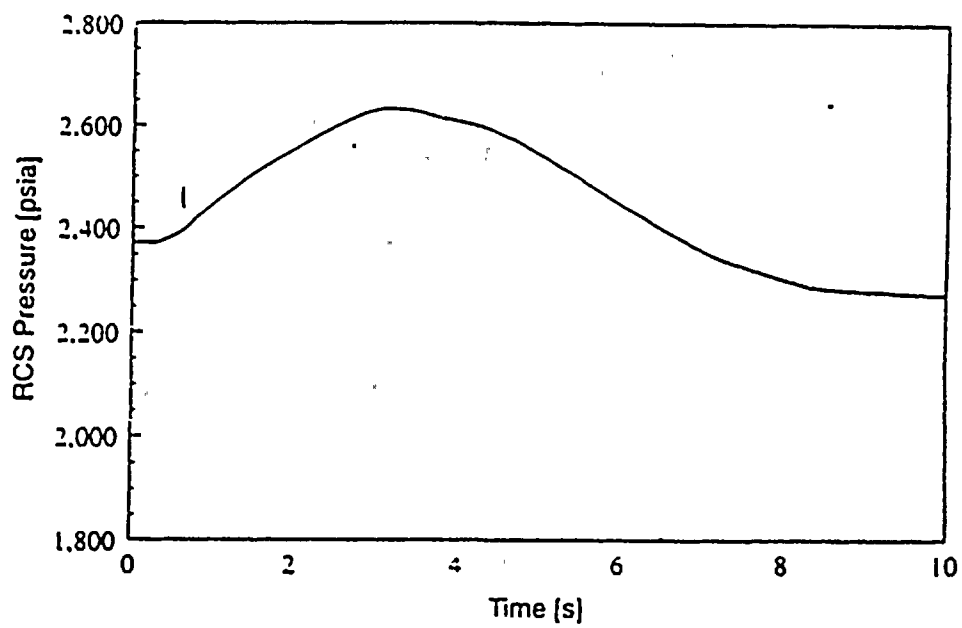
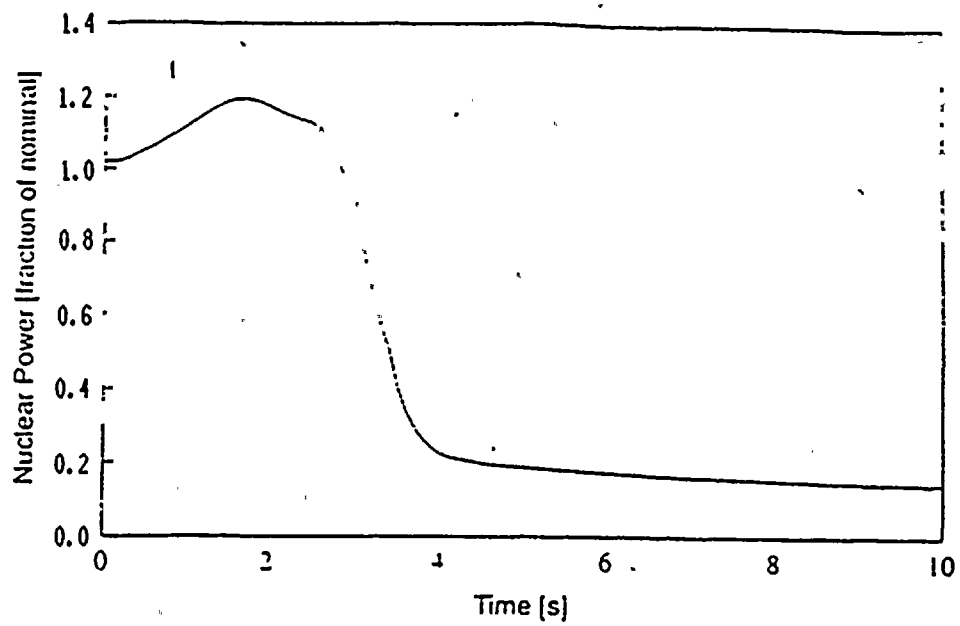
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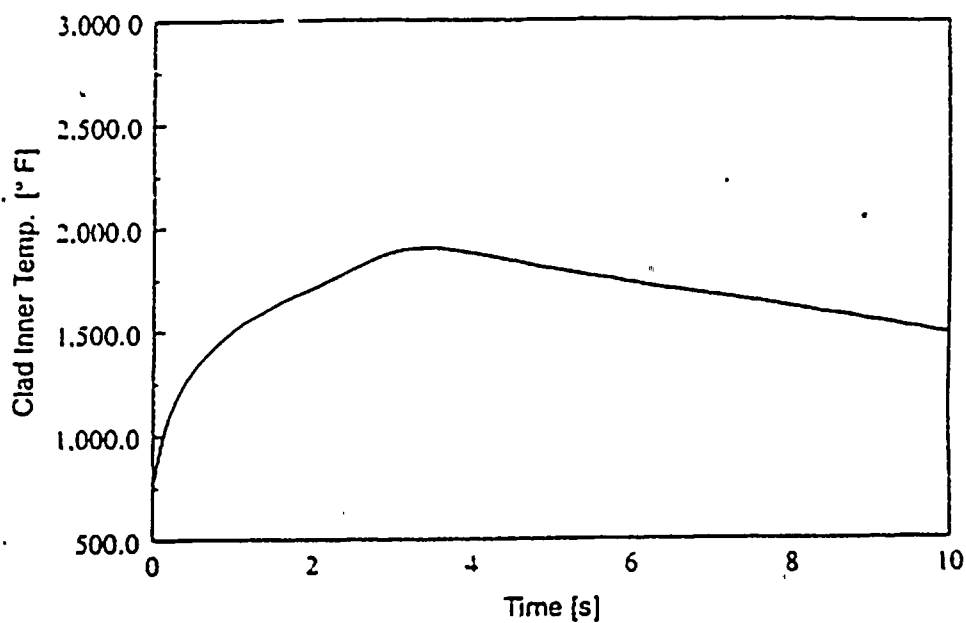
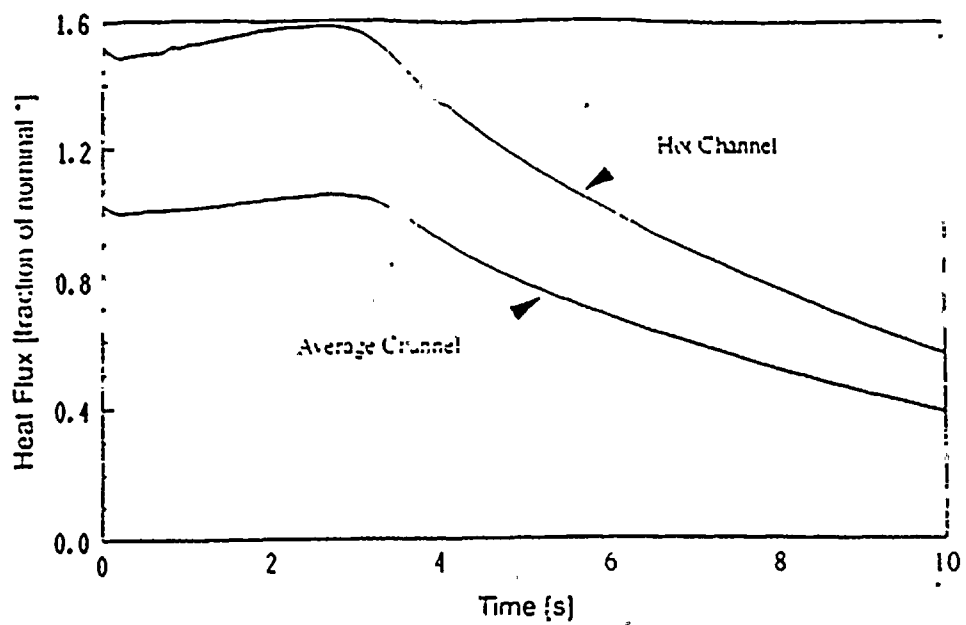
FIGURE 14.1.6-7
Total Core Flow and Faulted Loop Flow vs. Time
For The Locked Rotor Event

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FIGURE 14.1.6-8
Nuclear Power and RCS Pressure vs. Time
For The Locked Rotor Event



• Heat fluxes are shown as a fraction of the nominal average channel heat flux

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FIGURE 14.1.6-9
Average and Hot Channel Heat Fluxes vs. Time and
Clad Inner Temperature vs. Time
For The Locked Rotor Event

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average thermal heat flux reaches 107% of the nominal value at 10.6 seconds. The core average temperature, core inlet temperature, reactor coolant system pressure and DNBR during the transient are shown. The minimum DNBR during the transient is 1.78 and occurs at 10.5 seconds.

Reduced Temperature and Pressure Consideration

This accident was not reanalyzed for reduced temperature and pressure since the event cannot occur above the P-7 setpoint (10% power) as restricted by the Technical Specifications. The above analysis remains bounding with respect to the restriction to 10% power for the operation of 3 reactor coolant pumps imposed by the Technical Specification. The parameters assumed in the analysis bound the parameters associated with a 10% power condition. Thus, the conclusions presented above remain valid.

Amendment 120 to the Unit 1 Technical Specifications removed Mode 1 and Mode 2 three loop (i.e., N-1 Loop) operating specifications. Since, N-1 loop operation in Modes 1 and/or 2 is prohibited for Donald C. Cook Unit 1, the SUIL event does not need to be considered for the Unit 1 30% SGTP program. Therefore, no reanalysis was performed for the 30% SGTP program.

Conclusion

The transient results for the startup of an inactive reactor coolant loop show that there is a considerable margin to a limiting DNBR of 1.3.

Note that this event has not been reanalyzed as a result of the SGTP program. |

14.1.8 LOSS OF EXTERNAL ELECTRICAL LOAD

The loss of external electrical load may result from an abnormal variation in network frequency, or other adverse network operating conditions. It may also result from a trip of the turbine generator or in an unlikely opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large nuclear steam supply system load reduction by the action of the turbine control.

In the event the steam dump valves fail to open following a large load loss the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer water level signal. The steam generator shell side pressure and reactor coolant temperature will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor coolant system and steam generator against overpressure for all load losses without assuming availability of the steam dump system. The steam dump valves will not be opened for load reductions of 10% or less. For larger load reductions they may open depending on the capability of the reactor control system.

The most likely source of a complete loss of load in the nuclear steam supply system is a trip of the turbine-generator. In this case, there is a direct reactor trip signal (unless power is below approximately 10% power, i.e., below P-7) derived from the turbine emergency trip fluid pressure. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. However, in this analysis, the behavior of the unit is evaluated for a complete loss of load from 100% of full power without a direct reactor trip, primarily to show adequacy of the pressure relieving devices and also to show that no core damage occurs. The reactor coolant system and main steam system pressure relieving capacities are designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or steam dump control systems.

Method of Analysis

The loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN, as described in Section 14.1. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the Revised Thermal Design Procedure, as mentioned in Section 14.1. Plant characteristics and initial conditions are listed in Table 14.1-3.

Major assumptions are summarized below:

- A. Initial Operating Conditions - nominal initial conditions for reactor power, pressure, and RCS temperatures are assumed for statistical DNB analyses.
- B. Moderator and Doppler Coefficients of Reactivity - the loss of load is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a +5 pcm/^oF MTC and the least negative Doppler coefficients.
- C. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the minimum and maximum moderator feedback cases are analyzed:

1. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.
2. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

E. Steam Release - no credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoints where steam release through the safety valves limits secondary steam pressure.

F. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

G. Reactor trip is actuated by the first reactor protection system trip setpoint reached. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT , high pressurizer water level, and low-low steam generator water level.

Results

The transient responses for a loss of load from 100% full power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 14.1.8-1 through 14.1.8-12).

Figures 14.1.8-1 through 14.1.8-3 show the transient responses for the loss of load with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal.

The minimum DNBR remains well above the limit value. The pressurizer relief and safety valves prevent overpressurization of the primary system. The steam generator safety valves prevent overpressurization of the secondary system, maintaining pressure below 110 percent of design value.

Figures 14.1.8-4 through 14.1.8-6 show the responses for the total loss of steam load with maximum reactivity feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The reactor is tripped by the low-low steam generator water level signal. The pressurizer safety valves are not actuated for this case.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to terminate the transient and bring the plant to a stabilized condition. If no action were taken by the operator the reduced power operation would continue until the condenser hotwell was emptied. A low-low steam generator water level reactor trip would be generated along with auxiliary feedwater initiation signals. Auxiliary feedwater would then be used to remove decay heat with the results less severe than those presented in Section 14.1.9, Loss of Normal Feedwater Flow.

The loss of load accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 14.1.8-7 through 14.1.8-9 show the transient responses with minimum reactivity feedback. The

neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below its initial value throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

Figures 14.1.8-10 through 14.1.8-12 show the transient responses with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

The sequence of events following each of these transients is included in Table 14.1.8-1.

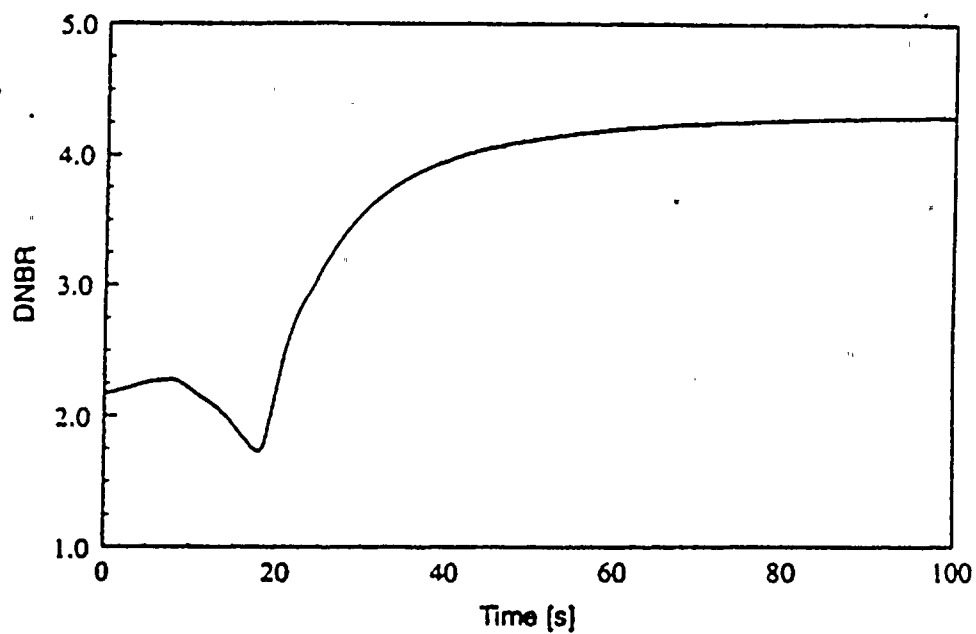
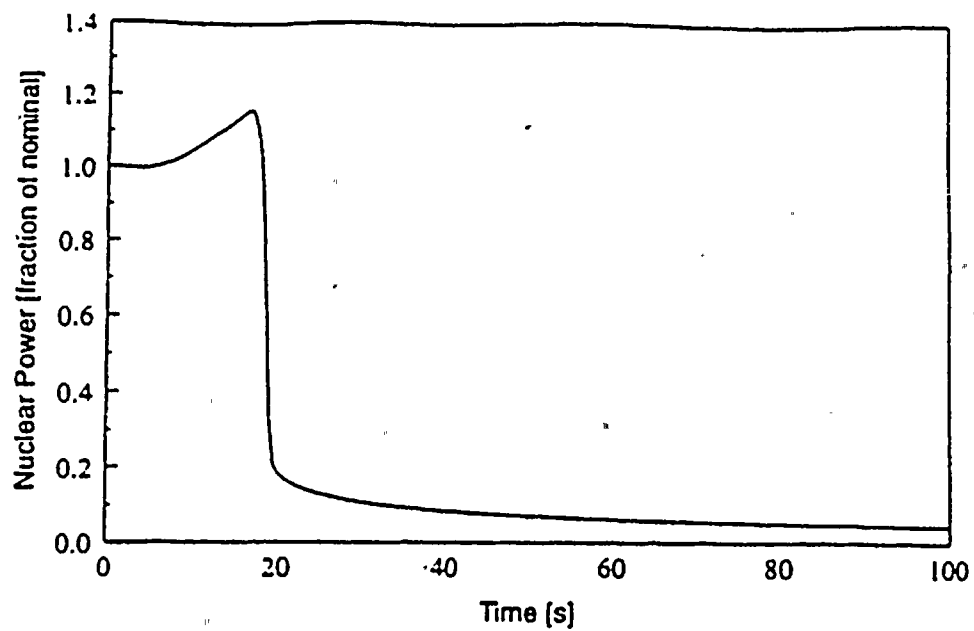
Conclusions

Results of the analyses show that the plant design is such that a loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the limit value.

Table 14.1.8-1
Sequence of Events for Loss
External Electrical Load

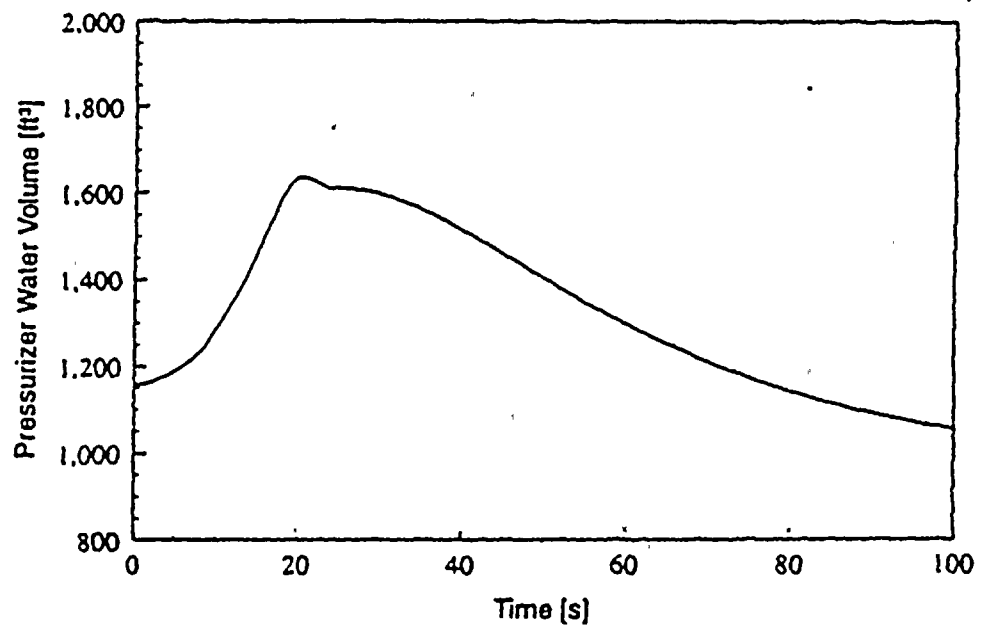
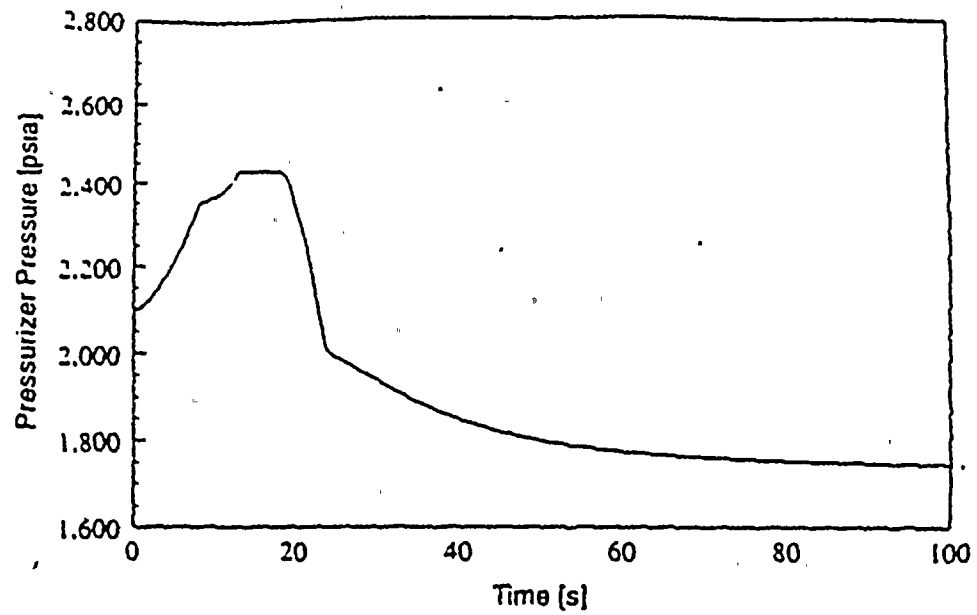
<u>Case</u>	<u>Event</u>	<u>Time (sec)</u>
Minimum Feedback with Pressure Control	Loss of external electrical load	0.0
	OTAT trip setpoint reached	14.2
	Peak RCS pressure occurs	15.5
	Rods begin to drop	16.2
	Minimum DNBR occurs	18.0
Maximum Feedback with Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	Peak RCS pressure occurs	10.0
	Low-low steam generator level trip setpoint reached	68.1
	Rods begin to drop	70.1
Minimum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	8.4
	Rods begin to drop	10.4
	Peak RCS pressure occurs	12.0
Maximum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	8.9
	Rods begin to drop	10.9
	Peak RCS pressure occurs	12.5



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FIGURE 14.1.8-1
Nuclear Power and DNBR vs. Time For
Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs

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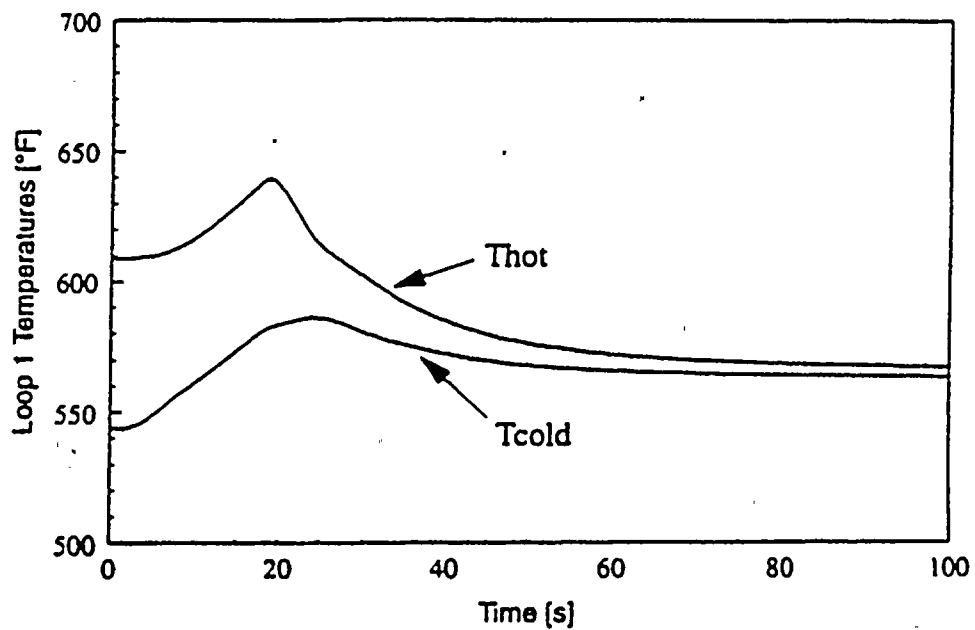
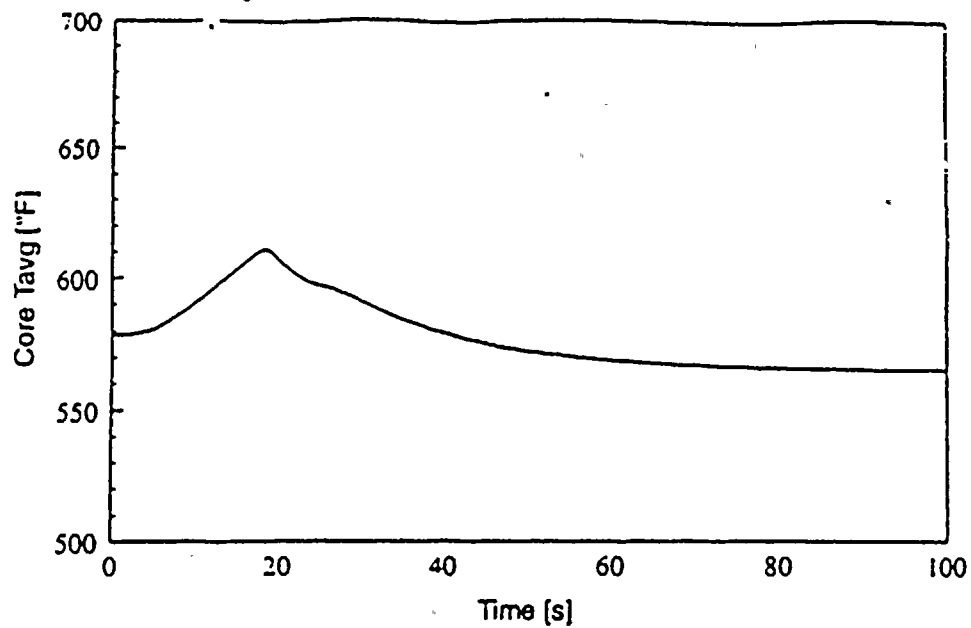


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FIGURE 14.1.8-2

Pressurizer Pressure and Pressurizer Water Volume vs.
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs

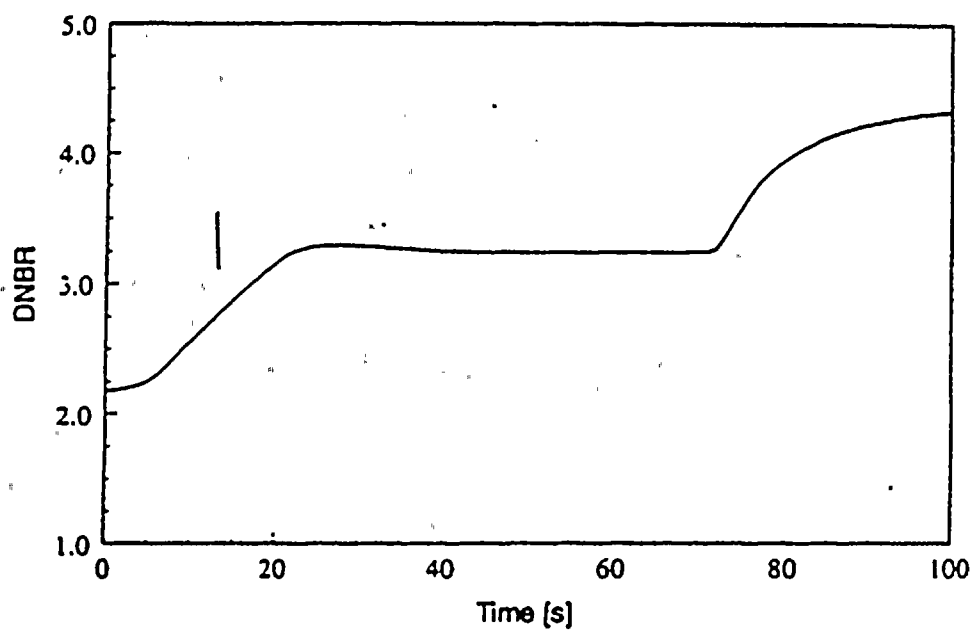
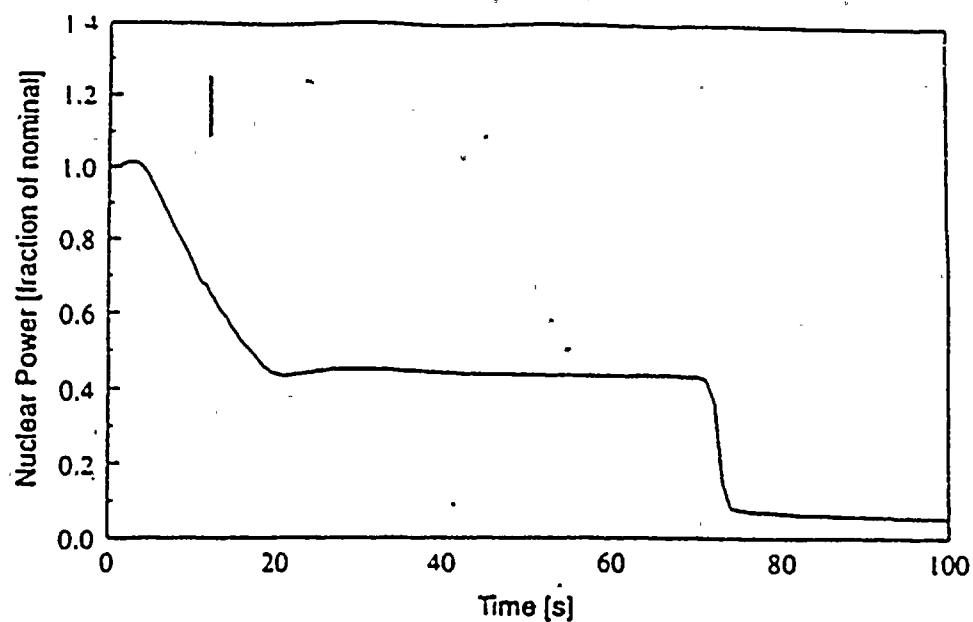
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FIGURE 14.1.8-3

Core Average and Loop 1 Temperatures vs. Time
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs

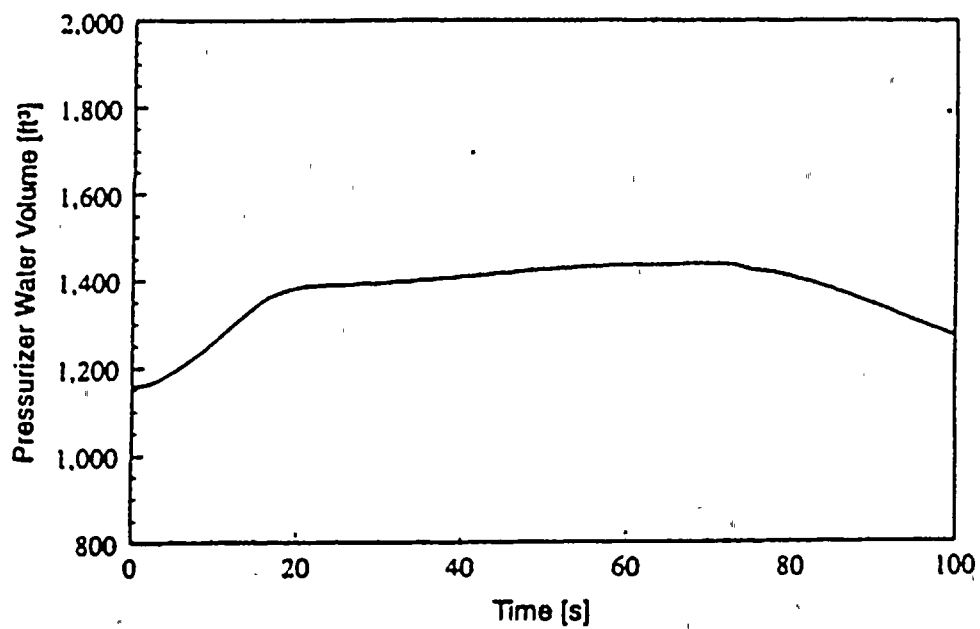
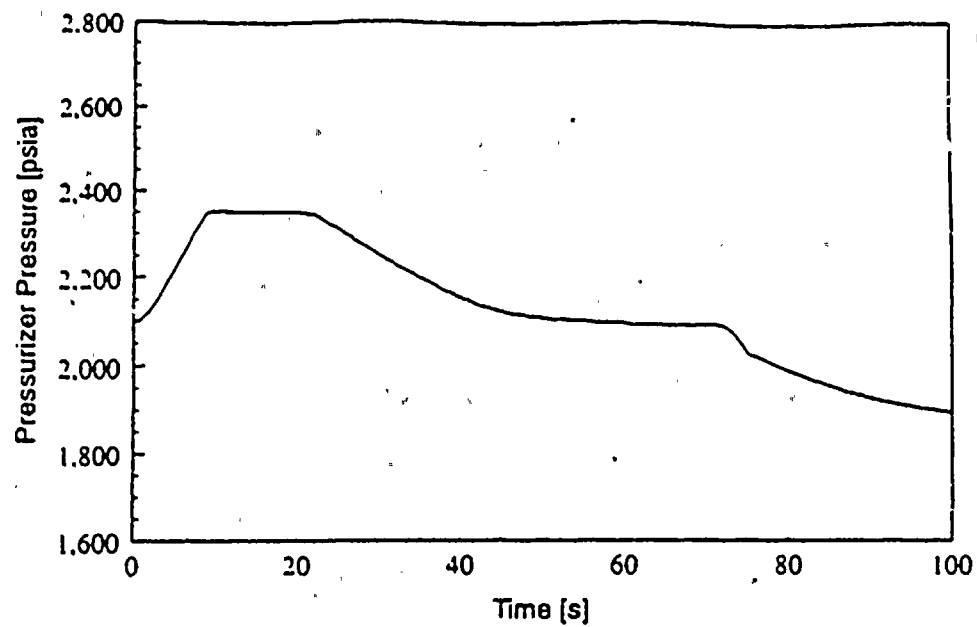


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FIGURE 14.1.8-4

Nuclear Power and DNBR vs. Time For
Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

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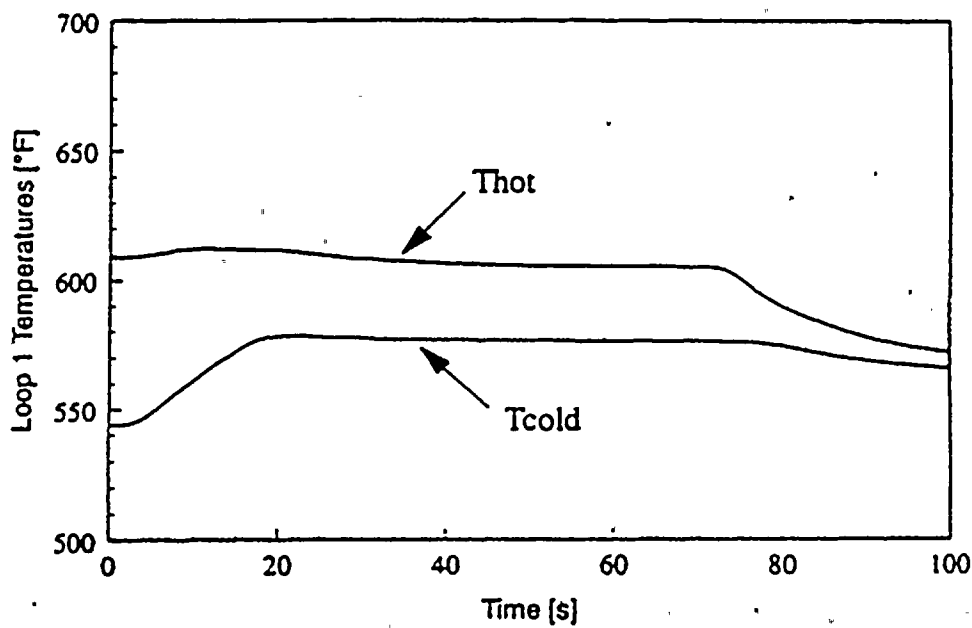
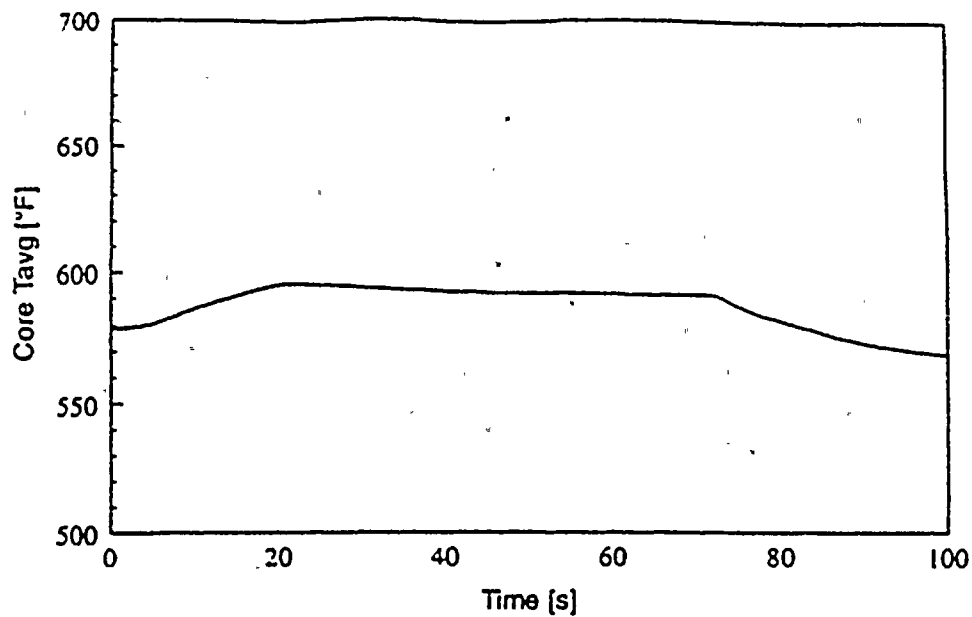


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FIGURE 14.1.8-5

**Pressurizer Pressure and Pressurizer Water Volume vs. Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs**

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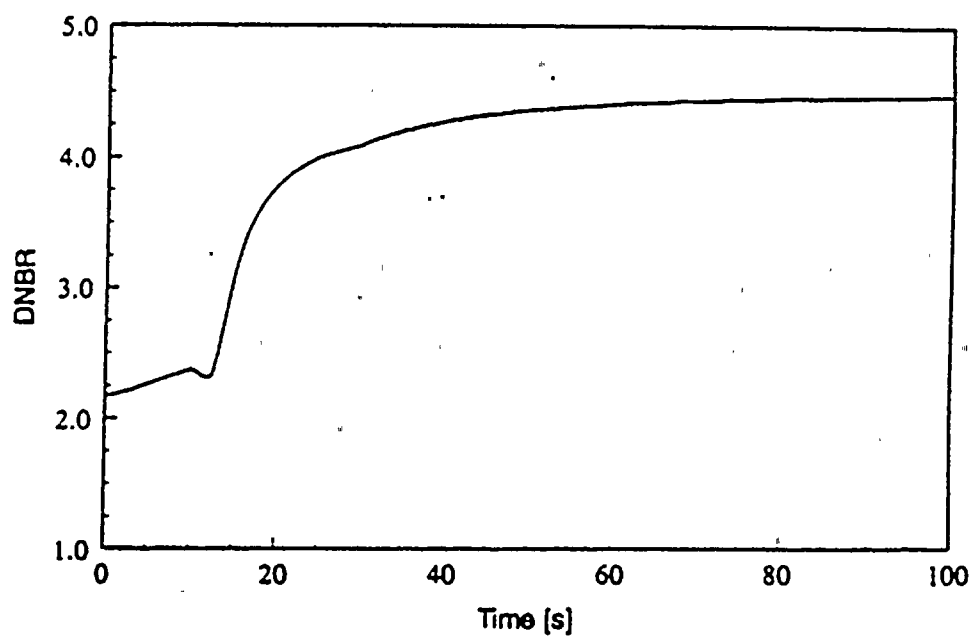
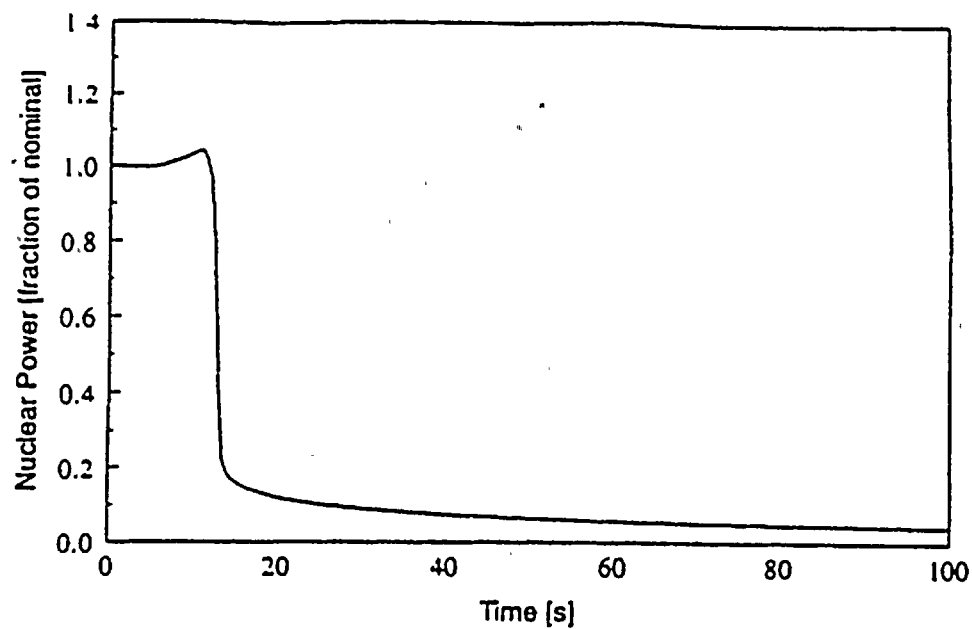


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FIGURE 14.1.8-6

Core Average and Loop 1 Temperatures vs. Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

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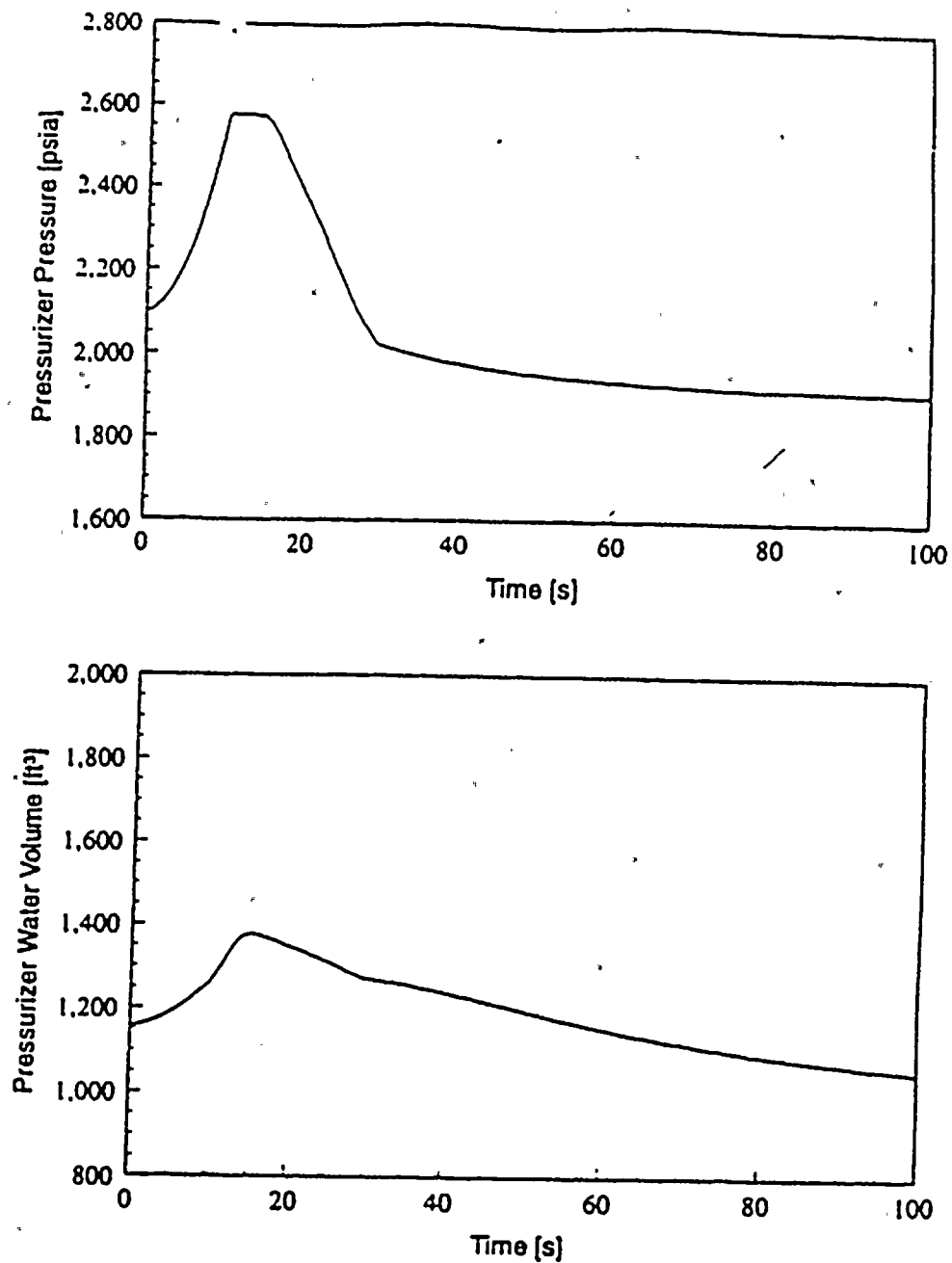


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FIGURE 14.1.8-7

Nuclear Power and DNBR vs. Time For
Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs

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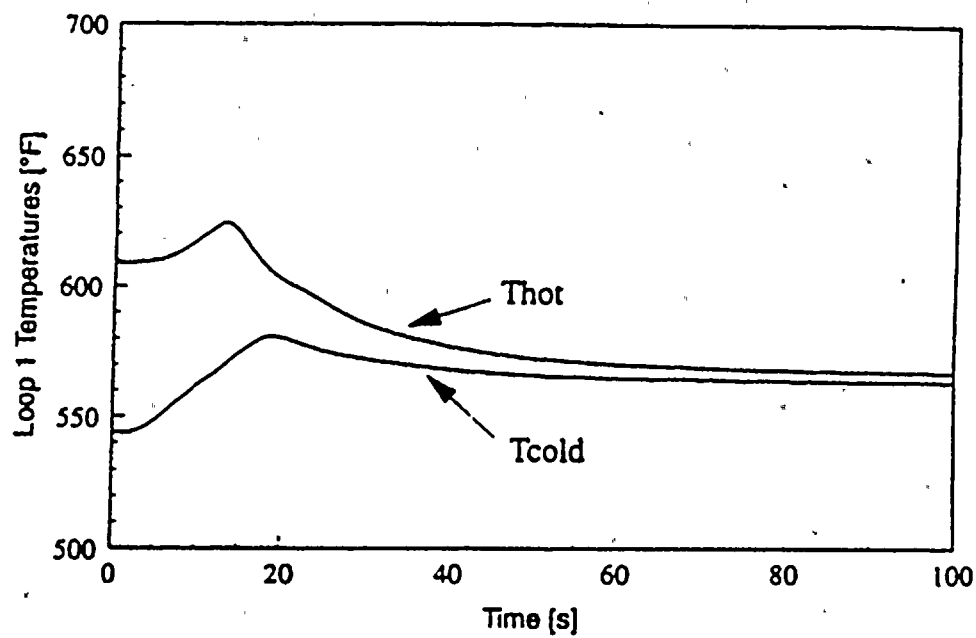
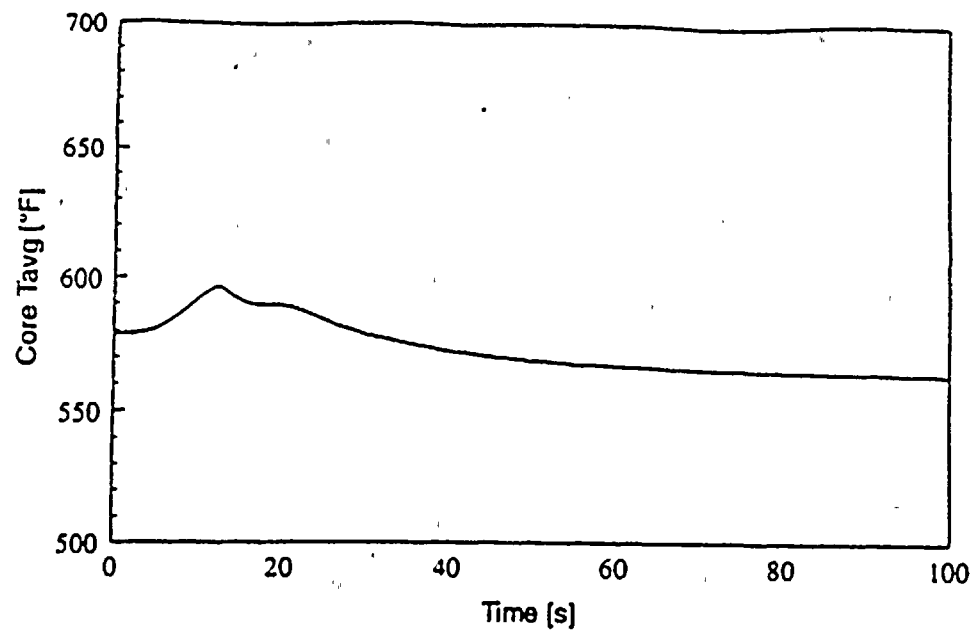


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FIGURE 14.1.8-8

Pressurizer Pressure and Pressurizer Water Volume vs. Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs

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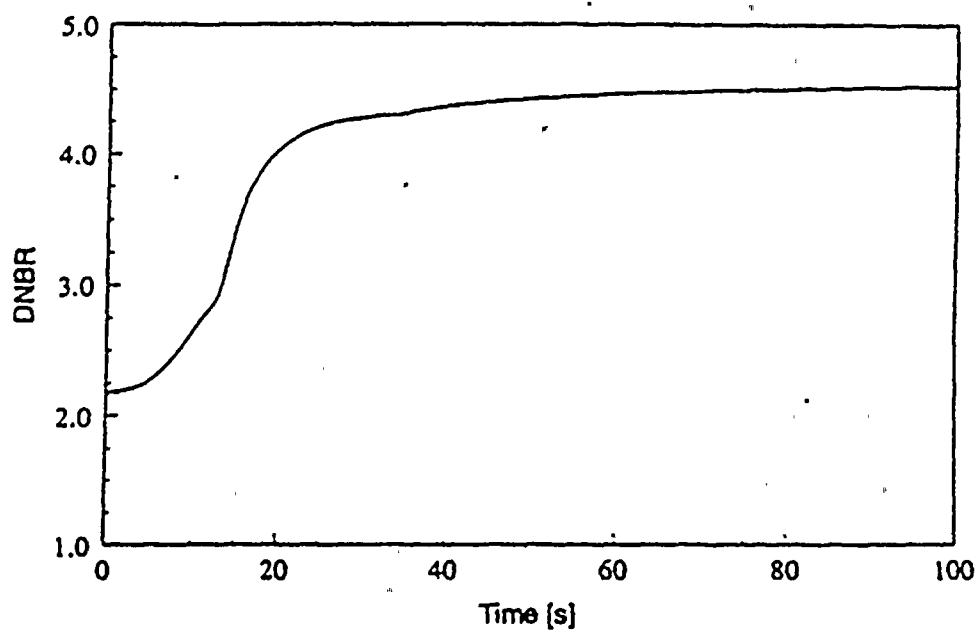
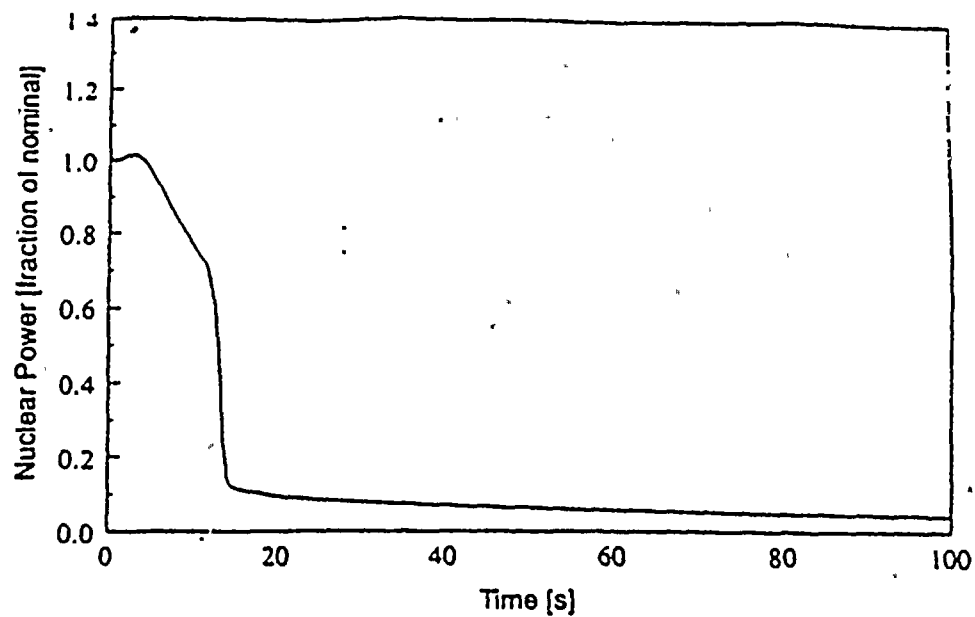


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FIGURE 14.1.8-9

**Core Average and Loop 1 Temperatures vs. Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs**

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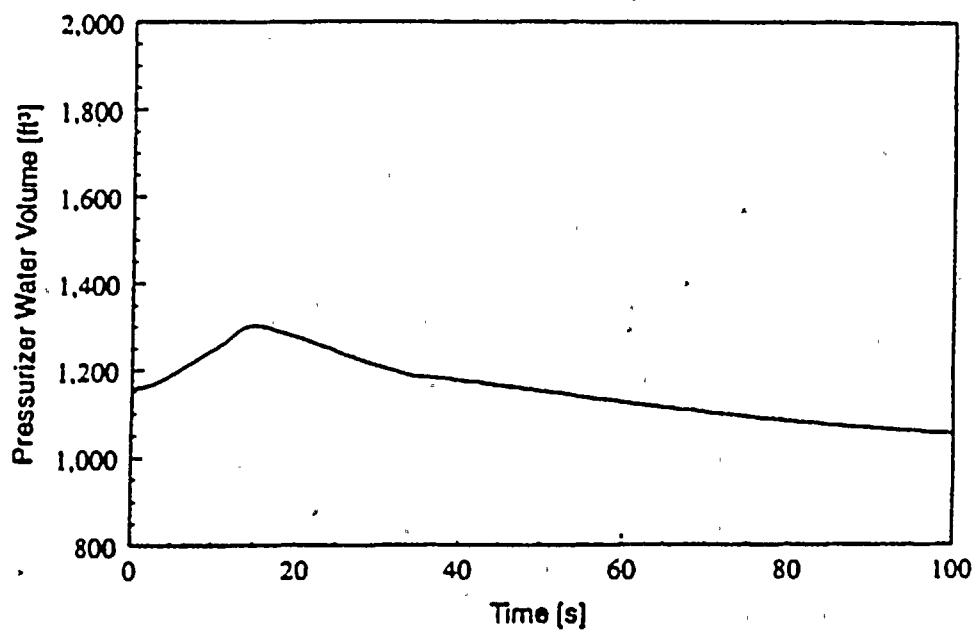
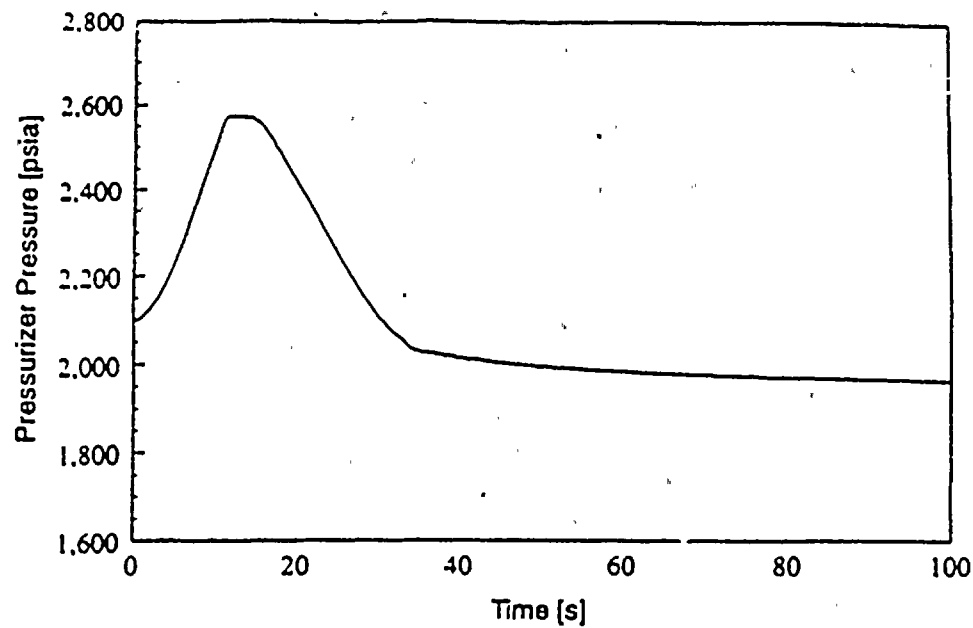


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FIGURE 14.1.8-10

Nuclear Power and DNBR vs. Time For
Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

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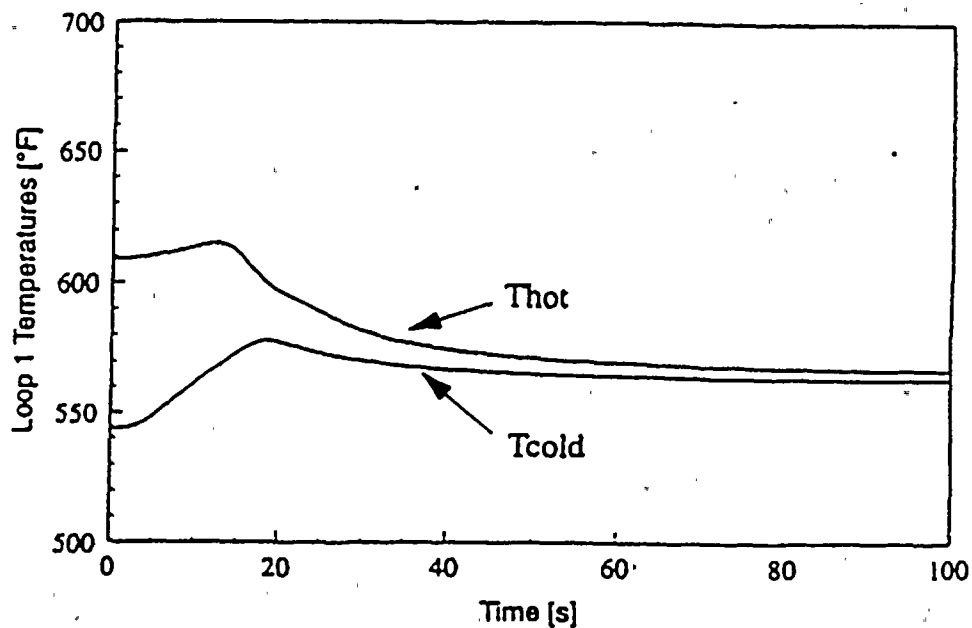
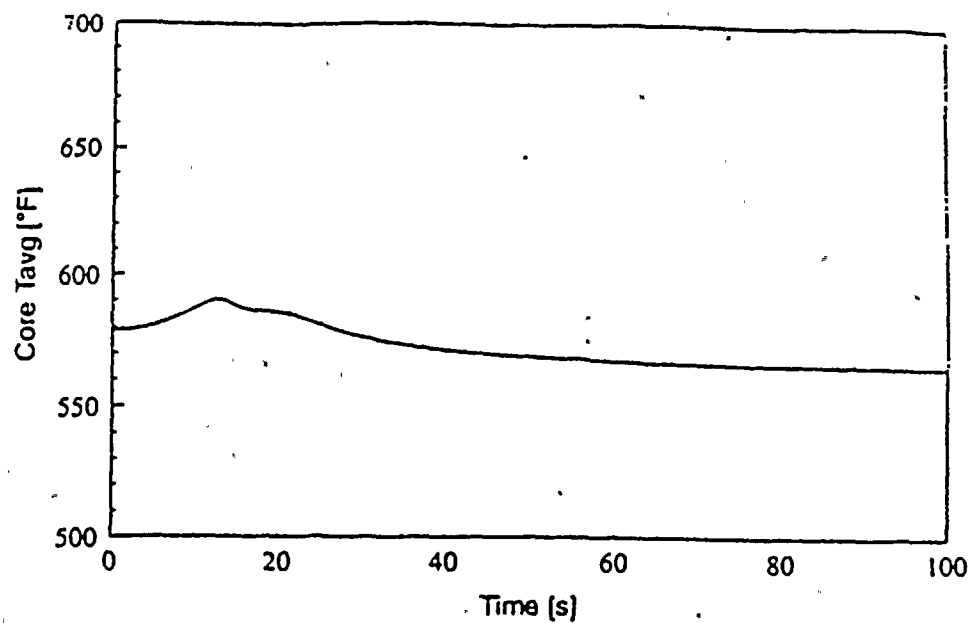


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FIGURE 14.1.8-11

Pressurizer Pressure and Pressurizer Water Volume vs. Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

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FIGURE 14.1.8-12

Core Average and Loop 1 Temperatures vs. Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

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14.1.9 LOSS OF NORMAL FEEDWATER FLOW

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The auxiliary feedwater system is started automatically. The turbine driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators if a loss of offsite power occurs. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

An analysis of the system transient is presented below to show that following a loss of normal feedwater when in reduced temperature and pressure operation, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or uncovering the core, and returning the plant to a safe condition.

Method of Analysis

A detailed analysis using the LOFTRAN code (described in Section 14.1) is performed in order to obtain the plant transient following loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS

including the natural circulation, pressurizer, steam generators and feedwater system. LOFTRAN computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

- A. The plant is initially operating at 102 percent of the Cook Nuclear Plant Unit 1 core power level of 3411 MWt, plus 20 MWt for reactor coolant pump heat. *
- B. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip. The ANS 1979 decay heat model plus two sigma uncertainty was assumed.
- C. Reactor trip occurs on steam generator low-low level.
- D. The worst single failure in the auxiliary feedwater system occurs (e.g., failure of turbine drive auxiliary feedwater pump).
- E. The event is modeled with auxiliary feedwater being delivered to four steam generators at a rate of 450 gpm. Automatic initiation of the auxiliary feedwater is assumed 60 seconds after a low-low steam generator signal is actuated.
- F. Secondary system steam relief is achieved through the steam generator safety valves.
- G. The initial reactor coolant average temperature is 4.5°F higher than the no load temperature, and initial pressurizer pressure is 35 psi higher than the nominal pressure of 2250 psia.
- H. The initial pressurizer water level is assumed to be at the maximum nominal setpoint (62% NRS) plus uncertainties (5% NRS).

* NOTE: Additional evaluations have demonstrated that this analysis bounds the case with a nominal core power level of 3250 MWt and assuming 30% tube plugging.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the fact that no fuel cladding integrity failures are expected to occur during fuel handling operations. However, the above analysis indicates that if the unlikely event of a fuel accident could occur, it would result from the dropping of a fuel assembly either in the containment or auxiliary buildings.

14.2.1.1 Auxiliary Building Accident

In the auxiliary building a fuel assembly could be dropped in the transfer canal or the spent fuel pool. However, supply air for the spent fuel pool area enters from both ends of the auxiliary building, is swept across the fuel pool and transfer canal, exhausted at the side of the fuel pool near the pool elevation, and then discharged through the Unit No. 1 vent.

Doors in the auxiliary building are administratively controlled to maintain controlled leakage characteristics in the spent fuel pool region during refueling operations involving irradiated fuel. Should a fuel assembly be dropped in the canal or in the pool and release radioactivity above a prescribed level the spent fuel pool radiation monitor sounds an alarm and automatically channels the spent fuel pool ventilation exhaust through charcoal filters to remove most of the halogens prior to discharging it to the Unit No. 1 vent. In the event the temporary portable radiation monitor on the bridge crane alarms, action shall be taken to manually align the spent fuel pool exhaust ventilation through the charcoal filters.

If the discharge vent radiation monitor indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the supply and exhaust ventilation systems servicing the spent fuel pool can be shut down, limiting the leakage to the atmosphere.

Any movement of the fuel cask in the spent fuel pool area is under administrative control. Interlocks prevent the crane from moving the cask over stored irradiated fuel and limit cask movement to one corner of the spent fuel pool away from the fuel assemblies. A cage or metal cylinder arrangement surrounds the cask during its movement in the pool to prevent it from toppling into any fuel assemblies. Interlocks prevent movement of the crane hook over any other portion of the spent fuel pool* at any other time except when it is absolutely necessary to service the pool and its equipment and instrumentation, and to add or remove any equipment associated with spent fuel handling, storage, or inspection. The crane hook is limited to the Technical Specification 3.9.7 value with the entire operation under strict administrative control. The detailed design and safety analysis of D.C. Cook Cask Drop Protection System, which has not yet been installed, is described in References (1), (2) and (3).

The probability of a fuel handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility as previously mentioned. The shock absorbing analyses presented above indicate that in most incidents where an assembly is struck against another object, the outer row of fuel rods would experience greater loads and stresses than the inner rows. Therefore, if a fuel assembly is dropped it does not necessarily mean that all the fuel rods break. Nevertheless, for a fuel handling accident analysis, the assumption is made that the cladding of all the fuel rods in one fuel assembly break suddenly, releasing all the gaseous fission products in the voids between the pellets.

* The main hoist load block of either auxiliary building crane and the auxiliary hoist load block of the east crane may be moved over the spent fuel pool if no load is being carried.

DF_f = effective Iodine decontamination factor for filters (= 10)

DF_p = effective Iodine decontamination factor for pool water (= 100)

The gap inventories listed in Table 14.2.1-1 are the product of I_i (core inventory) and F_g (the fraction existing in the gap).

The function used to calculate the external whole body dose from beta (D_β) or gamma (D_γ) radiation in the cloud uses many of the terms defined above and is given by:

$$D_\beta = \sum_i 0.23 (x/Q) F P G_i E_{\beta_i}$$

and

$$D_\gamma = \sum_i 0.25 (x/Q) F P G_i E_{\gamma_i}$$

where G_i is the gap inventory of the gaseous radionuclides of Xe and Kr and the functions above are summed over all the noble gases. E_β and E_γ are the average energies of decay (beta and gamma radiation respectively) for the various radionuclides. These functions assume the noble gas decontamination factors in water and the charcoal filters are 1.0. The gap inventories of radioiodine make a negligible contribution to the whole body doses, D_β or D_γ , because of the large decontamination factors appropriate to the iodines.

RESULTS

A summary of the assumptions used to evaluate the fuel handling accident is given in Table 14.2.1-2. The minimum time after shutdown when fuel assemblies would be moved was conservatively assumed to be 100 hours. At 100 hours after shutdown, the two-hour dose at the site boundary, for a fuel handling accident releasing all of the gaseous fission product radioactivity in the gaps of all rods in the highest power assembly, are as follows:

Two-Hour Site Boundary Dose

	<u>NUREG/CR-5009</u> <u>Method</u>	<u>Reg. Guide</u> <u>1.25</u>
Inhalation thyroid dose =	7.07 Rads	5.97 Rads
Whole body beta dose, D, =	0.36 Rads	0.70 Rads
Whole body gamma dose, D, =	0.31 Rads	0.58 Rads

These doses are well within the limits of 10 CFR Part 100 in conformance with the acceptance criteria of SRP 15.7.4. (Rev. 1, July 1981)^o.

14.2.1.2 FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

During fuel handling operations, the containment is kept in an isolated condition with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from a radiation monitor indicating that radioactivity is above prescribed limits. During core alterations, one containment airlock door shall be closed or, appropriate administrative controls shall be in place to allow both airlock doors to remain open.

Should a fuel assembly be dropped and release activity above a prescribed level, the upper and/or lower containment area radiation monitors sound an alarm, the containment is isolated, and personnel evacuated.

Potential consequences of a fuel handling accident were evaluated using (1) the conservative assumptions listed in Regulatory Guide 1.25 and (2) the realistic assumptions given in Regulatory Guide 4.2 Appendix I. The analysis was done for a core power level of 3391 MWt, which was representative of the original Unit 2 licensed power level. Unit 2 is currently licensed for a core power level of 3411 MWt. See Unit 2 UFSAR Section 14.3.5 for a reevaluation at a bounding core power level of 3588 MWt.

TABLE 14.2.1-1
FUEL HANDLING ACCIDENT AUXILIARY BUILDING
INVENTORIES AND CONSTANTS OF SIGNIFICANT FISSION PRODUCT RADIONUCLIDES

NUCLIDE	SHUTDOWN CORE INVENTORY CURIES	DECAY CONST. λ , 1/HRS	TOTAL GAP INVENTORY, CURIES		DOSE CONVERSION Ri	E, (MEV)	E, (MEV)
			NUREG/CR-5009	Reg. Guide 1.25			
			100 hrs	100 hrs			
I-131	9.0 E+7	3.591E-3	7.5 E+6	6.3 E+6	1.48 E+6	0.186	0.389
I-132	1.3 E+8	3.013E-1	Negligible *	Negligible	5.35 E+4	-	-
I-133	1.8 E+8	3.332E-2	6.3 E+5 *	6.3 E+5	4.0 E+5	0.419	0.597
I-134	1.9 E+8	7.905E-1	Negligible *	Negligible	2.5 E+4	-	-
I-135	1.7 E+8	1.048E-1	Negligible *	Negligible	1.24 E+5	0.394	1.456
Kr-85M	1.9 E+7	1.547E-1	Negligible *	Negligible		-	-
Kr-85	1.4 E+6	7.376E-6	2.0 E+5	4.2 E+5		0.251	0.002
Kr-87	3.6 E+7	5.451E-1	Negligible	Negligible		-	-
Kr-88	5.0 E+7	2.442E-1	Negligible	Negligible		-	-
Xe-131M	1.0 E+6	2.427E-3	7.9 E+4 *	7.9 E+4		-	0.163
Xe-133M	5.6 E+6	1.319E-2	1.5 E+5 *	1.5 E+5		-	0.233
Xe-133	1.8 E+8	5.506E-3	5.1 E+6	1.0 E+7		0.102	0.081
Xe-135	3.9 E+7	7.626E-2	Negligible	Negligible		0.309	0.262

* No release fraction given - assumed same as Reg. Guide 1.25
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TABLE 14.2.1-2

DATA AND ASSUMPTIONS FOR THE EVALUATION
OF THE FUEL HANDLING ACCIDENT
IN THE AUXILIARY BUILDING

1.	<u>Source Term Assumptions</u>	<u>VALUES</u>
	Core power level, MWT	3411
	Fuel burnup, MWD/MTU	60,000
	Analytical method	ORIGEN
2.	<u>Release Assumptions</u>	
	Number of failed fuel rods	all rods in 1 of 193 assemblies
	Fraction of core inventory released to gap (NUREG/CR-5009 % release of Iodine-131 is reported to be 20% higher)	<u>Reg. Guide 1.25</u> % of the Iodine - 10 % of the Xenon - 10 % of Kr-85 - 30
	Assumed power peaking factor	1.65
	Inventory in gap available for release	Table 14.2.1-1
	Pool decontamination factors	
	For Iodines	100
	For noble gases	1
	Filter decontamination factors	
	For Iodine	10
	For noble gases	1
	Atmospheric Dispersion, (χ/Q)	3.15×10^{-4} sec/m'
	Breathing rate	3.47×10^{-4} m'/sec

14.2.2 Accidental Release of Radioactive Liquids

The inadvertent release of radioactive liquid to the environment is not considered a credible accident. Any radioactive liquids must ultimately be diverted to the monitor tanks, and any tritium from the CVCS to the monitor tanks also, prior to discharge. (Liquids from these tanks are sampled and monitored for acceptable radioactive levels before being released to the lake.) Erroneous sampling and malfunction of the radiation monitor would have to occur sequentially to discharge radioactive liquid inadvertently, and this series of events is not considered credible.

Waste Evaporator Condensate and Monitor Tanks

Any spillage of radioactive fluid due to equipment leaks or ruptures would drain directly to either the sump tank or waste holdup tanks, or would accumulate in the area sumps prior to being pumped to the waste holdup tanks. Radioactive liquids to be processed by the waste disposal system are ultimately stored in the waste holdup tanks.

Periodically the contents of the waste holdup tanks and the laundry tanks are analyzed and if the radioactive level is within discharge limits, the liquid is transferred to the waste evaporator condensate tanks and then to the monitor tanks for release.

Effluents from the waste disposal system and monitor tanks 3 and 4 are released, not recycled. Distillate from the CVCS boric acid evaporator is discharged to monitor tanks. The contents of monitor tanks 1 and 2 are analyzed before being pumped to the primary water storage tanks.

Occasionally it may be necessary to dispose of some of the boric acid distillate for tritium control. (If analysis of the contents of the monitor tank is within prescribed limits for discharge to the environment, the liquid is pumped directly to the waste liquid discharge line after the normally locked-closed valve in this line is opened.) The radiation monitor downstream prevents discharge of fluids above prescribed levels as explained in the preceding paragraph.

A representative sample is obtained from the monitor tank to determine appropriate release setpoints. Administrative clearance must be granted to open a locked-closed valve. In the highly unlikely event that the locked-closed valve is opened and the tank contents are inadvertently pumped to the discharge tunnel for release to the lake without being previously analyzed for activity, the radiation monitors setpoint is set such that the release will not exceed release limits. If it did, the radiation monitor would trip the second valve downstream of the monitor and terminate the release. Therefore, a pumping accident having radiological consequences is not considered credible.

Condensate Storage Tank, Primary Water Storage Tank, and Refueling
Water Storage Tank

The condensate storage tank and the primary water storage tank are essentially free from radionuclides. The refueling water storage tank contains a relatively low level of radioactivity. These tanks are not connected to the radwaste system. In the unlikely event of loss of water from any of these tanks the water will percolate down the underground water table, which is estimated to be at elevation 590', that is, about 20 feet below ground level. The hydraulic gradient of the ground is very low; less than 4%. Our studies show a minimum of 50 years would be required for the water to reach the nearest ground water well. The spilled water would preferentially follow the very small natural ground gradient toward the lake and would be eventually diluted in the lake water. By the time any radioactive materials reach the nearest drinking water intake from the lake,

Bridgman is 2.5 miles away from the plant discharge, resultant dilution, dispersion, and radioactive decay will have reduced the radiological consequences to insignificance.*

- * The information presented here refers to the original Unit 1 studies. Later results of studies on this subject are included in Section 14.2 of the Unit 2 Updated FSAR.

Auxiliary Building Liquid Waste Storage Tanks

The inadvertent release of radioactive liquid waste to the environment is not considered a credible accident. Any spillage of radioactive fluid due to equipment leaks or ruptures would drain directly to either sumps or waste holdup tanks. Radioactive liquid wastes are diverted to tanks to be processed for release. Tanks are sampled and analyzed to determine that the concentration of radioactive nuclides can be released within discharge limits. The release must pass through a normally locked closed valve, a radiation monitor and another valve in series prior to reaching the discharge tunnels for release to the lake. Administrative clearance must be granted to open the locked closed valve. In the highly unlikely event that the locked-closed valve is opened and the tank contents are inadvertently pumped to the discharge tunnel for release to the lake without being previously analyzed for activity, the radiation monitors setpoint is set such that the release will not exceed release limits. If it did, the radiation monitor would trip the second valve downstream of the monitor and terminate the release. . Therefore, a pumping accident involving radioactive waste releases having radiological consequences is not considered credible.

Piping

The pipes running from the refueling water storage tank, the primary water storage tank, and the condensate storage tank to the auxiliary building are installed in a pipe tunnel. In case of a break in any of these pipes, the water will enter the auxiliary building sump, from where it will be processed as described in the Auxiliary Building liquid waste tanks. No pipes from these tanks are directed toward the containment building.

CVCS Holdup Tanks

The analysis of a CVCS holdup tank rupture is presented in Section 14.2.2, Unit 2.

14.2.3 Accidental Waste Gas Release

Radioactive gases are introduced into the reactor coolant by the escape of fission products if defects and contamination existed in the fuel cladding. The processing of the reactor coolant by auxiliary systems results in the accumulation of radioactive gases in various tanks. The two main sources of any significant gaseous radioactivity that could occur would be the volume control tank (VCT) and the gas decay tanks. These tanks, located in the lower elevations of the auxiliary building which is a seismically designed structure, are also designed to withstand a seismic event (DBE) without failure. For the purposes of an accidental waste gas release analysis, it is assumed that a tank ruptures by an unspecified mechanism after the reactor has been operating for one core cycle with 1% defects in the fuel cladding.

Volume Control Tank

Noble gases in the reactor coolant accumulate in the volume control tank throughout a core cycle by the stripping action of the entering spray. Gases retained in this tank are vented to the gas decay tanks when the reactor is shutdown for refueling. A rupture of the volume control tank just prior to venting would release all the accumulated gases in the liquid and gas phases, plus that amount in the 75 gpm flow from the letdown line which is assumed to continue flowing up to fifteen minutes until isolation is accomplished. The equilibrium activities which are associated with a release of the gases at this time are based on 1% defects in the fuel cladding, and are listed in Table 14.2.3-1. The total represents 18,080 curies equivalent Xe-133.

Gas Decay Tanks

The gas decay tanks accumulate radioactive gases from three major sources processed by the waste disposal system: the gas stripper, the liquid holdup tanks, and the volume control tank. Of these three, only the volume control tank is capable of introducing large amounts of activity to the gas decay tanks in a relatively short period of time. After

shutting down the reactor for refueling, the reactor coolant system is purified and degassed. The gases accumulated in the volume control tank are periodically vented to the waste gas compressor prior to being stored in the gas decay tanks. For an accident analysis, it is assumed that the entire equilibrium inventory of Kr-85 and Xe-133 in the reactor coolant system and the volume control tank vapor space is contained in a single gas decay tank at the time of rupture. The other noble gas isotopes are not considered because they are present in negligible amounts in the reactor coolant or become negligible through decay during the processing period. This approach is conservative since no credit is taken for the decay of Xe-133 while being stripped from the reactor coolant and transferred to the gas decay tanks. The maximum activities available for release are based on 1% fuel cladding defects and are listed in Table 14.2.3-2. The total represents 83,300 curies equivalent Xe-133.

Dose Evaluation

Off-site radiation exposure evaluated for noble gases released is based on the meteorological model and radiation dose equation described in Section 14.3.5, including the effect of dilution in the wake of one containment building, a 2 m/sec wind velocity and a dispersion factor, $\chi/Q = 3.15 \times 10^{-4}$ sec/m³ at the site boundary, 610 meters from the containment. Assuming the incident occurred immediately after a refueling shutdown following operation with 1% fuel cladding defects, the maximum two-hour integrated whole body dose at the site boundary during passage of escaped gases would be:

Volume Control Tank	0.27 rem
Gas Decay Tank	1.26 rem

It is, therefore, concluded from the above analysis that an unlikely event of an accidental waste gas release would present no hazard to the health and safety of the public, since the maximum two-hour integrated whole body dose at the site boundary is well below the 25 rem guideline for accidental exposure as set forth in 10 CFR 100.

14.2.4 Steam Generator Tube Rupture

General

The accident examined is the complete severance of a single steam generator tube (SGTR). The accident is assumed to take place at a reactor power level of 3262 MWt¹ with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power operated relief valves (and safety valves if their setpoint is reached).

The steam generator tube material is Inconel 600 and, as the material is highly ductile, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance and an accumulation of minor leaks which exceed the Technical Specification limits is not permitted during unit operation.

The operator is expected to determine that a steam generator tube rupture (SGTR) has occurred, to identify and isolate the ruptured steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize the contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured steam generator. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale that ensures that break flow to the ruptured steam generator is terminated before the water level in the affected steam

¹ Assumes 30% tube plugging.

generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Description of Accident

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low level alarms are actuated, and prior to plant trip, charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip, as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that steam generator.
2. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer until a reactor trip signal is generated by low pressurizer pressure or overtemperature ΔT . A safety injection signal, initiated by low pressurizer pressure follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.
3. The steam generator blowdown liquid monitor and the air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.
4. The reactor trip automatically trips the turbine, and if outside power is available, the steam dump valves open, permitting steam dump to the condenser. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator power operated relief valves (and the steam generator safety valves if their setpoint is reached).

5. Following plant trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank (RWST)) provide a heat sink. Thus, steam bypass to the condenser, or in the case of loss of outside power, steam relief to atmosphere, is attenuated during the time in which the recovery procedure leading to isolation is being carried out.
6. Safety injection flow results in restoration of pressurizer water level.

Results

In estimating the mass transfer from the reactor coolant system through the broken tube, the following assumptions were made:

- a. Plant trip occurs automatically as a result of low pressurizer pressure.
- b. Following the initiation of the safety injection signal, both centrifugal charging pumps are actuated and continue to deliver flow.
- c. After reactor trip the break flow equilibrates to the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.2.4-1. In the original accident analysis, the resultant break flow is assumed to persist from plant trip until 30 minutes after initiation. An assessment has been made of the impact on the original analysis of allowing the operator longer than 30 minutes to terminate break flow to the faulted steam generator. That assessment has shown that the break flow termination could be increased up to two hours (provided the steam generator does not overfill) without exceeding the offsite dose radiological guidelines discussed in the "Conclusion" portion of this section. (Reference 3)

- d. The steam generators are controlled at the safety valve setting minus 3% tolerance rather than the power operated relief valve setting.
- e. The original analysis assumed that the operator identifies the accident type and terminates break flow to the ruptured steam generator within 30 minutes of accident initiation. An assessment has been made of the impact on the original analysis of allowing the operator longer than 30 minutes to terminate break flow to the faulted steam generator. That assessment has shown that the break flow termination could be increased up to two hours (provided the steam generator does not overfill) without exceeding the offsite dose radiological guidelines discussed in the "Conclusion" portion of this section. (Reference 3)

The above assumptions lead to a conservative upper bound of 140,264 pounds for the total amount of reactor coolant transferred to the ruptured steam generator and 56,525 pounds for the total amount of steam released to the atmosphere via the ruptured steam generator as a result of the steam generator tube rupture accident.

Recovery Procedure

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the Emergency Operating Procedures (EOPs). The EOPs are based on guidance in the Westinghouse Owner's Group Emergency Response Guidelines (Reference 1) which addresses the recovery from a SGTR with and without offsite power available. The major operator actions include identification and isolation of the rupture steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

High secondary side activity, as indicated by the secondary side radiation monitors will typically provide the initial indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator level, a high radiation indication on the main air ejector monitor, or from the steam generator blowdown liquid monitor. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease off-scale on the narrow range for all of the steam generators. The auxiliary feedwater flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will return to the narrow range earlier in that steam generator and will continue to increase more rapidly. This response, as indicated by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cook down the RCS using the intact steam generators.

After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam

from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power operated relief valves (PORVs) on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

- Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

30 Percent Tube Plugging Analysis

Reference 2 addresses the recent analysis for to support steam generator tube plugging for unit 1 cycle 16. This included an evaluation of the steam generator tube rupture accident to determine the impact on dose releases associated with the analysis.

The primary thermal hydraulic parameters which affect the calculated offsite radiation doses for a steam generator tube rupture event are the assumed radioactivity in the reactor coolant, the reactor coolant released through the ruptured tube to the secondary steam volume, and the steam released from the ruptured tube to the atmosphere. The change in steam generator tube plugging did not impact the reactivity level of the reactor coolant. However, both the primary coolant release to the secondary and the secondary system release to the atmosphere were impacted by the assumed tube plugging level.

To evaluate the effect of 30% steam generator tube plugging level, the mass releases from the RCS and from the secondary volume to the atmosphere were calculated. Four cases were considered assuming a nominal RCS temperature of 533°F and 576.3°F with both symmetric and asymmetric RCS flow conditions. A nominal full power level of 3262 MWt was also assumed. The results of the 30% steam generator tube plugging analysis resulted in offsite radiological consequences less than those previously provided.

Conclusion

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming a simultaneous loss of offsite power such that liquid does not enter the steam piping space. The doses to the public as a result of a steam generator tube rupture have been shown to be less than the permissible limits of 10 CFR Part 100. These limits are: for pre-accident iodine spike, the thyroid dose in 10 CFR 100 or 300 rem, for accident initiated iodine spike, 10% (small fraction) of 10 CFR 100 or 30 rem thyroid and 2.5 rem gamma body.

References

1. Westinghouse Owners Group; Emergency Response Guidelines, Published by Westinghouse Electric Corporation for the Westinghouse Owners Group.
2. Westinghouse Electric Co.; Donald C. Cook Nuclear Plant, Unit 1; Steam Generator Tube Plugging Licensing Report, May, 1995 (Westinghouse report WCAP-14285, Revision 1)
3. Westinghouse letter to AEP.NSD-NT-ESI-97-388, (AEP 97-102); American Electric Power Donald C. Cook Nuclear Plant Units 1 and 2, Revised SGTR FSAR Section 14.2.4; dated June 26, 1997.

14.2.5 Rupture of a Steam Pipe

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative coolant temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high hot channel factors which exist when the most reactive assembly is assumed stuck in its fully withdrawn position. The core is ultimately shut down by boric acid delivered by the emergency core cooling system.

The analysis of a steam pipe rupture is performed to demonstrate that:

Assuming a stuck assembly, with or without offsite power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

- A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN Code, described in Section 14.1, has been used.

- B. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC, described in Section 14.1, has been used to determine if DNB occurs for the core conditions computed in item A above.

The following conditions were assumed to exist at the time of a main steam line break accident:

- A. End-of-life shut down margin ($1.30\% \Delta k/k$) at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the RCCA banks during core burnup is restricted in such a way (to not violate the rod insertion limits presented in the Technical Specifications) that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.
- B. A negative moderator coefficient of reactivity corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1050 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 14.2.5-1. The Doppler power feedback assumed for this analysis is presented in Figure 14.2.5-2.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the

- F. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and thus are different for each case studied.

The analyses assumed initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than steam line breaks occurring at power.

- G. In computing the steam flow during a steam line break, the Moody Curve⁽²⁾ for $f_1/D = 0$ is used.

- H. The fast acting steam line isolation valves are assumed to close in less than eleven seconds from receipt of actuation signal. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Results

The limiting case of cases a through e was shown to be the doubled-ended rupture located upstream of the flow restrictor with offsite power available. Table 14.2.5-1 lists the limiting statepoints for this worst case. The results presented are a conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously. The sequence of events for this transient is presented in Table 14.2.5-2.

Figures 14.2.5-4 through 14.2.5-7 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe at the exit of the steam generator nozzle) at initial no-load condition.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steamline and the remaining steamlines or by high steam flow signals in coincidence with either low low RCS temperature or low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high containment pressure signals or low steam line pressure or high steam flow coincident with low-low T_{avg} . Even with the failure of one valve, release is limited to approximately 13 seconds for the other steam generators while the one generator blows down. The steam line stop valves are designed to be fully closed in less than eleven seconds from receipt of a closure signal.

As shown in Figure 14.2.5-7, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2400 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

The assumed steam release for an accidental depressurization of the main steam system (Case e) is the maximum capacity of any single steam dump, relief, or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 2400 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about five minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown. The DNB transient is bounded by the limiting case for a steamline rupture.

The DNB analysis for the limiting case (double-ended rupture located upstream of the flow restrictor) showed that the minimum DNBR remained above the limit value.

Conclusions

The analysis has shown that the criteria stated earlier are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for the rupture (or an accidental depressurization of the main steam system) assuming the most reactive RCCA stuck in its fully withdrawn position.

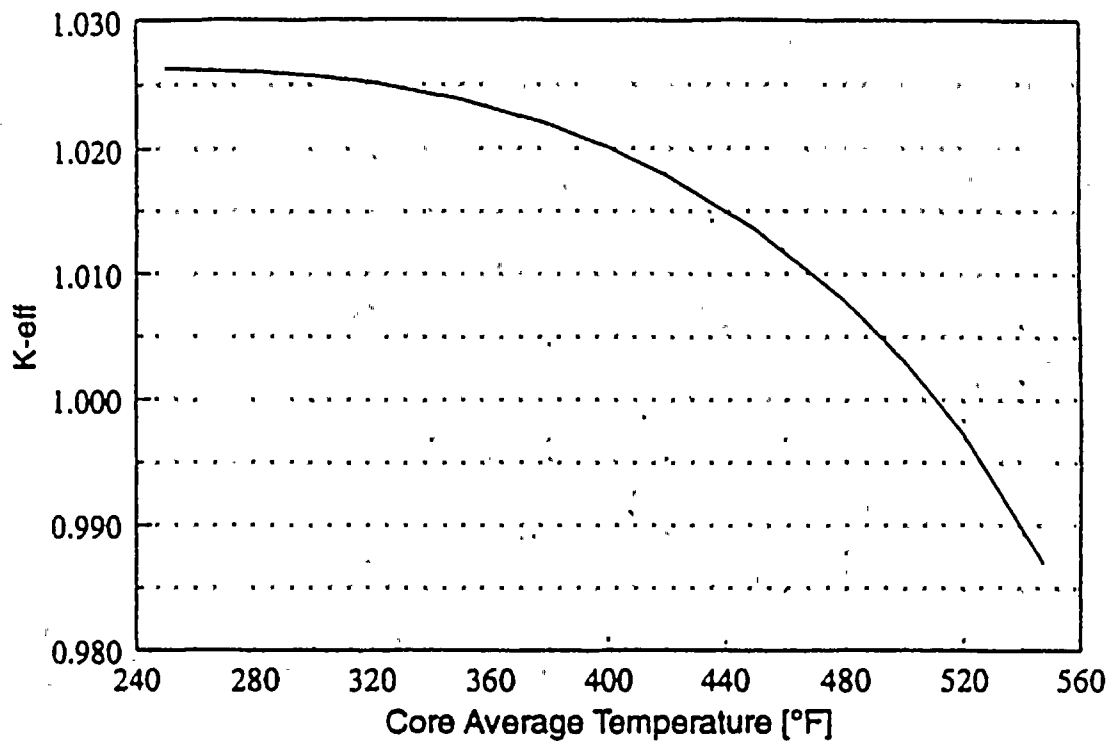
TABLE 14.2.5-1

LIMITING STEAMLINE BREAK STATEPOINT
DOUBLE ENDED RUPTURE INSIDE CONTAINMENT
WITH OFFSITE POWER AVAILABLE

Time sec	Pressure psia	Heat Flux Fraction	Inlet Cold °F	Temp. Hot °F	Flow Fraction	Boron PPM	Reactivity Percent	Density gm/cc
180.2	601.93	.228	336.6	463.3	1.0	7.13	-.001	.849

Table 14.2.5-2
TIME SEQUENCE OF EVENTS
DOUBLE ENDED RUPTURE INSIDE CONTAINMENT
WITH OFFSITE POWER AVAILABLE

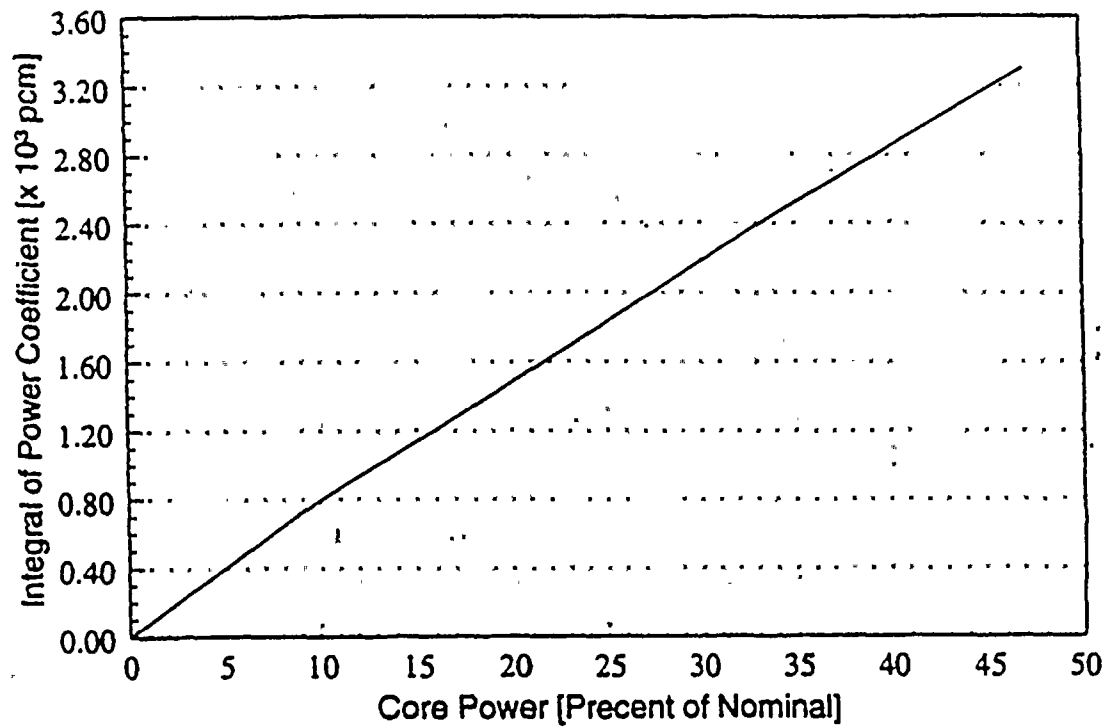
<u>Event</u>	<u>Time (sec)</u>
Steam line rupture occurs	0.00
Low steam line pressure coincident with high steam flow in two steam lines reached	2.06
Feedwater Isolation (All loops)	10.06
Criticality attained	12.40
Steamline Isolation (Loops 2, 3, and 4)	13.06
Pressurizer empties	13.20
SI flow starts	29.06
Boron from SI reaches the core	39.80
Peak heat flux attained	179.2
Core becomes subcritical	180.0



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FIGURE 14.2.5-1
Variation of Reactivity With Core Temperature At
1050 psia For The End Of Life Rodded Core With
One Control Rod Assembly Stuck (Zero Power)
For The Steamline Break Double Ended Rupture Event

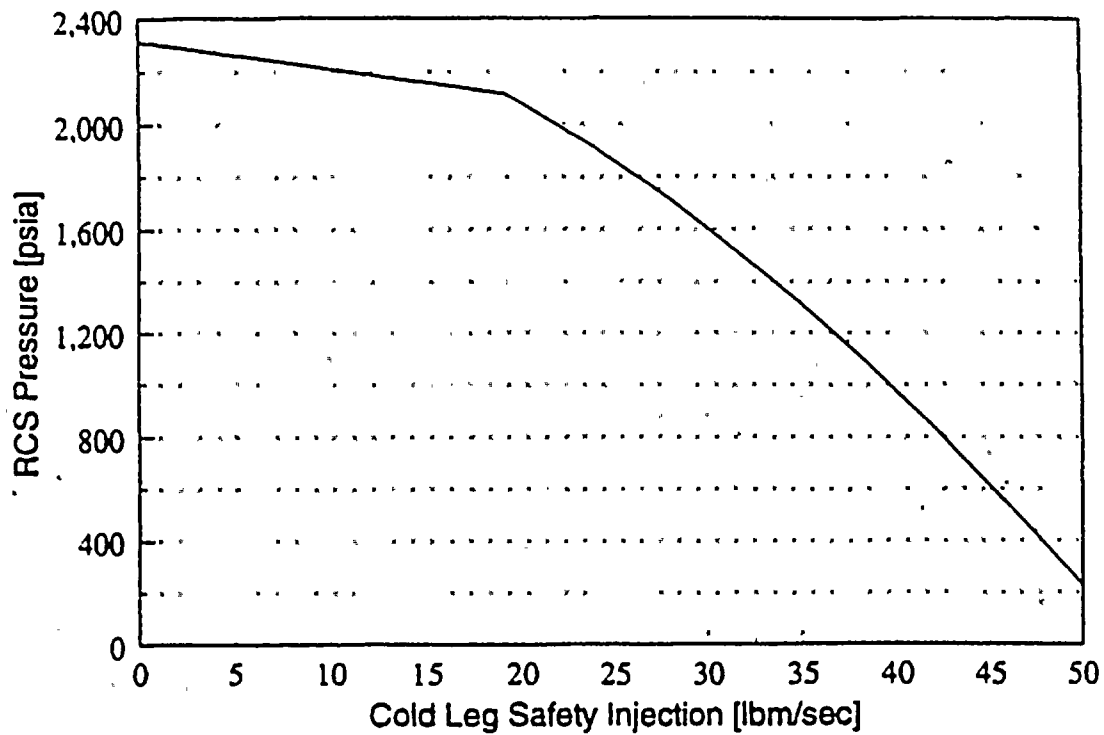
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FIGURE 14.2.5-2
Doppler Power Feedback
For The Steamline Break Double Ended Rupture Event

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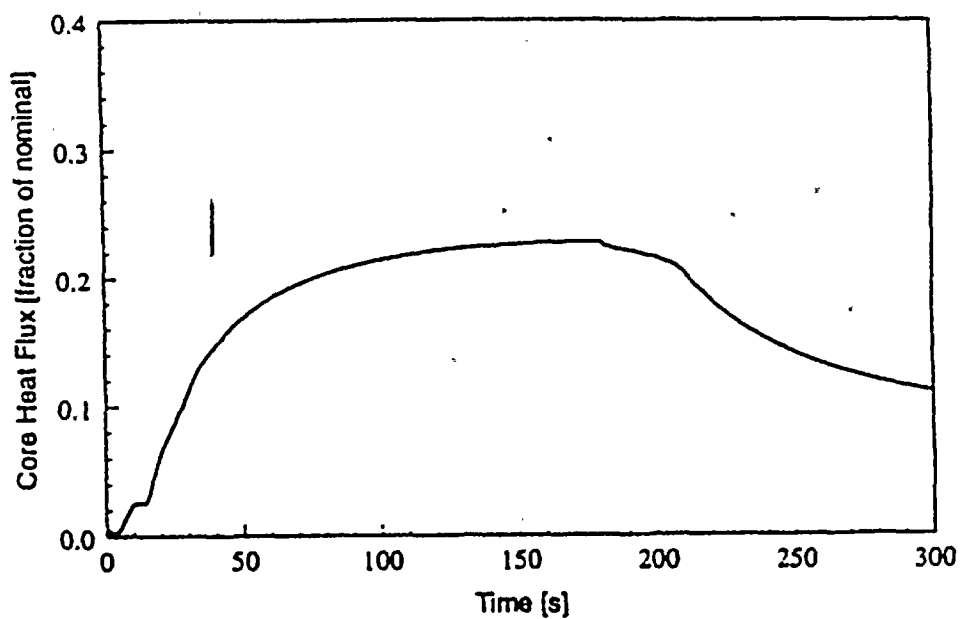
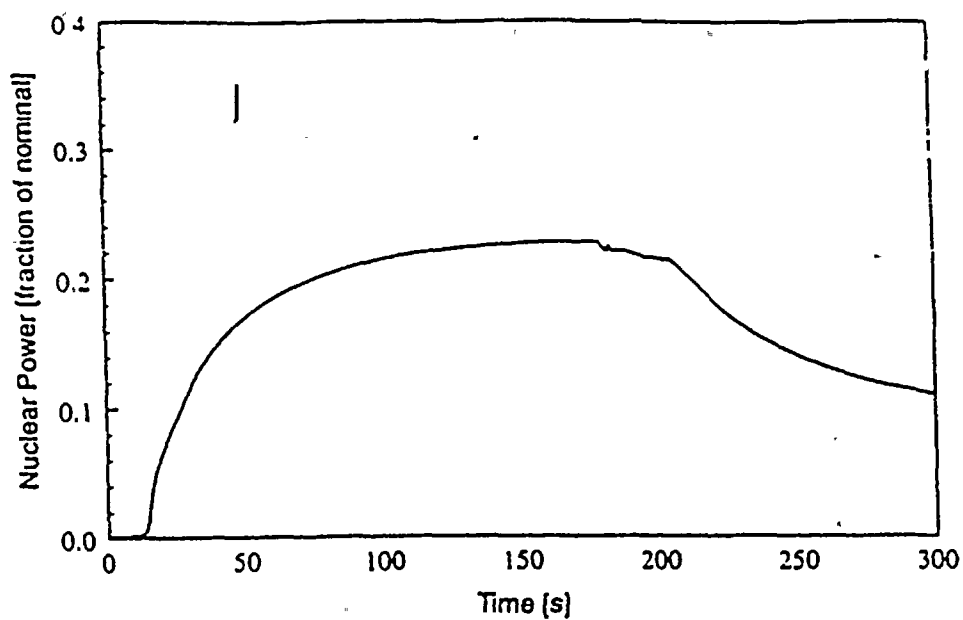


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FIGURE 14.2.5-3

**Safety Injection Flow Supplied By One Charging Pump
For The Steamline Break Double Ended Rupture Event**

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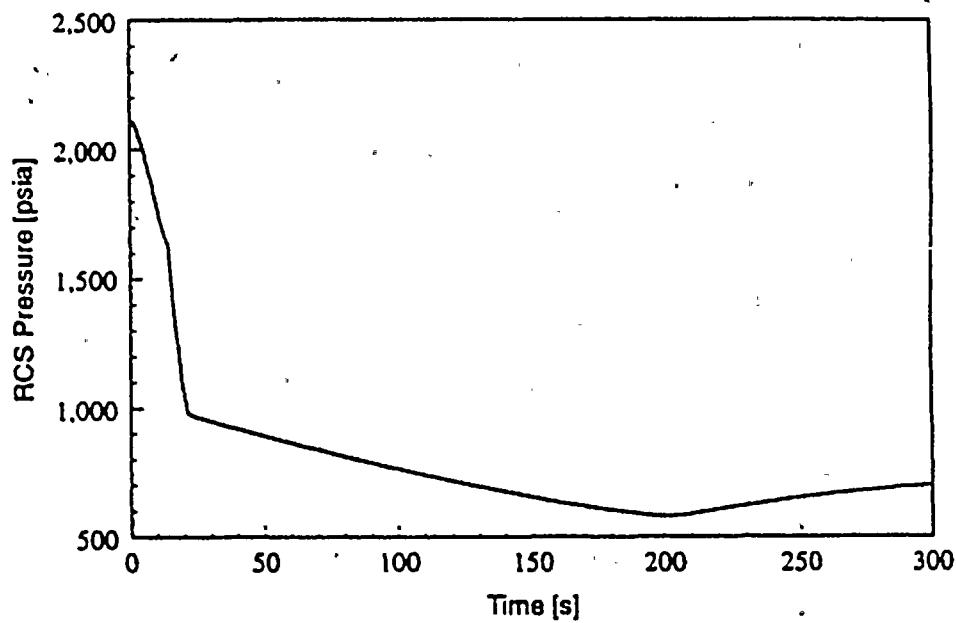
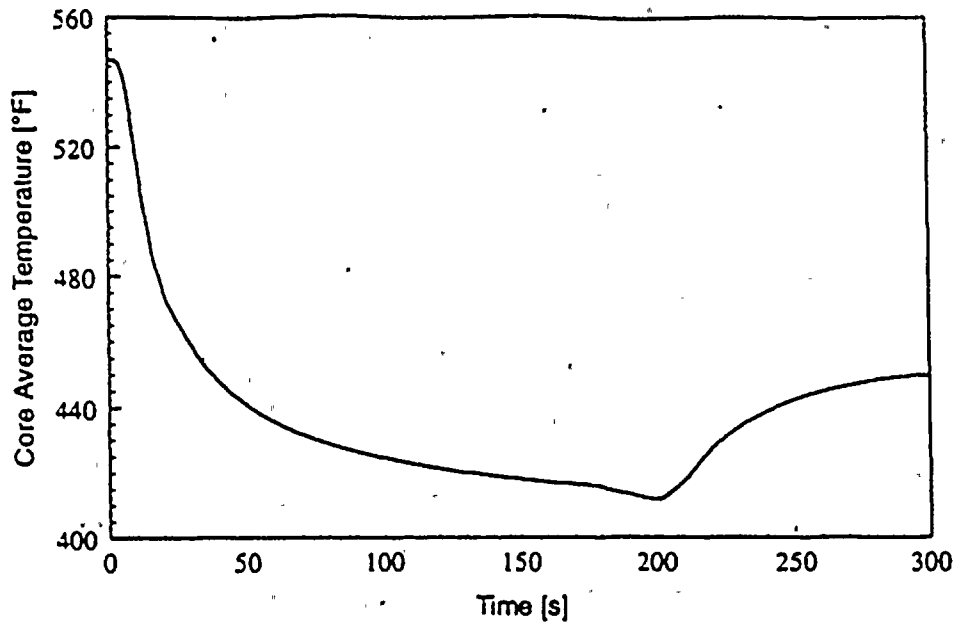


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FIGURE 14.2.5-4

**Nuclear Power and Core Heat Flux vs. Time For
The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]**

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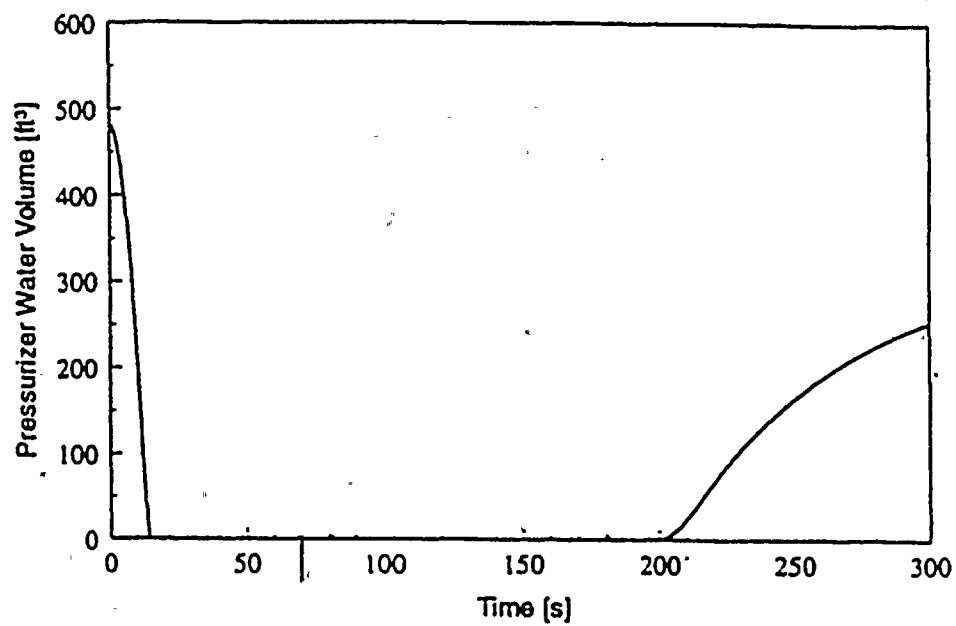


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FIGURE 14.2.5-5

**Core Average Temperature and RCS Pressure vs. Time
For The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]**

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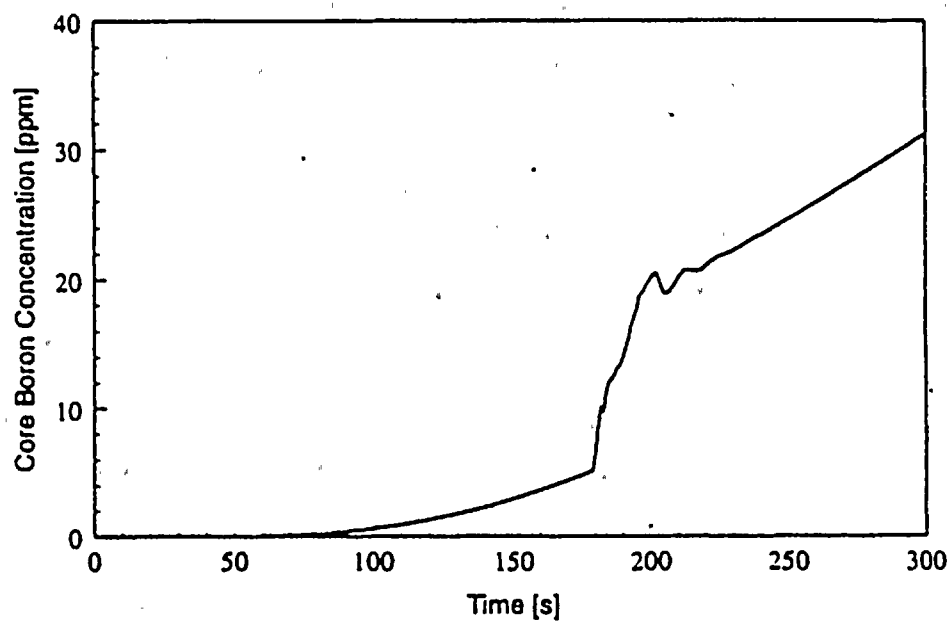
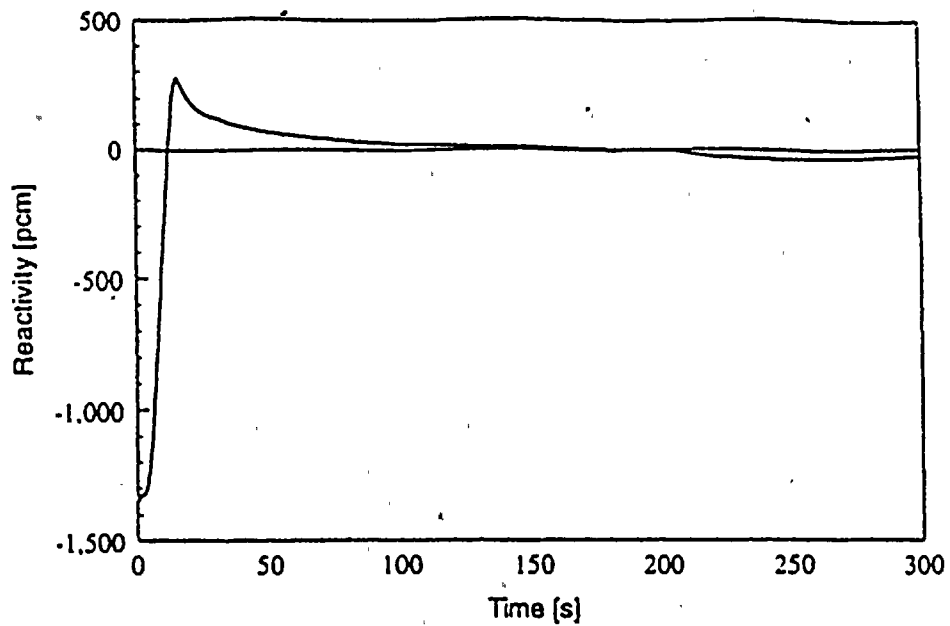


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FIGURE 14.2.5-6

Pressurizer Water Volume vs. Time
For The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]

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FIGURE 14.2.5-7

**Reactivity and Core Boron Concentration vs. Time
For The Steamline Break Double Ended Rupture Event
[Inside Containment With Power]**

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In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves.

The limiting criteria is described in Reference 4 and summarized below:

- A. Average fuel pellet enthalpy at hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.
- B. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (3000°F).
- C. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.
- D. Fuel melting will be limited to less than ten percent (10%) of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion A above.

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average channel core calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed investigation using this method, and a demonstration of the conservativeness of the calculation compared to three-dimensional spatial kinetics, is presented in WCAP-7588.⁽⁴⁾

Average Core Analysis

The spatial kinetics computer code, TWINKLE, is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial effects code since it allows a more realistic representation of the spatial kinetics of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 14.1.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the RCCA. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the fuel assembly with the ejected RCCA, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative radial power distribution is used within the fuel rod.

significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% $\Delta k/k$ due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow (supplied from the RWST) starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

Results

Table 14.2.6-1 summarizes the results. Cases are presented for both beginning and end of life at zero and full power.

A. Beginning of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The worst ejected RCCA worth and hot channel factor were conservatively calculated to be 0.15% $\Delta k/k$ and 6.8 respectively. The peak clad average temperature was 2299°F. The peak spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

B. Beginning of Cycle, Zero Power

For this condition, Control Bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in Control Bank D and has a worth of 0.65% $\Delta k/k$ and a hot channel factor of 12.0. The peak clad average temperature reached 2130°F, the fuel center temperature was 3120°F.

C. End of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.19% $\Delta k/k$ and 7.1 respectively. This resulted in a peak clad average temperature of 2245°F. The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10% of the pellet.

D. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case was obtained assuming Control Bank D to be fully inserted and banks B and C at their insertion limits. The results were 0.75% $\Delta k/k$ and 19.0 respectively. The peak clad average and fuel center temperatures were 2322°F and 3258°F. The Doppler weighting factor for this case is significantly higher than for other cases due to the very large transient hot channel factor.

For all the cases analyzed, average fuel pellet enthalpy at the hot spot remains below 200 cal/gm.

The nuclear power and hot spot fuel and clad temperature transients for two cases (end of life zero power and end of life full power) are presented in Figures 14.2.6-1 through 14.2.6-4.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents (LOCA) are discussed in Section 14.3.1 and 14.3.2. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

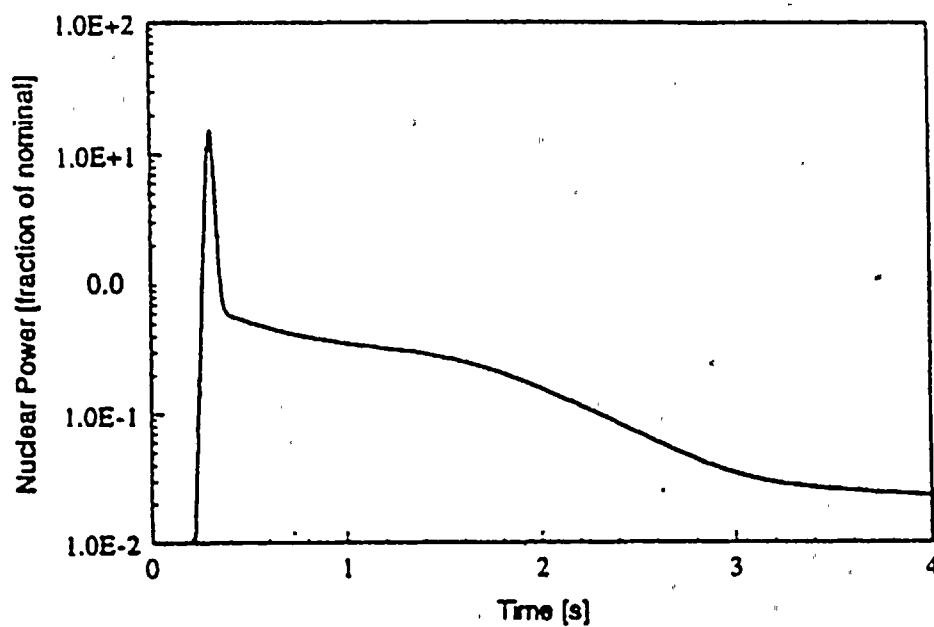
REFERENCES, SECTION 14.2.6

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2. Taxelius, T. G., ed. "Annual Report - Spert Project, October 1968 September 1969," Idaho Nuclear Corporation TID-4500, June 1970.
3. Liimatainen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO_2 -Core Simulated Fuel Elements," Argonne National Laboratory Chemical Engineering Division Semi-Annual Report, ANL-7225, January-June 1966.
4. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse PWR's Using Spatial Kinetics Methods," WCAP-7588, Revision 1A.
5. Bishop, A. A., Sandberg, R. O., and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

TABLE 14.2.6-1

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL
ASSEMBLY EJECTION ACCIDENT

<u>Time in Life</u>	<u>HZP</u> <u>Beginning</u>	<u>HFP</u> <u>Beginning</u>	<u>HZP</u> <u>End</u>	<u>HFP</u> <u>End</u>
Power Level (%)	0	102	0	102
Ejected Rod Worth (%Δk)	0.65	0.15	0.75	0.19
Delayed Neutron Fraction (%)	0.0050	0.0050	0.0040	0.0040
Feedback Reactivity Weighting	2.071	1.30	2.755	1.30
Trip Reactivity (%Δk)	2.	4.	2.	4.
F _q Before Rod Ejection	2.50	2.50	2.50	2.50
F _q After Rod Ejection	12.	6.8	19.	7.1
Number of Operational Pumps	2.	4.	2.	4.
Maximum Fuel Pellet Average Temperature (°F)	2764	4056	2963	3969
Maximum Fuel Center Temperature (°F)	3120	4968	3258	4872
Maximum Clad Average Temperature (°F)	2130	2299	2322	2245
Maximum Fuel Stored Energy (cal/gm)	112.7	177.3	122.2	172.7
Fuel Melt in Hot Pellet, %	0	<10	0	<10

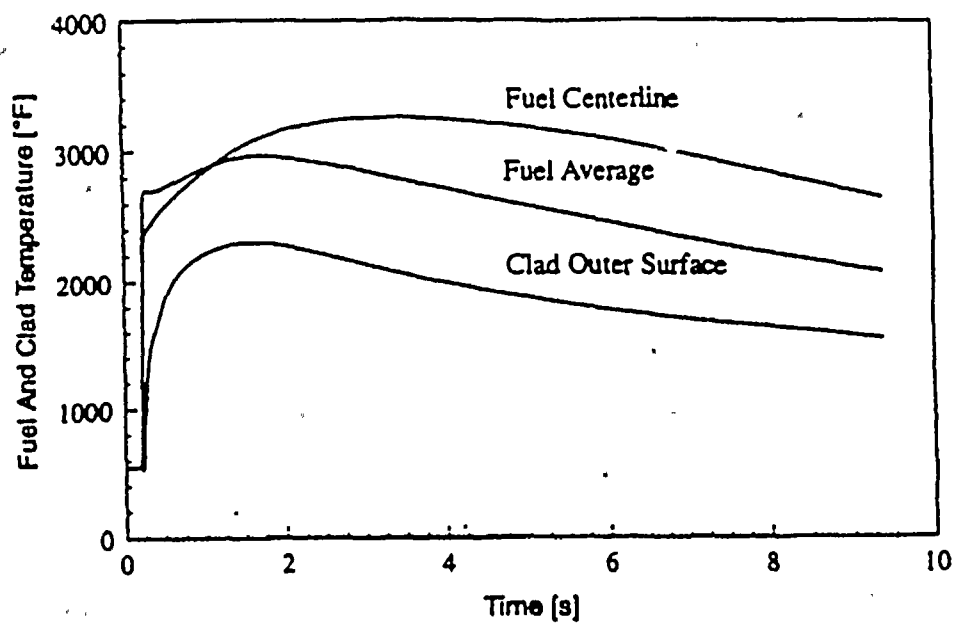


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FIGURE 14.2.6-1

Nuclear Power vs. Time For The Rod Ejection Event,
Hot Zero Power, End Of Life

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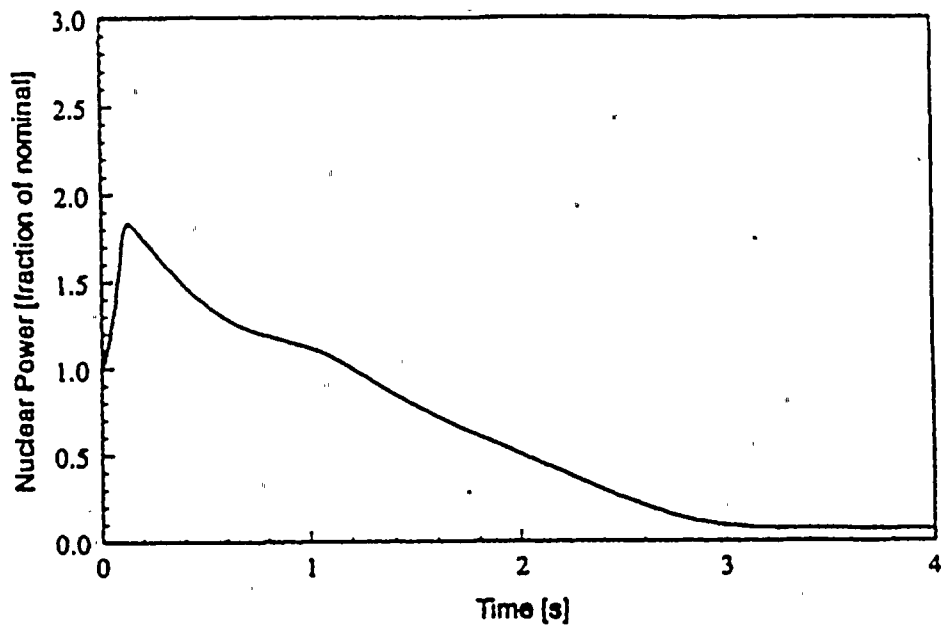


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FIGURE 14.2.6-2

**Fuel Centerline, Fuel Average, and
Clad Outer Surface Temperature vs. Time For The Rod
Ejection Event, Hot Zero Power, End Of Life**

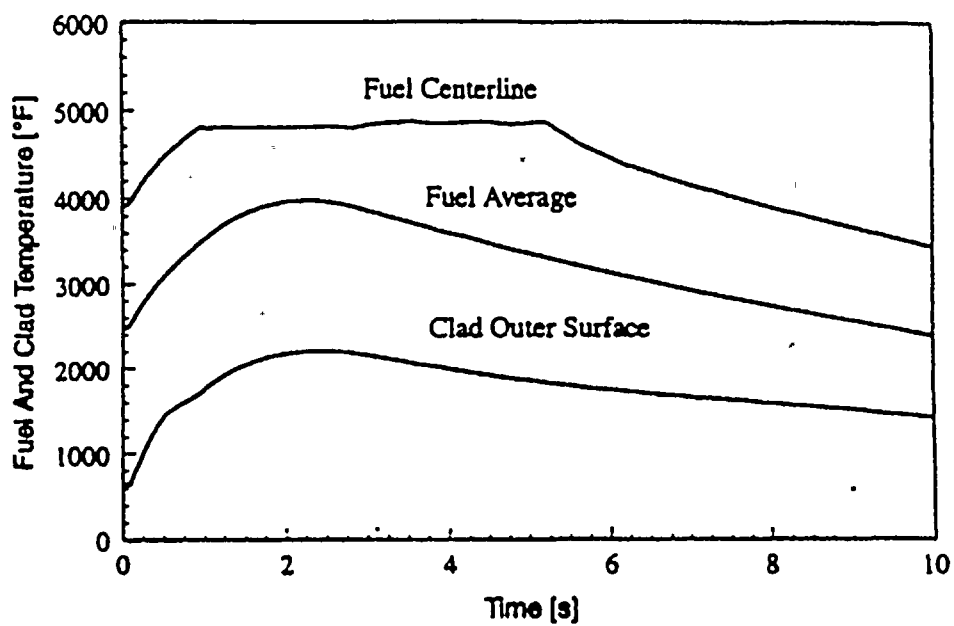
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FIGURE 14.2.6-3
Nuclear Power vs. Time For The Rod Ejection Event.
Hot Full Power, End Of Life

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FIGURE 14.2.6-4

**Fuel Centerline, Fuel Average, and
Clad Outer Surface Temperature vs. Time For The Rod
Ejection Event, Hot Full Power, End Of Life**

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TABLE 14.2.7-3

STEAM GENERATOR TUBE RUPTURESTEAM RELEASE

Steam release from defective steam generator	56,525 lbs (0-30 min)	
Steam release from 3 non-defective steam generators	413,000 lbs (0-2 hr) 978,000 lbs (2-8 hr)	
Feedwater flow to 3 non-defective steam generators	613,000 lbs (0-2 hr) 1,074,000 lbs (2-8 hr)	
Reactor coolant released to the defective steam generator	140,264 lbs (0-30 min)	

This event is not part of the Unit 1 license basis. Formerly, an informational purpose only analysis summary of this event had been included in Unit 1 Section 14.2.8.

As documented in references 1 and 2, an evaluation was performed which concluded that the results presented in Unit 2 Section 14.2.8 of the UFSAR bound Unit 1. This evaluation is based on the changes to the Unit 1 steamline break protection logic which made it identical to that installed in Unit 2. This conclusion is recognized in reference 3.

References

1. WCAP-14285, Donald C. Cook Nuclear Plant Unit 1, Steam Generator Tube Plugging Program Licensing Report, May 1995.
2. Letter AEP:NRC:1207, Donald C. Cook Nuclear Plant Units 1 and 2 License Nos. DPR-58 and DPR-74 Proposed Technical Specification Changes Supported by Analyses to Increase Unit 1 Steam Generator Tube Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Related Analyses, E. E. Fitzpatrick to USNRC Document Desk, May 26, 1995.
3. Amendment No. 214 to Facility Operating License No. DPR-58, March 13, 1997.

Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft². This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of Cook Nuclear Plant Unit 1, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50, 1974)⁽¹⁾ as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.
3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.
4. The core remains amenable to cooling during and after the break.
5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975)⁽²⁾ presents a study in regards to the probability of occurrence of RCS pipe ruptures.

Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.
2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the present Westinghouse design, the large break single failure is the loss of one RHR (low head) pump. This means that credit could be taken for two high head charging pumps, two safety injection pumps, and one low head pump. The following is a discussion of the modeling procedure for the minimum safeguards and the flow spilling from a break of an ECCS branch injection line (i.e., the spilling line assumptions).

The current procedure for large break analyses assumes that at least one train of ECCS is available for delivery of water to the RCS. Although the single failure is an RHR pump, only one pump in each subsystem is assumed to deliver to the primary loops. However, both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment deck fans and sprays. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 and is conservative for the large break LOCA. The high head charging pump starts and delivers flow through the injection lines (one per loop) with one branch injection line spilling to the containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. When one safety injection pump and one low head

residual heat removal pump start, flow is delivered to the reactor coolant system through the accumulator injection lines. Again, one line, with the minimum resistance, is assumed to spill to containment backpressure. In addition, the safety injection pump and low head residual heat removal pump performance curves were degraded by 15%. For the high head charging pumps, the performance curves were degraded by 10% and a 25 gpm flow imbalance was assumed.

Therefore, in the large break ECCS analysis performed by Westinghouse, single failure is conservatively accounted for via the loss of an ECCS train, and the spilling of the minimum resistance injection line despite full containment active heat removal system operation (i.e., two EDGs).

The time sequence of events following a large break LOCA is presented in Table 14.3.1-1.

Before the break occurs, the unit is in an equilibrium condition; that is, the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50⁽¹⁾. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves, and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. Since loss of offsite power (LOOP) is assumed, the RCPs are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (values with uncertainty assumed to be 2317 psia or 2033 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the emergency core cooling water bypassing the core are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with the dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold recirculation phase of operation in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the low head safety injection (residual heat removal) pumps and

returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 12 hours after the initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel. Long-term cooling includes long-term criticality control. Criticality control is achieved by determining the RWST and accumulator concentration necessary to maintain subcriticality without credit for RCCA insertion. The necessary RWST and accumulator concentration is a function of each core design and is checked each cycle. The current Technical Specification value is 2400 ppm to 2600 ppm boron.⁽³⁾

An evaluation has been performed to determine the effect of a 3 minute SI interruption during the switchover to sump recirculation on the LBLOCA analysis. This scenario could occur if the RHR pump which was first switched over to recirculation fails at the time the other RHR pump is secured for switchover. Using a conservatively short estimate of the RWST draindown time and a bounding scenario for the availability of pumped injection, it was shown (Reference 21) that the short-term peak clad temperature results are not challenged by a three minute interruption of all ECCS flow.

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974)⁽¹⁾.

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the

core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974)⁽⁴⁾. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)⁽⁵⁾; Kelly et al. (1974)⁽⁶⁾; Young et al. (1987)⁽⁷⁾; and Bordelon et al. (1974)⁽⁴⁾. Code modifications are specified in References 8, 9, 10 and 11. It is noted that the WREFLOOD code, which was previously used to calculate the RCS behavior during vessel lower plenum refill, has been replaced by the REFILL code as reported in Reference 18. The REFILL code is identical to the section of the WREFLOOD code that modeled the refill phase.

These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling throughout and subsequent to the blowdown,

refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the REFILL computer code calculates this transient during the refill phase of the accident. The BASH code is used to determine the system response during the reflood phase of the transient. The LOTIC computer code, described by Hsieh and Raymund in WCAP-8355 (1975) and WCAP-8345 (1974)⁽¹²⁾, calculates the containment pressure transient.

The containment pressure transient is input to BASH for the purpose of calculating the reflood transient. The LOCBART computer code calculates the thermal transient of the hottest fuel rod in the three phases. The Revised PAD Fuel Thermal Safety Model, described in References 13, generates the initial fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, information on the state of the system is transferred to the REFILL code which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriated to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH⁽⁷⁾ computer code.

Input Parameters and Initial Conditions:

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature and a uniform steam generator tube plugging level of 30%. The analysis is also based on plant operation with the RHR cross-tie valves closed, and a diesel generator start time of 30 seconds which results in a safety injectin delay time of 47 seconds. A list of plant input parameters used in the large break LOCA analysis is provided in Table 14.3.1-2.

A range of reactor operating temperatures were analyzed in order to justify plant operation at a reactor power level of 3250 MWt between 609.1°F to 586.8°F in the hot legs and 543.5°F and 519.2°F in the cold legs. In addition to the temperature range analyzed, initial RCS pressure was also varied to justify plant operation at 2250 and 2100 psia. A full spectrum break analysis was done at the nominal RCS conditions (initial RCS pressure of 2250 psia and initial hot leg temperature of 609.1°F) from which the limiting break size was determined. The limiting break was then reanalyzed at the reduced hot leg temperature of 586.8°F and nominal RCS pressure of 2250 psia. The limiting break was also reanalyzed at the nominal hot leg temperature of 609.1°F and RCS pressure of 2100 psia. Table 14.3.1-1 identifies the cases analyzed.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974⁽¹⁴⁾; Salvatori 1974⁽¹⁵⁾; Johnson, Massie, and Thompson 1975⁽¹⁶⁾). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

Another input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. Large break LOCA analyses have been traditionally performed using a symmetric, chopped cosine axial power shape. Recent calculations have shown that there was a potential for top-skewed power distributions to result in peak cladding temperatures (PCT) greater than those calculated with a chopped cosine axial power distribution. Westinghouse previously developed a process, called the power shape sensitivity model (PSSM), that reasonably ensured that the cosine remains the limiting power distribution by defining appropriate power distribution surveillance data. The PSSM process was applied for the Cycle 13 and 14 reloads for Cook Nuclear Plant Unit 1. However, PSSM was subsequently replaced by an alternate axial power shape methodology designated ESHAPE, which is based on explicit analysis of a set of skewed axial power shapes. The explicit use of skewed power shapes has previously been approved by the NRC as part of the Westinghouse Large Break LOCA Evaluation Model. The ESHAPE methodology was utilized for the Cycle 15 reload, and has also been applied for this analysis for Cycle 16. The application of the ESHAPE methodology demonstrated that the cosine axial power shape used for the current large break LOCA analysis is more limiting than potential top-skewed power shapes. The ESHAPE methodology has been implemented in the reload design process to ensure that top-skewed axial power distributions that are potentially more limiting than the power distribution used in the ECCS analysis are precluded for future cycles.

A meeting was held at the Westinghouse Licensing Office in Bethesda on December 17, 1981, between members of the U. S. Nuclear Regulatory Commission and members of the Westinghouse Nuclear Safety Department to discuss the impact of maximum safety injection on the large break ECCS analysis on a generic basis. Further discussion of this issue is provided in a letter from E. P. Rahe, Manager of Westinghouse Nuclear Safety Department, to Robert L. Tedesco of the U. S. Nuclear Regulatory Commission⁽¹⁷⁾. A brief description of this issue is given below.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one residual heat removal (RHR) pump as the most limiting single failure. This is conservatively modeled as a loss of one train of

safety injection, including RHR pump, safety injection pump and centrifugal charging pump. Both containment spray pumps are assumed operable. This is the limiting single failure assumption when offsite power is unavailable for most Westinghouse plants. However, for some Westinghouse plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assume minimum injection line resistances, enhanced ECCS pump performance, and no single failure, result in the highest amount of flow delivered to the RCS.

The worst break for Cook Unit 1 (CASE F) was reanalyzed, assuming maximum safeguards. The results of the large break LOCA analyses are given in Table 14.3.1-1.

Results:

Based on the results of the LOCA sensitivity studies (Westinghouse 1974⁽¹⁴⁾; Salvatori 1974⁽¹⁵⁾; Johnson, Massie, and Thompson 1975⁽¹⁶⁾) the limiting large break was found to be the double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Table 14.3.1-1.

The containment data used to generate the LOTIC backpressure transient are shown in Table 14.3.1-3. The mass and energy release data used for the limiting minimum safeguards case are shown in Table 14.3.1-4. Nitrogen release rates to the containment are given in Table 14.3.1-5.

Figures 14.3.1-1a through 14.3.1-19 present the results of the cases analyzed for the large break LOCA. The alpha designation in the figure number corresponds to the cases as described in Table 14.3.1-1.

Figures 14.3.1.1a-f The system pressure shown is the calculated core pressure.

Figures 14.3.1.2a-f The flow rate from the break is plotted as the sum of both ends of the guillotine break.

Figures 14.3.1.3a-f The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.

Figures 14.3.1.4a-f The core flow is shown during the blowdown phase of the transient.

Figures 14.3.1.5a-f The accumulator flow during blowdown is plotted as the sum of that injected into the intact cold legs.

Figures 14.3.1.6a-f The core and downcomer collapsed liquid water level, and the core quench front are plotted during the reflood phase of the transient.

Figures 14.3.1.7a-f The core inlet flow is shown as it is calculated during the reflood phase.

Figures 14.3.1.8a-f The total accumulator and pumped ECCS flow injected into the intact cold legs during reflood is shown.

Figures 14.3.1.9a-f The integral of the core inlet flow as calculated with BASH is plotted.

Figures 14.3.1.10a-f The mass flux is plotted at the hot spot (the node which produced the peak clad temperature) on the hot rod.

Figures 14.3.1.11a-f The heat transfer coefficient is plotted at the hot spot on the hot rod.

Figures 14.3.1.12a-f The vapor temperature at the hot spot on the hot rod is plotted.

Figures 14.3.1.13a-f The clad temperature at the hot spot is shown for the hot rod.

Figure 14.3.1.14 The containment pressure transient used in the analysis is provided for the minimum SI case.

Figures 14.3.1.15-18 These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray.

Figure 14.3.1.19 This figure shows the temperature transients in both the lower and upper compartments of containment.

The maximum clad temperature calculated for a large break is 2164°F, which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 14.30 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction for all breaks is less than the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

10 CFR 50.46(a)(3) requires the record keeping and reporting of changes in LOCA evaluation models and of changes in the application of these models. References 19 and 20 report the following permanent changes that apply to Cook Nuclear Plant Unit 1:

1. During migration of the LOCA codes from the Cray computer to Unix-based platforms, programming errors were made in two library routines related to improper specification of double precision variables (Reference 19). This resulted in a 5°F benefit applied to the peak cladding temperature for Case E.
2. An error was discovered in the coding related to the translocation of fluid conditions between the SATAN blowdown hydraulics code and the LOCTA code used for subchannel analysis of the fuel rods (Reference 20). This resulted in a 15°F penalty applied to the peak cladding temperature for Case E.

3. An error was discovered in the LOCBART code related to improper modeling of fuel rod cladding creep and burst (Reference 20). This resulted in a 9°F benefit applied to the peak cladding temperature for Case E.

Tabulations of the assessments against peak cladding temperature due to these changes for the limiting Case E are shown in Table 14.3.1-1. In each case, the peak cladding temperature result remains less than 2200°F.

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TABLE 14.3.1-1
LARGE BREAK LOCA
RESULTS

	Case A C _D =0.4 T _{HOT} =609.1°F P=2250 psia Min. SI	Case B C _D =0.6 T _{HOT} =609.1°F P=2250 psia Min. SI	Case C C _D =0.8 T _{HOT} =609.1°F P=2250 psia Min. SI	Case D C _D =0.4 T _{HOT} =586.8°F P=2250 psia Min. SI	Case E C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI	Case F C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI
Peak Clad Temperature (°F)	2069	1993	1965	2036	2164	2149
Computed in Analysis	2069	1993	1965	2036	2164	2149
Model Assessments						
Salibrary Double Precision Errors					-5.0	
Translation of Fluid Conditions from SATAN					+15.0	
LOCBART Clad Creep and Burst Error					-9.0	
Current Licensing Basis					2165*	
Peak Clad Location (ft)	5.75	6.25	6.25	6.00	6.25	6.25
Local Zr/H ₂ O Reaction (Max %)	7.59	8.19	6.62	8.45	14.30	12.01
Local Zr/H ₂ O Location (ft)	5.75	6.00	6.00	6.00	6.25	6.25

*See the LOCA evaluation log maintained by the Nuclear Safety Section for temporary margin allocations.

TABLE 14.3.1-1 (Cont.)
LARGE BREAK LOCA
RESULTS

	Case A C _D =0.4 T _{HOT} =609.1°F P=2250 psia Min. SI	Case B C _D =0.6 T _{HOT} =609.1°F P=2250 psia Min. SI	Case C C _D =0.8 T _{HOT} =609.1°F P=2250 psia Min. SI	Case D C _D =0.4 T _{HOT} =586.8°F P=2250 psia Min. SI	Case E C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI	Case F C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI
Total Zr/H ₂ O Reaction (%)	<10.	<1.0	<1.0	<1.0	<1.0	<1.0
Hot Rod Busrt Time (s)	43.6	41.8	45.7	46.5	42.0	42.0
Hot Rod Burst Location (ft)	5.75	6.00	6.00	6.00	6.25	6.25

TABLE 14.3.1-1 (Cont.)
LARGE BREAK LOCA
TIME SEQUENCE OF EVENTS

	Case A C _D =0.4 T _{HOT} =609.1°F P=2250 psia Min. SI	Case B C _D =0.6 T _{HOT} =609.1°F P=2250 psia Min. SI	Case C C _D =0.8 T _{HOT} =609.1°F P=2250 psia Min. SI	Case D C _D =0.4 T _{HOT} =586.8°F P=2250 psia Min. SI	Case E C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI	Case F C _D =0.4 T _{HOT} =609.1°F P=2100 psia Min. SI
Start	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.64	0.64	0.63	0.55	0.49	0.49
Safety Injection Signal	4.80	4.60	4.50	4.40	4.10	4.10
Accumulator Injection	18.70	13.90	11.60	17.80	18.70	18.70
End of Blowdown	40.75	31.77	28.05	40.61	39.96	39.96
Pump Injection	51.80	51.60	51.50	51.50	51.10	51.10
Bottom of Core Recovery	54.30	44.60	41.80	55.30	54.20	54.00
Accumulator Empty	69.09	62.30	48.75	70.09	68.96	69.78

TABLE 14.3.1-2

PLANT INPUT PARAMETERS USED IN LARGE BREAK LOCA ANALYSIS

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.434
Total Core Peaking Factor, F_Q	2.15
Hot Channel Enthalpy Rise Factor, F_{AH}	1.55
Maximum Assembly Average Power, P_{HA}	1.38
Fuel Assembly Array	15 X 15 OFA
Steam Generator Tube Plugging Level (%)	30
Accumulator Water Volume (ft ³ /tank)	946
Accumulator Tank Volume (ft ³ /tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	100
Refueling Water Storage Tank Temperature (°F)	70 - 105
Thermal Design Flowrate (gpm/loop)	83,200
RCS Loop Average Temperature (°F)	553.0 and 576.3
Nominal Initial RCS Pressure (psia)	2100 and 2250
Nominal Steam Pressure (psia)	595 and 749
Safety Injection Delay Time (sec)	47
RHR Pump Head Degradation (%)	15
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
RHR Cross-Tie Valve Position	Closed

TABLE 14.3.1-3

LARGE BREAK CONTAINMENT DATA
(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME

(Includes Distribution Between Upper, Lower, and Dead-Ended Compartments)	UC	746,829 ft ³
	LC	249,446 ft ³
	DE	116,168 ft ³
	IC	163,713 ft ³

Initial Conditions

Pressure 14.7 psia

Maximum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC	100°F
	LC	120°F
	DE	120°F

Minimum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC	60°F
	LC	60°F
	DE	60°F

RWST Temperature 70°F

Temperature Outside Containment -22°F

Initial Spray Temperature 70°F

Spray System

Runout Flow for a Spray Pump 3600 gpm

Number of Spray Pumps Operating 2

Post-Accident Initiation of Spray
System 36 sec

Distributuion of Spray Flow to the Upper and Lower Compartments	LC	2700 gpm
	UC	4500 gpm

Deck Fan

Post-Accident Initiation of Deck Fans 480 sec

Flow Rate per Fan 43,890 cfm per fan

Assumed Spray Efficiency of Water from Ice
Condenser Drains 100%

TABLE 14.3.1-3 (cont'd)

STRUCTURAL HEAT SINKS

<u>wall</u>	<u>compartment</u>	<u>area (ft²)</u>	<u>thickness (ft)</u>	<u>material</u>
1	LC	12,105	0.0469/2.0	steel/concrete
2	LC	11,701	2.0	concrete
3	LC	65,979	4.0	concrete
4	LC	5,462	0.0833	steel
5	LC	5,273	0.0103	steel
6	LC	290	0.25	lead
7	LC	14,896	0.0078	steel
8	LC	4,515	0.1042	steel
9	LC	5,775	0.009	steel
10	LC	57,317	0.00833	steel
11	LC	9,404	0.0313	steel
12	LC	2,623	0.0313	steel
13	UC	378	0.0365/0.1667	steel/concrete
14	UC	34,895	0.0078	steel
15	UC	8,060	0.0208	steel
16	UC	420	0.0052	steel
17	UC	29,332	2.0	concrete
18	UC	34,125	0.0469/2.0	steel/concrete
19	UC	420	0.0052	steel

UC: Upper Compartment
 LC: Lower Compartment
 DE: Dead-End Compartment
 IC: Ice Compartment

TABLE 14.3.1-4

MASS AND ENERGY RELEASE RATES, MAXIMUM SI

<u>time (sec)</u>	<u>mass (lbm/sec)</u>	<u>energy (BTU/sec)</u>
0	57910	3.081(10 ⁷)
2	48870	2.542(10 ⁷)
4	33500	1.762(10 ⁷)
6	25260	1.357(10 ⁷)
8	22660	1.223(10 ⁷)
10	19580	1.096(10 ⁷)
12	16980	9.838(10 ⁶)
12.4	16000	9.346(10 ⁶)
14	14530	8.608(10 ⁶)
16	12140	7.313(10 ⁶)
18	10410	6.254(10 ⁶)
20	9170	5.472(10 ⁶)
24	7010	3.871(10 ⁶)
28	6750	2.839(10 ⁶)
32	5640	1.757(10 ⁶)
36	3580	7.951(10 ⁵)
40	4390	9.057(10 ⁵)
52	230	1.267(10 ⁴)
65	280	6.321(10 ⁴)
75	390	2.073(10 ⁵)
86	810	2.884(10 ⁵)
95	420	2.464(10 ⁵)
124	400	1.666(10 ⁵)
206	430	1.452(10 ⁵)
294	330	1.314(10 ⁵)

TABLE 14.3.1-5
NITROGEN MASS AND ENERGY RELEASE RATES

<u>time (sec)</u>	<u>flow rate (lbm/sec)</u>
69.2	231.8
73.2	166.4
77.2	120.8
81.2	87.3
85.2	62.1
89.2	42.9
93.2	28.8
97.2	19.5
101.2	14.1
105.2	11.1
109.2	9.0
113.2	7.3
117.2	5.9
121.2	4.8
125.2	3.9
129.2	3.2
137.2	2.1
141.2	1.8
145.2	1.4
153.2	1.0
161.2	0.7
169.2	0.5
177.2	0.3

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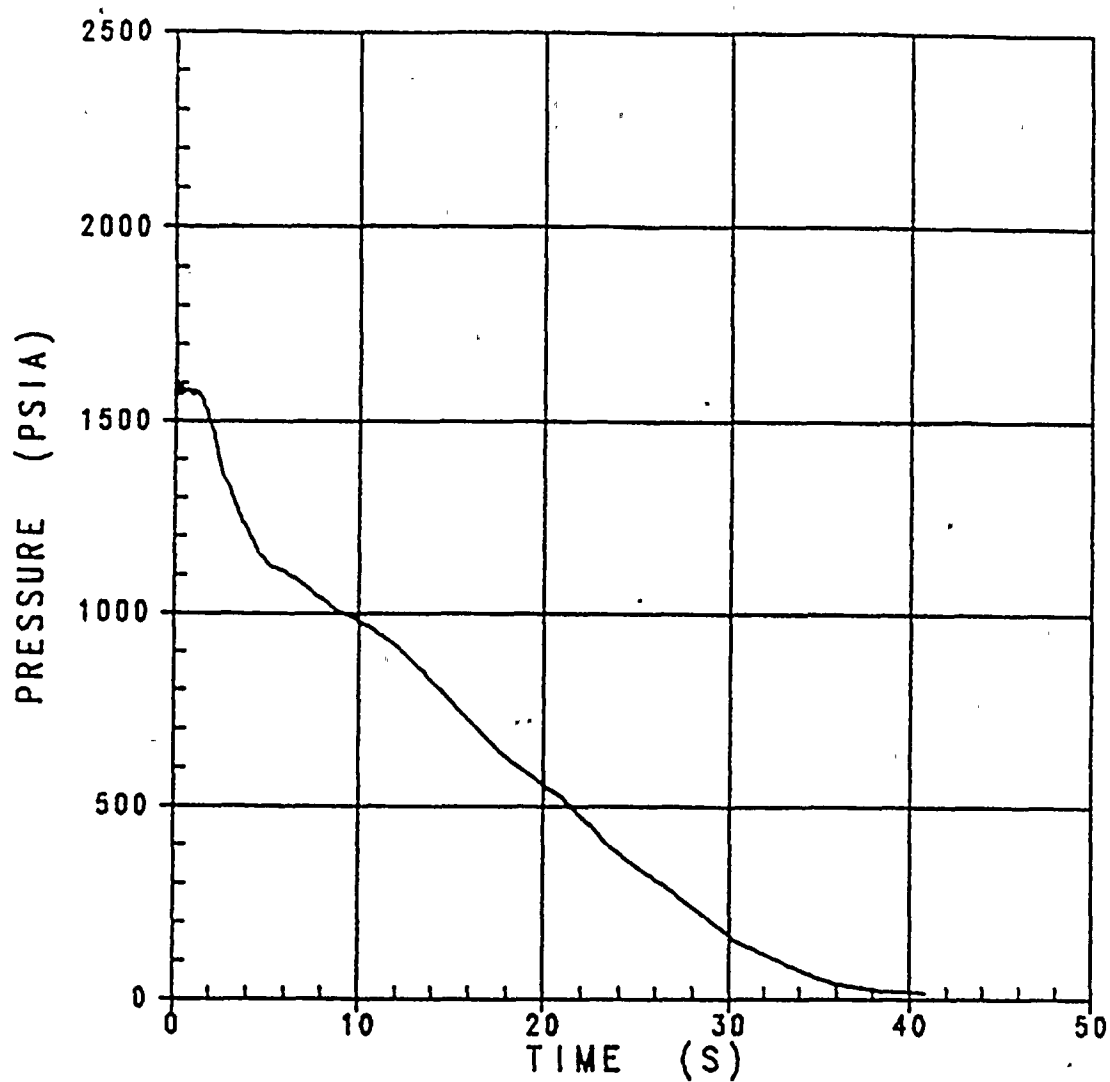


Figure 14.3.1-1a Reactor Coolant System Pressure
Case A, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

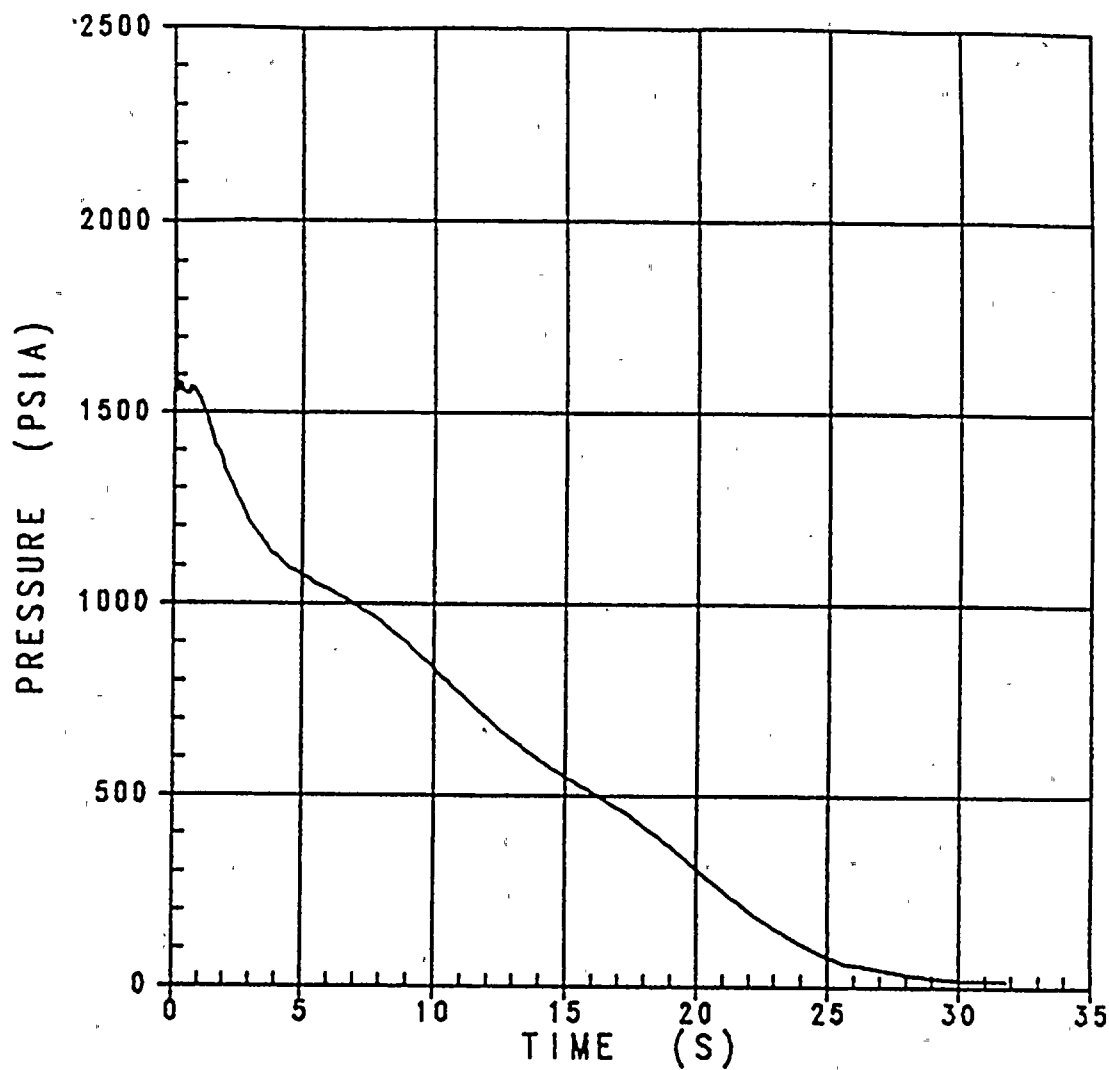


Figure 14.3.1-1b Reactor Coolant System Pressure
Case B, CD=0.6, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

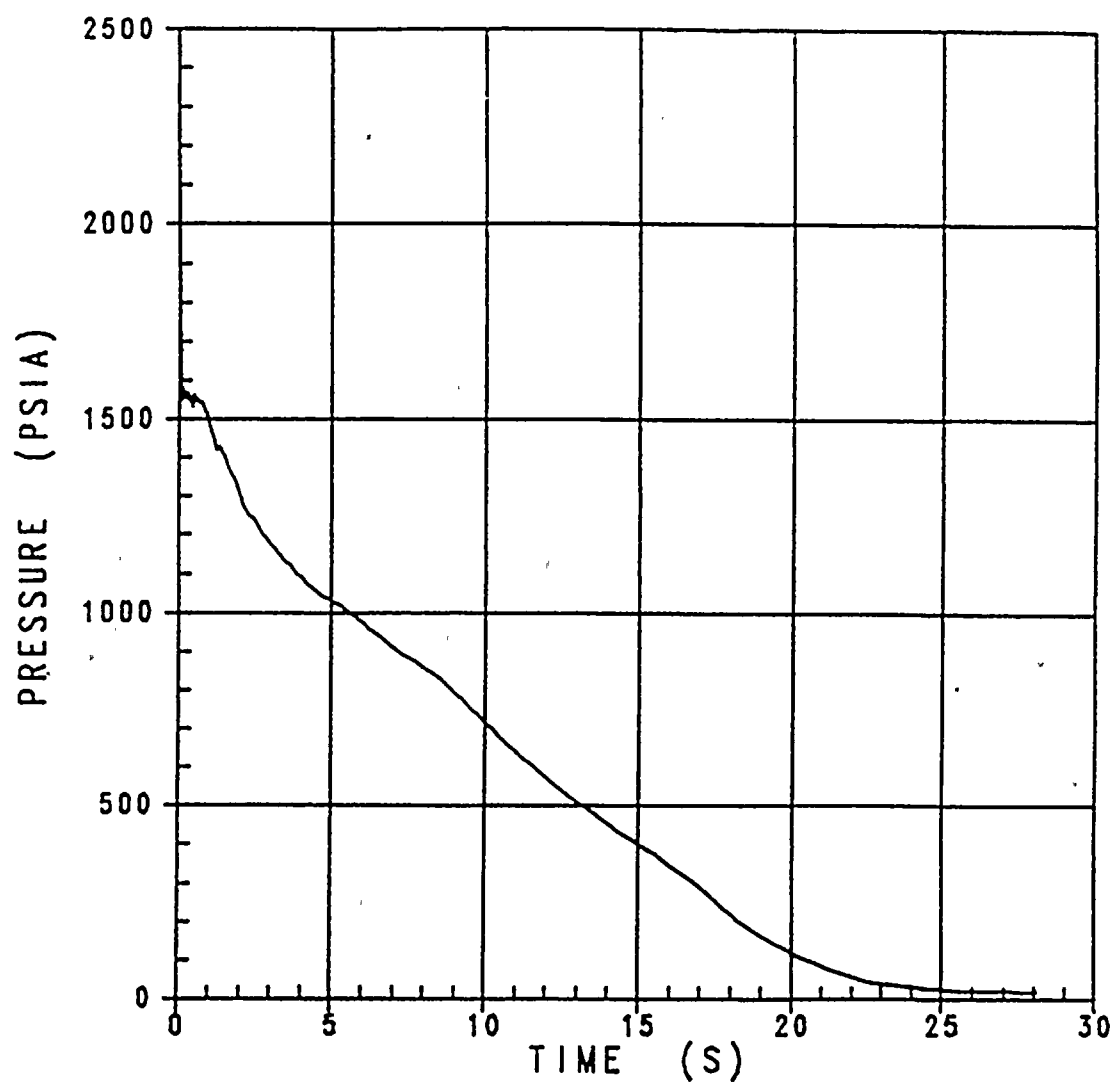


Figure 14.3.1-1c Reactor Coolant System Pressure
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

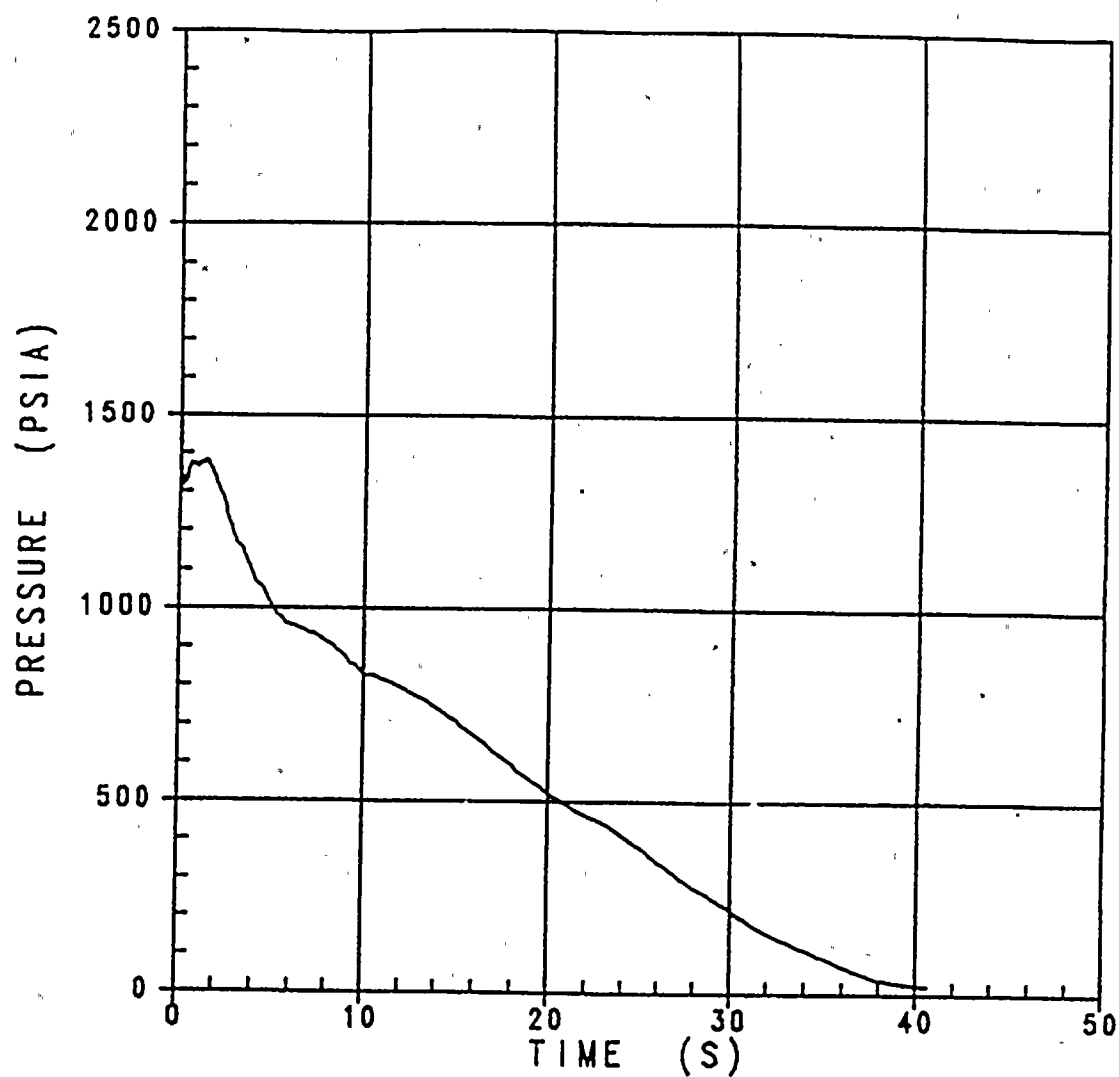


Figure 14.3.1-1d Reactor Coolant System Pressure
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

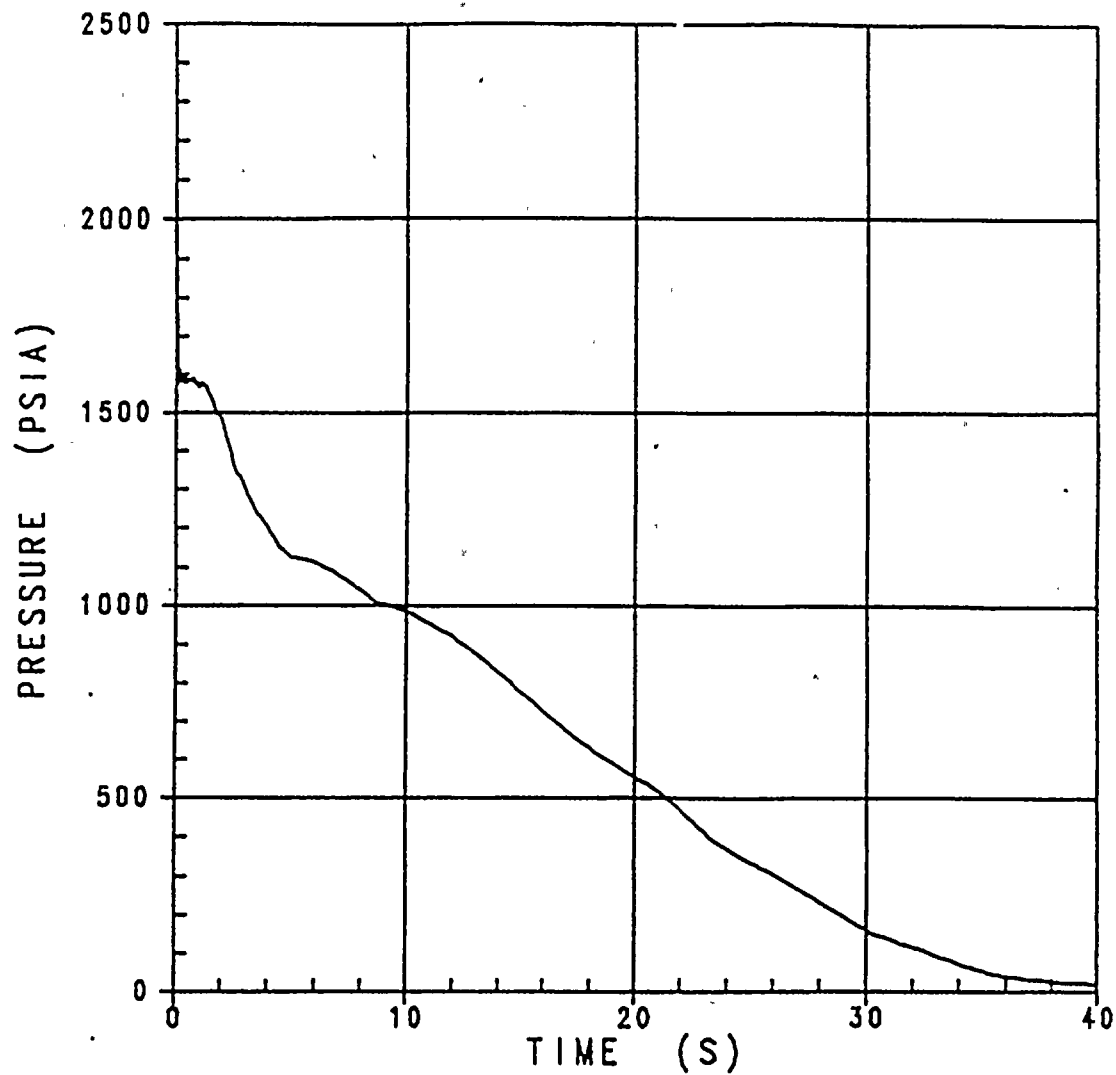


Figure 14.3.1-1e Reactor Coolant System Pressure
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

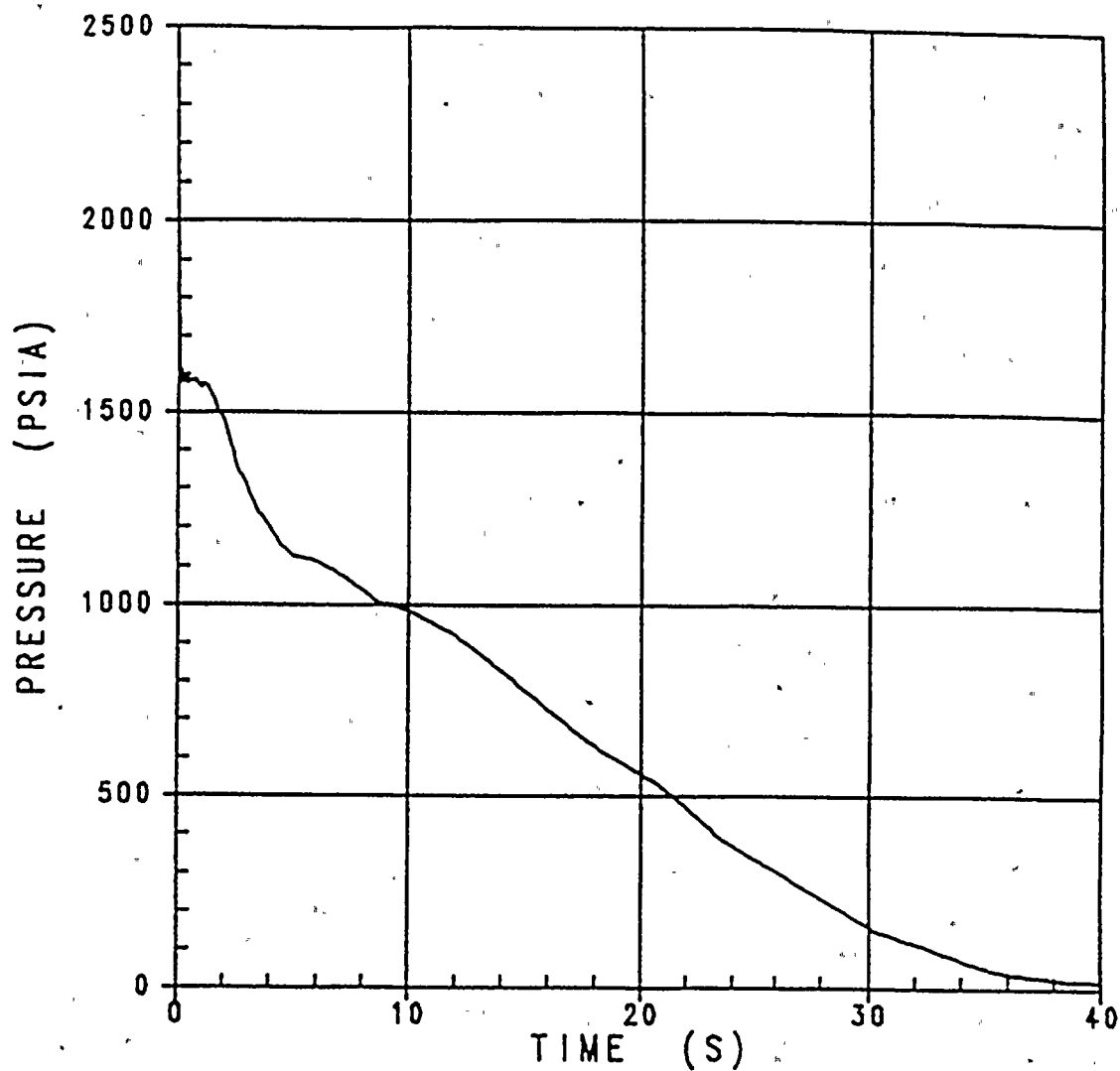


Figure 14.3.1-1f Reactor Coolant System Pressure
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

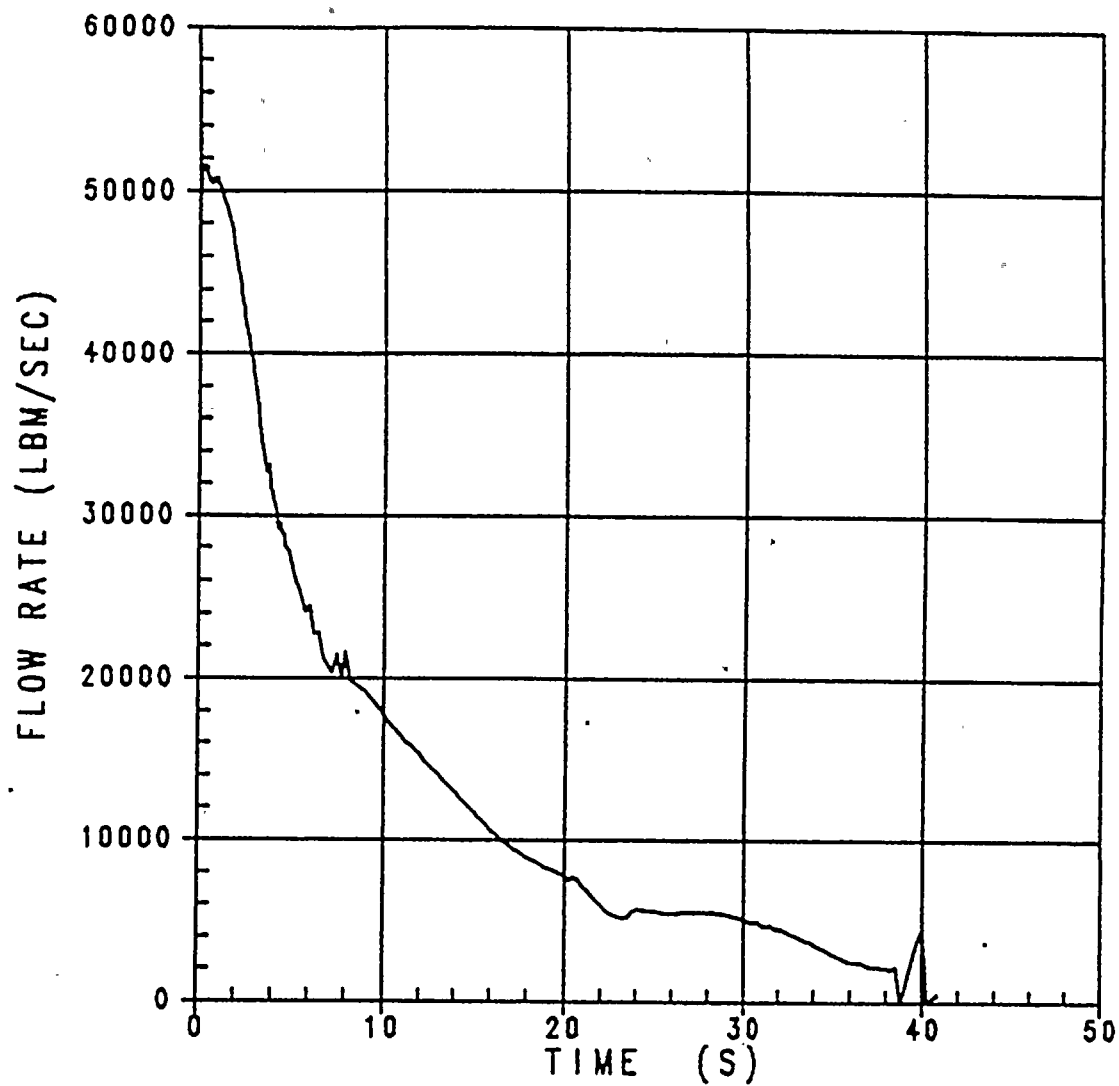


Figure 14.3.1-2a Break Flow During Blowdown
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

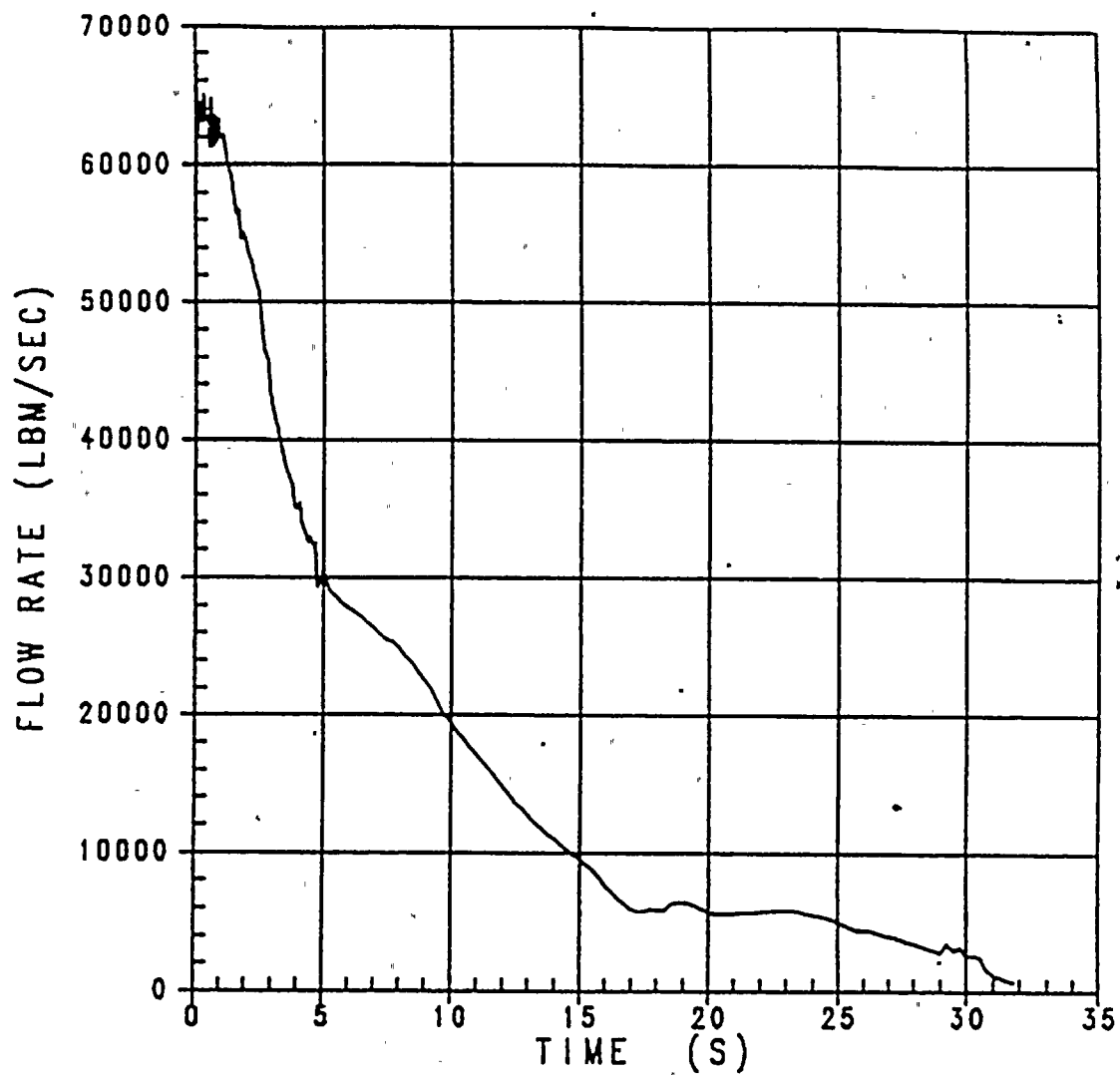


Figure 14.3.1-2b Break Flow During Blowdown
Case B, $CD=0.6$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

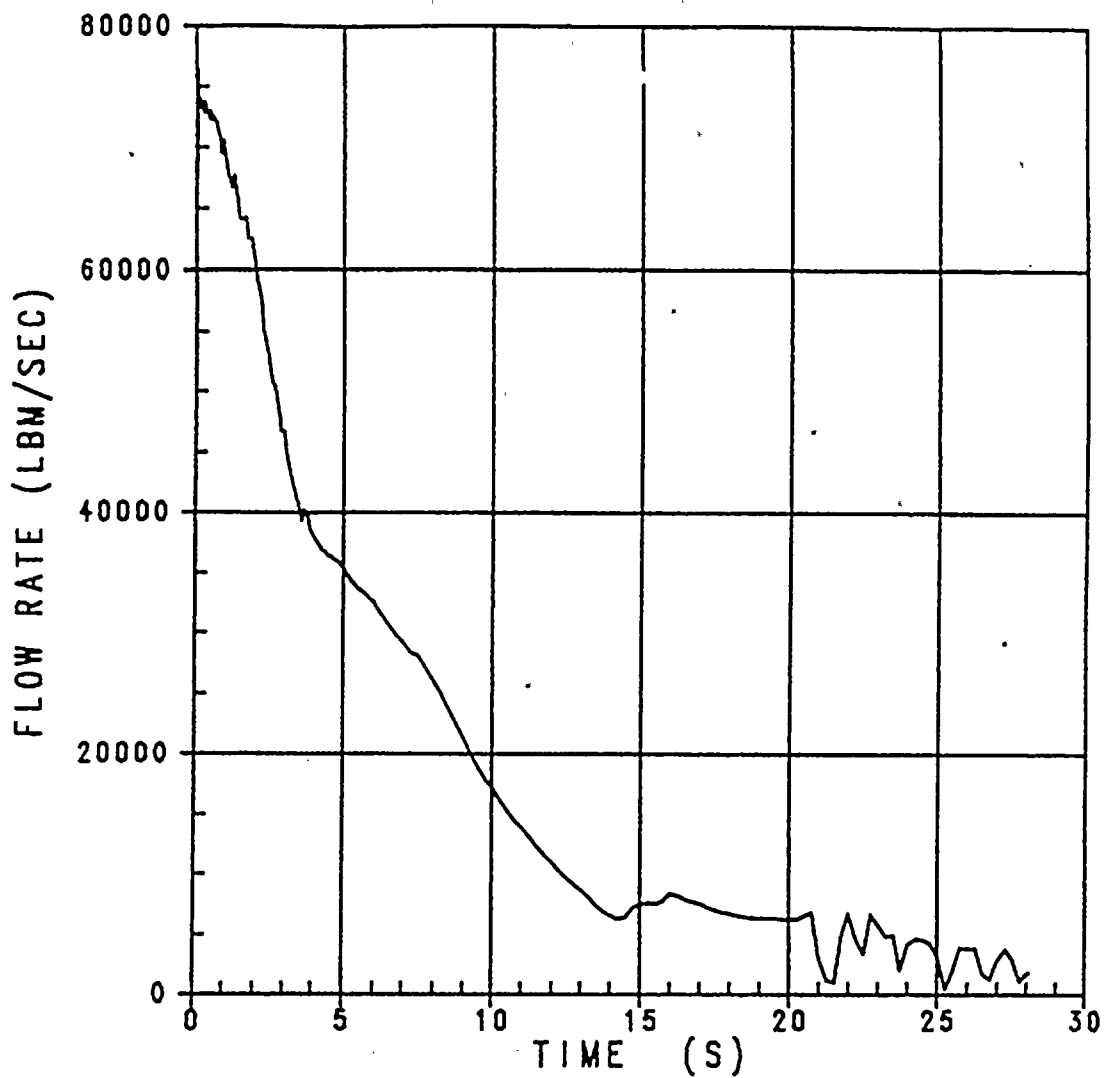


Figure 14.3.1-2c Break Flow During Blowdown
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

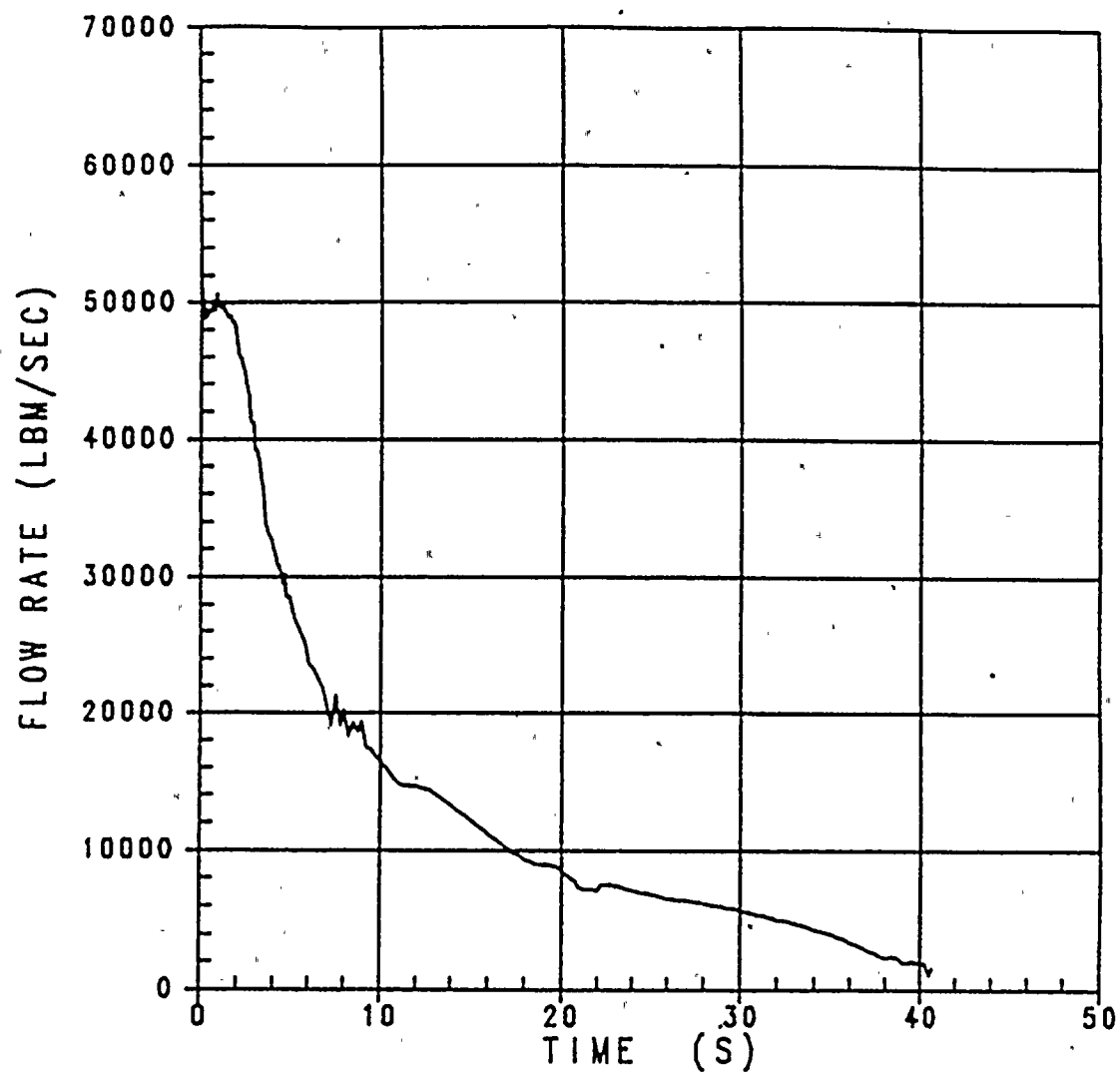


Figure 14.3.1-2d Break Flow During Blowdown
Case D, $CD=0.4$, $T_{hot}=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

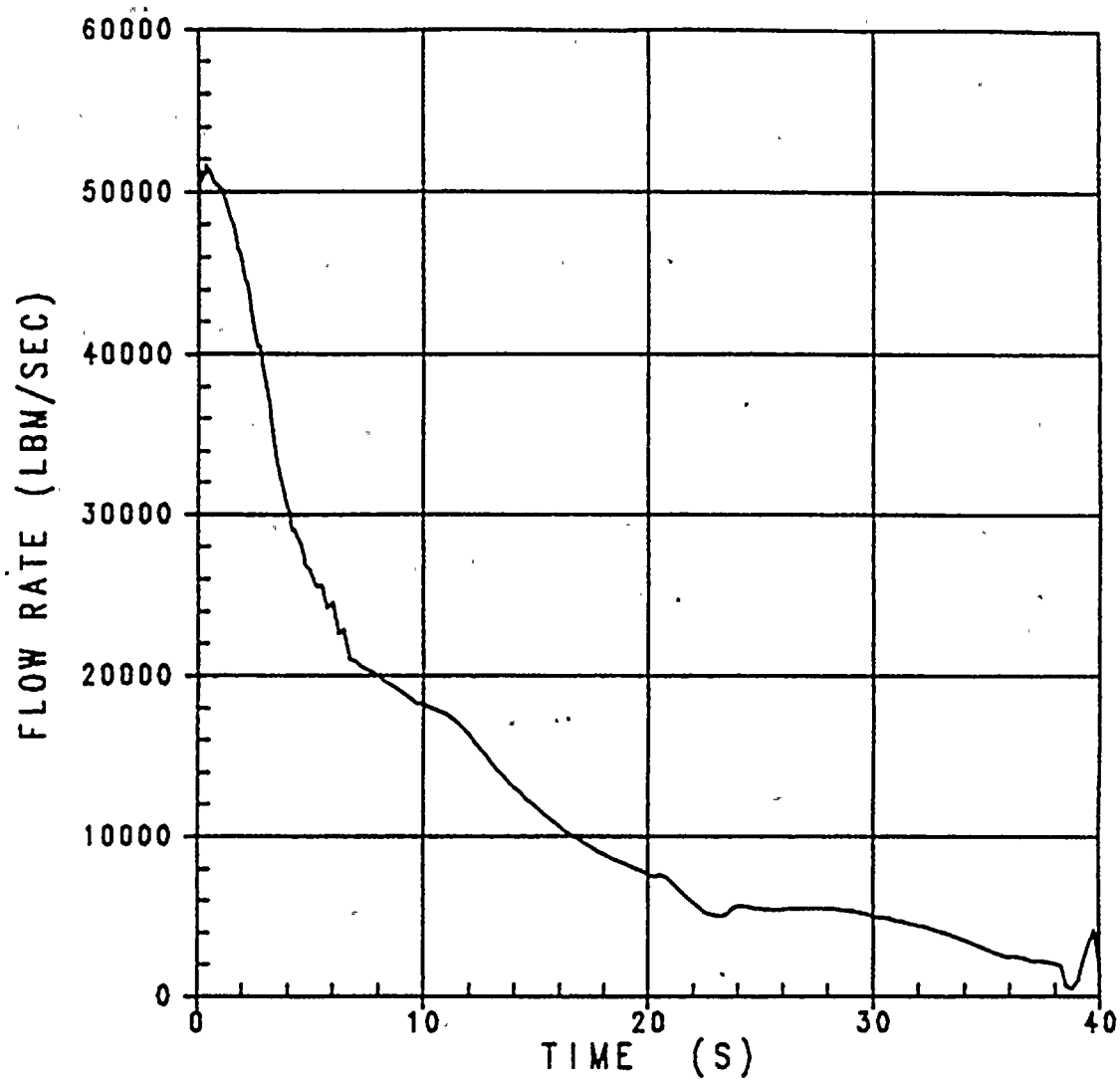


Figure 14.3.1-2e Break Flow During Blowdown
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

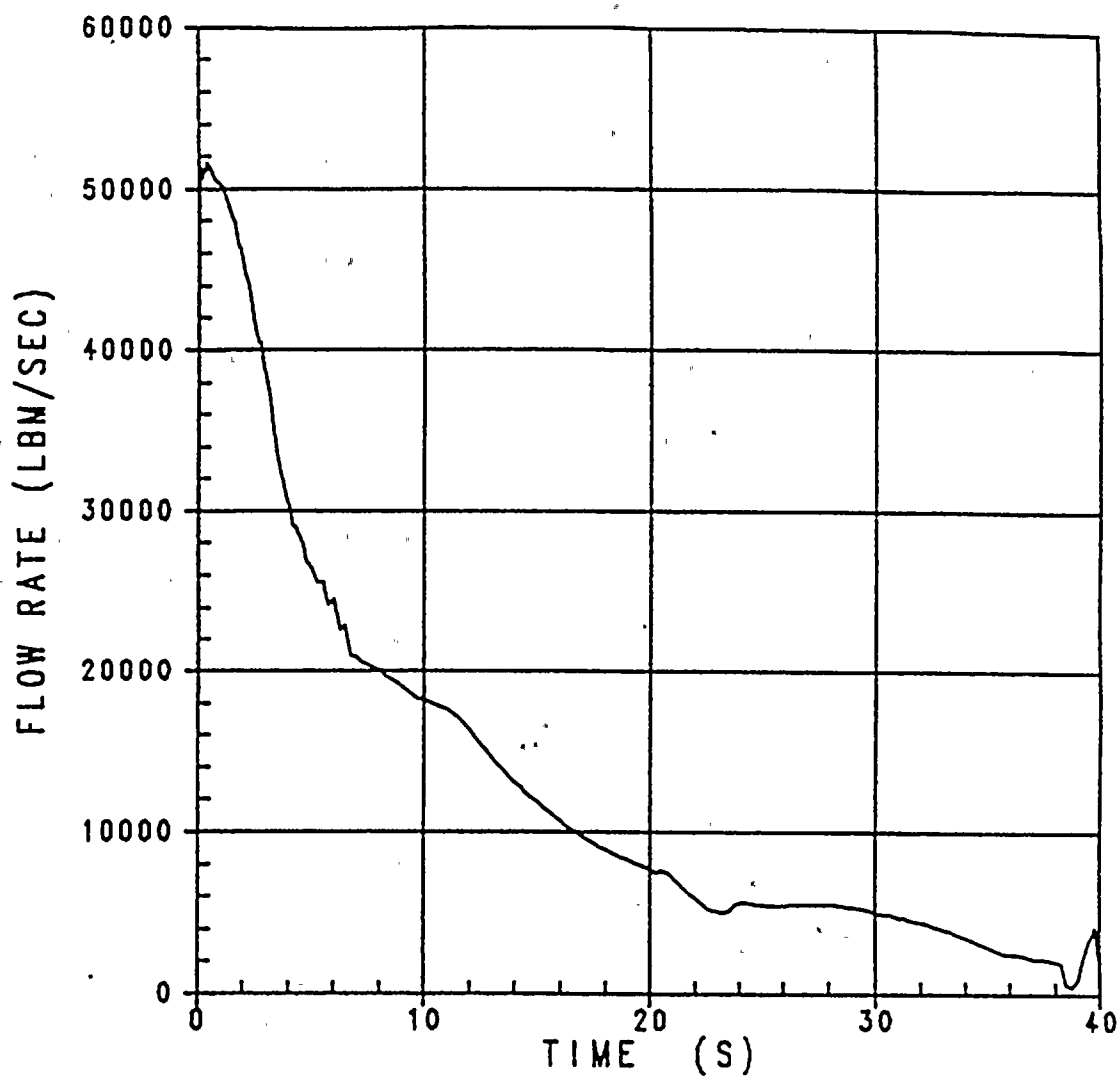


Figure 14.3.1-2f Break Flow During Blowdown
Case F, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

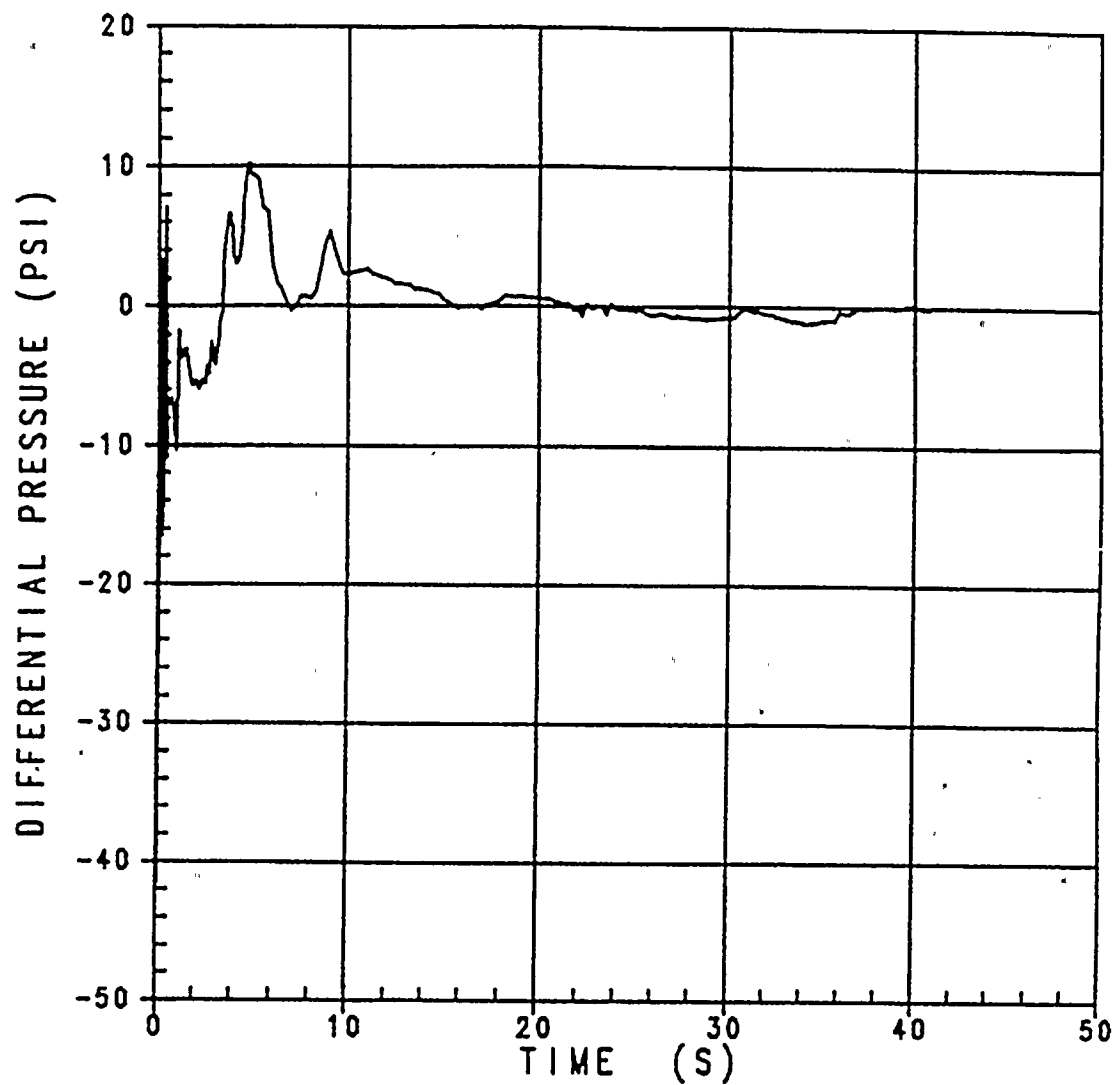


Figure 14.3.1-3a Core Pressure Drop
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

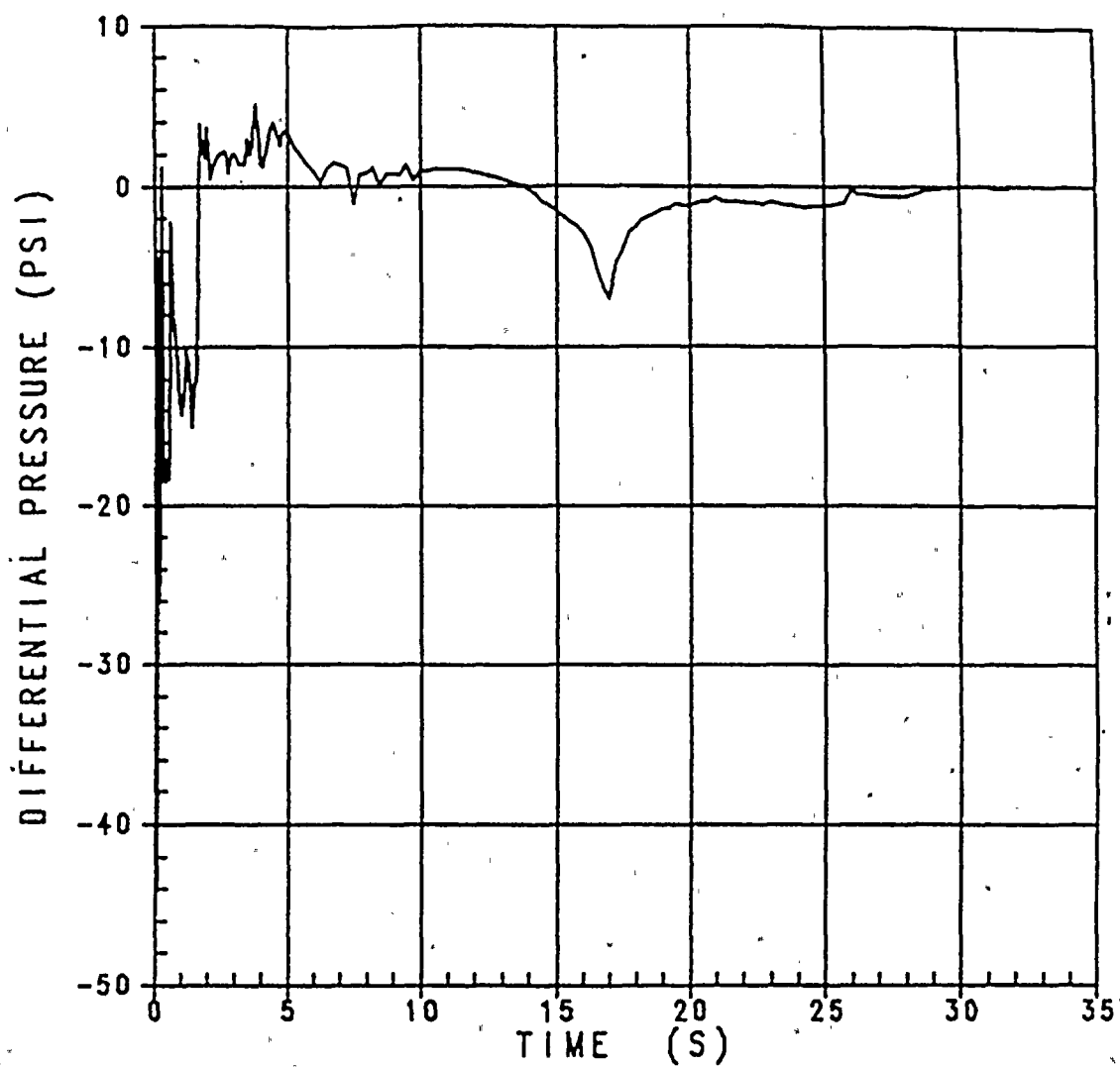


Figure 14.3.1:3b

Core Pressure Drop

Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia

Donald C. Cook Unit 1

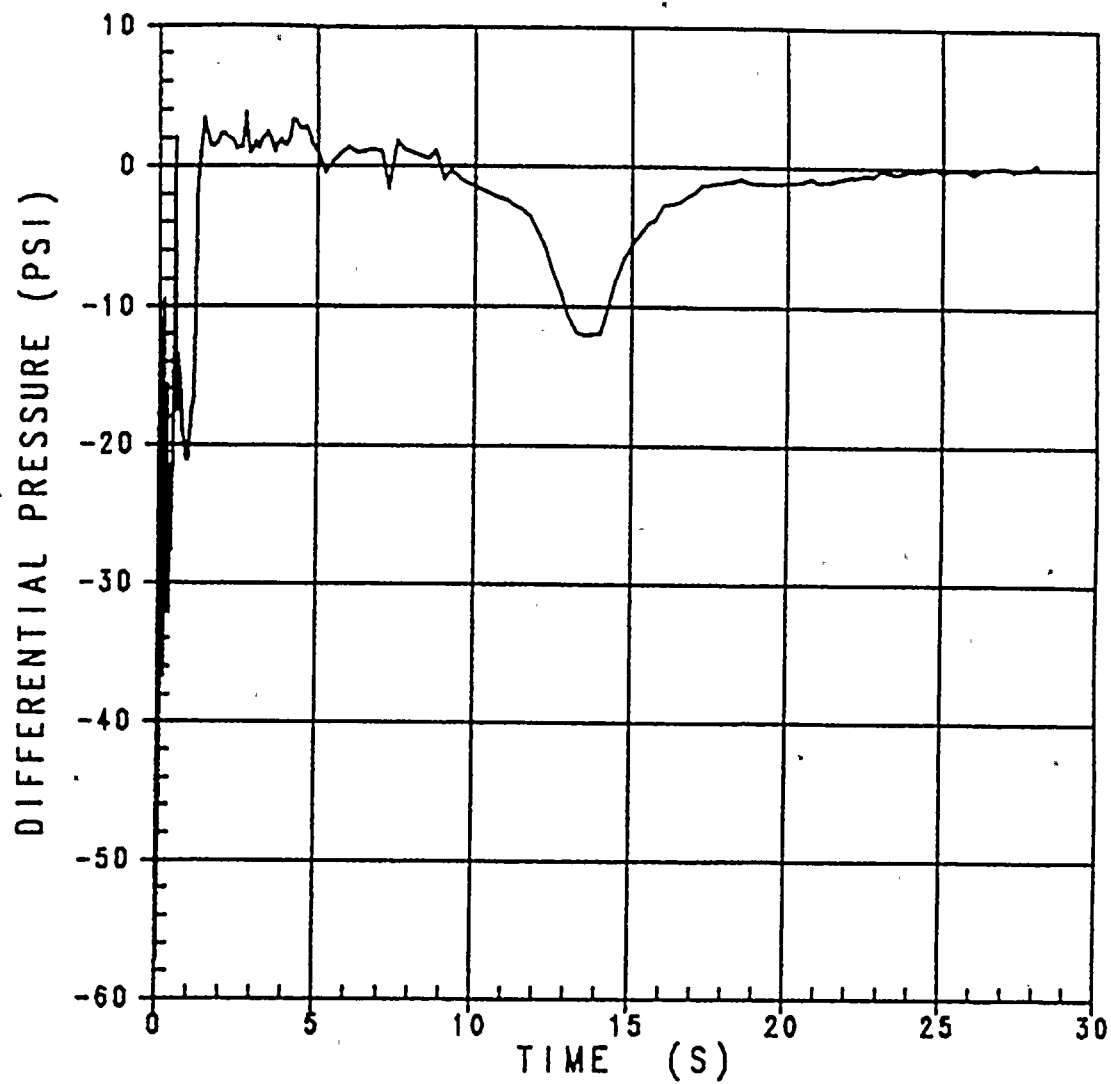


Figure 14.3.1-3c Core Pressure Drop
Case C, $CD=0.8$, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

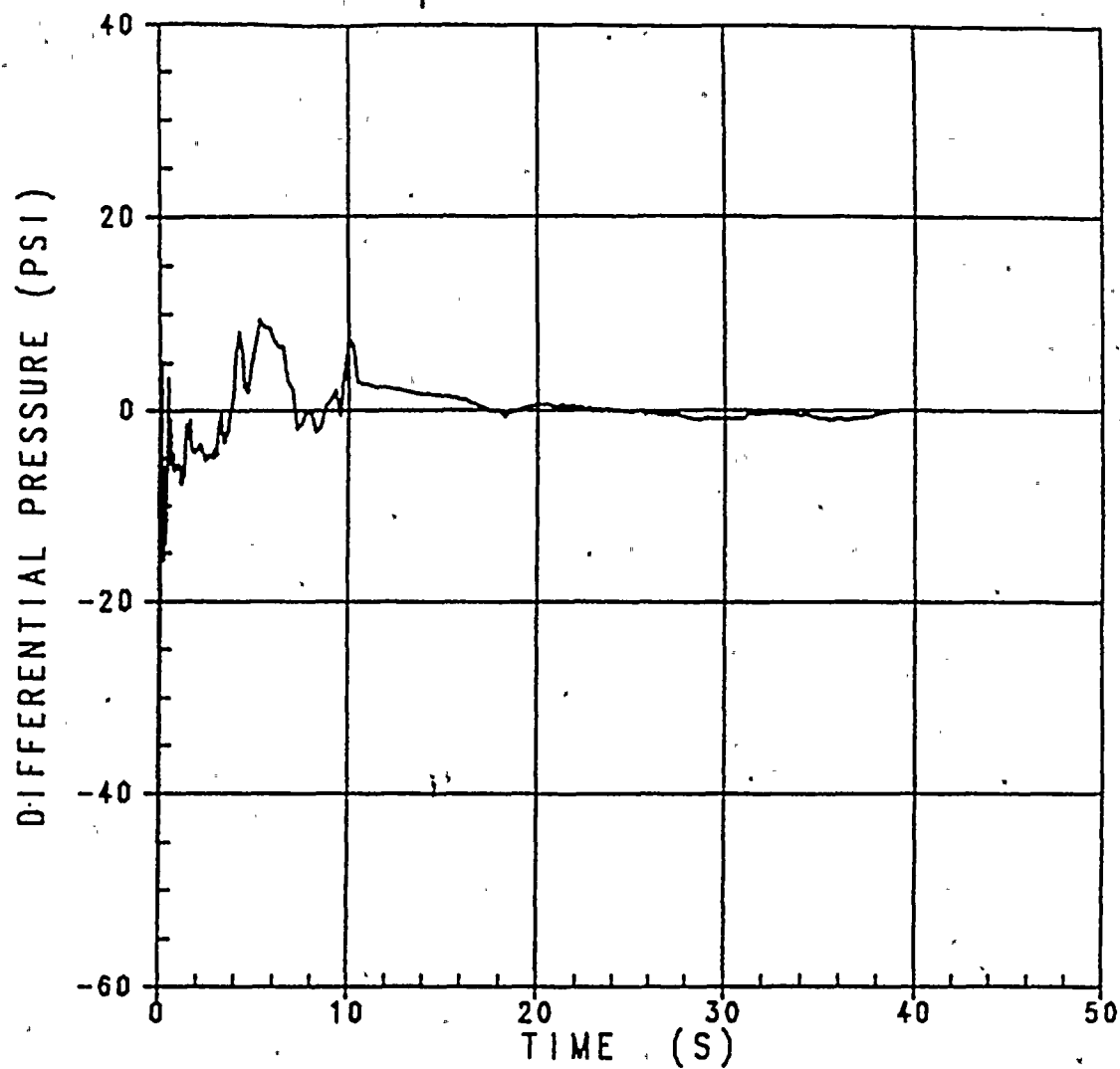


Figure 14.3.1-3d Core Pressure Drop
Case D, $CD=0.4$, $T_{hot}=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

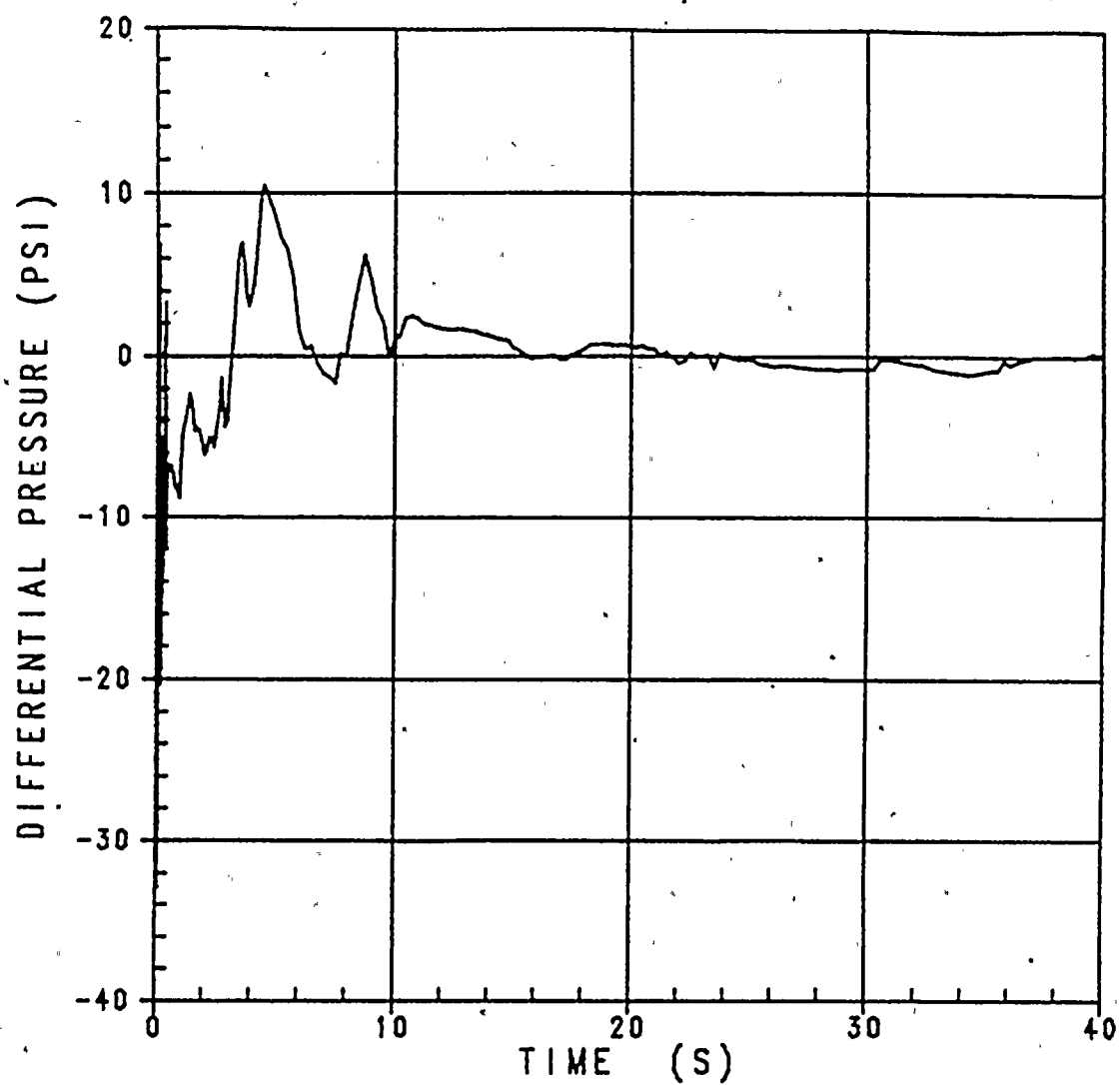


Figure 14.3.1-3e Core Pressure Drop
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

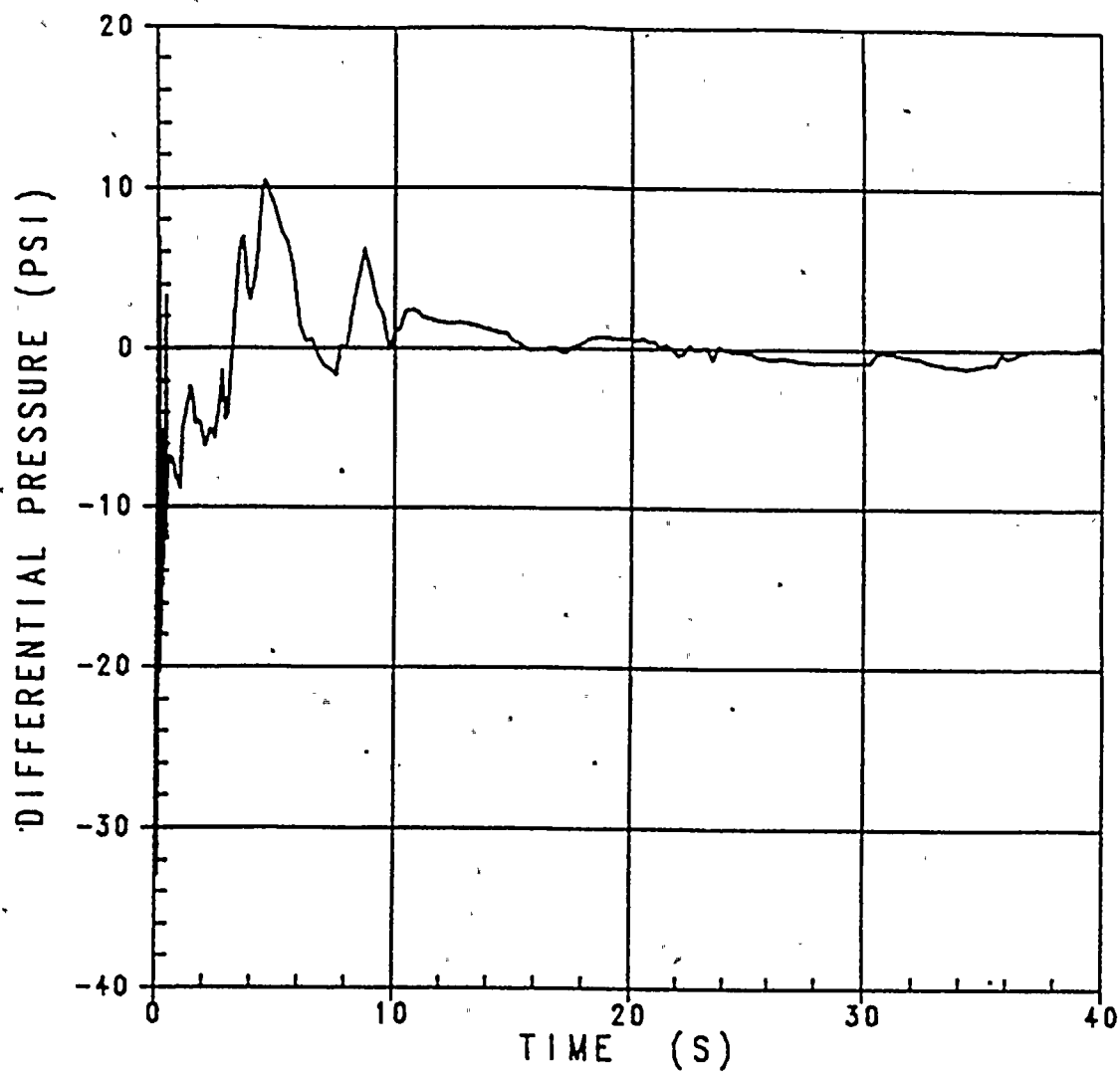


Figure 14.3.1-3f Core Pressure Drop
Case F, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

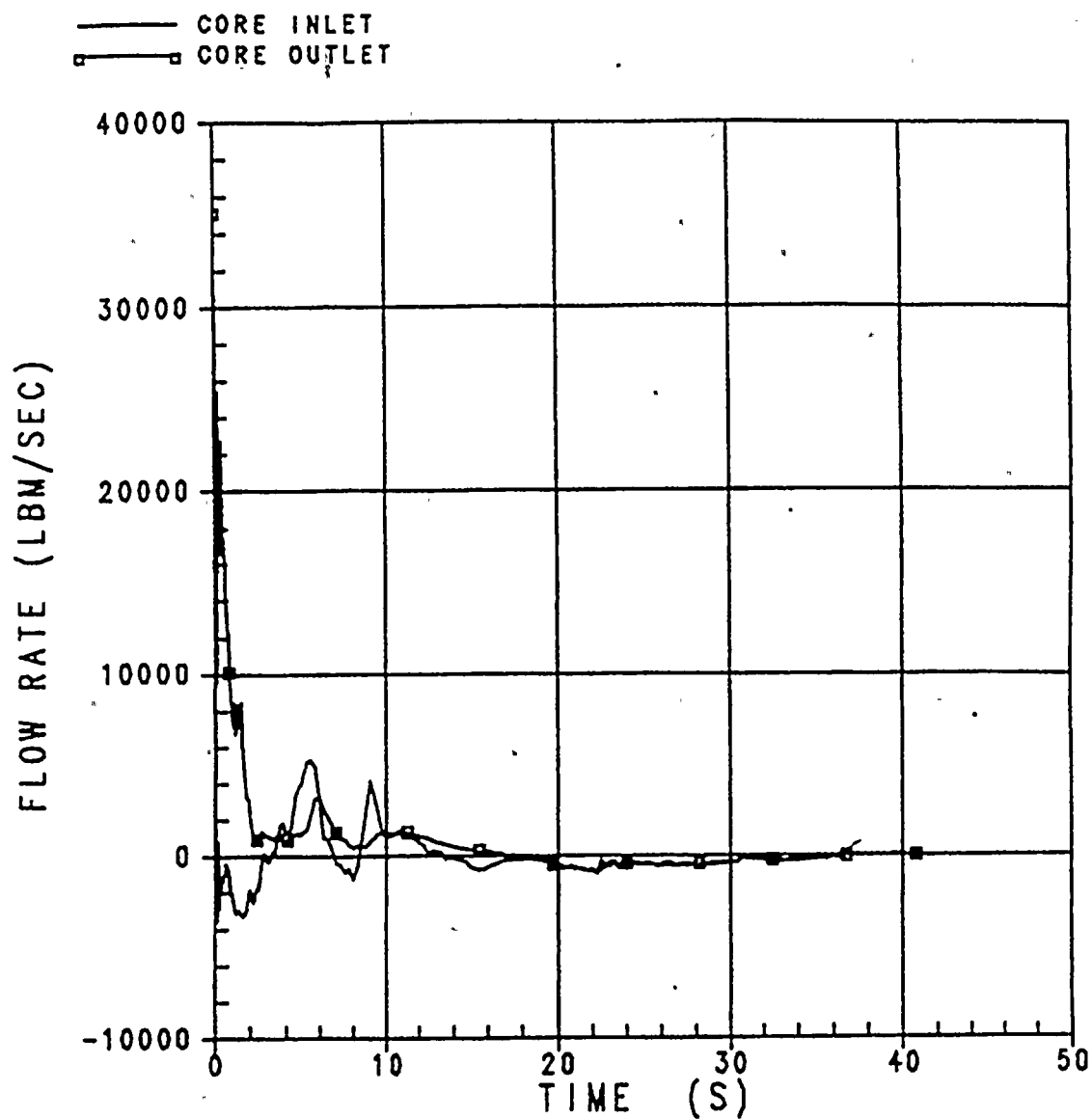


Figure 14.3.1-4a

Core Flowrate
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

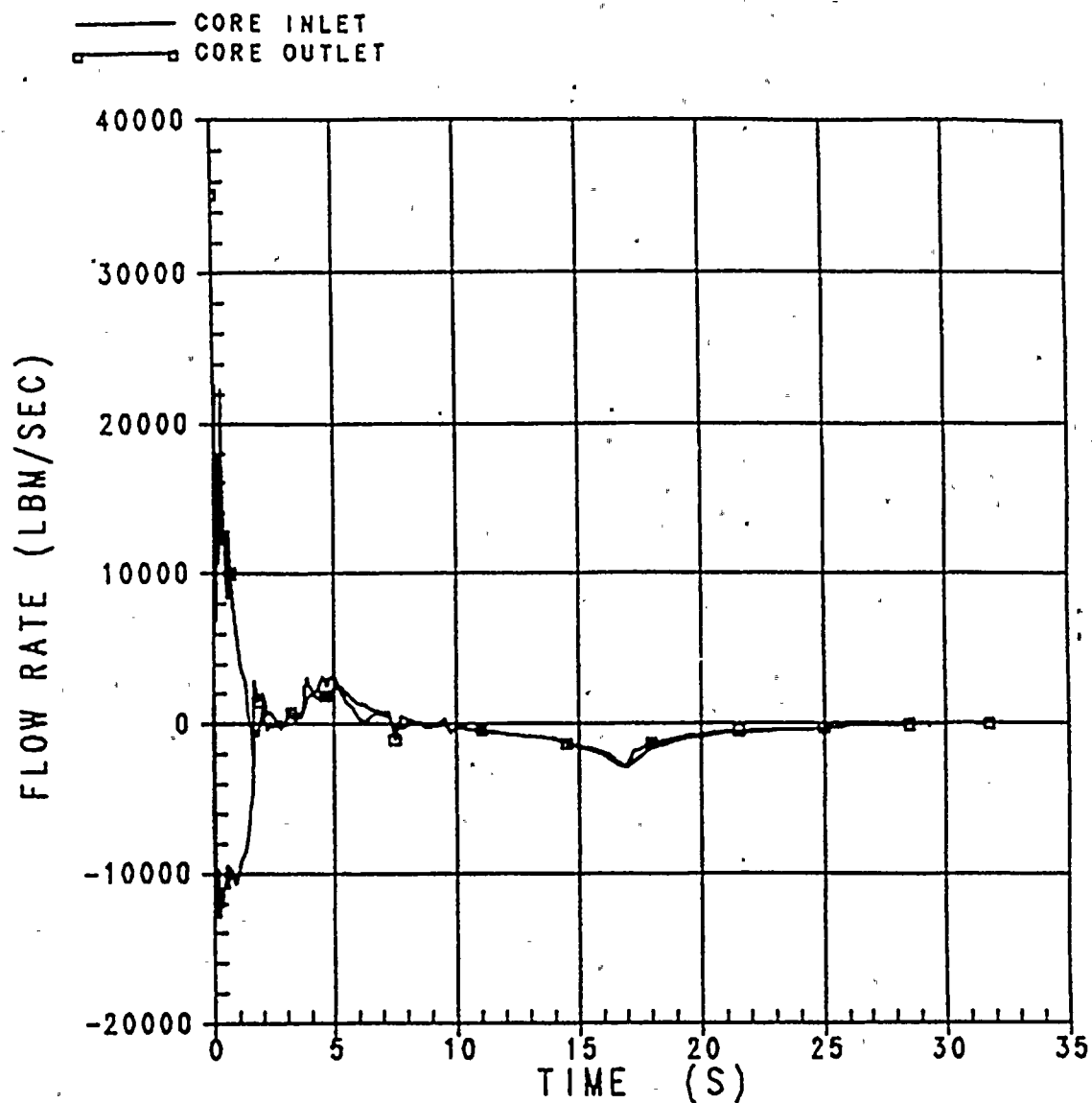


Figure 14.3.1-4b

Core Flowrate

Case B, $CD=0.6$, $Thot=609.1^{\circ}F$, $P=2250$ psia

Donald C. Cook Unit 1

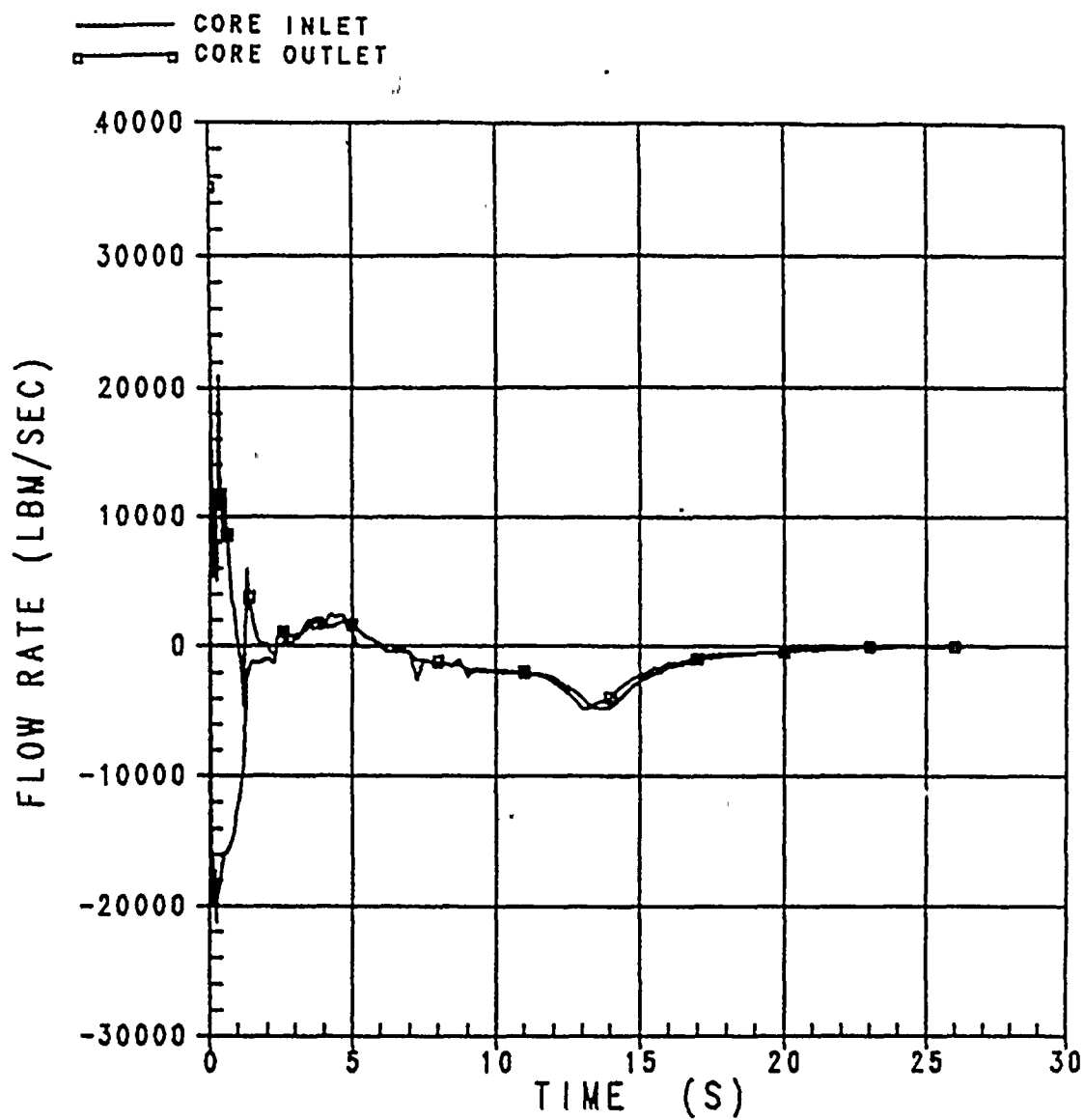


Figure 14.3.1-4c

Core Flowrate

Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia

Donald C. Cook Unit 1

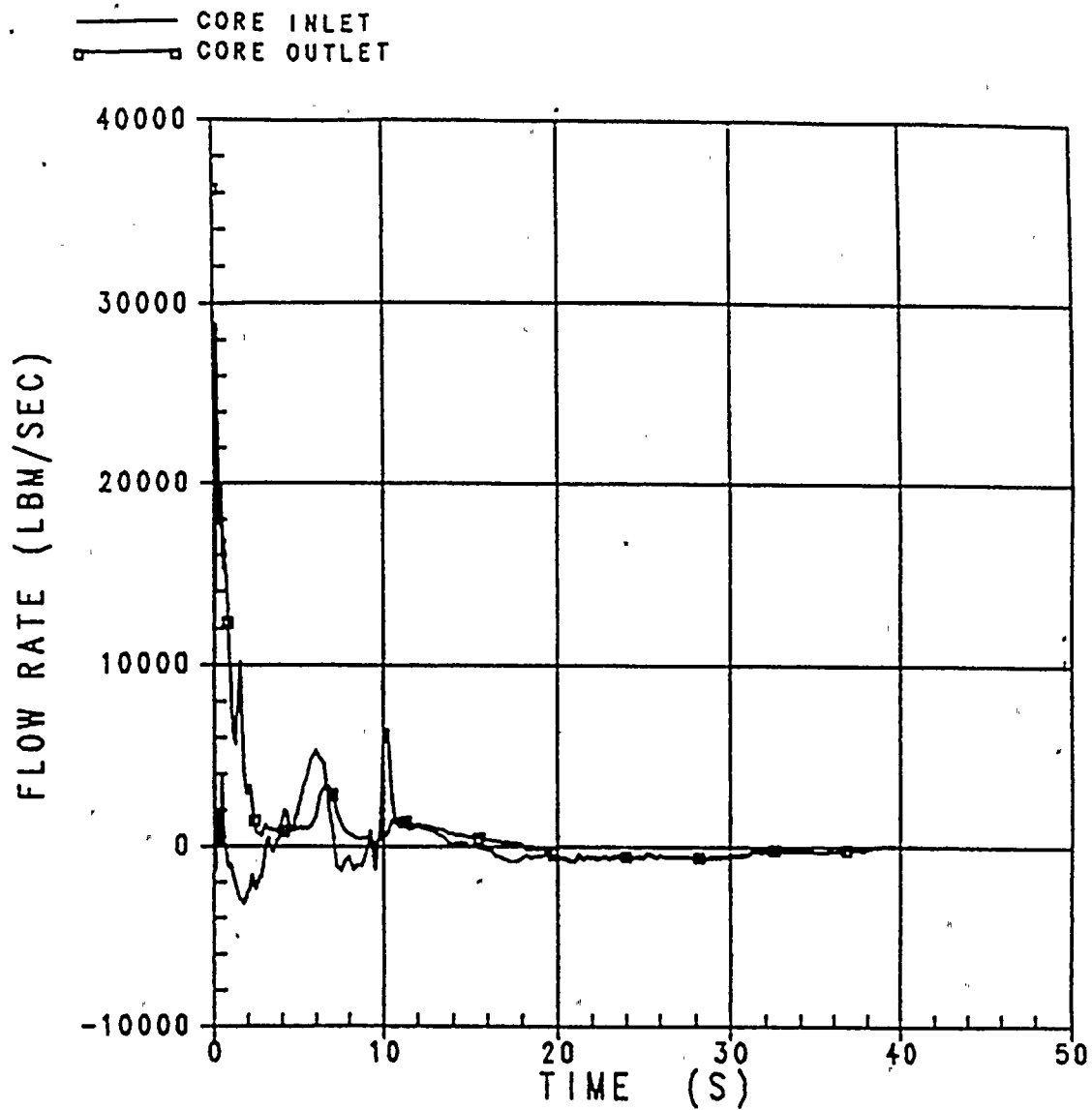


Figure 14.3.1-4d

Core Flowrate

Case D, CD=0.4, Thot=586.8°F, P=2250 psia

Donald C. Cook Unit 1

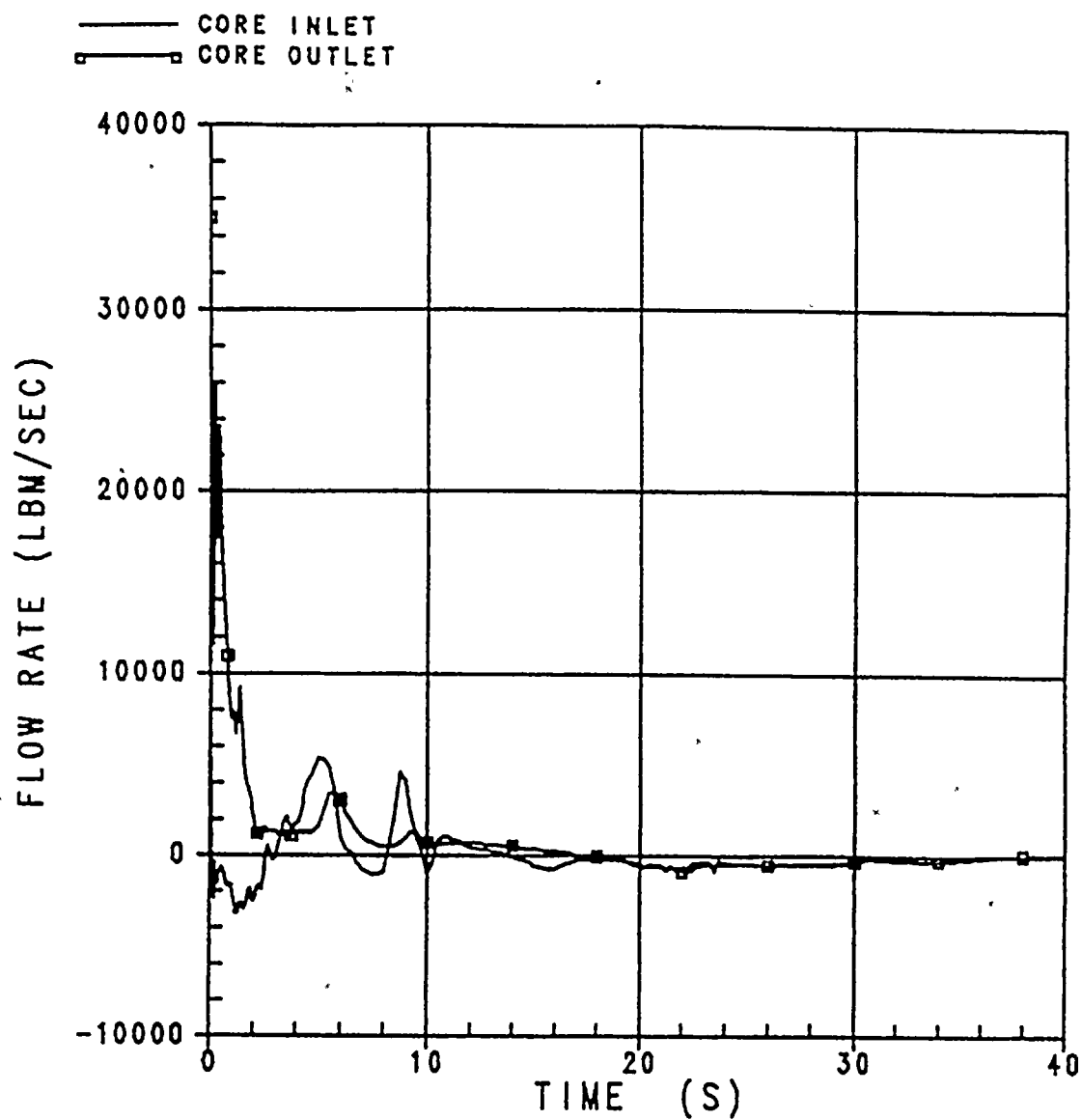


Figure 14.3.1-4e

Core Flowrate

Case E, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia

Donald C. Cook Unit 1

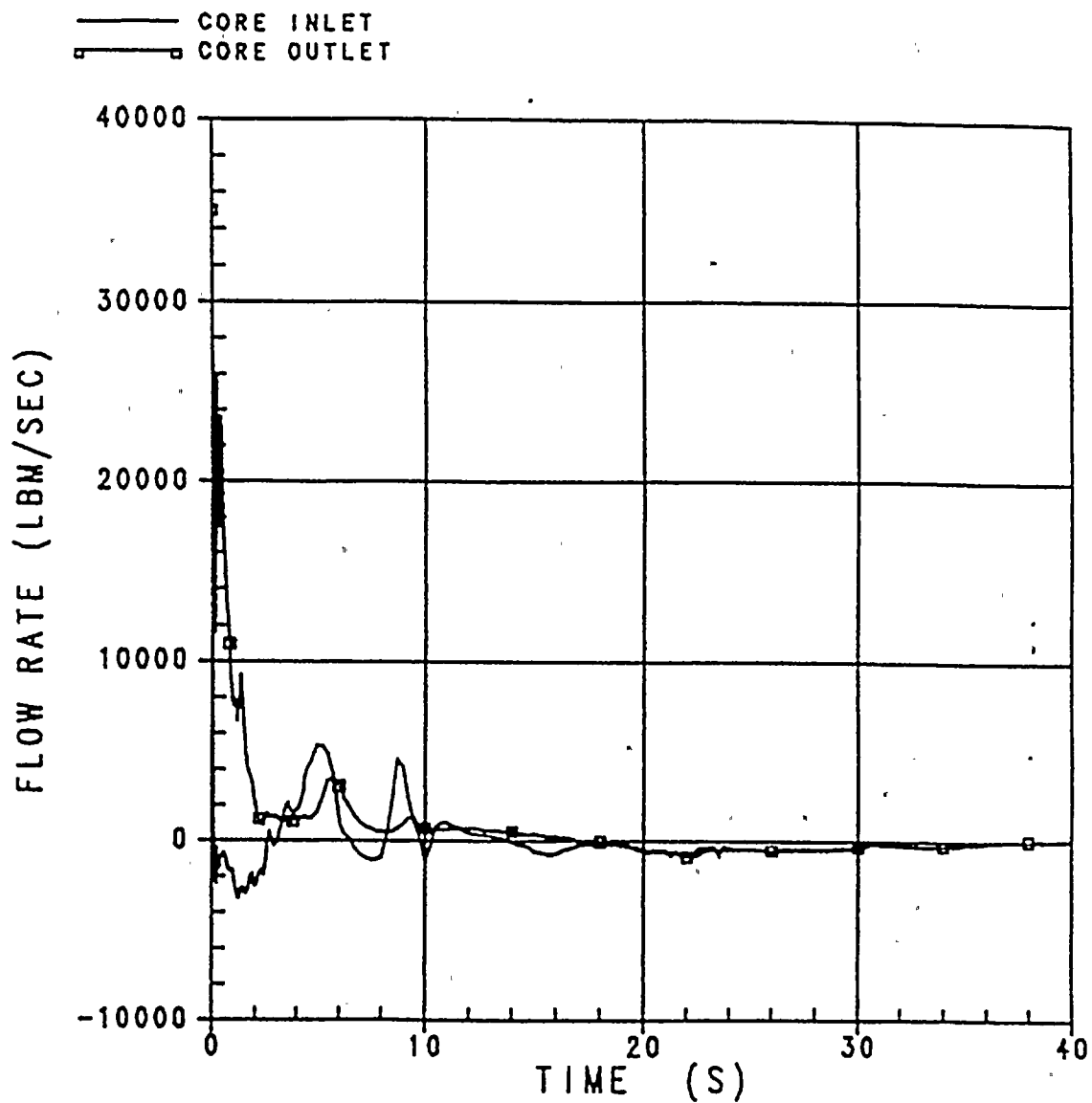


Figure 14.3.1-4f Core Flowrate
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

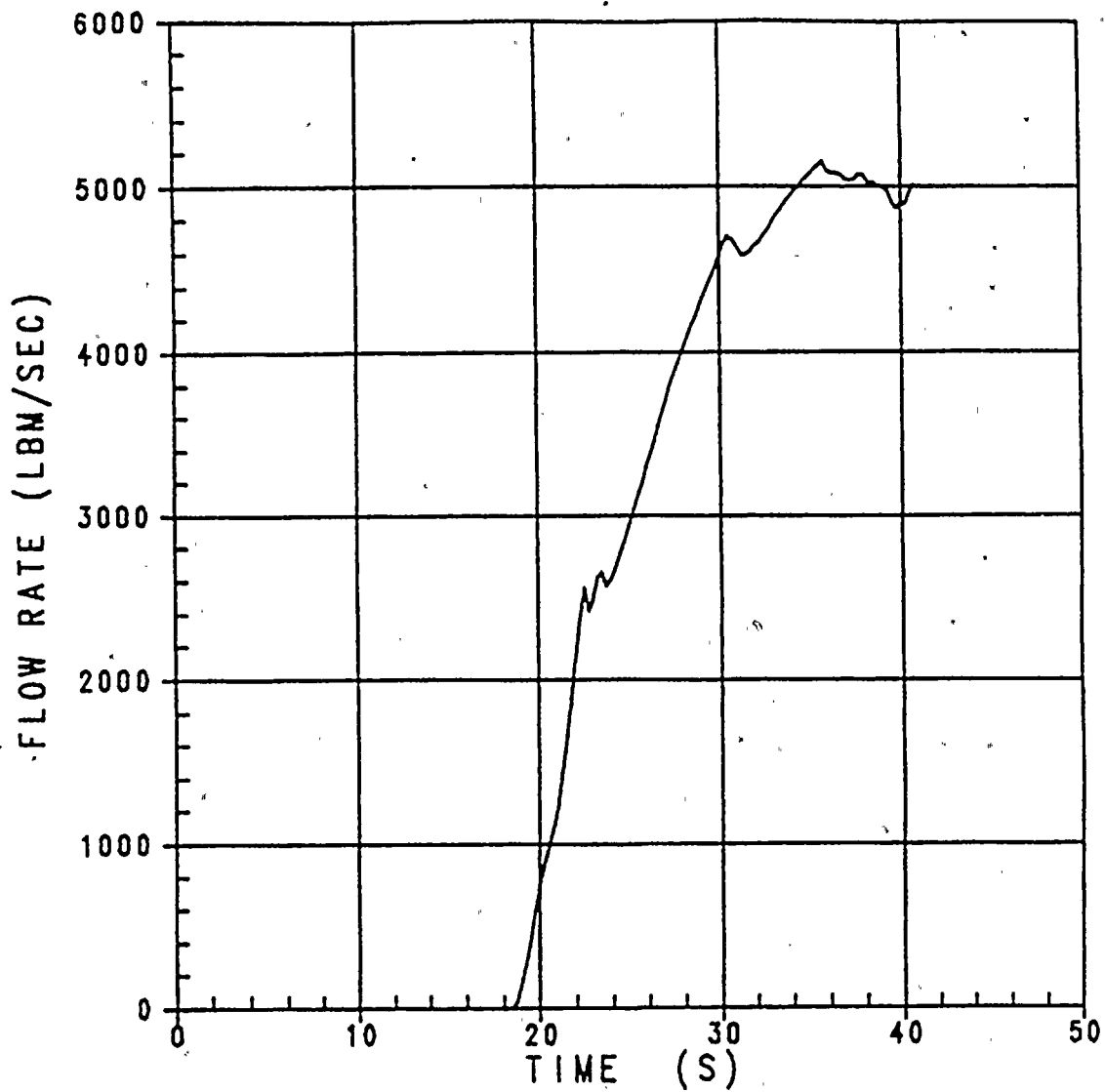


Figure 14.3.1-5a

Accumulator Flow During Blowdown
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

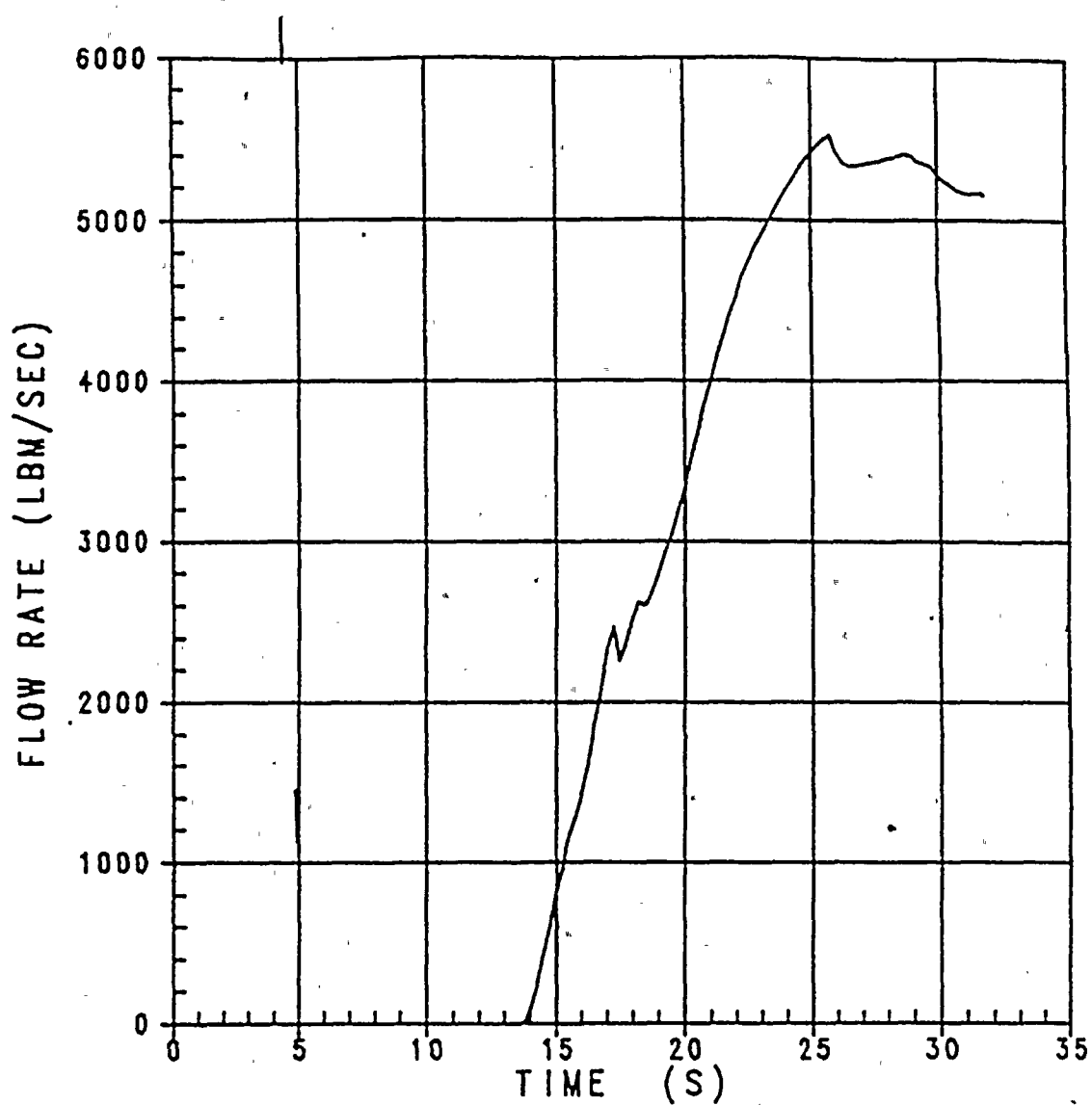


Figure 14.3.1-5b

Accumulator Flow During Blowdown
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

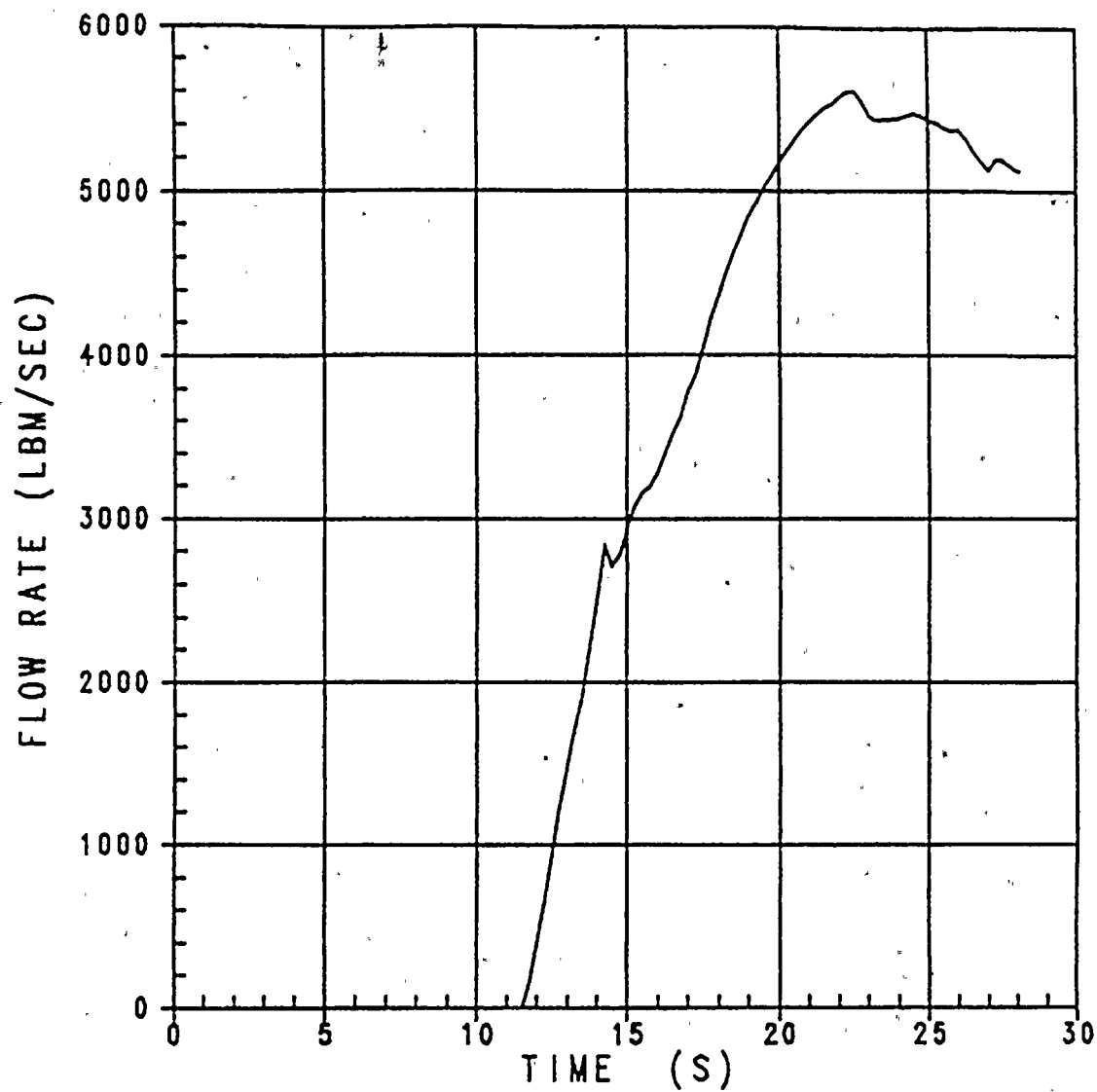


Figure 14.3.1-5c Accumulator Flow During Blowdown
Case C, $CD=0.8$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

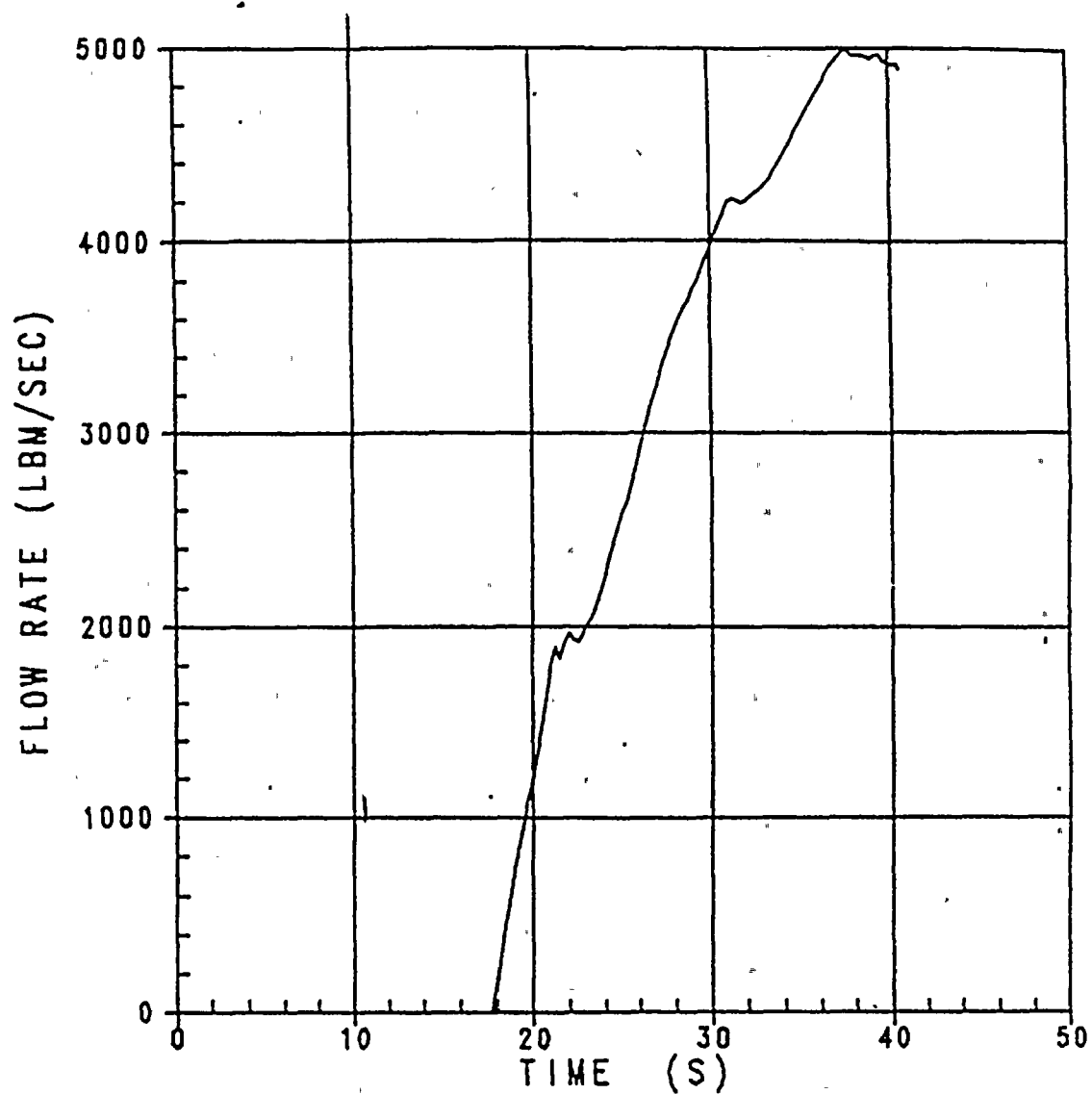


Figure 14.3.1-5d Accumulator Flow During Blowdown
Case D, $CD=0.4$, $T_{hot}=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

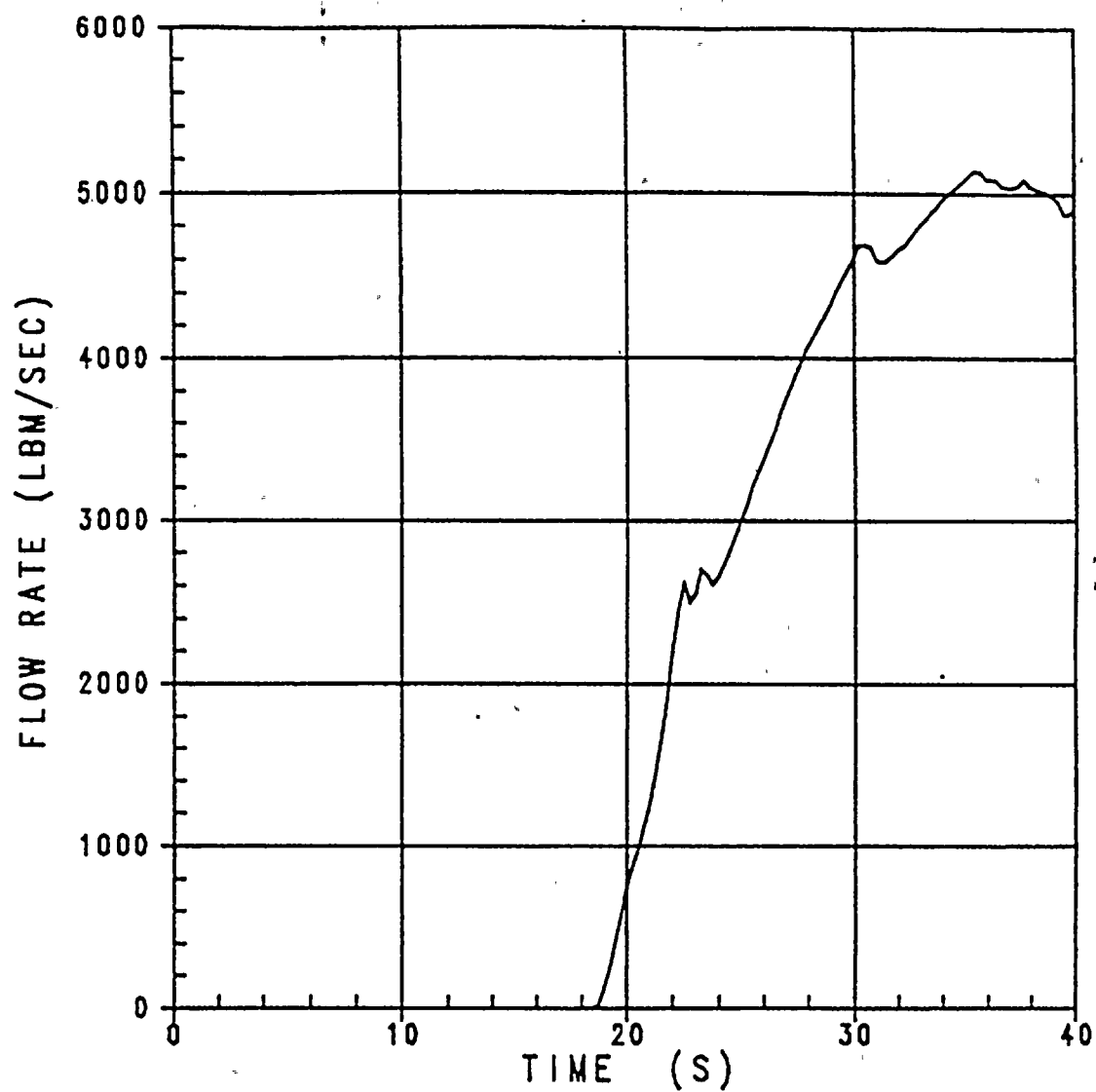


Figure 14.3.1-5e Accumulator Flow During Blowdown
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

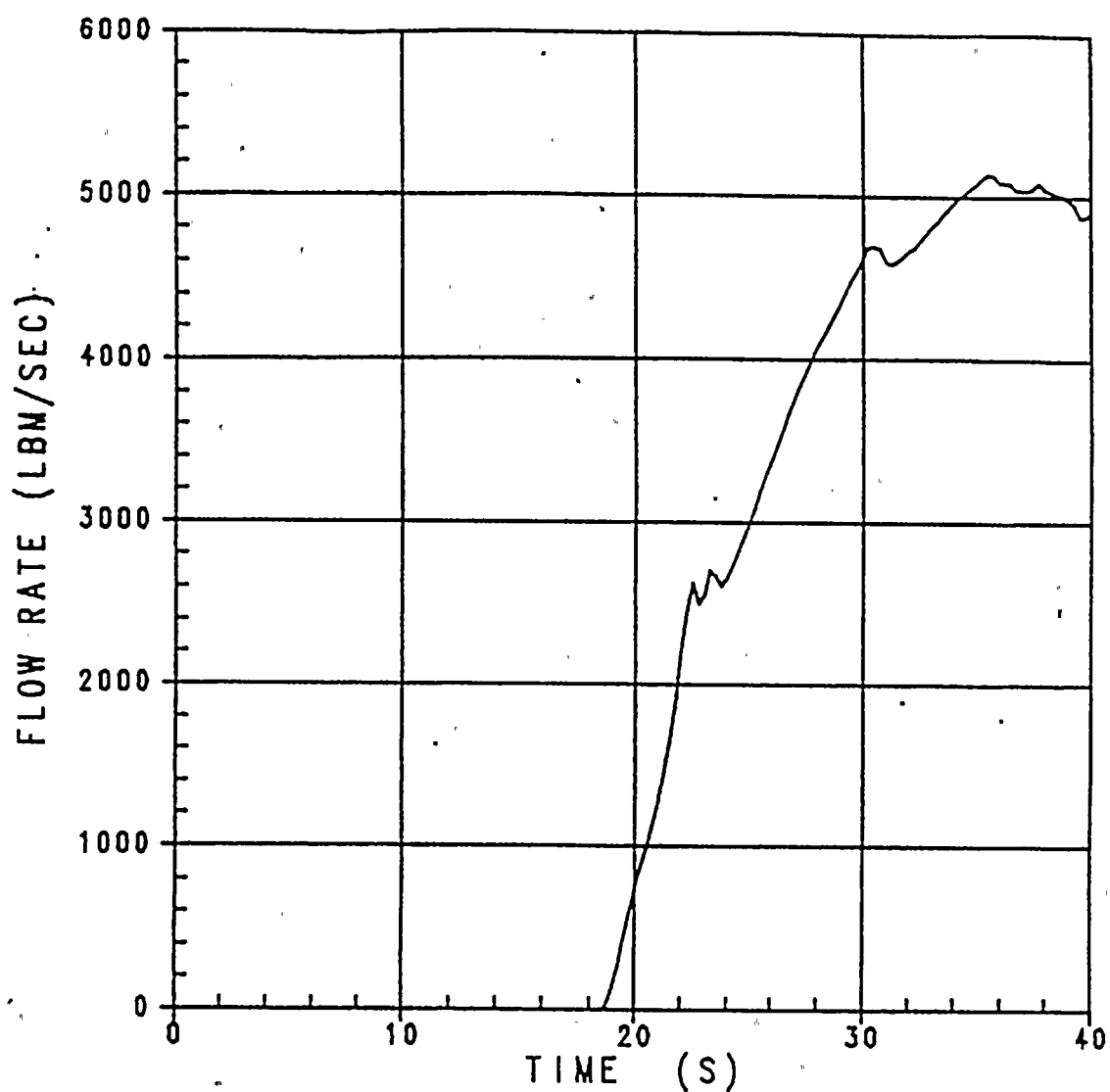


Figure 14.3.1-5f

Accumulator Flow During Blowdown

Case F, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia, max SI

Donald C. Cook Unit 1

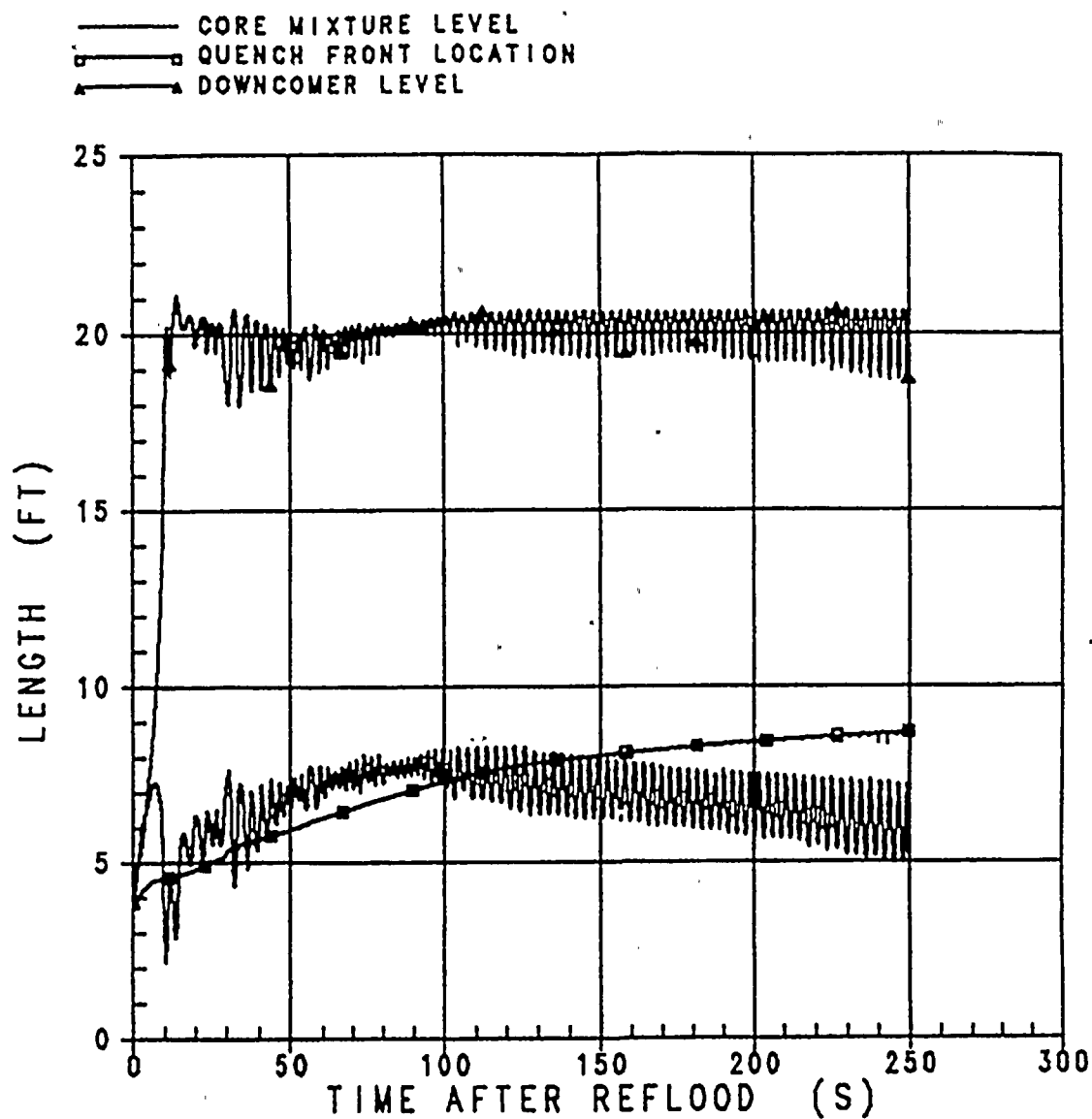


Figure 14.3.1-6a Vessel Liquid Levels During Reflood
 Case A, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2250$ psia
 Donald C. Cook Unit 1

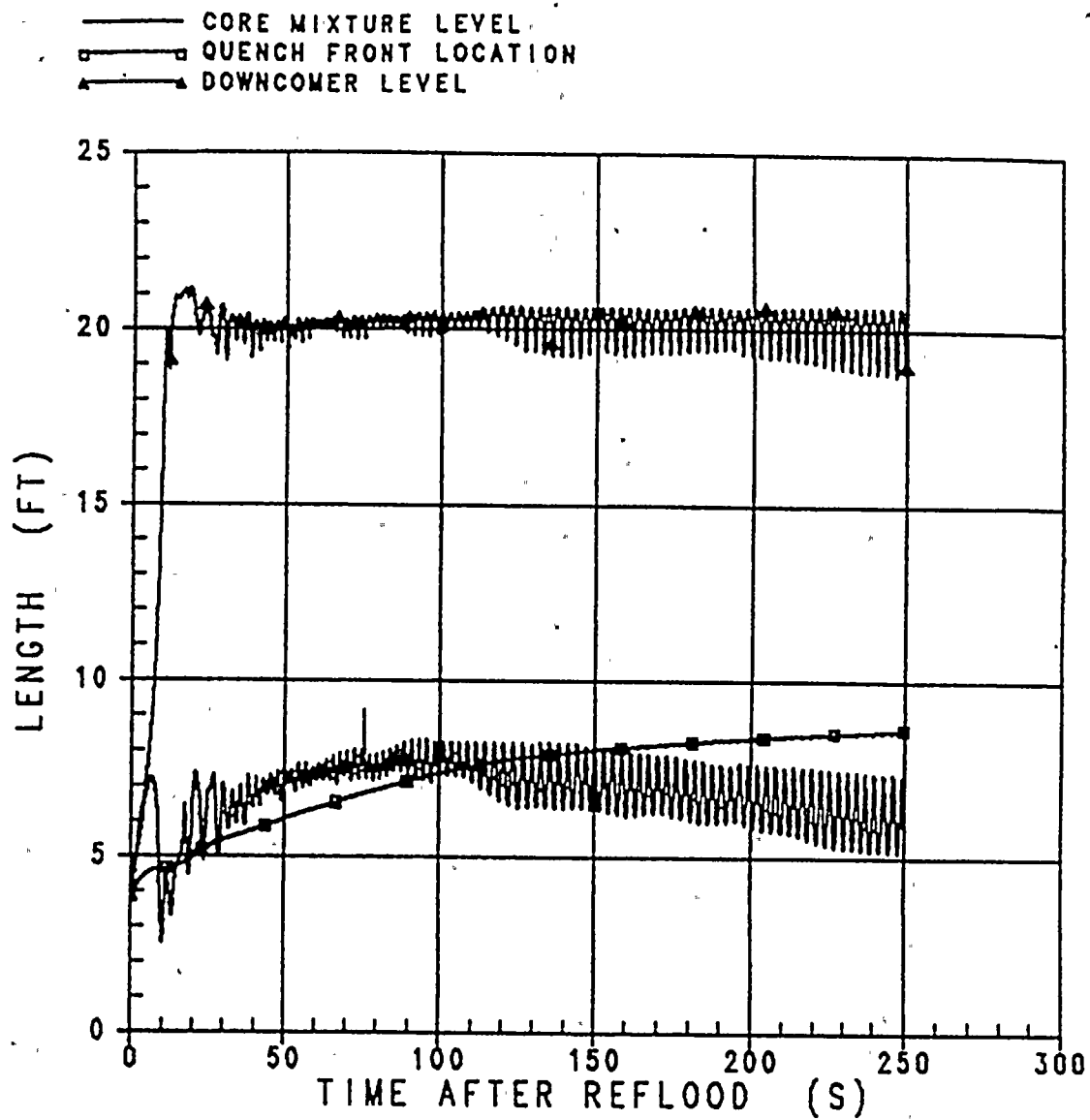


Figure 14.3.1-6b Vessel Liquid Levels During Reflood
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

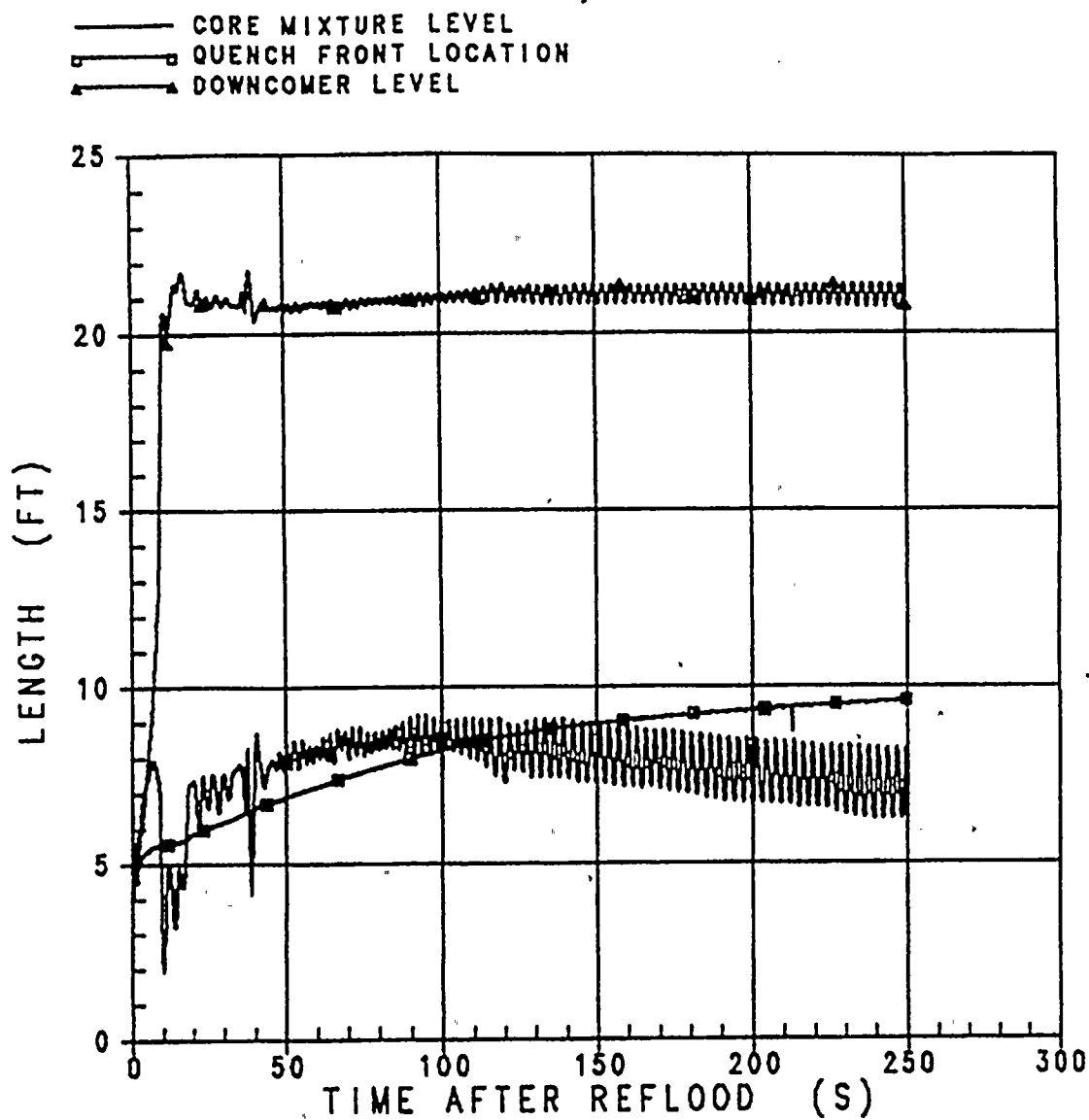


Figure 14.3.1-6c Vessel Liquid Levels During Reflood
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

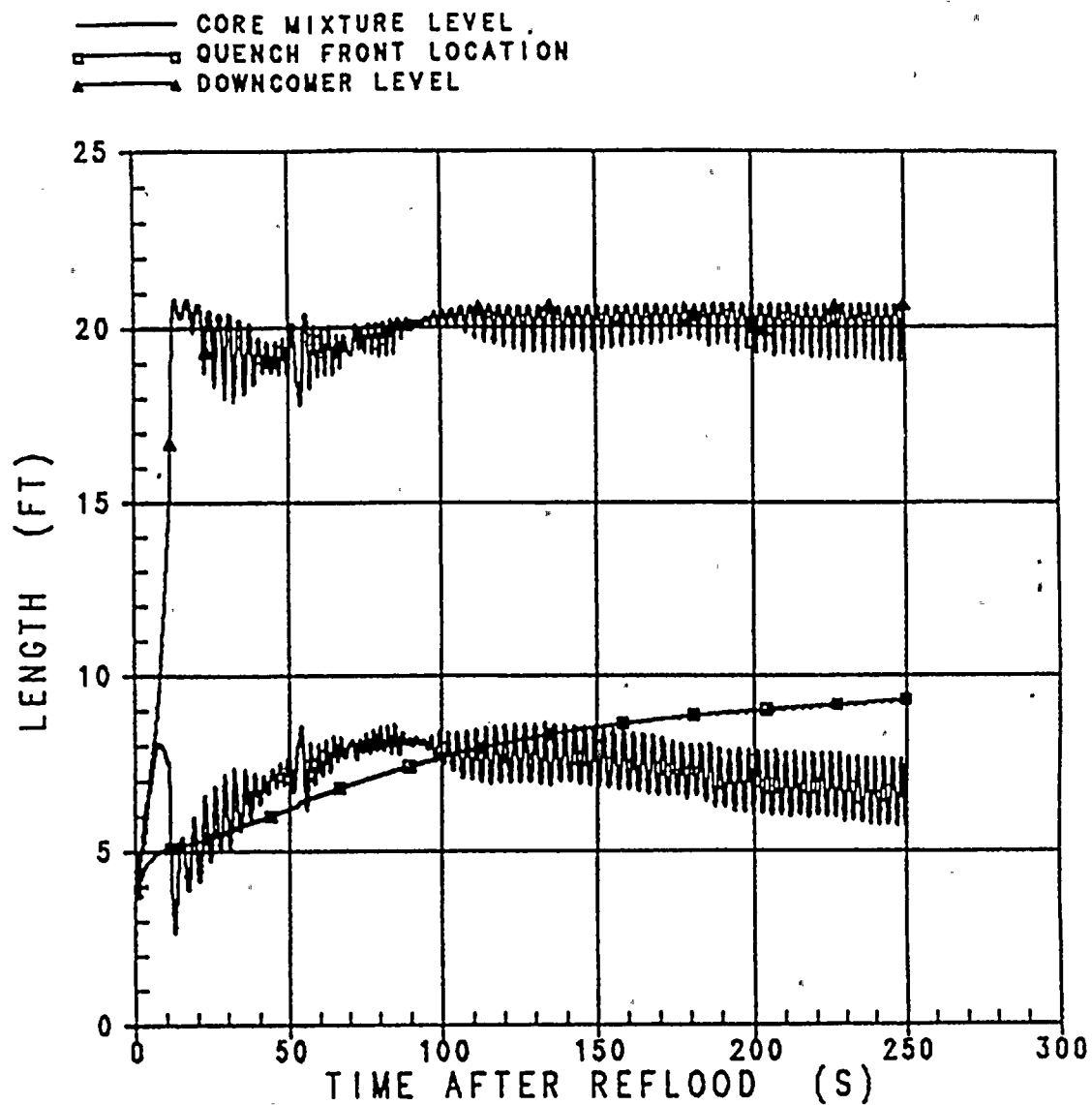


Figure 14.3.1-6d Vessel Liquid Levels During Reflood
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

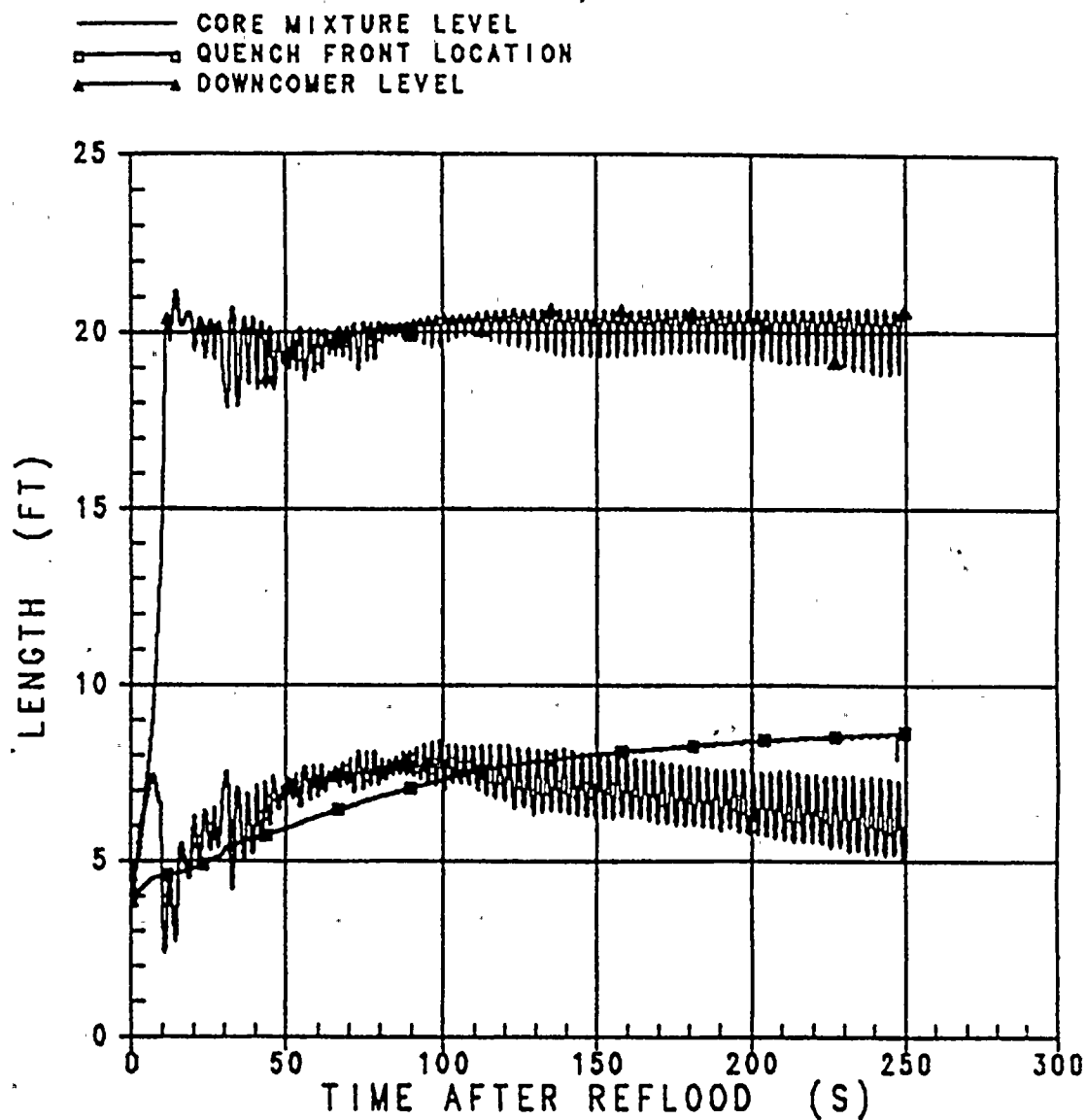


Figure 14.3.1-6e Vessel Liquid Levels During Reflood
Case E, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

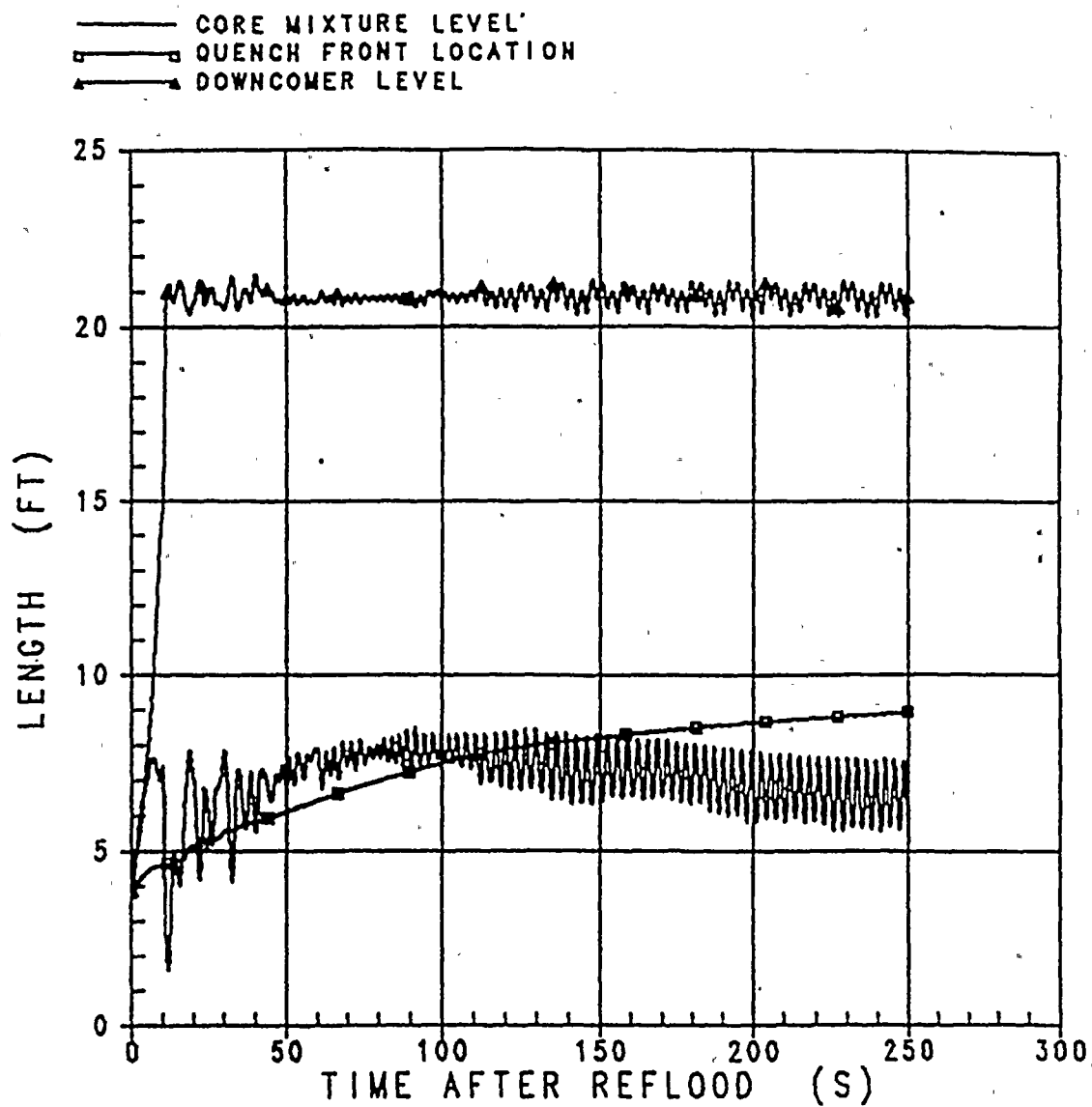


Figure 14.3.1-6f

Vessel Liquid Levels During Reflood
 Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
 Donald C. Cook Unit 1

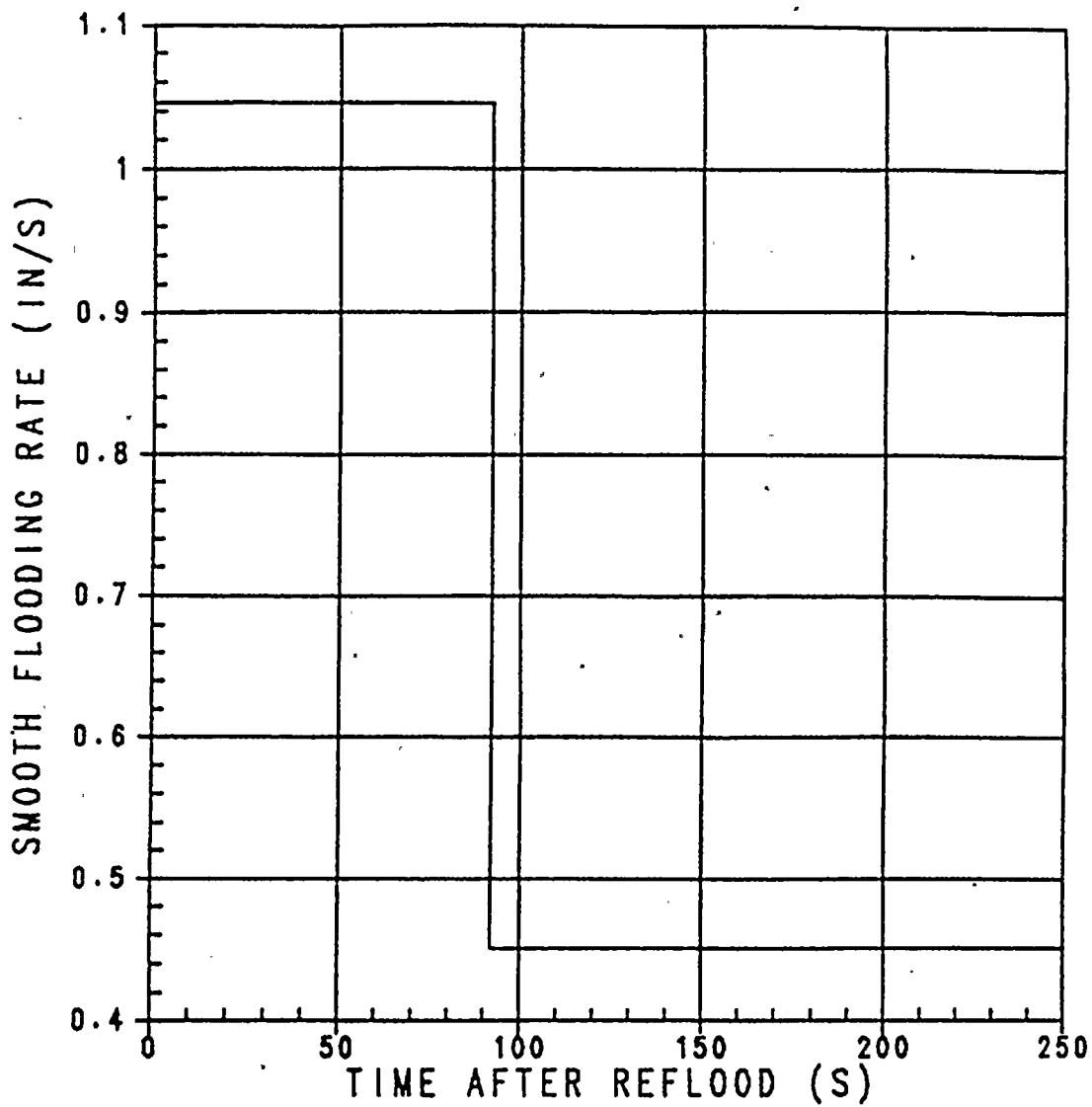


Figure 14.3.1-7a Core Inlet Flow During Reflood
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

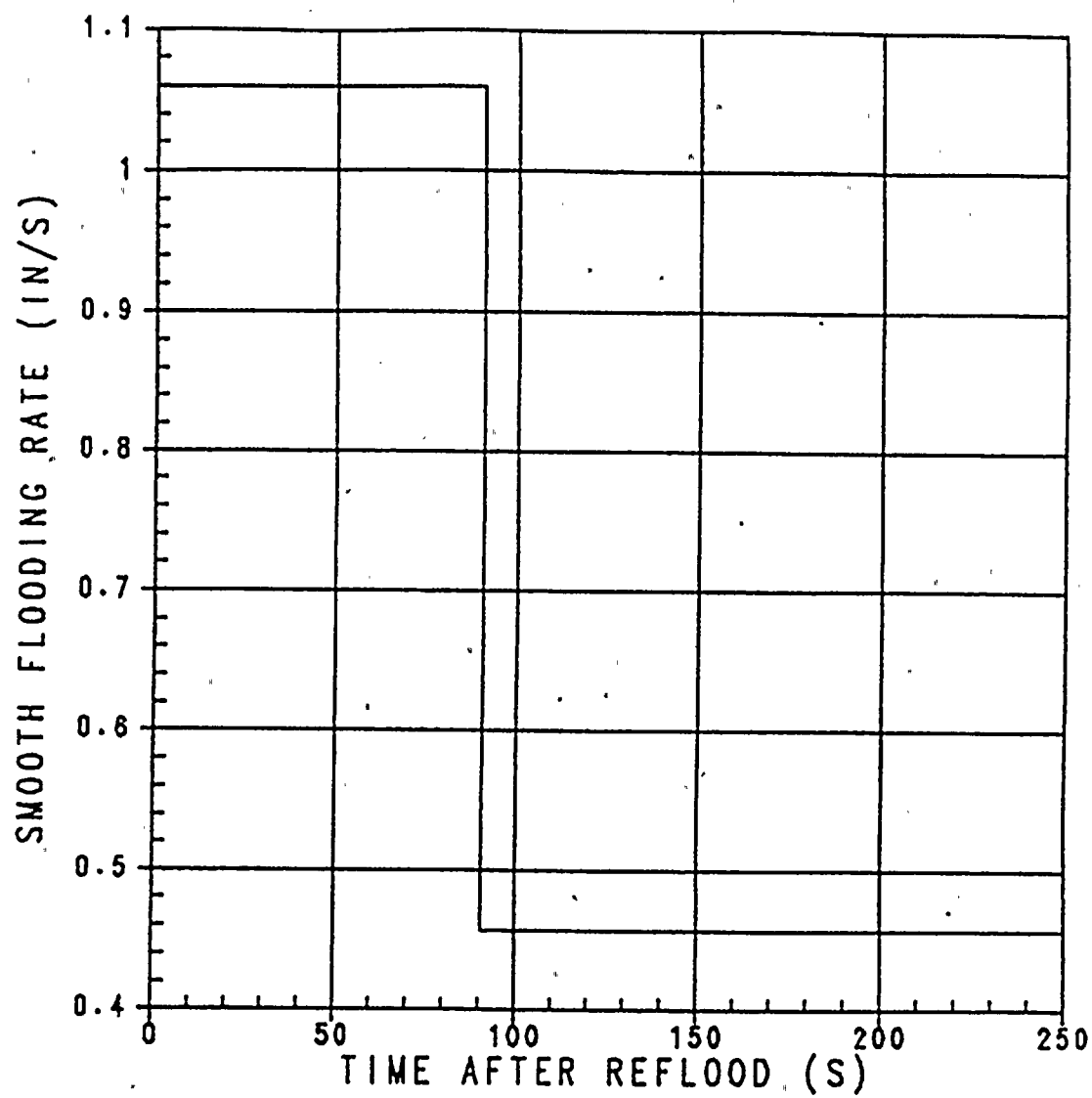


Figure 14.3.1-7b Core Inlet Flow During Reflood
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

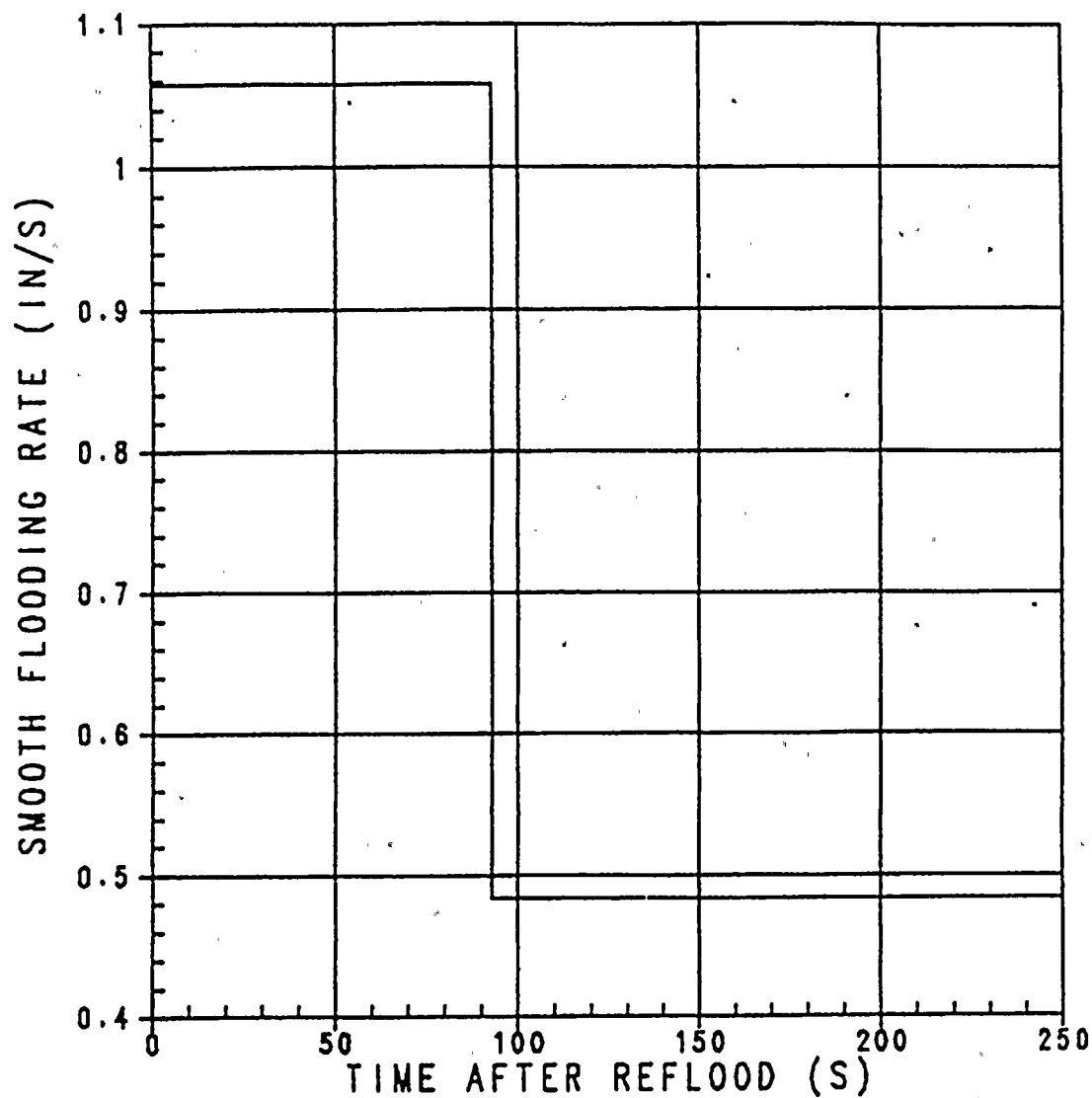


Figure 14.3.1-7c Core Inlet Flow During Reflood
Case C, $CD=0.8$, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

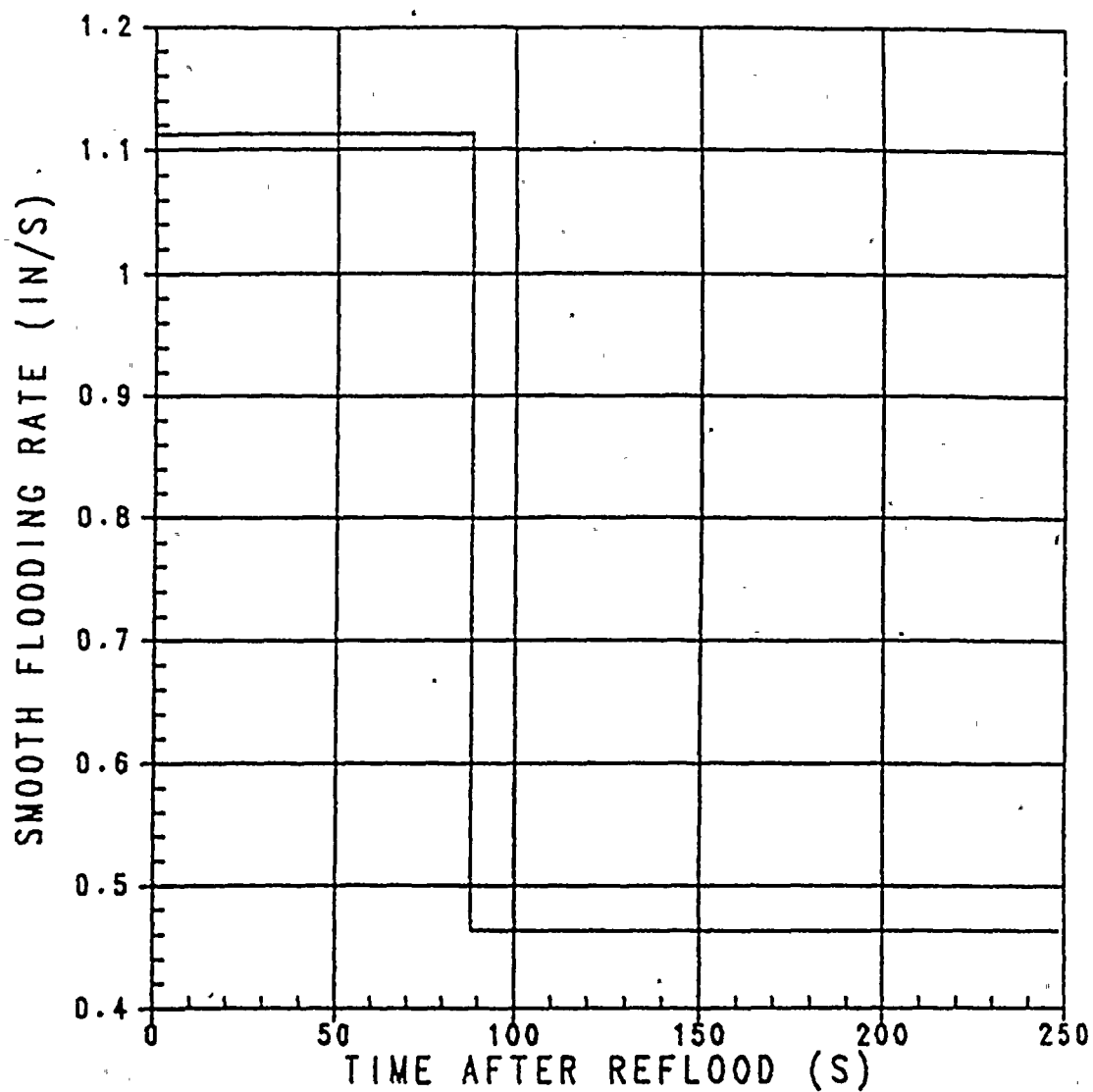


Figure 14.3.1-7d Core Inlet Flow During Reflood
Case D, $CD=0.4$, $T_{hot}=586.8^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

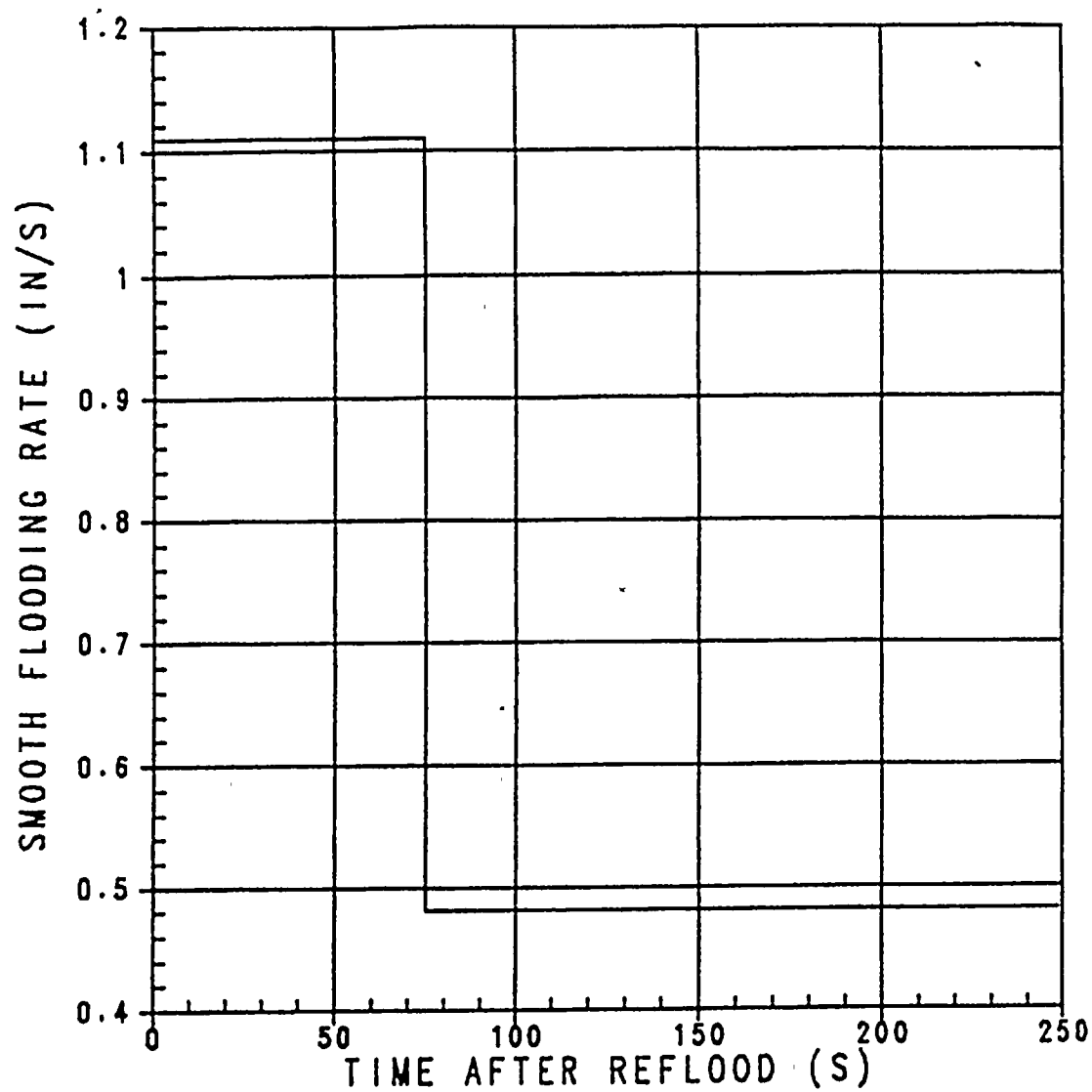


Figure 14.3.1-7e Core Inlet Flow During Reflood
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

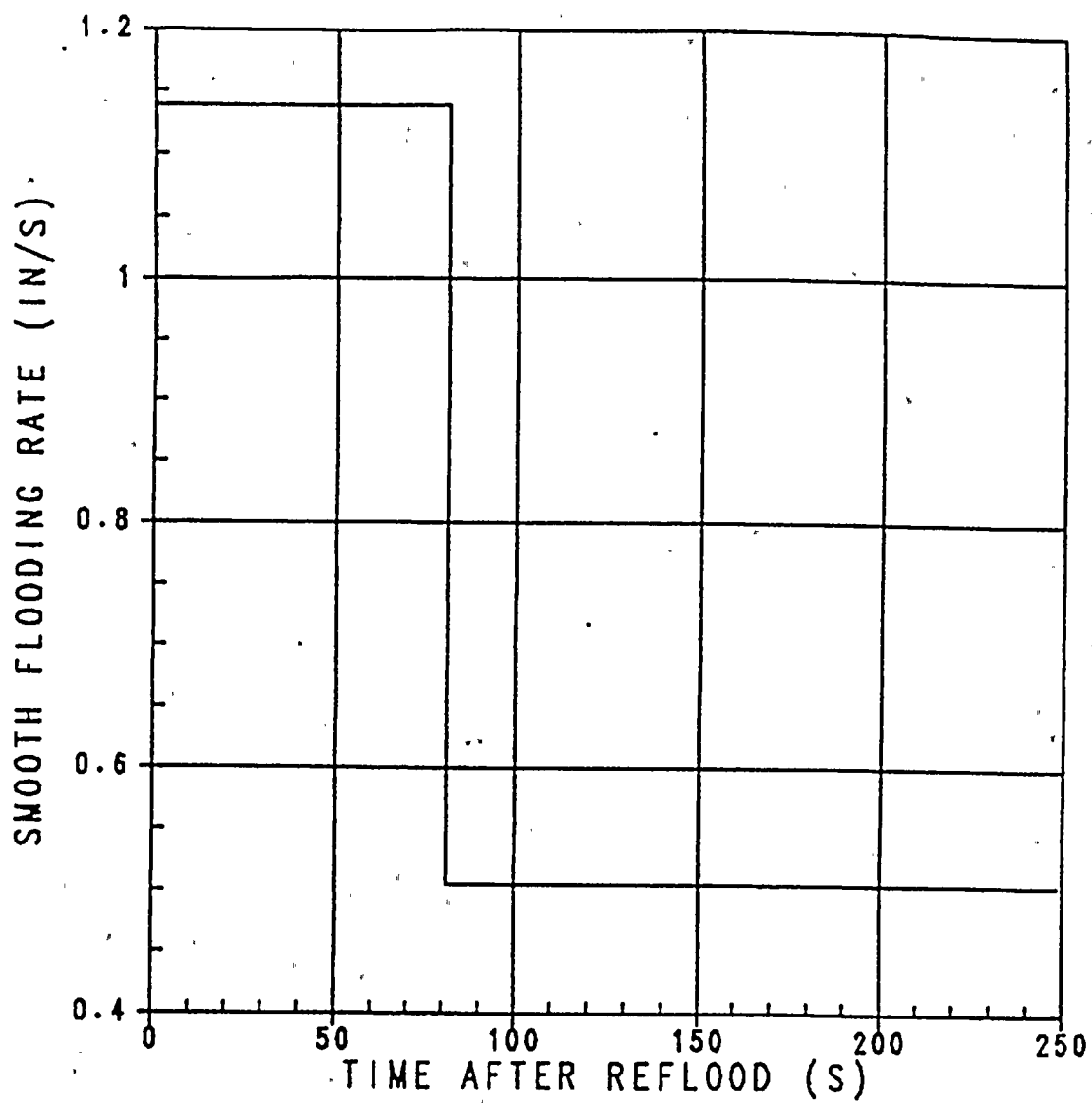


Figure 14.3.1-7f

Core Inlet Flow During Reflood

Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI

Donald C. Cook Unit 1

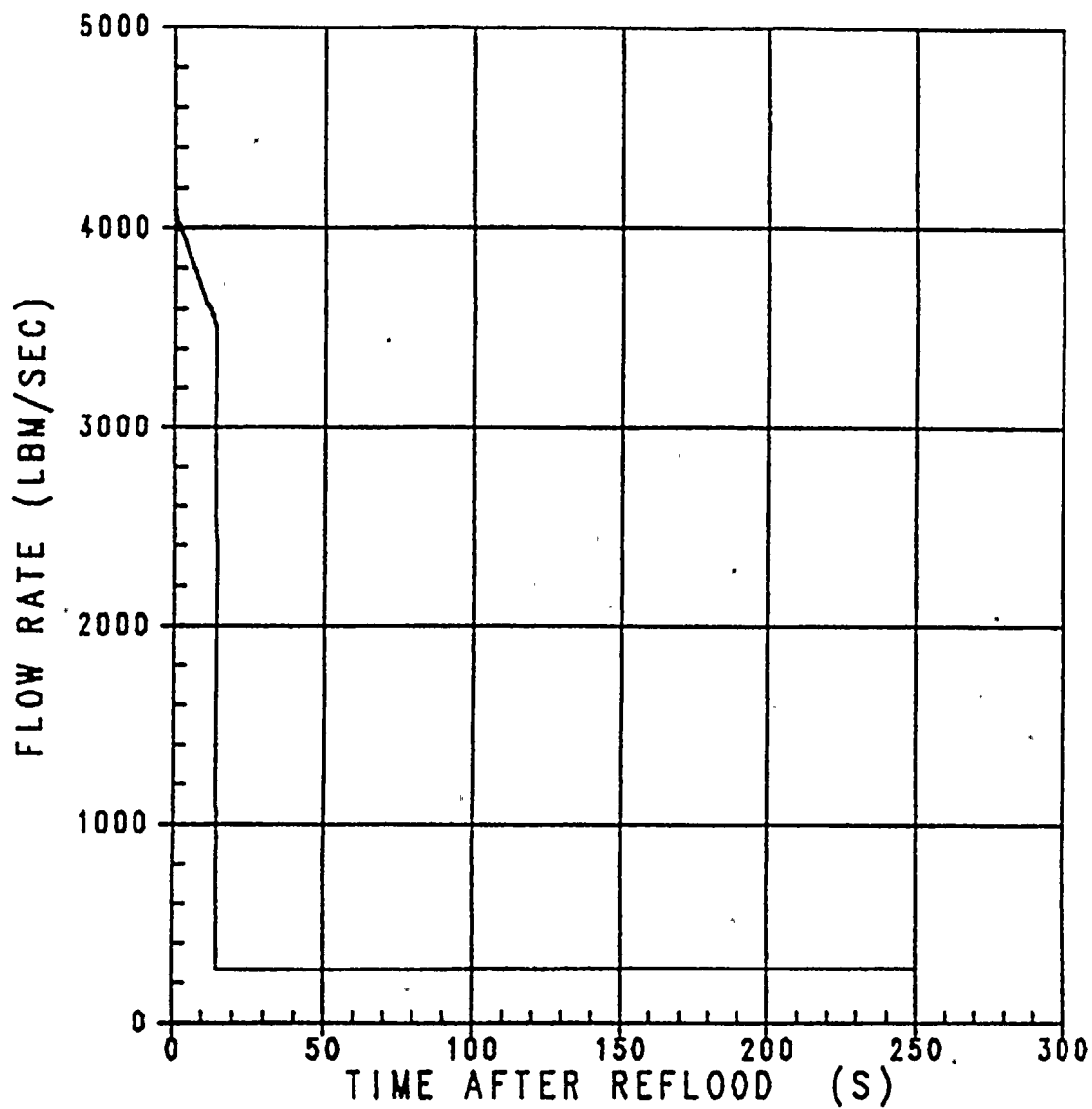


Figure 14.3.1-8a

Accumulator and SI Flow During Reflood
Case A, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

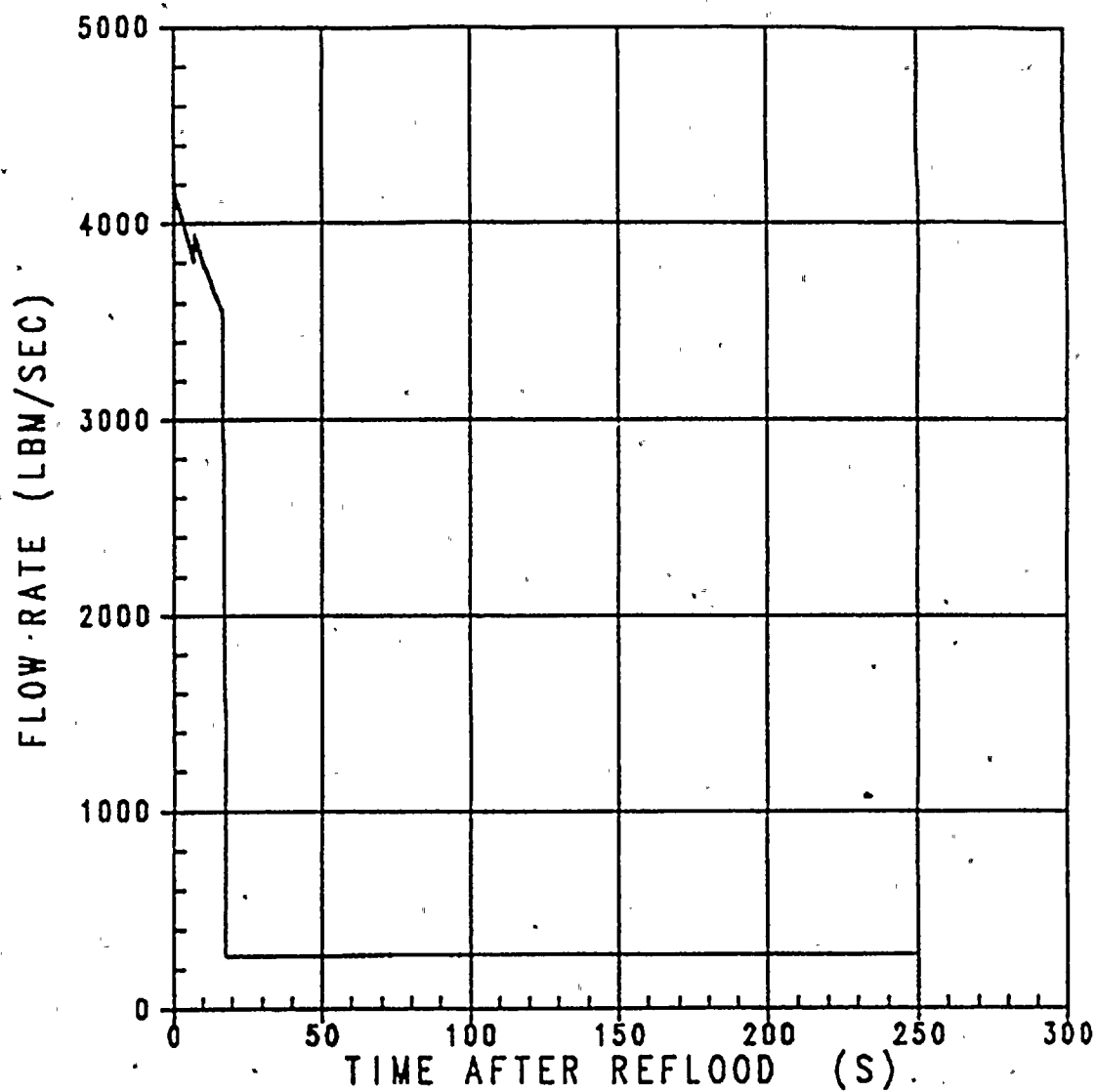


Figure 14.3.1-8b Accumulator and SI Flow During Reflood
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

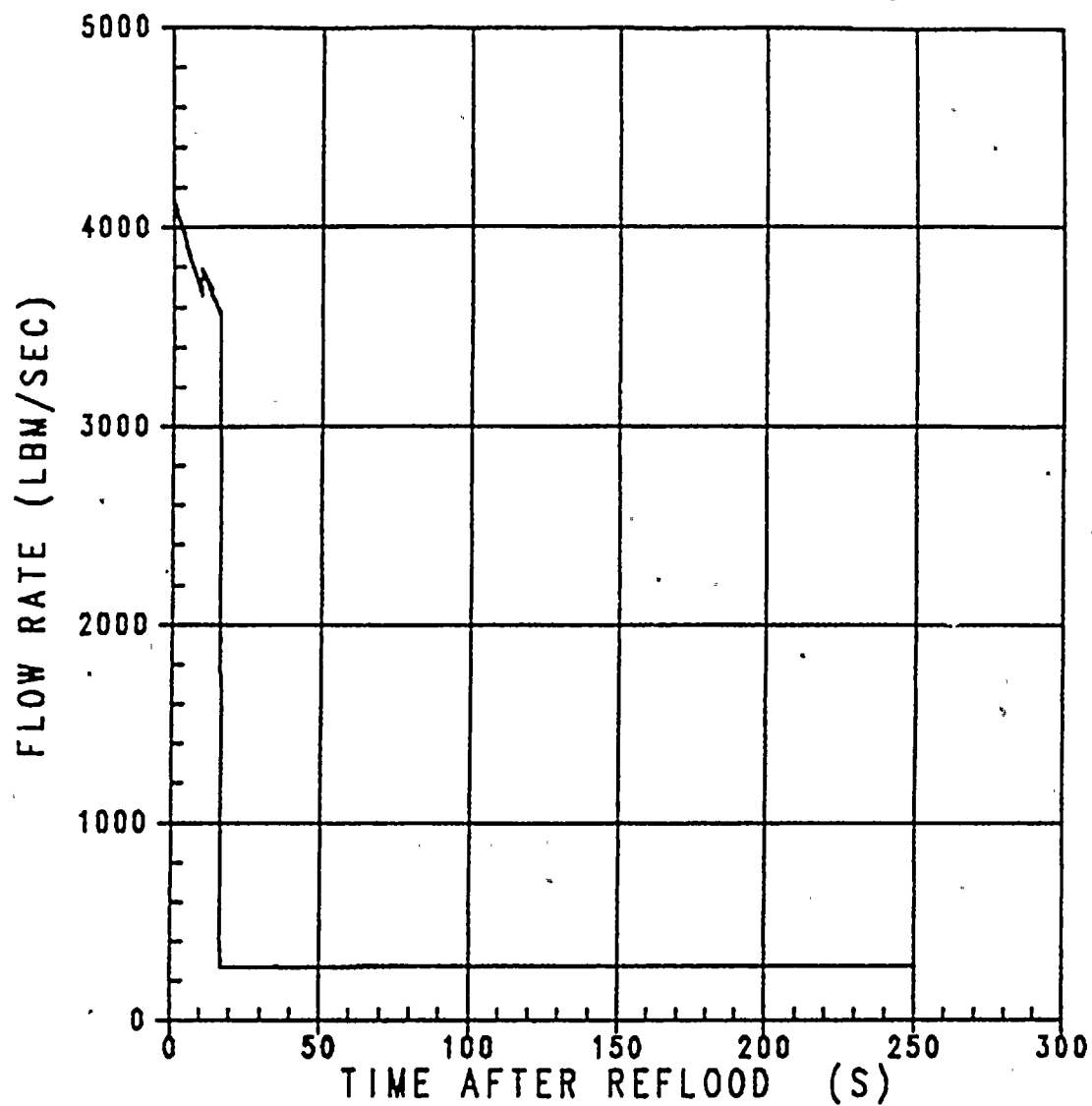


Figure 14.3.1-8c Accumulator and SI Flow During Reflood
Case C, $CD=0.8$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

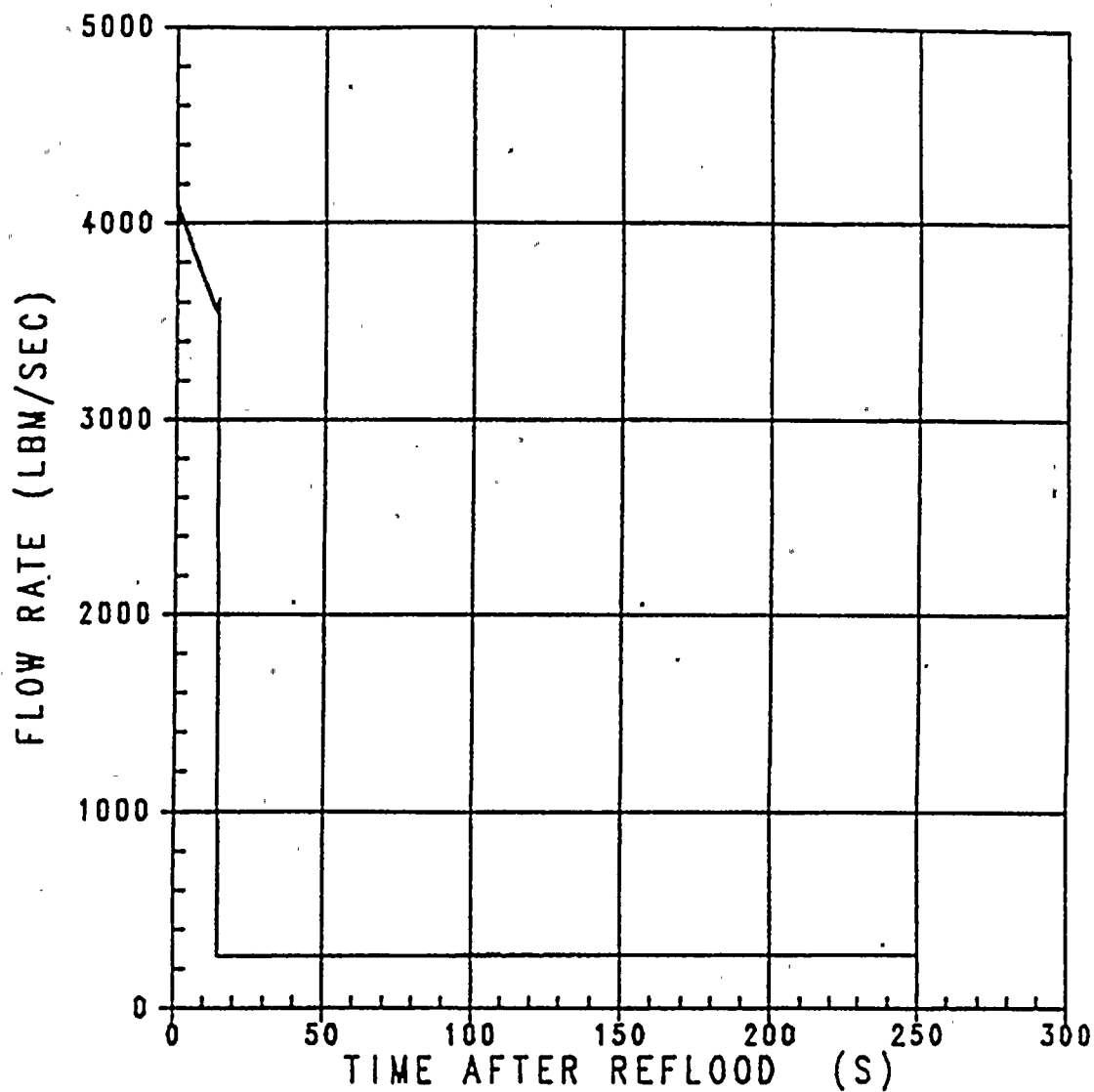


Figure 14.3.1-8d Accumulator and SI Flow During Reflood
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

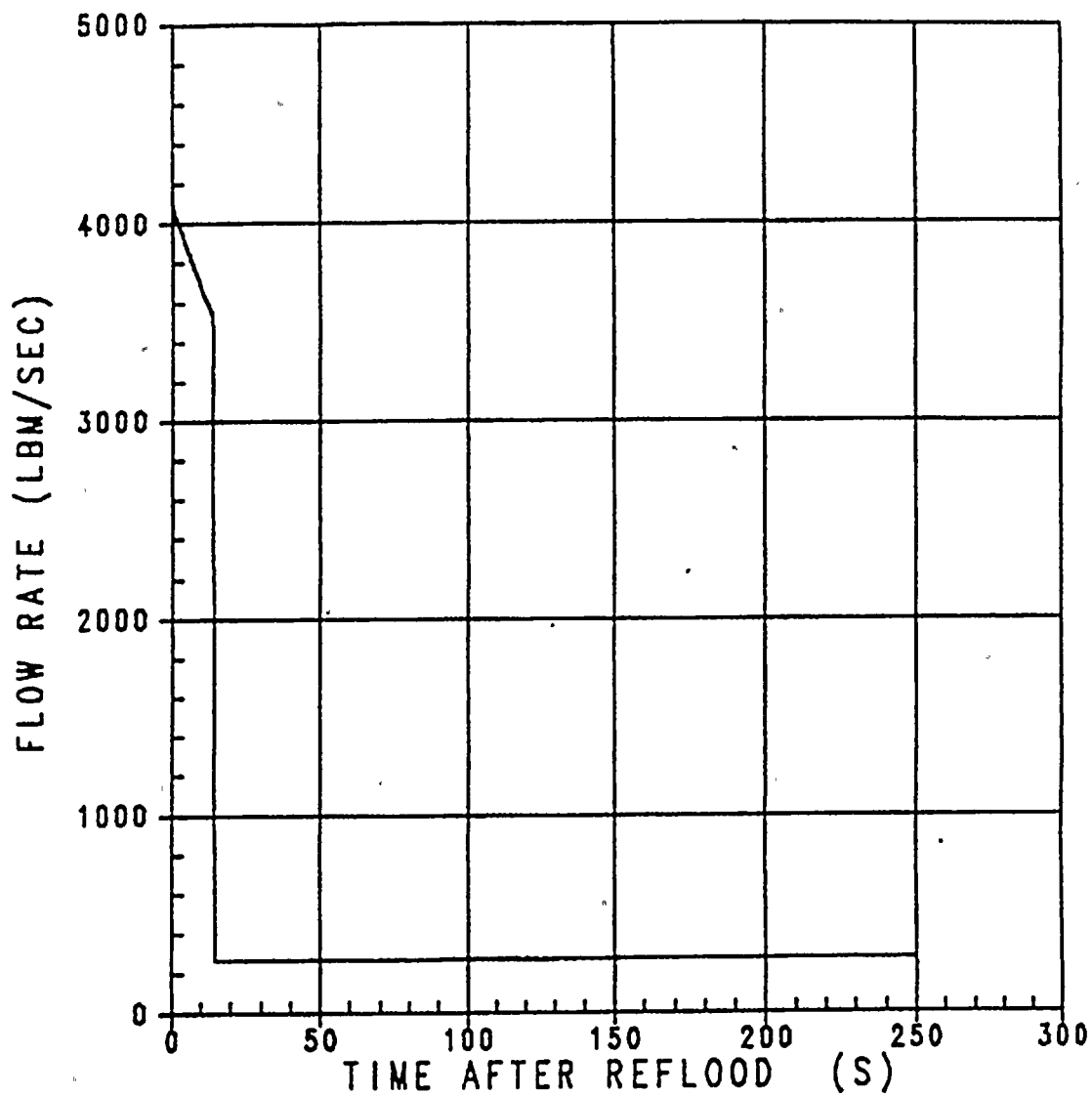


Figure 14.3.1-8e Accumulator and SI Flow During Reflood
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

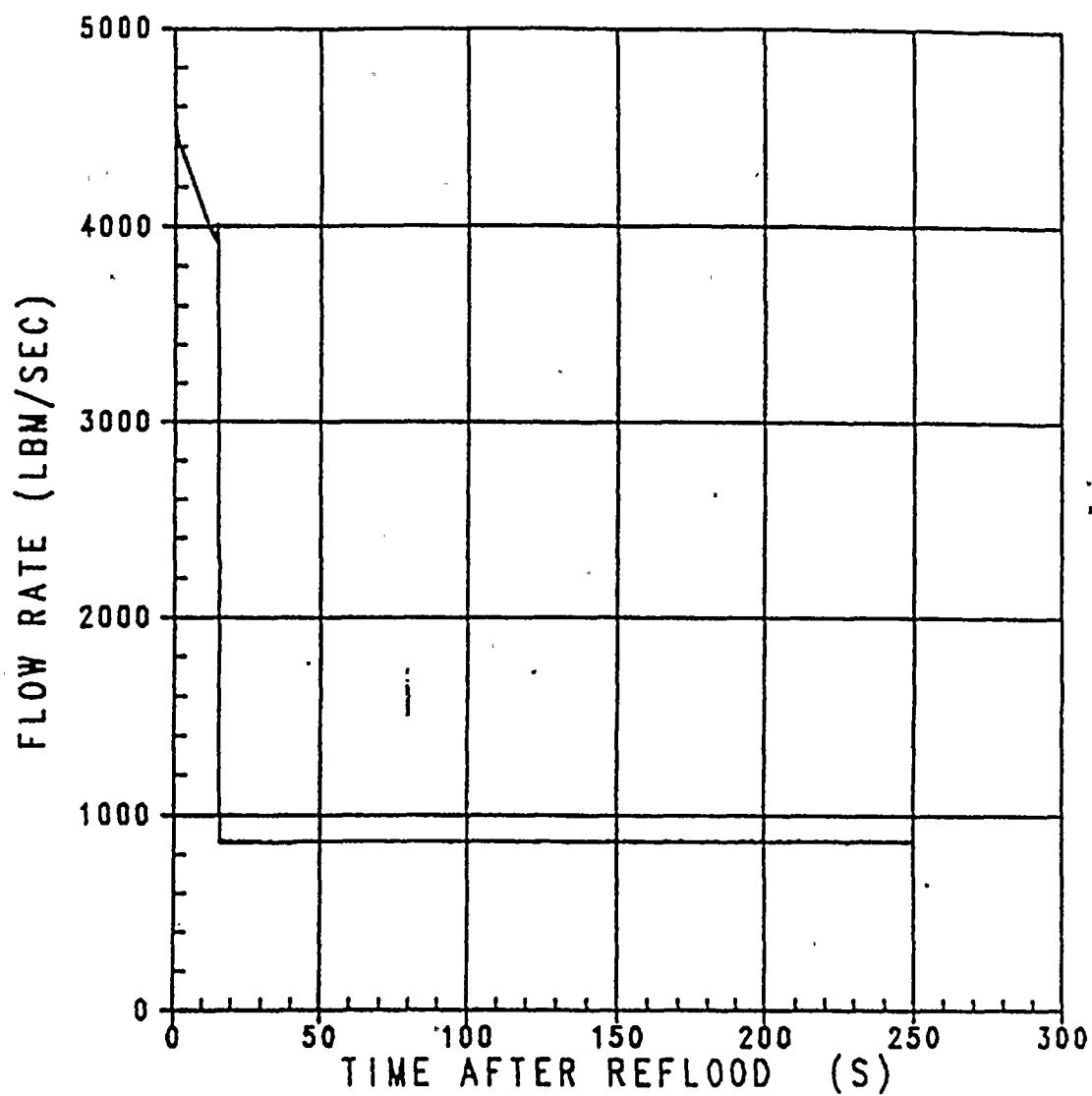


Figure 14.3.1-8f Accumulator and SI Flow During Reflood
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

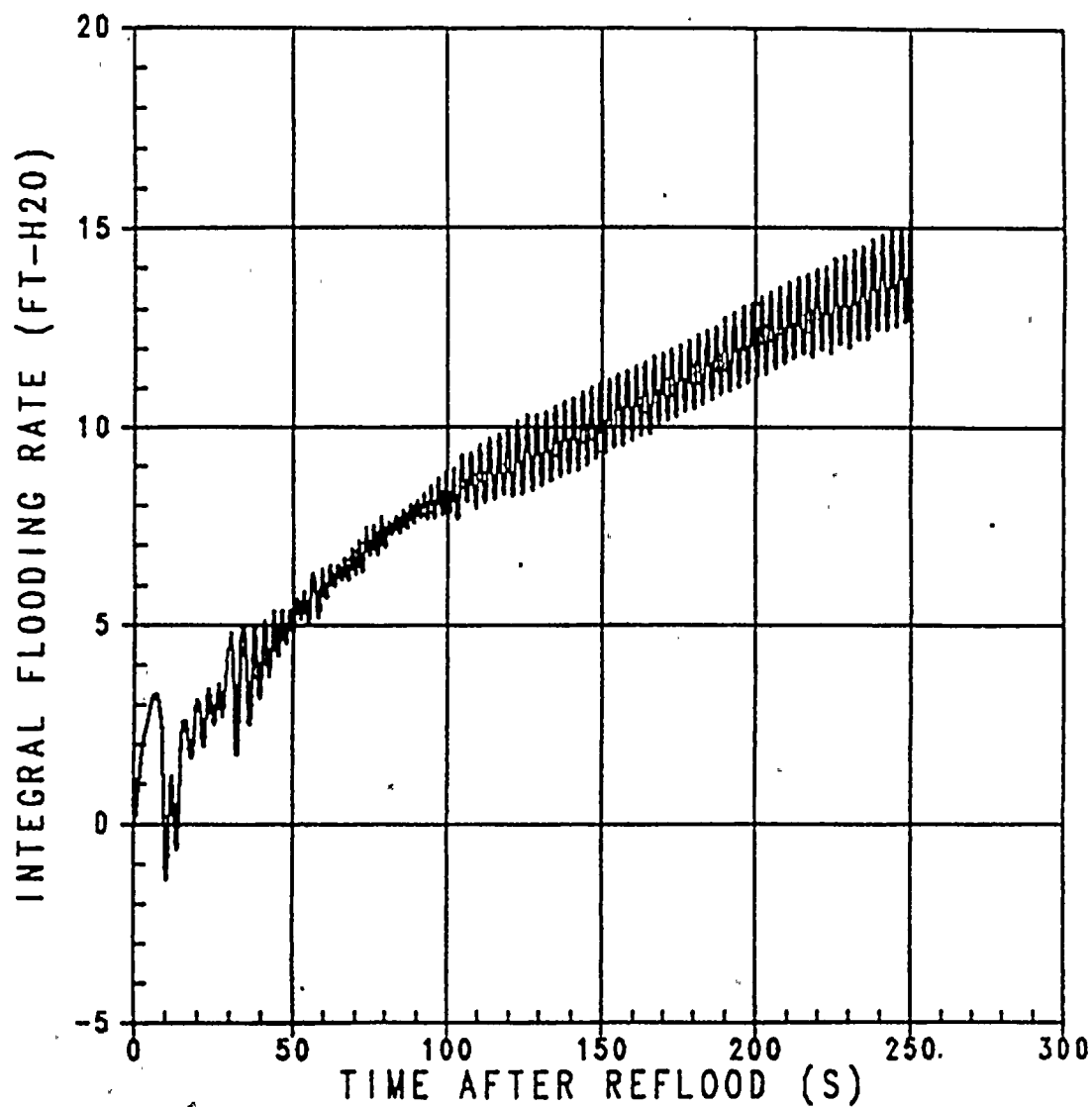


Figure 14.3.1-9a Integral of Core Inlet Flow
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

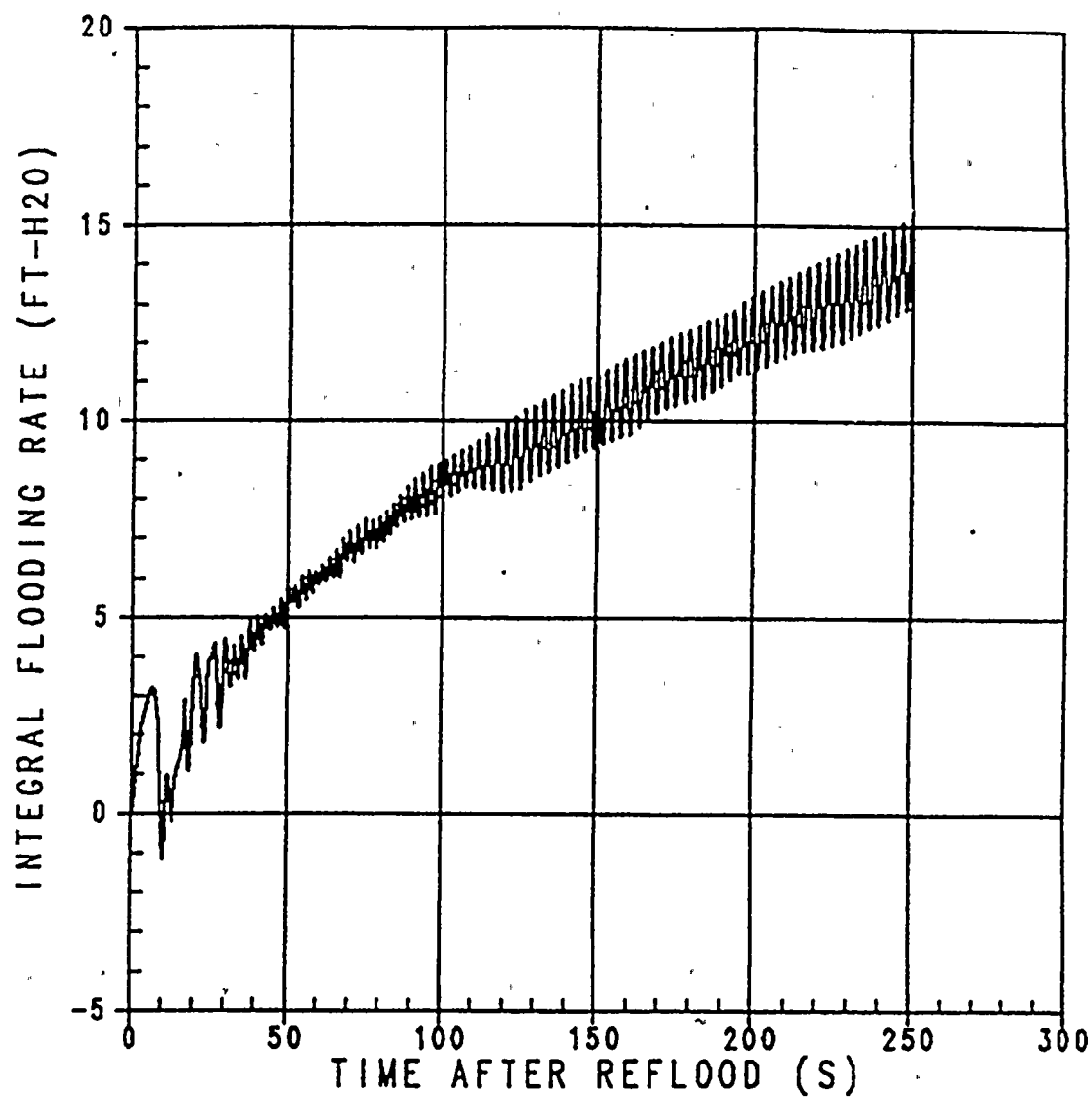


Figure 14.3.1-9b

Integral of Core Inlet Flow

Case B, $CD=0.6$, $Thot=609.1^{\circ}F$, $P=2250$ psia

Donald C. Cook Unit 1

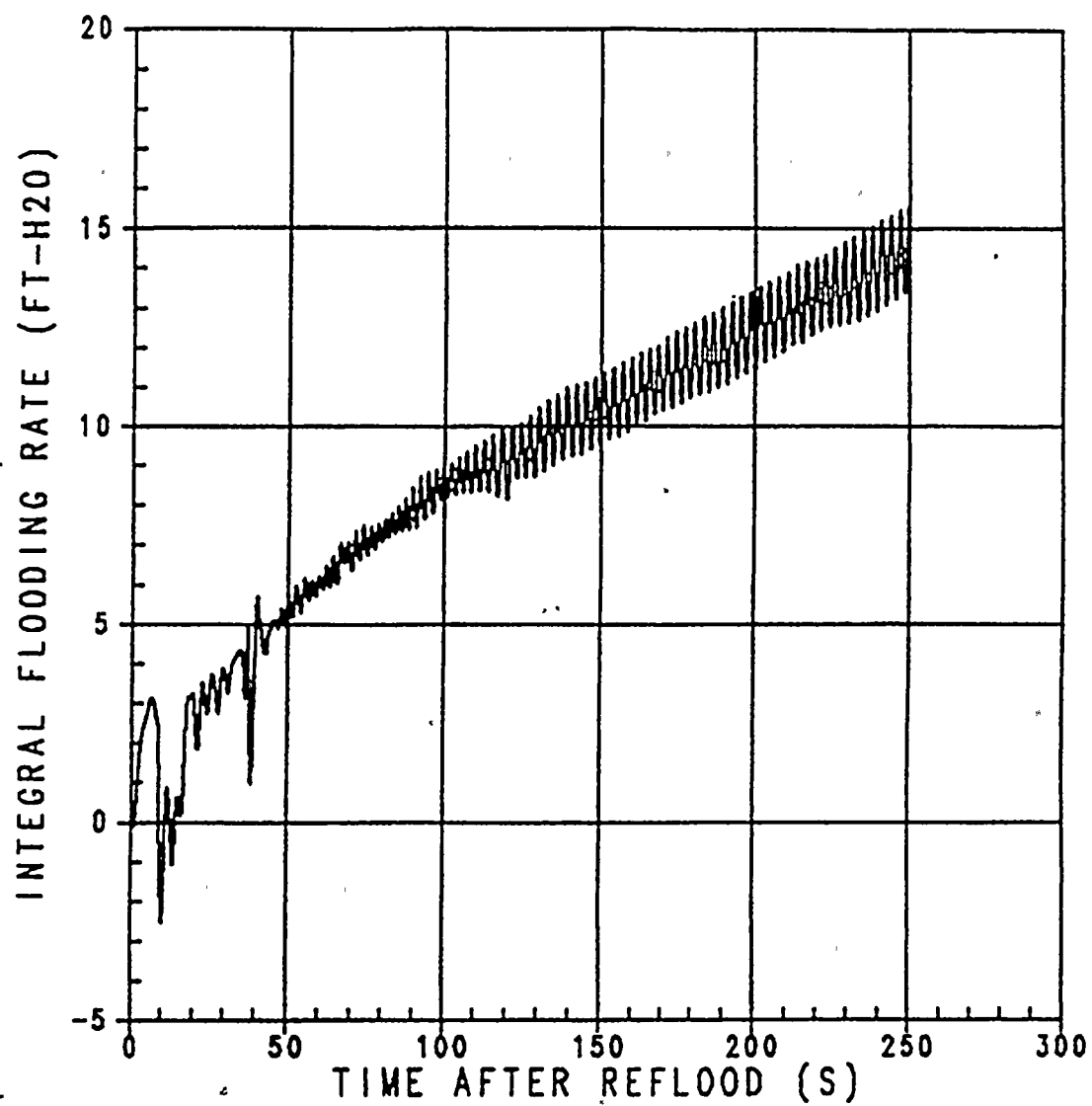


Figure 14.3.1-9c Integral of Core Inlet Flow
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

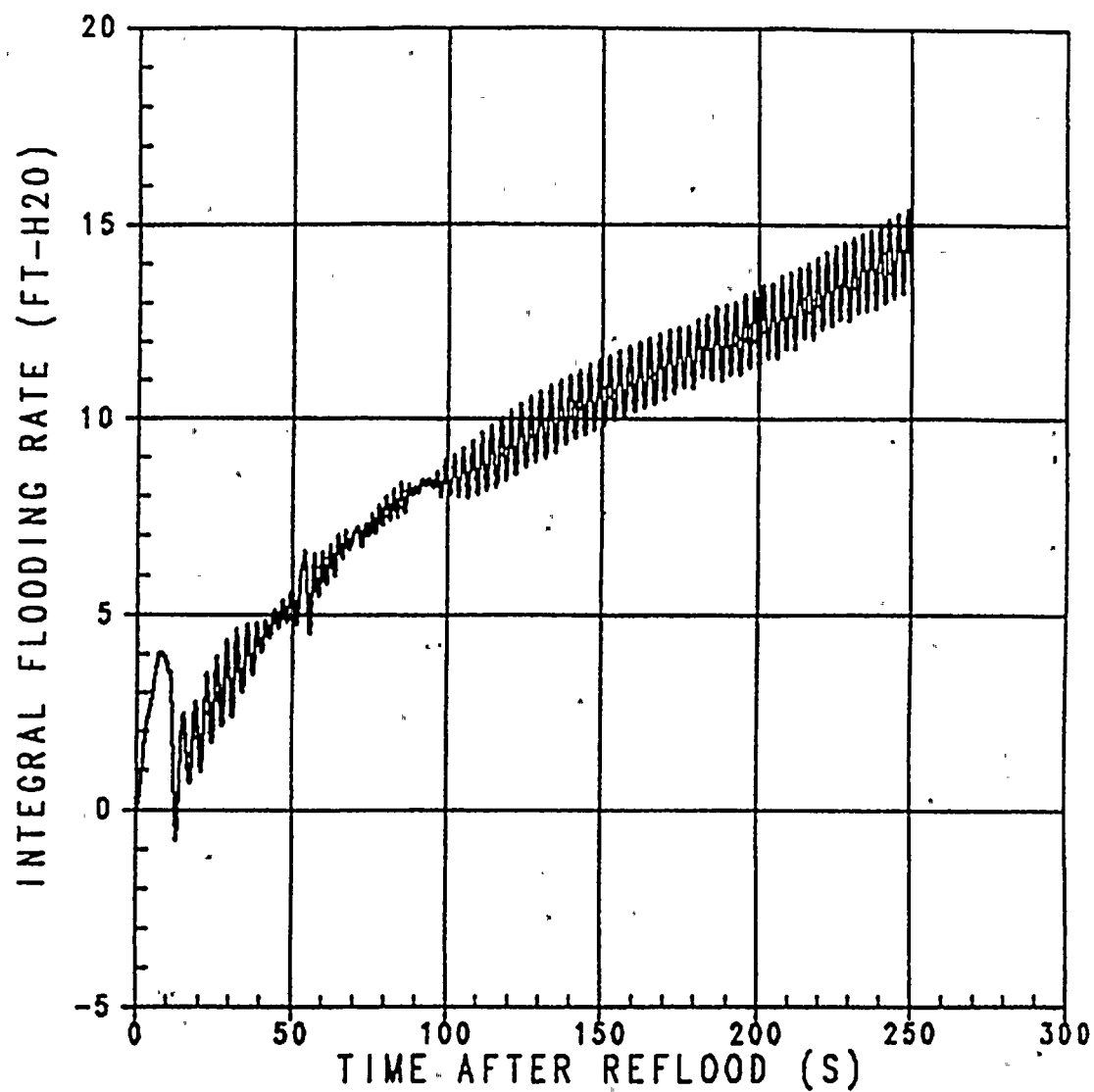


Figure 14.3.1-9d Integral of Core Inlet Flow
Case D, $CD=0.4$, $T_{hot}=586.8^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

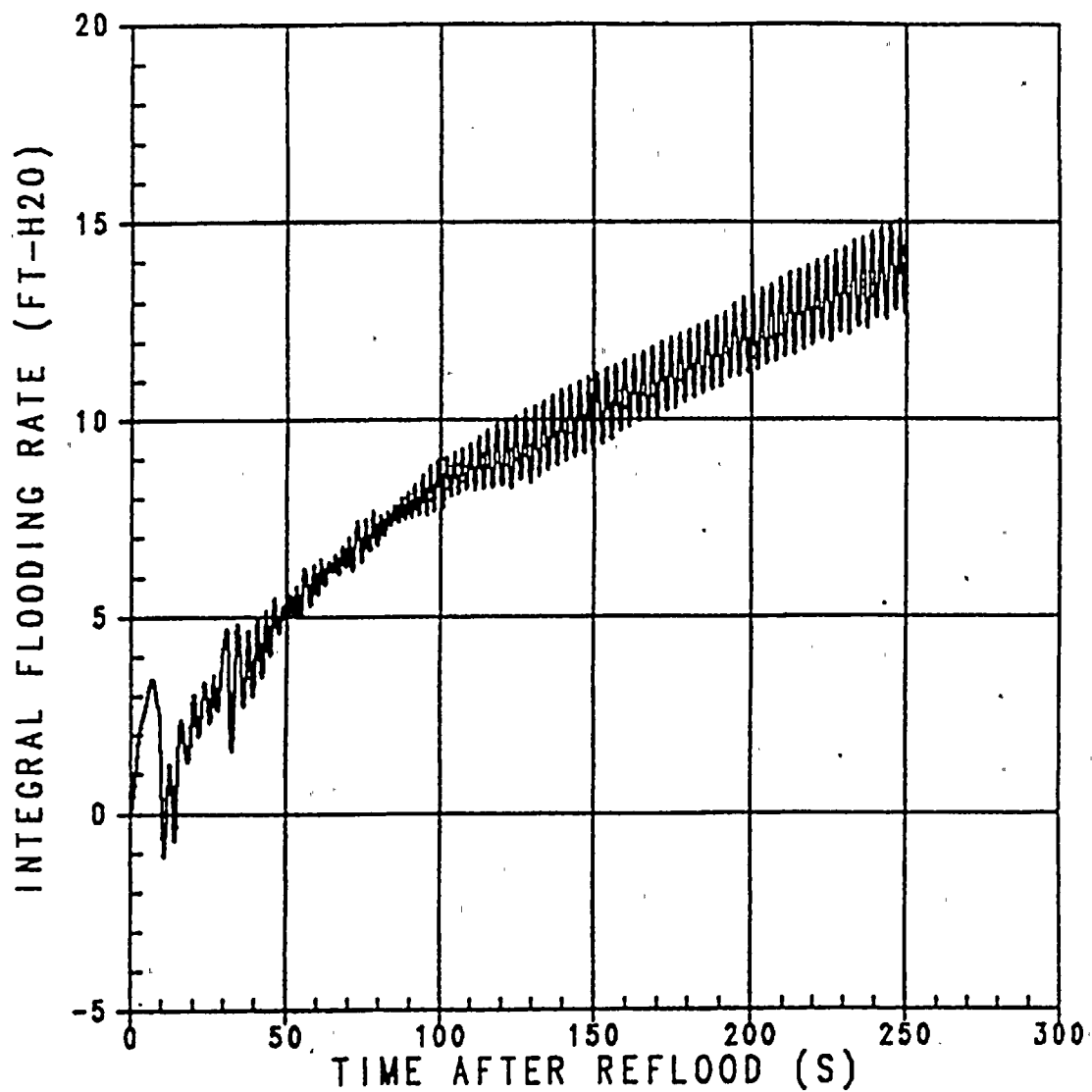


Figure 14.3.1-9e

Integral of Core Inlet Flow
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

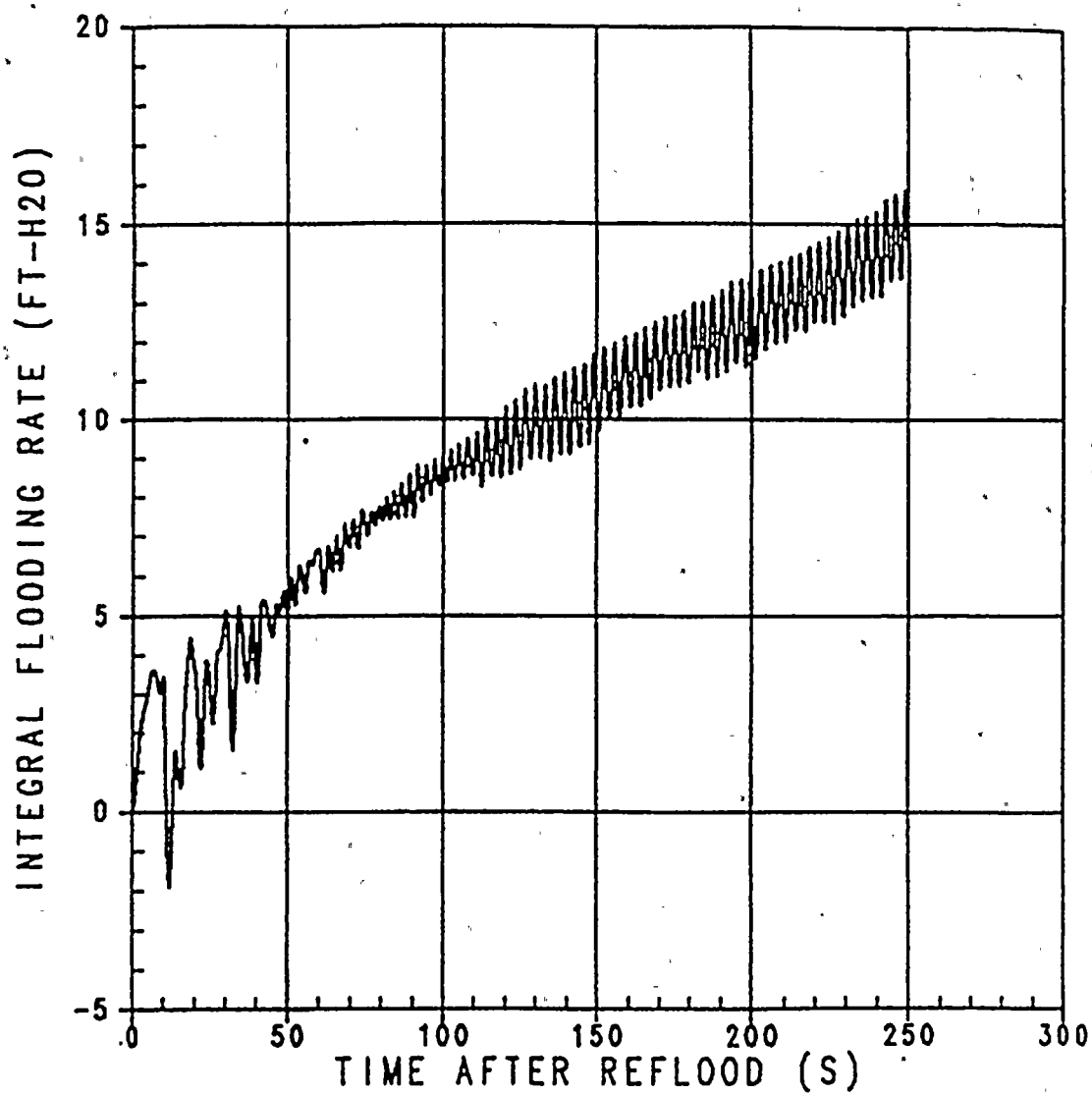


Figure 14.3.1-9f

Integral of Core Inlet Flow

Case F, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia, max SI

Donald C. Cook Unit 1

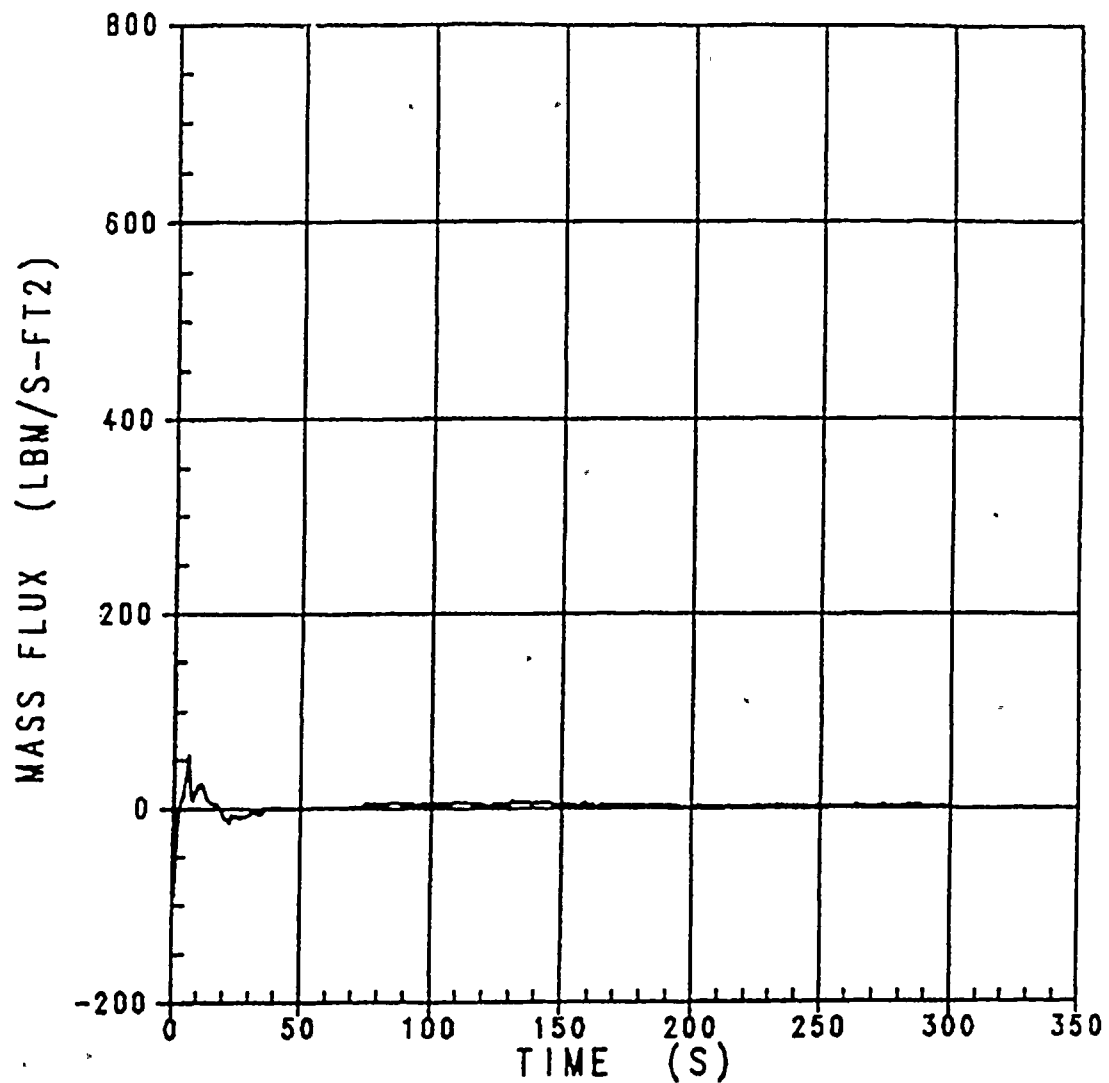


Figure 14.3.1-10a Mass Flux at Peak Temperature Elevation
Case A, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

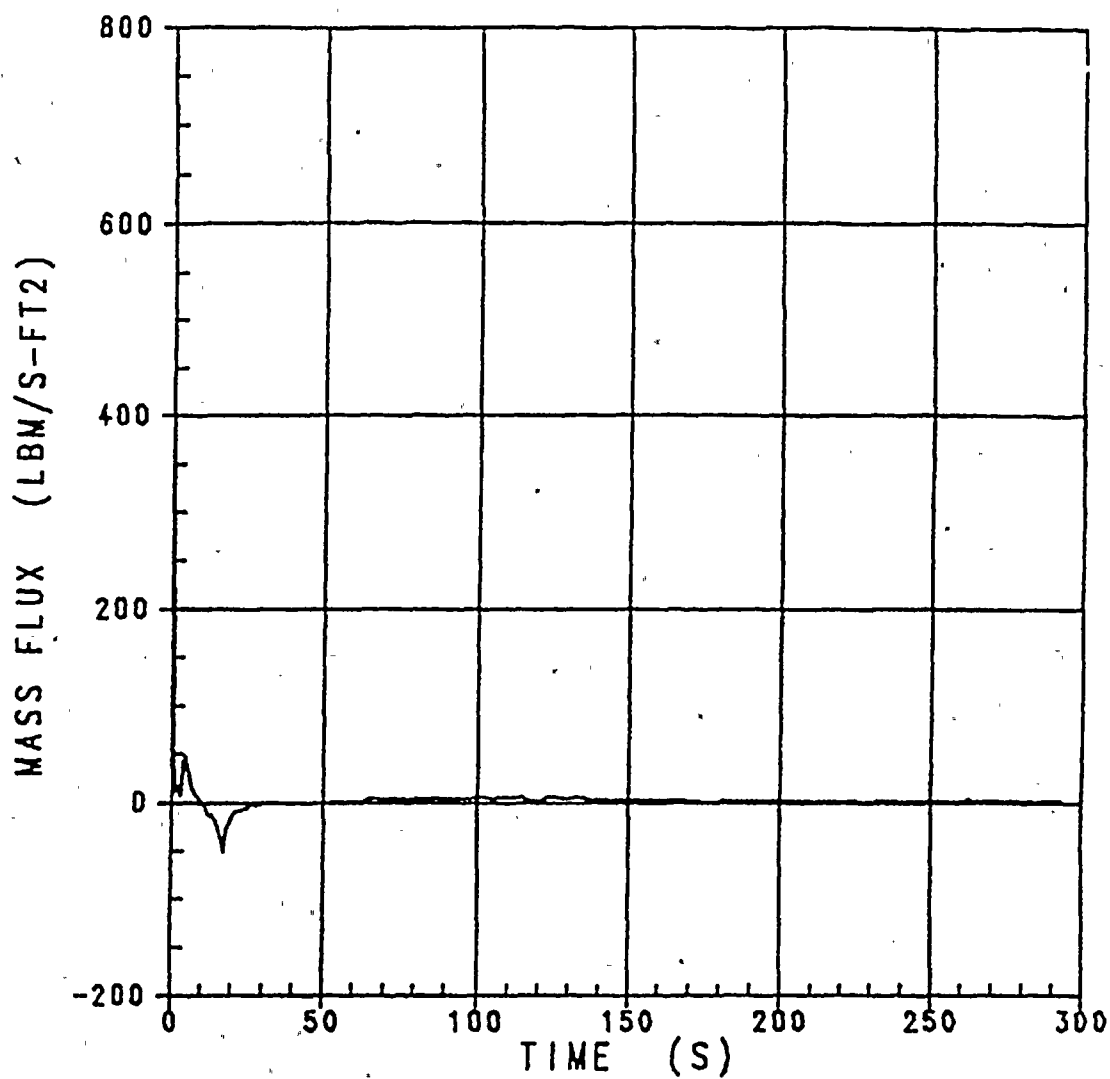


Figure 14.3.1-10b Mass Flux at Peak Temperature Elevation
Case B, $CD=0.6$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

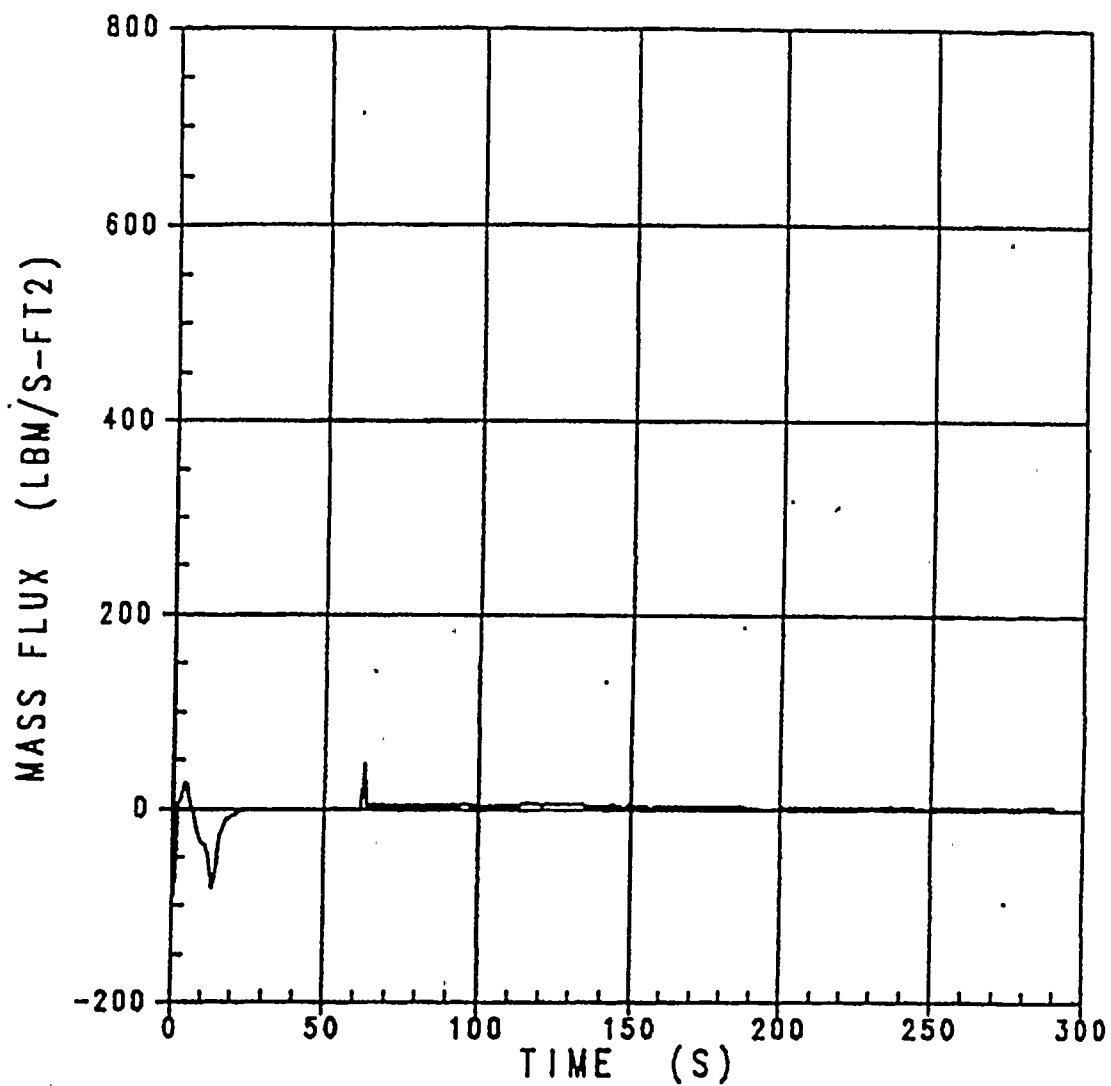


Figure 14.3.1-10c Mass Flux at Peak Temperature Elevation
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

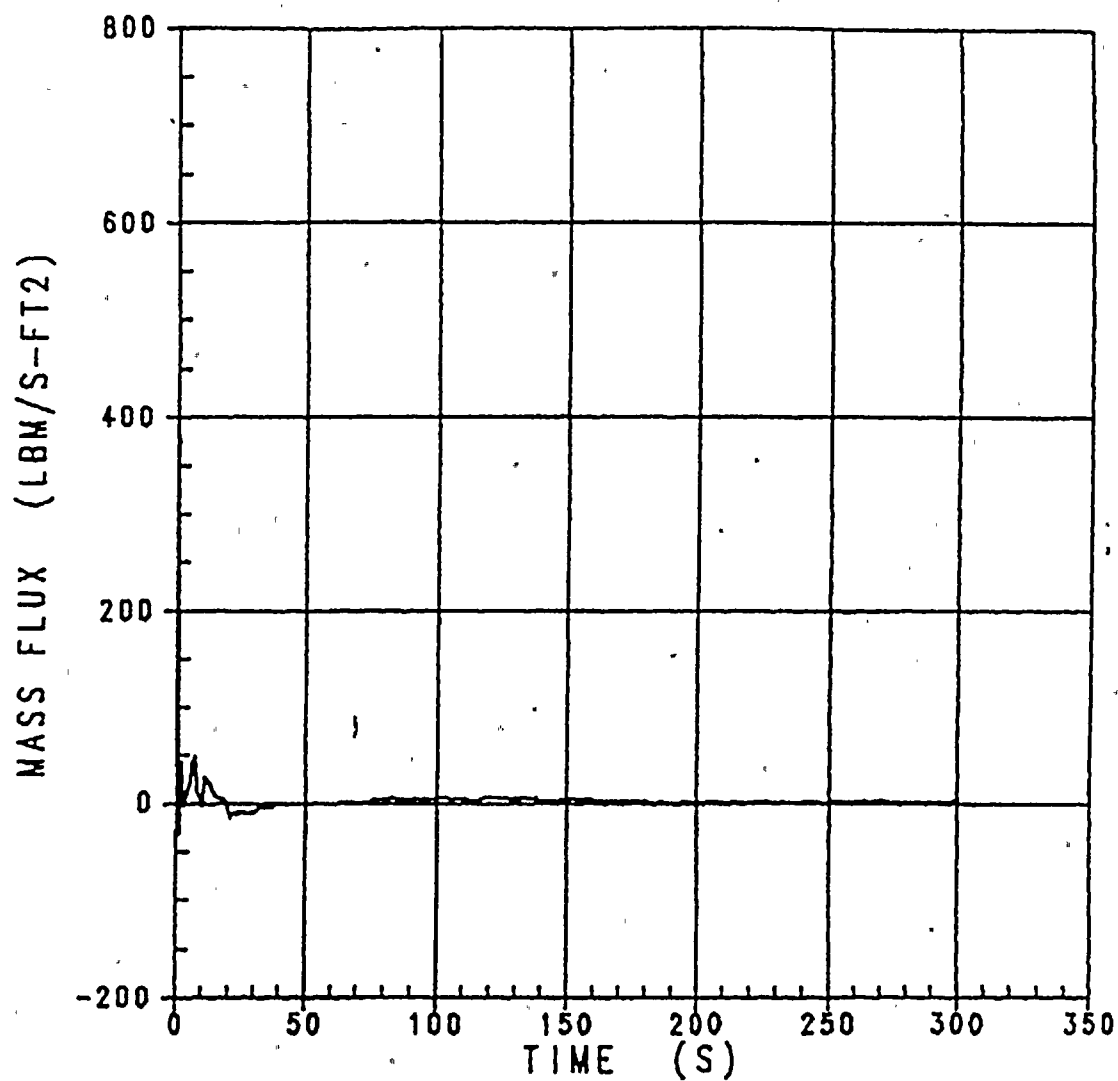


Figure 14.3.1-10d Mass Flux at Peak Temperature Elevation
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

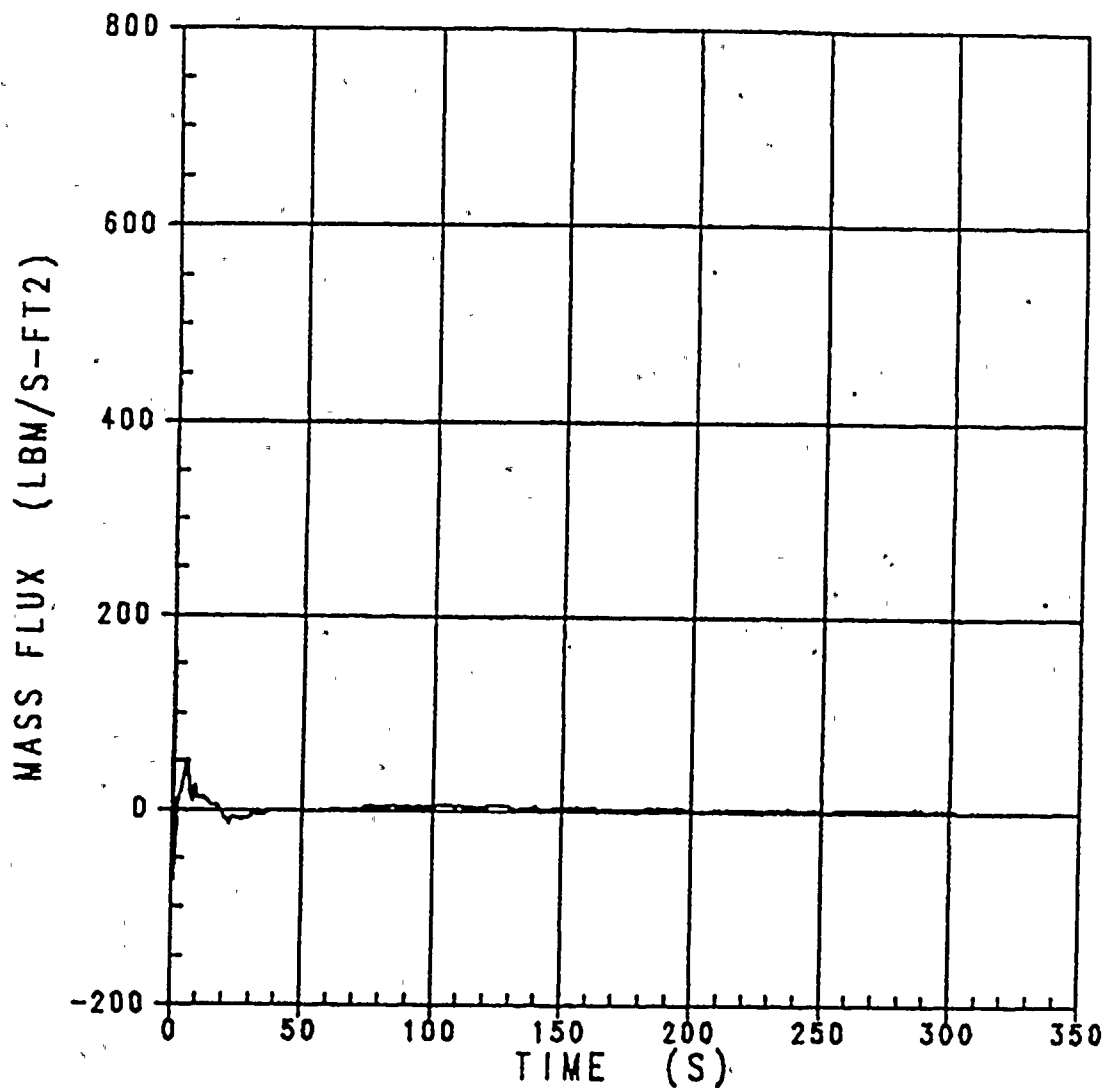


Figure 14.3.1-10e Mass Flux at Peak Temperature Elevation
Case E, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

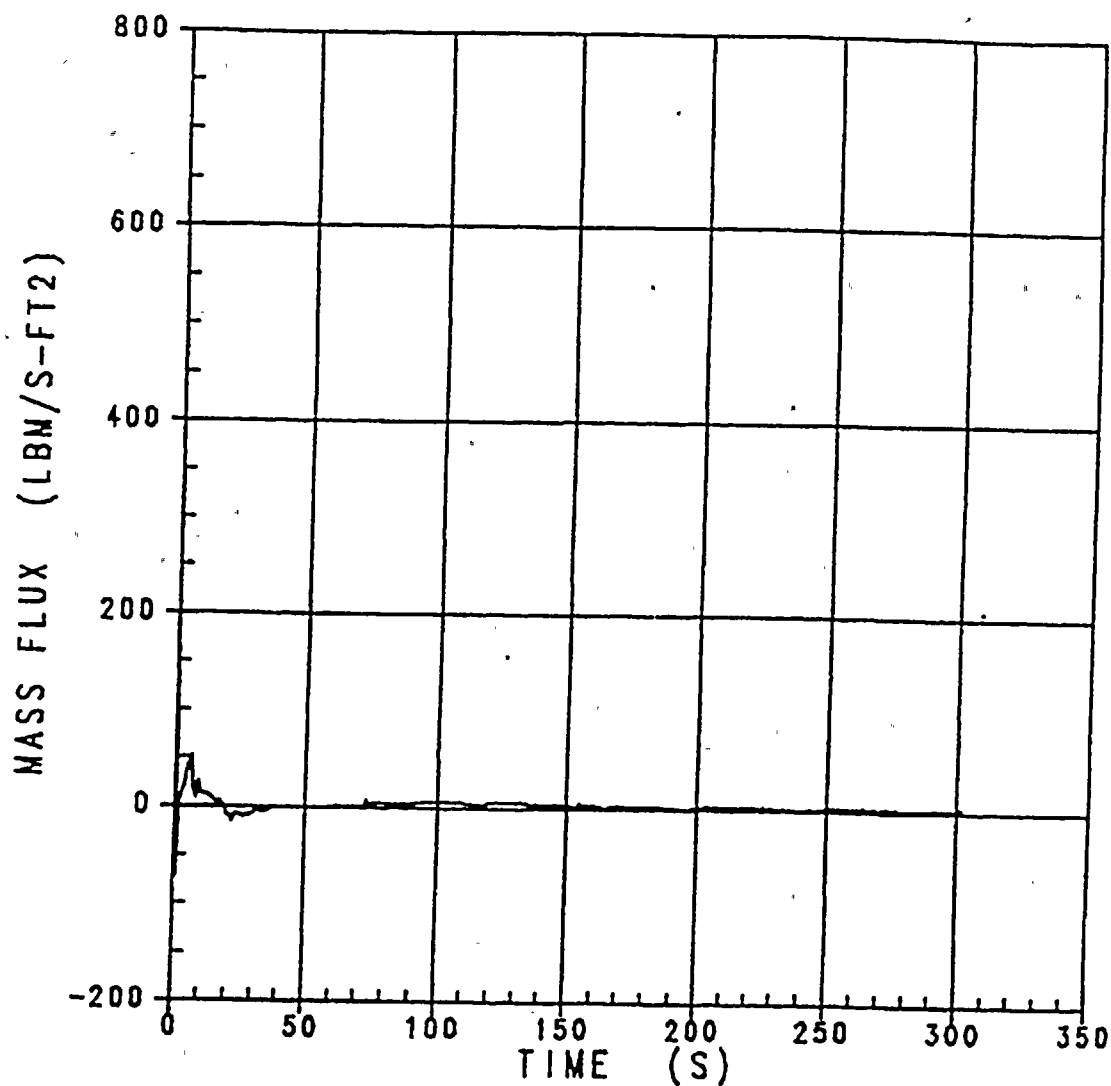


Figure 14.3.1-10f Mass Flux at Peak Temp. Elevation
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

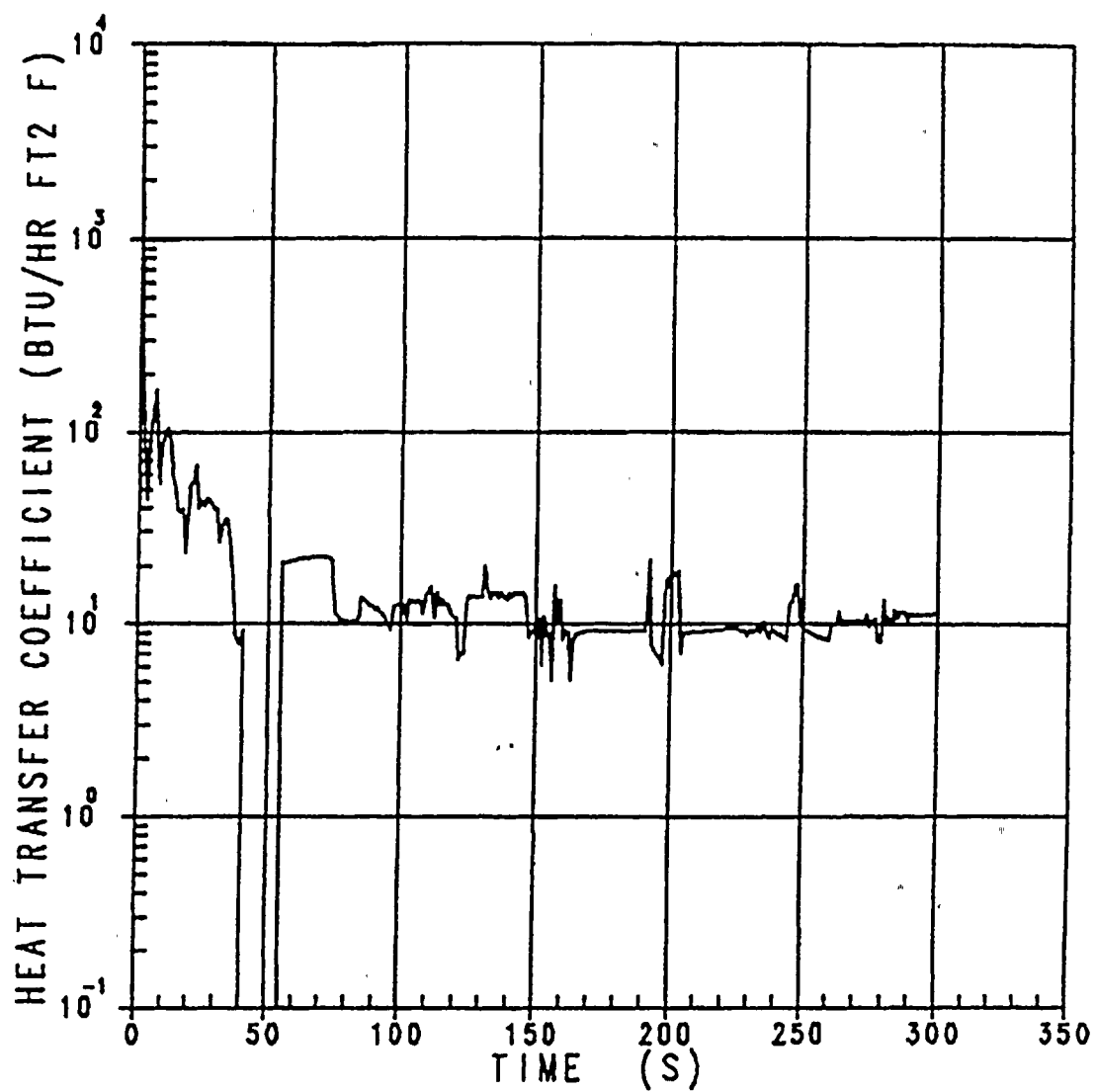


Figure 14.3.1-11a Rod H.T.C. at Peak Temperature Elevation
Case A, CD=0.4, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

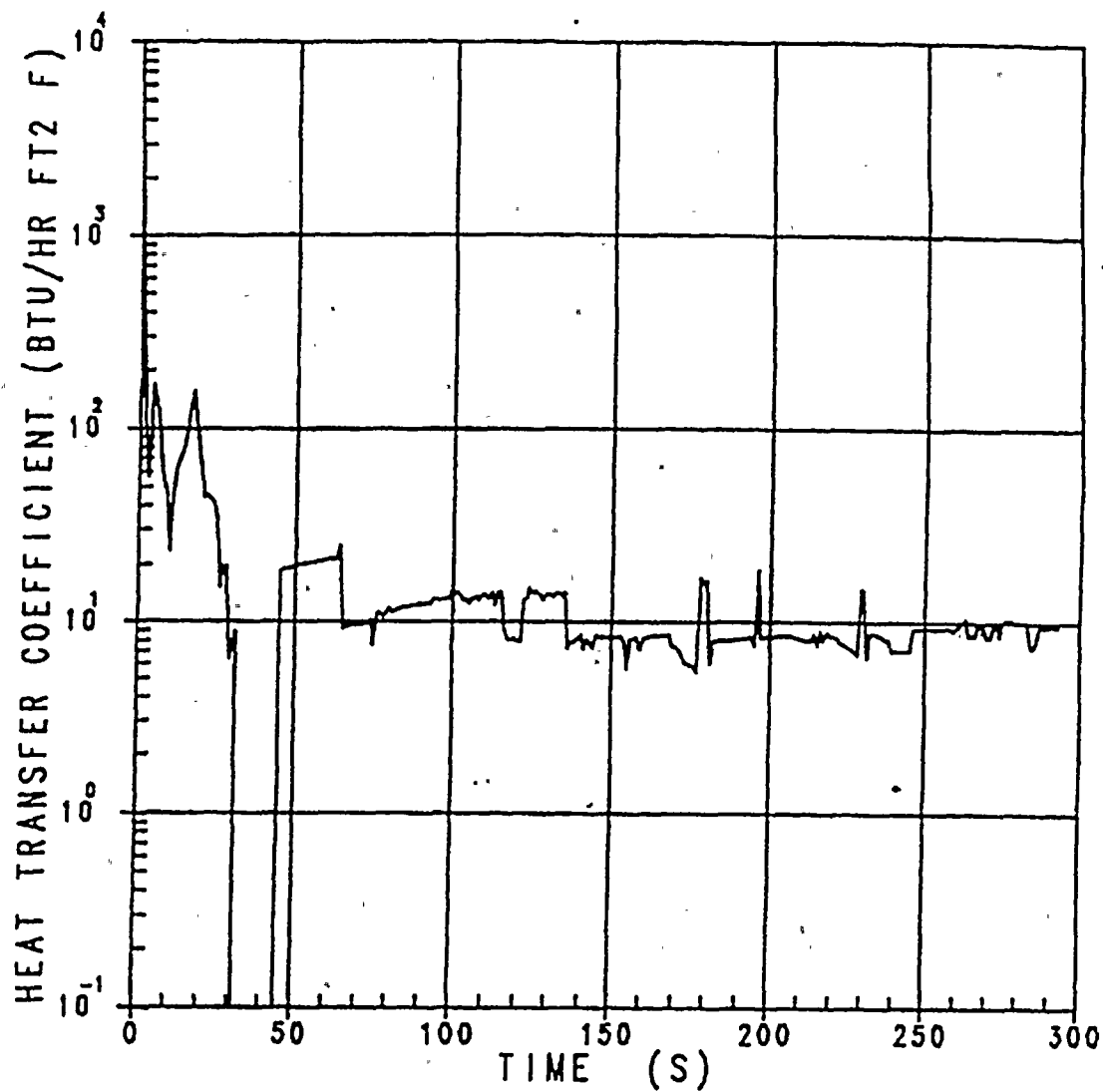


Figure 14.3.1-11b Rod H.T.C. at Peak Temperature Elevation
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1.

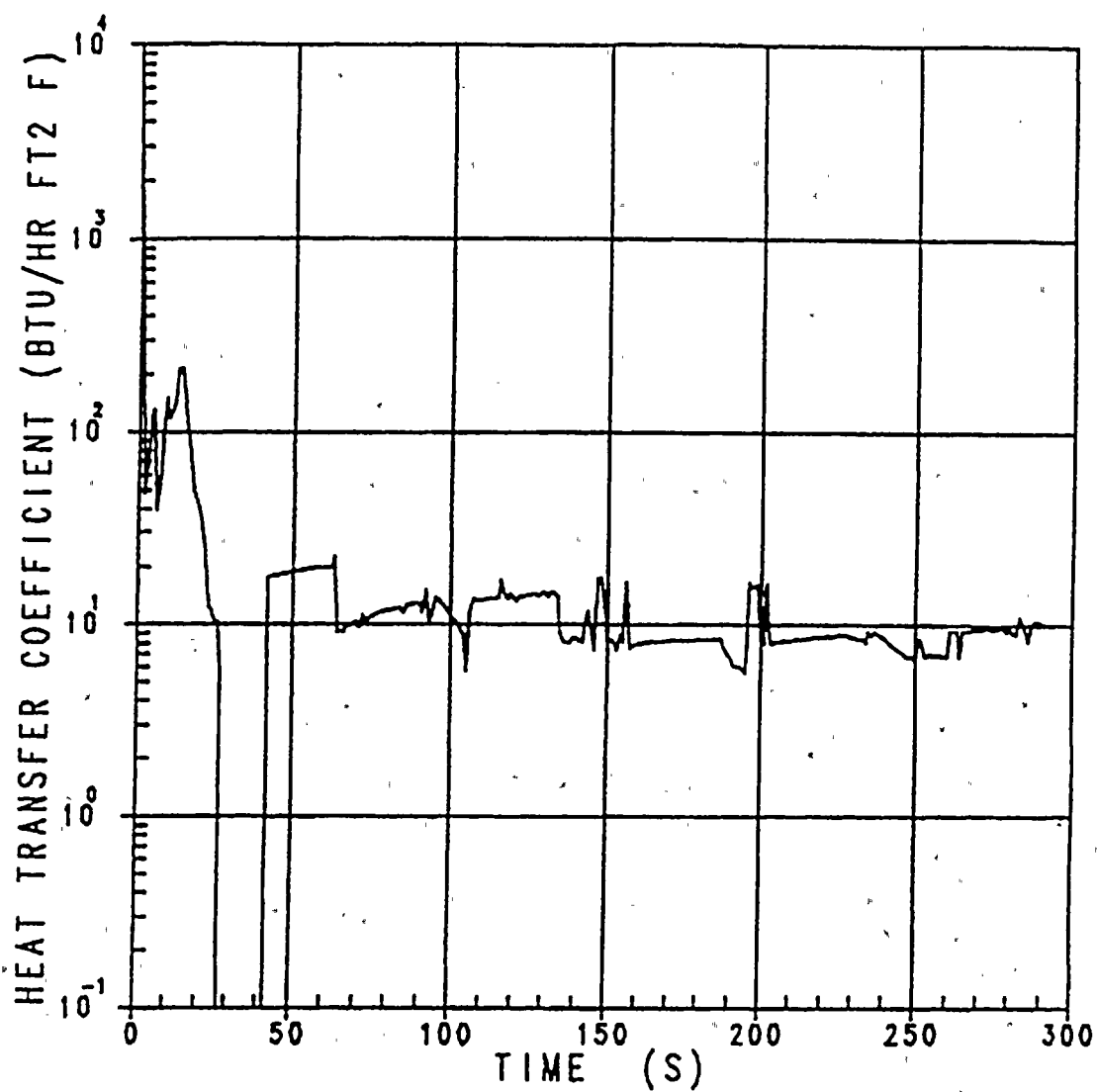


Figure 14.3.1-11c Rod H.T.C. at Peak Temperature Elevation
Case C, $CD=0.8$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

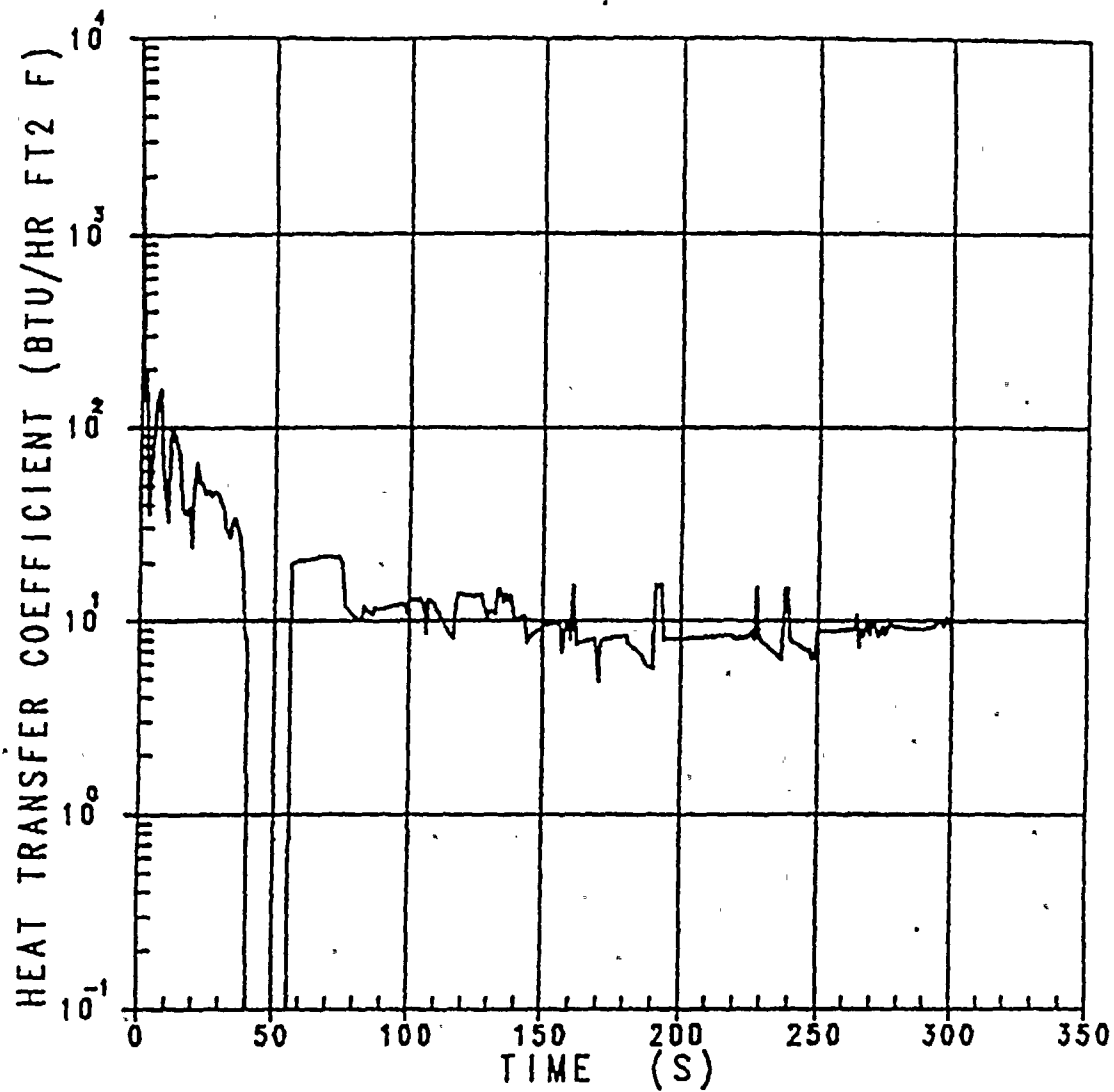


Figure 14.3.1-11d Rod H.T.C. at Peak Temperature Elevation
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

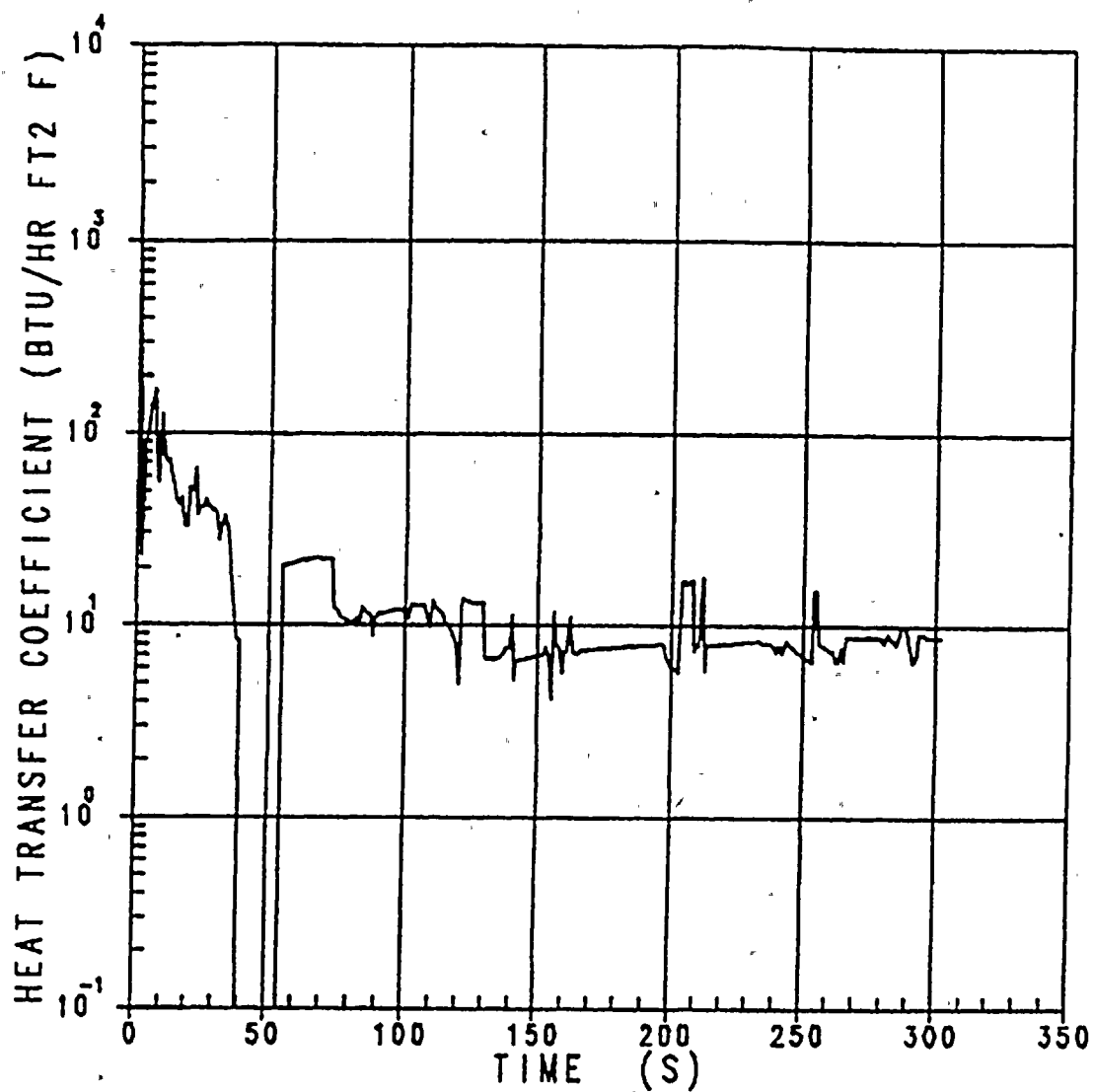


Figure 14.3.1-11e Rod H.T.C. at Peak Temperature Elevation
Case E, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1

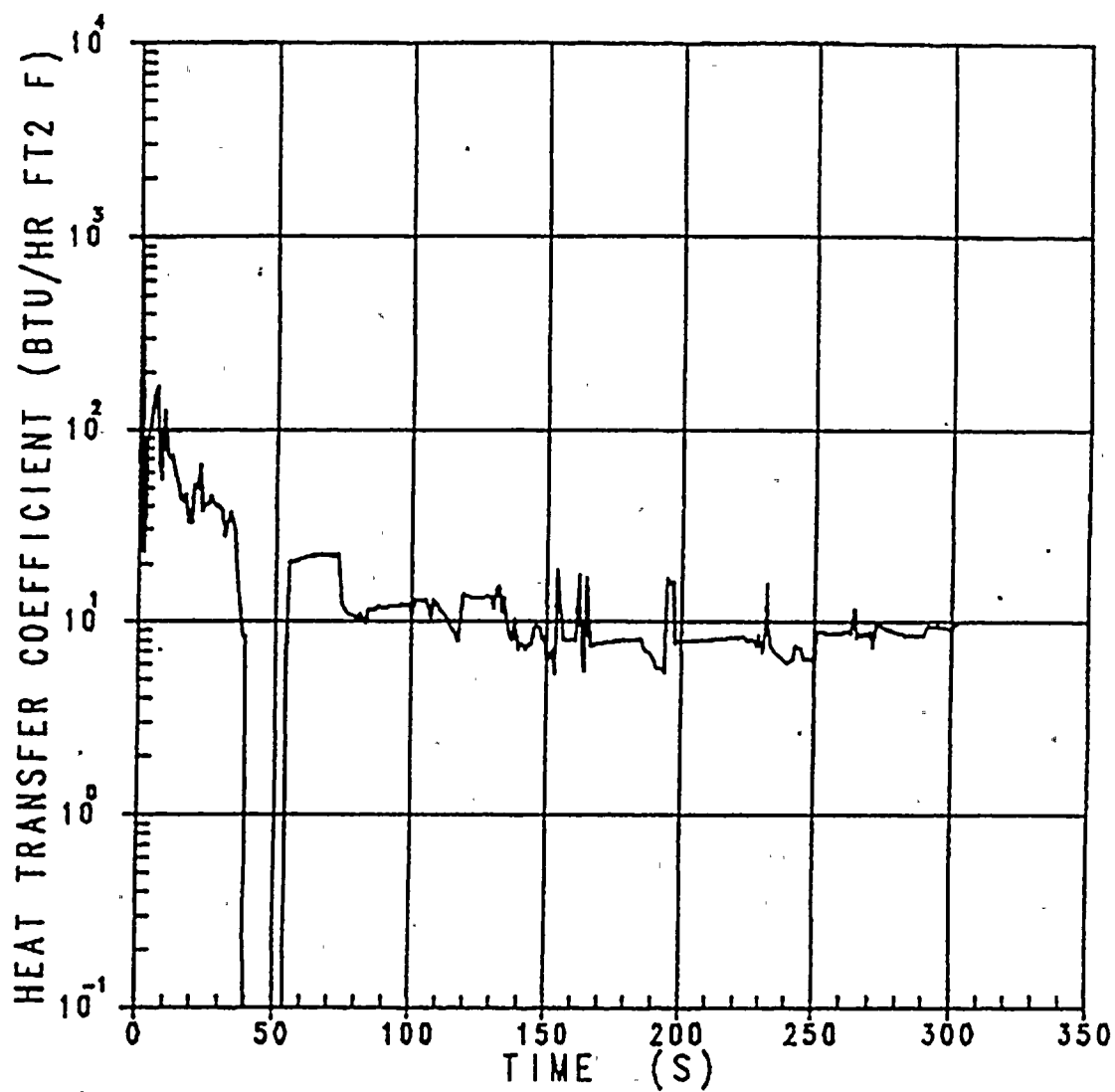


Figure 14.3.1-11f Rod H.T.C. at Peak Temp. Elevation
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

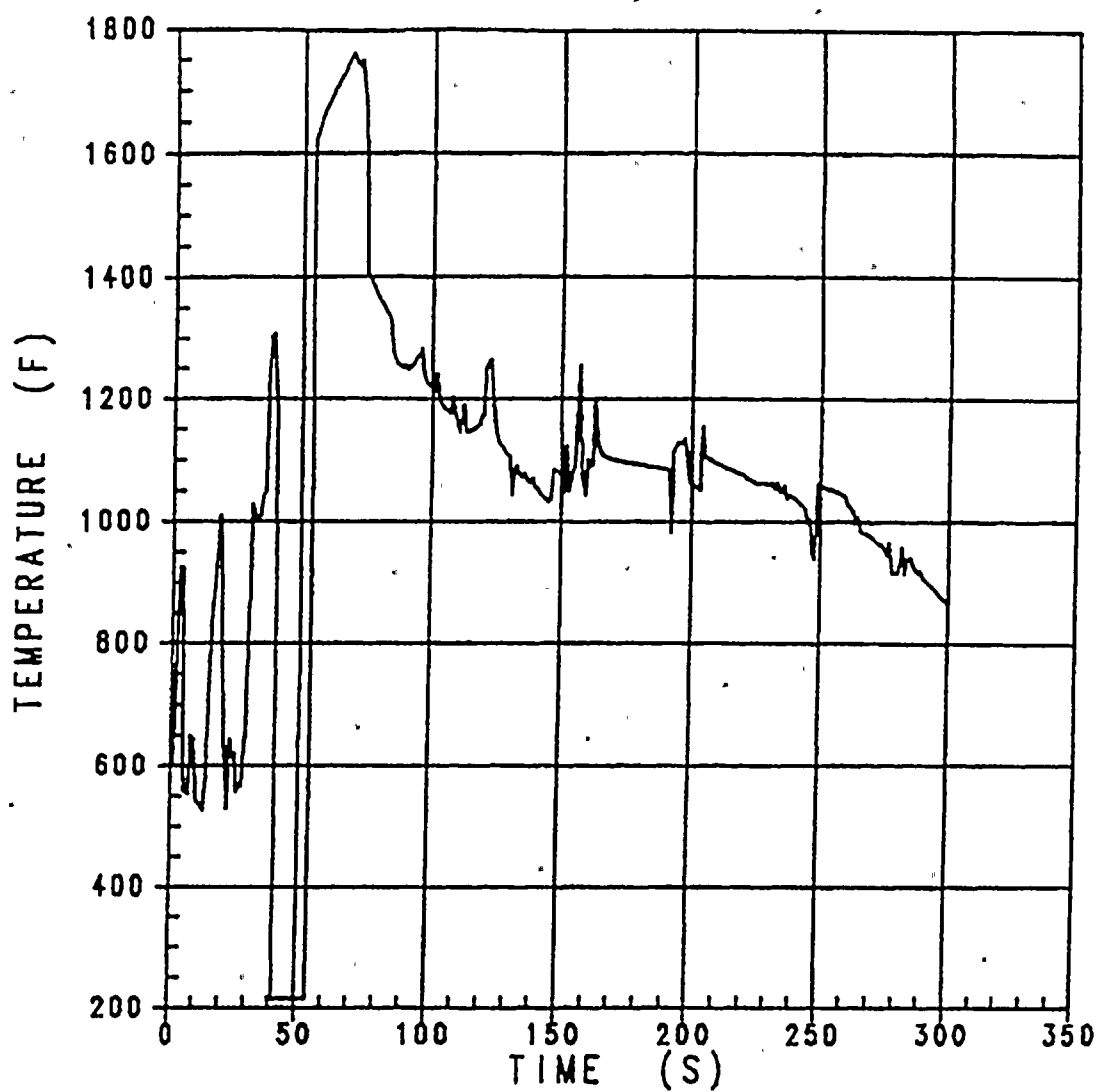


Figure 14.3.1-12a Vapor Temperature
Case A, $CD=0.4$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

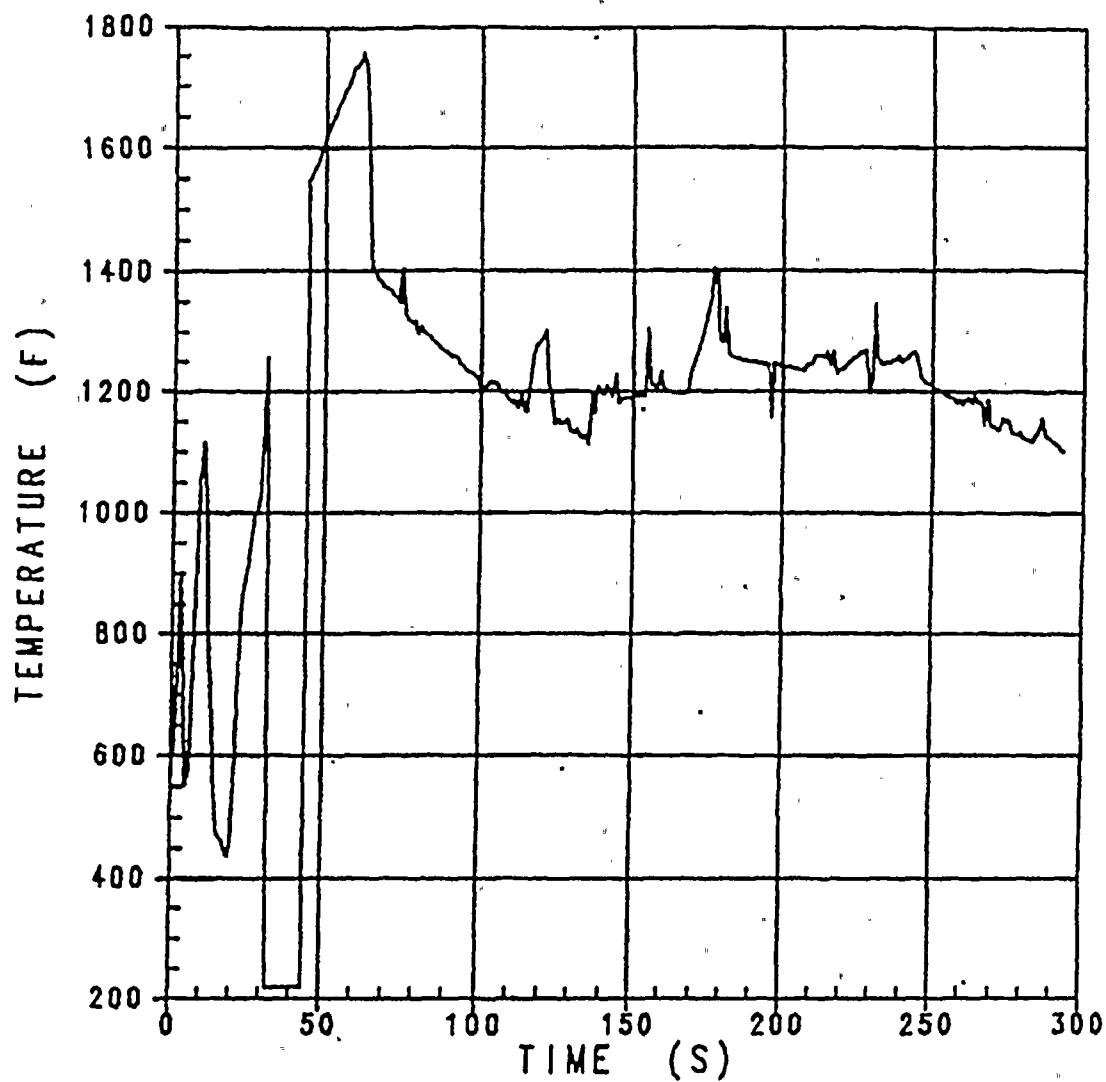


Figure 14.3.1-12b Vapor Temperature
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

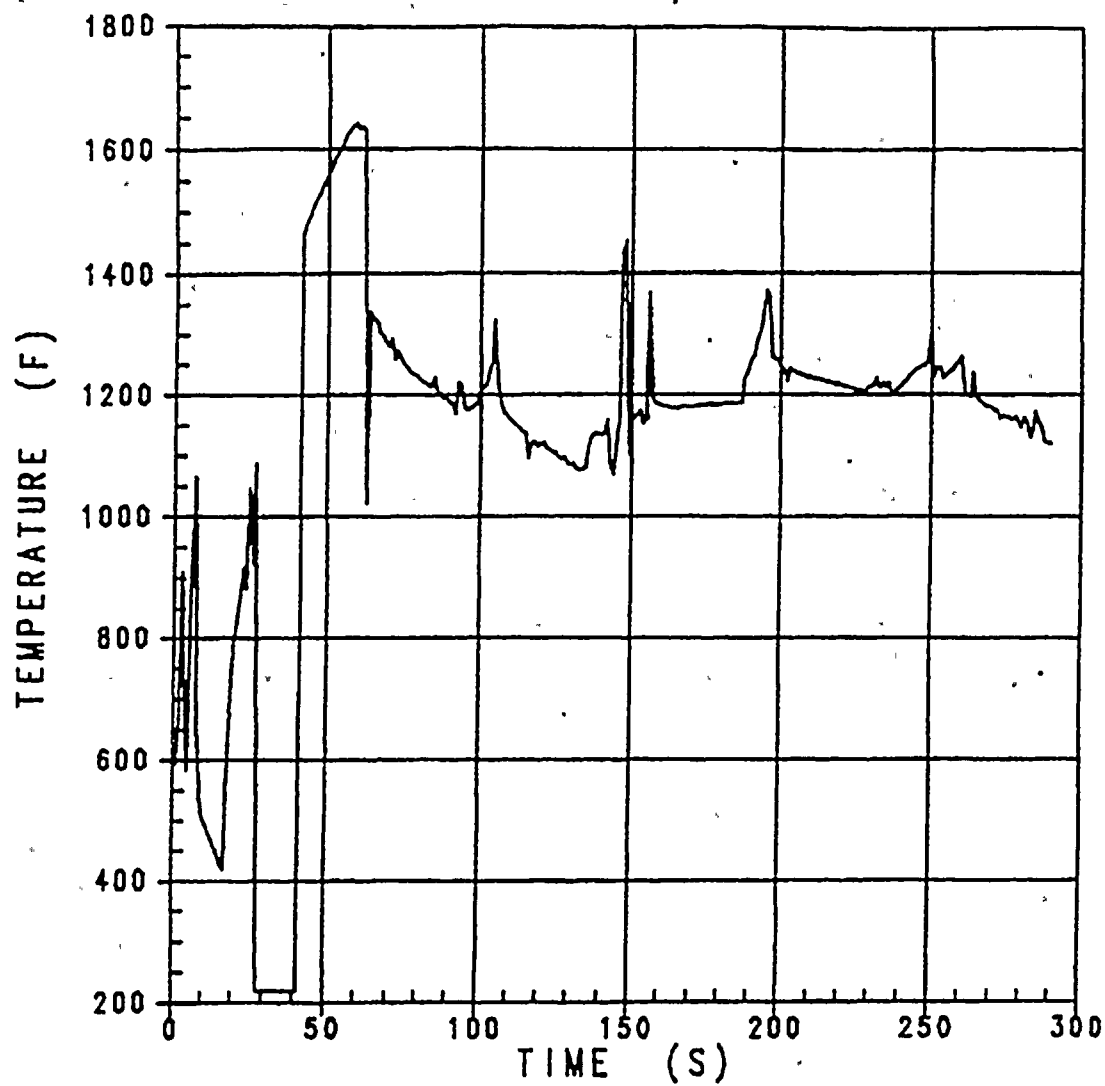


Figure 14.3.1-12c Vapor Temperature
Case C, $CD=0.8$, $Thot=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

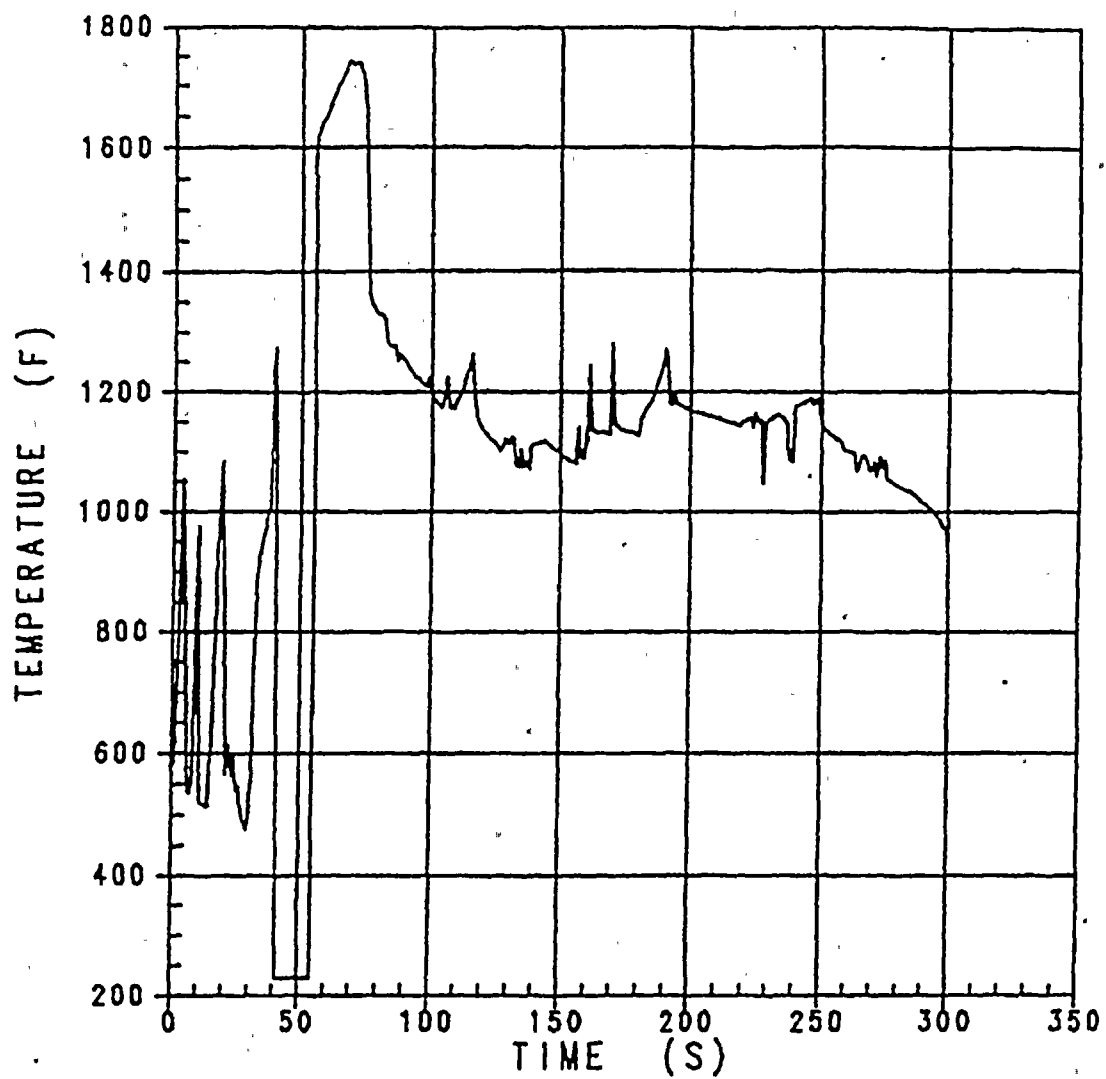


Figure 14.3.1-12d Vapor Temperature
Case D, $CD=0.4$, $Thot=586.8^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

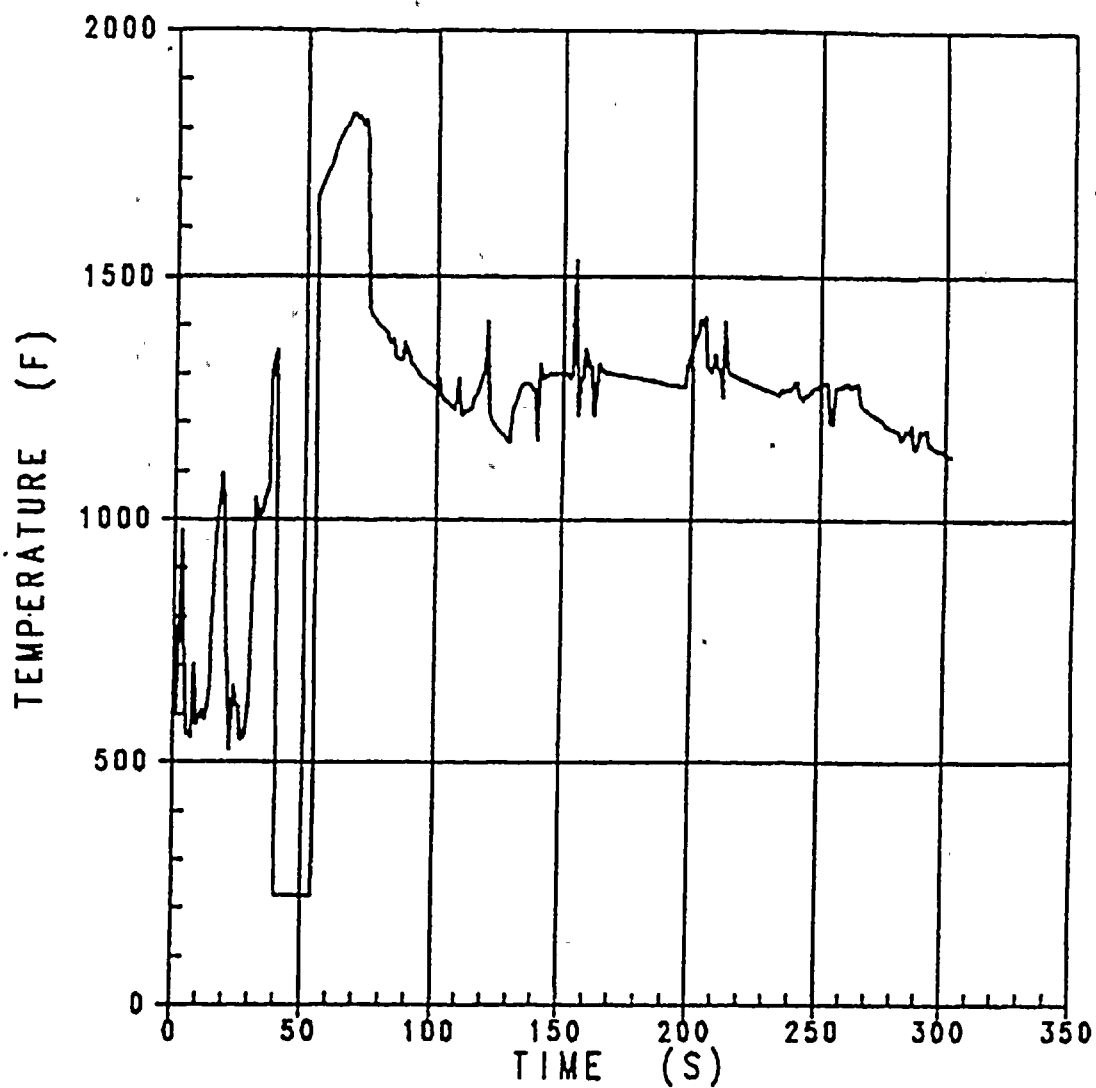


Figure 14.3.1-12e Vapor Temperature
Case E, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

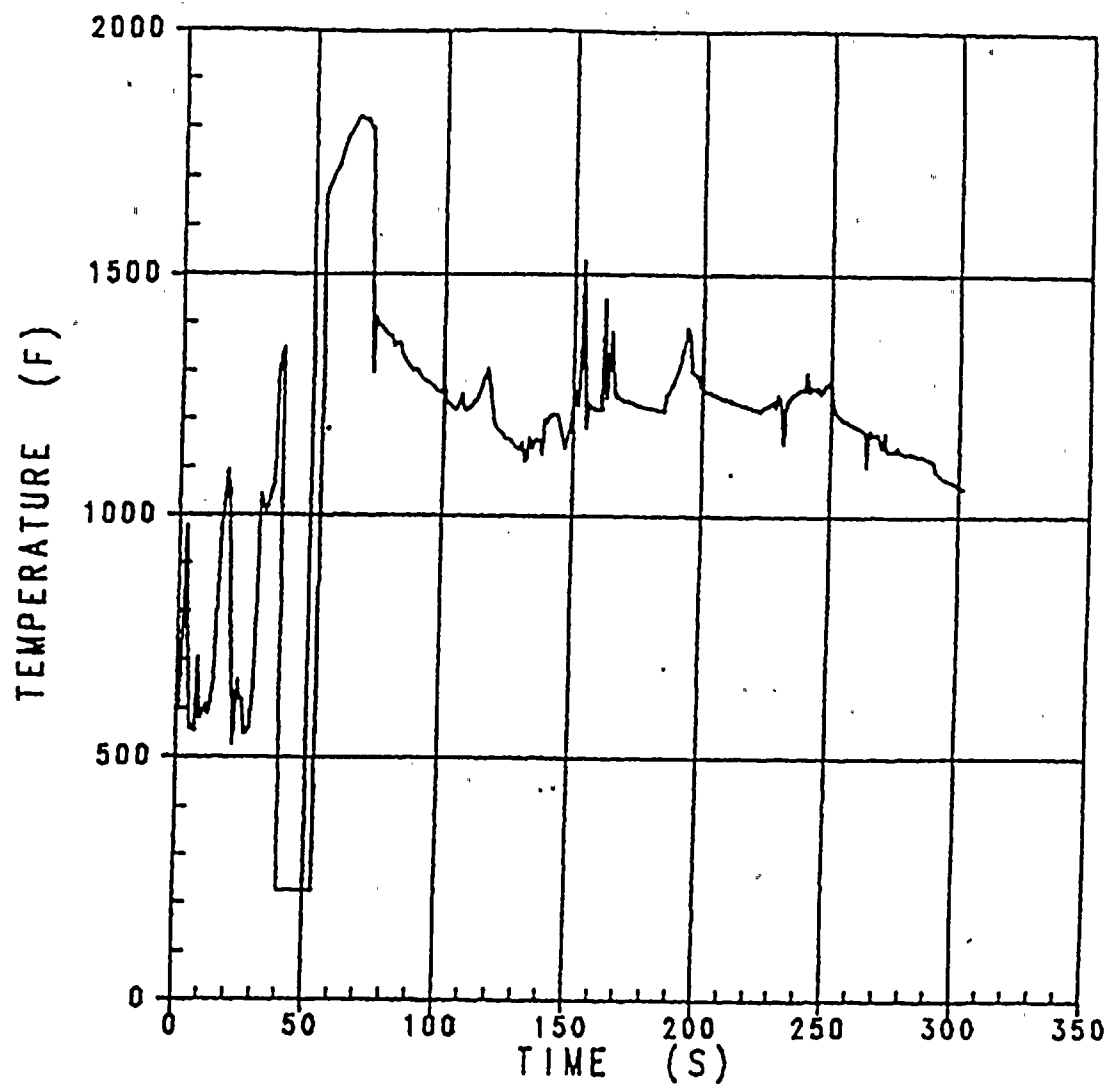


Figure 14.3.1-12f Vapor Temperature
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

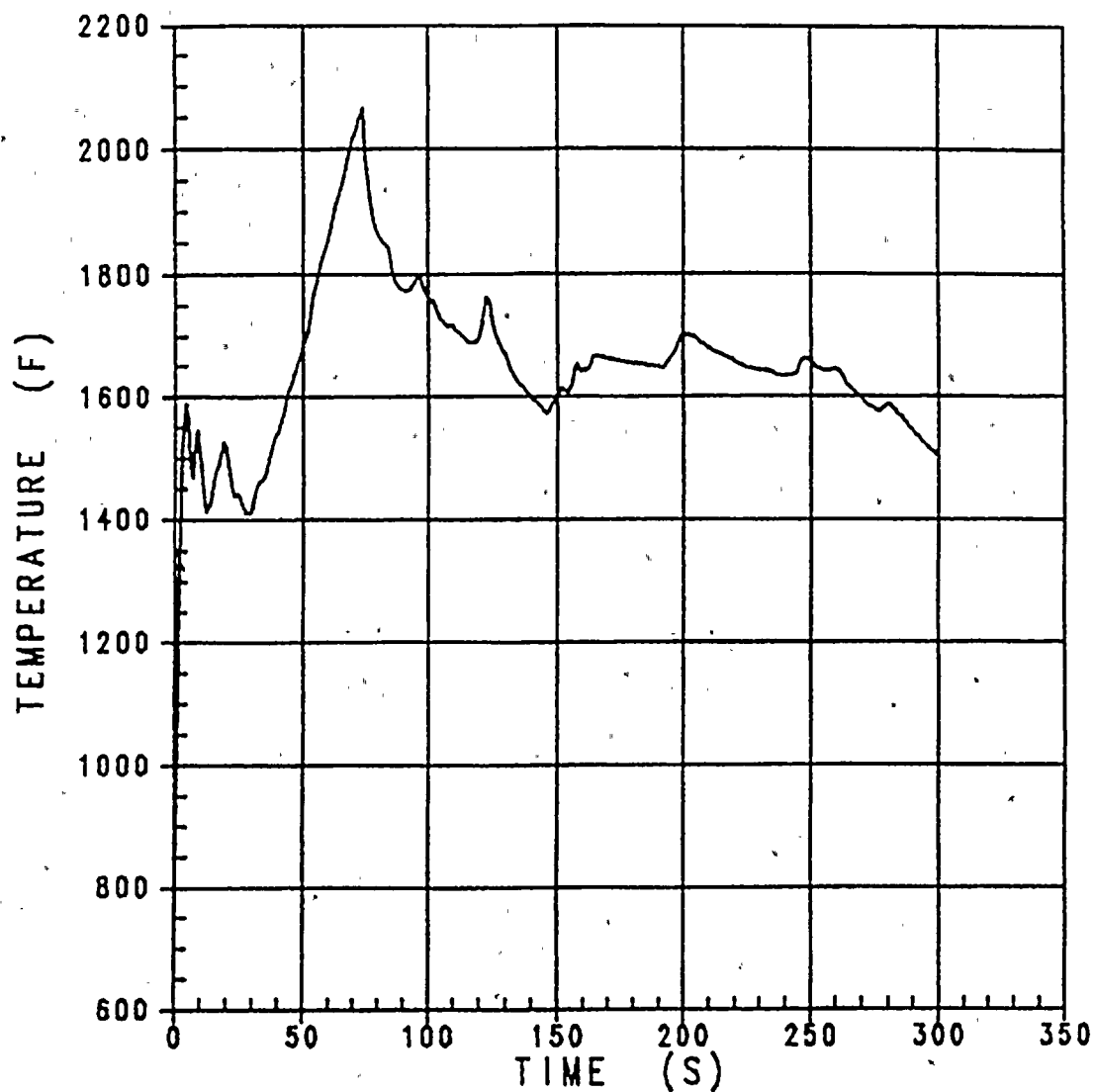


Figure 14.3.1-13a Fuel Rod Peak Clad Temperature
Case A, CD=0.4, $T_{hot}=609.1^{\circ}\text{F}$, $P=2250$ psia
Donald C. Cook Unit 1

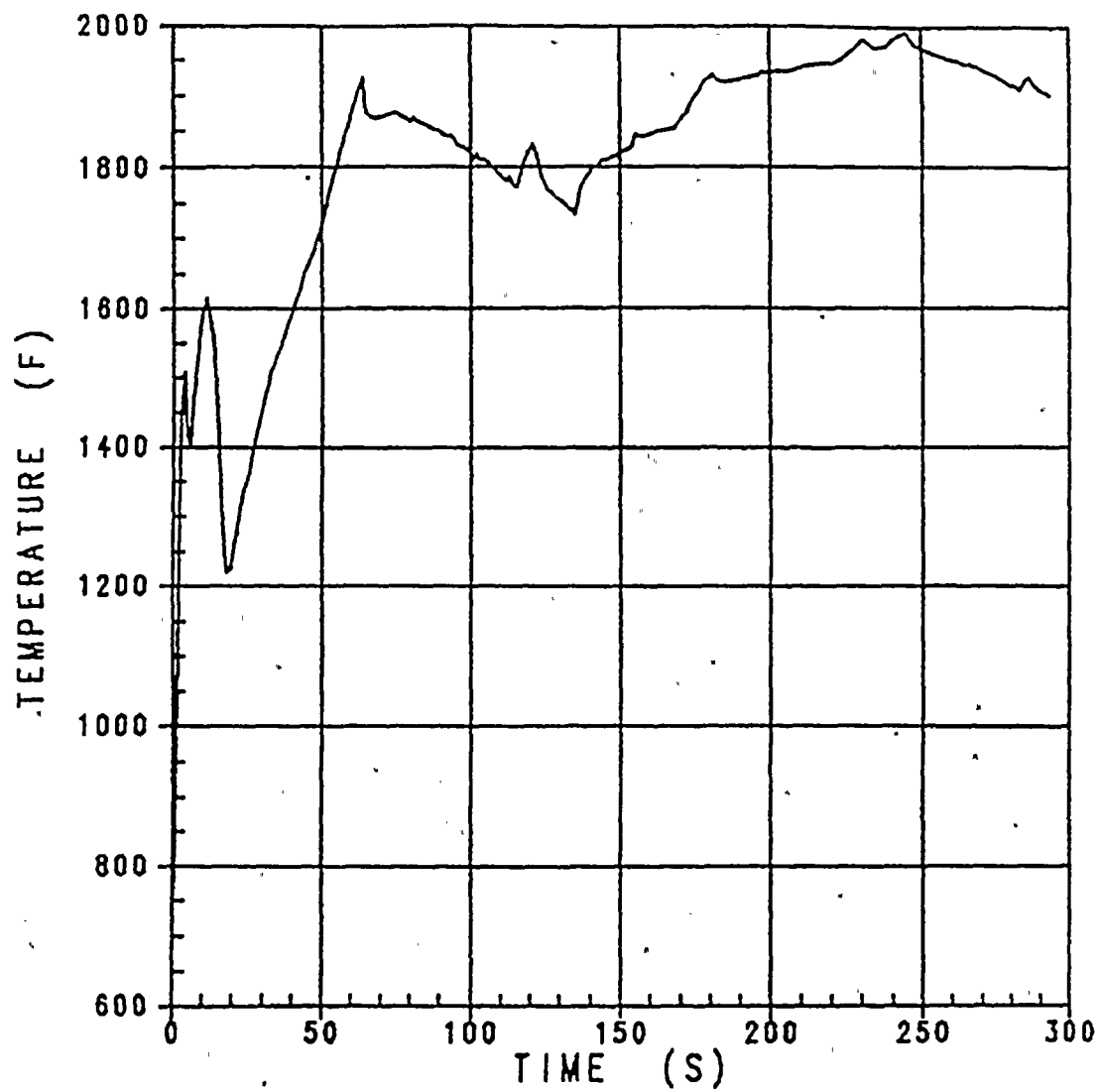


Figure 14.3.1-13b Fuel Rod Peak Clad Temperature
Case B, $CD=0.6$, $T_{hot}=609.1^{\circ}F$, $P=2250$ psia
Donald C. Cook Unit 1

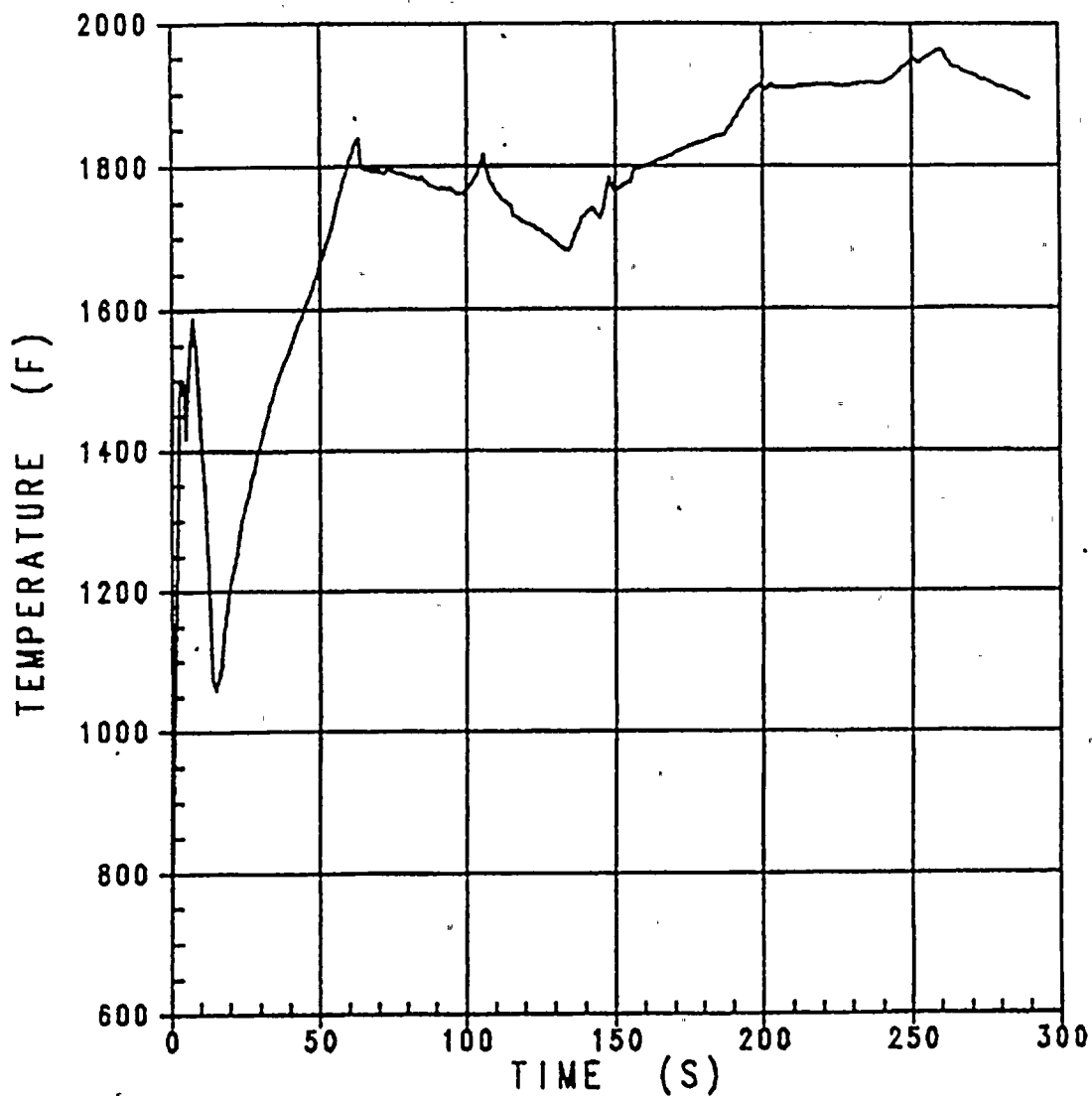


Figure 14.3.1-13c Fuel Rod Peak Clad Temperature
Case C, CD=0.8, Thot=609.1°F, P=2250 psia
Donald C. Cook Unit 1

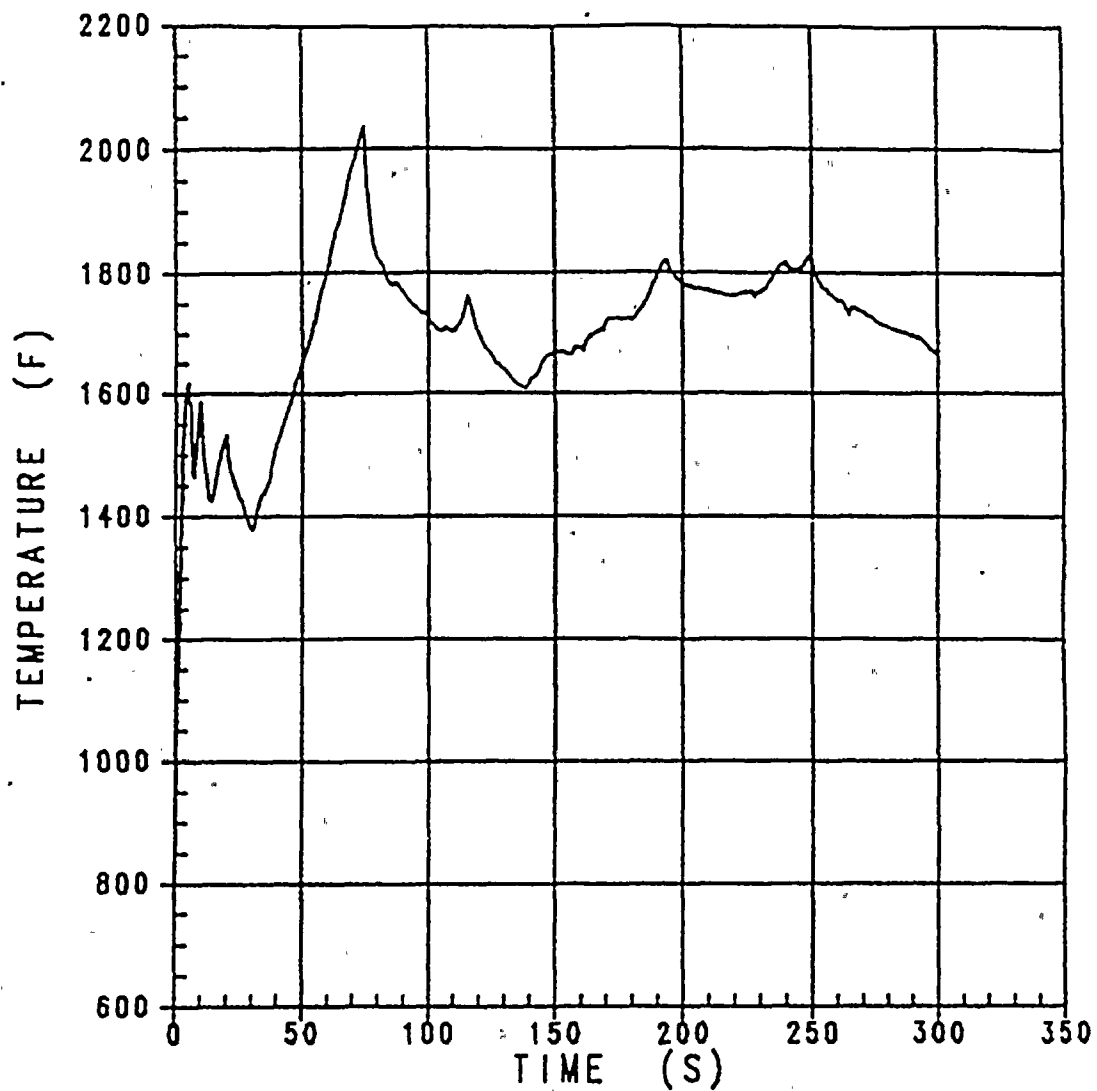


Figure 14.3.1-13d Fuel Rod Peak Clad Temperature
Case D, CD=0.4, Thot=586.8°F, P=2250 psia
Donald C. Cook Unit 1

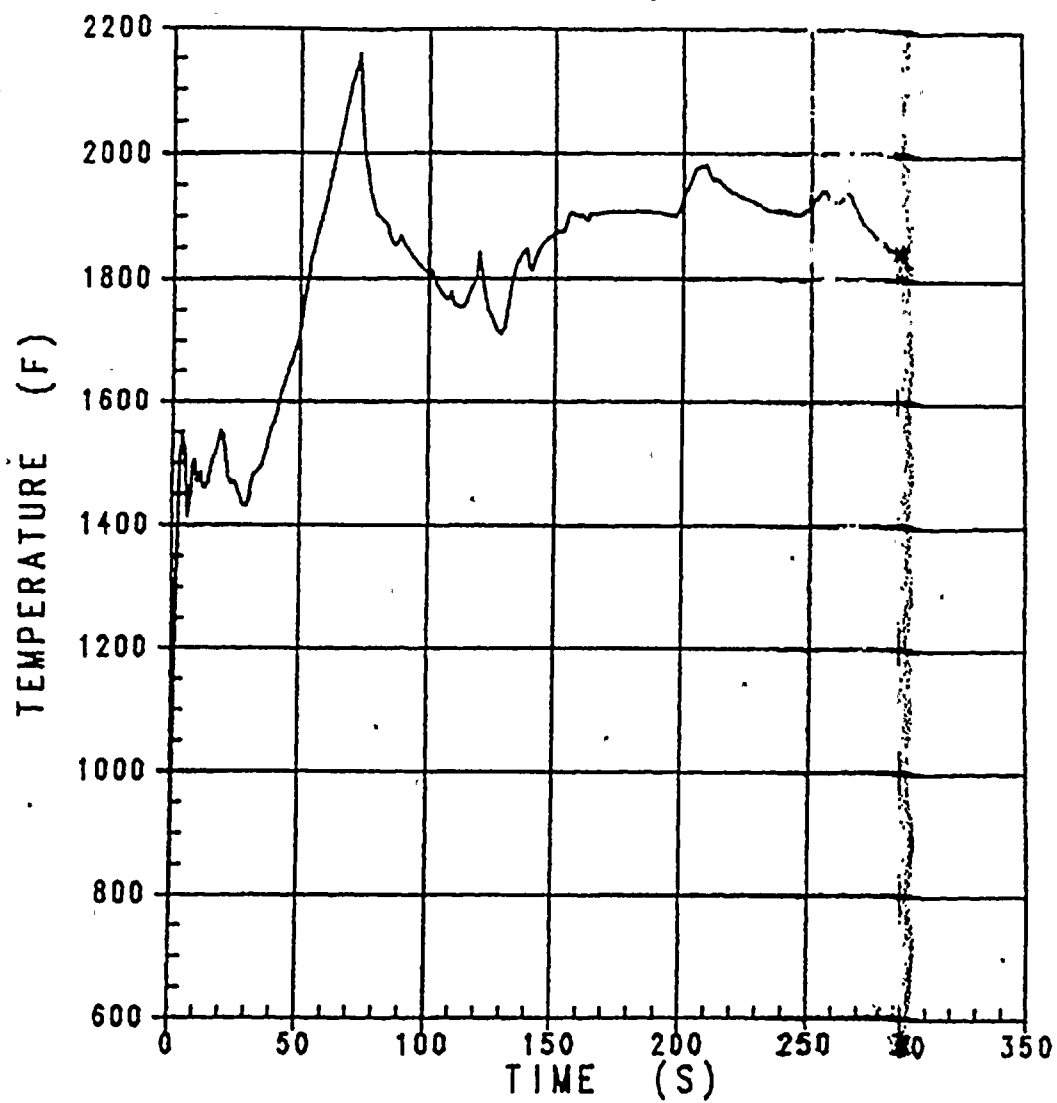


Figure 14.3.1-13e Fuel Rod Peak Clad Temperature
Case E, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia
Donald C. Cook Unit 1

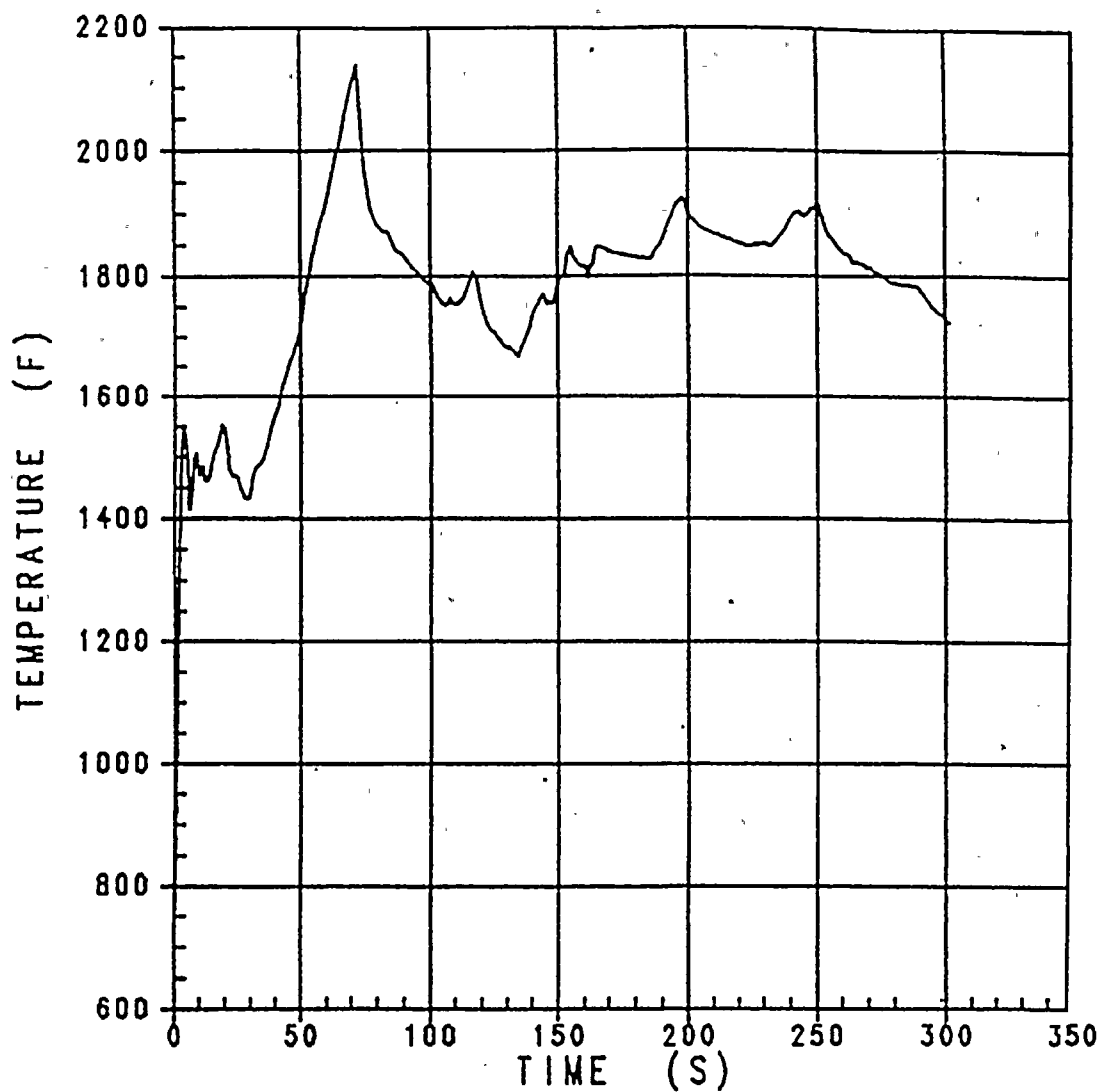


Figure 14.3.1-13f Fuel Rod Peak Clad Temperature
Case F, $CD=0.4$, $Thot=609.1^{\circ}F$, $P=2100$ psia, max SI
Donald C. Cook Unit 1

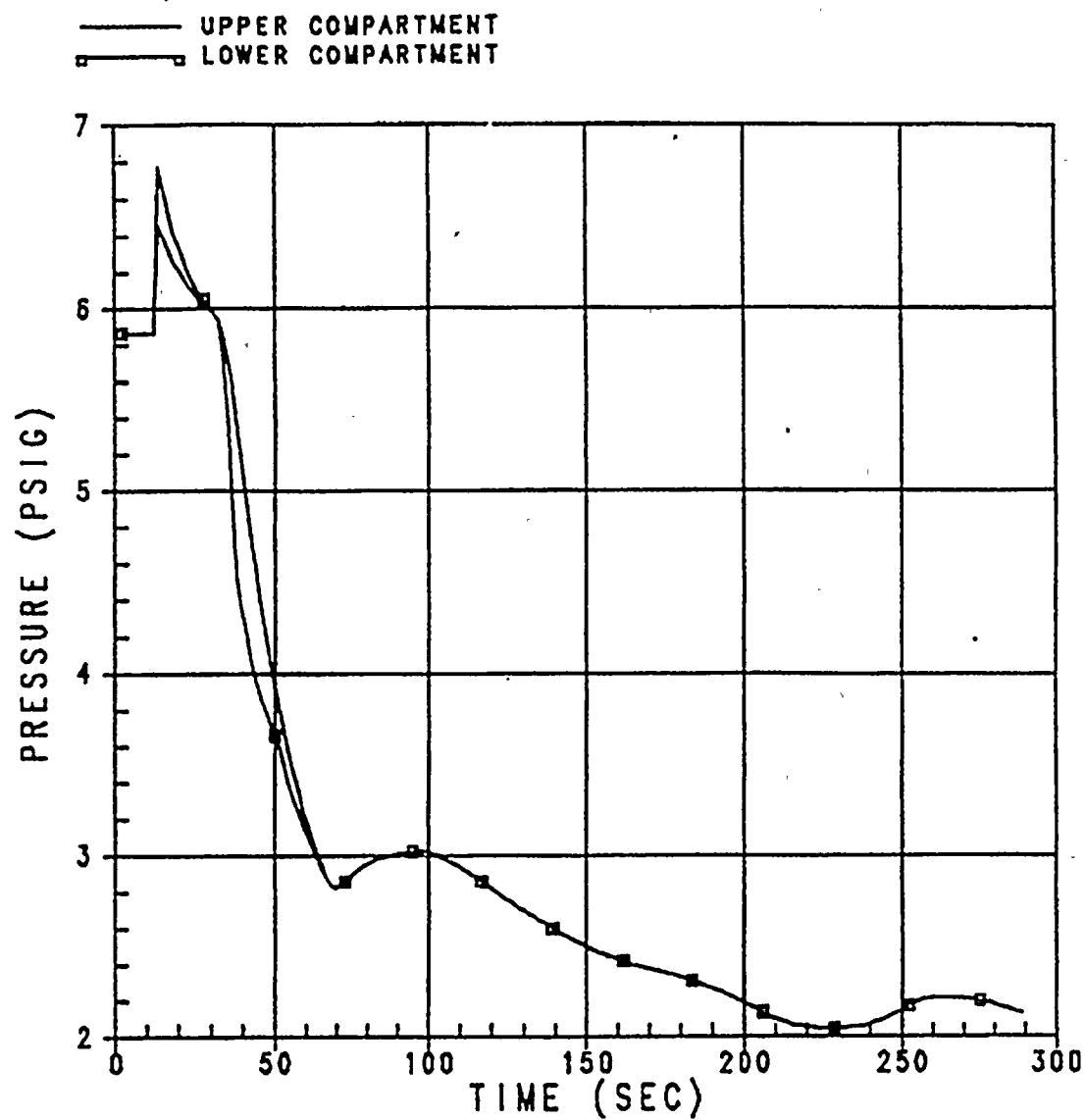


Figure 14.3.1-14 Containment Pressure
CD=0.4, Min SI
Donald C. Cook Unit 1

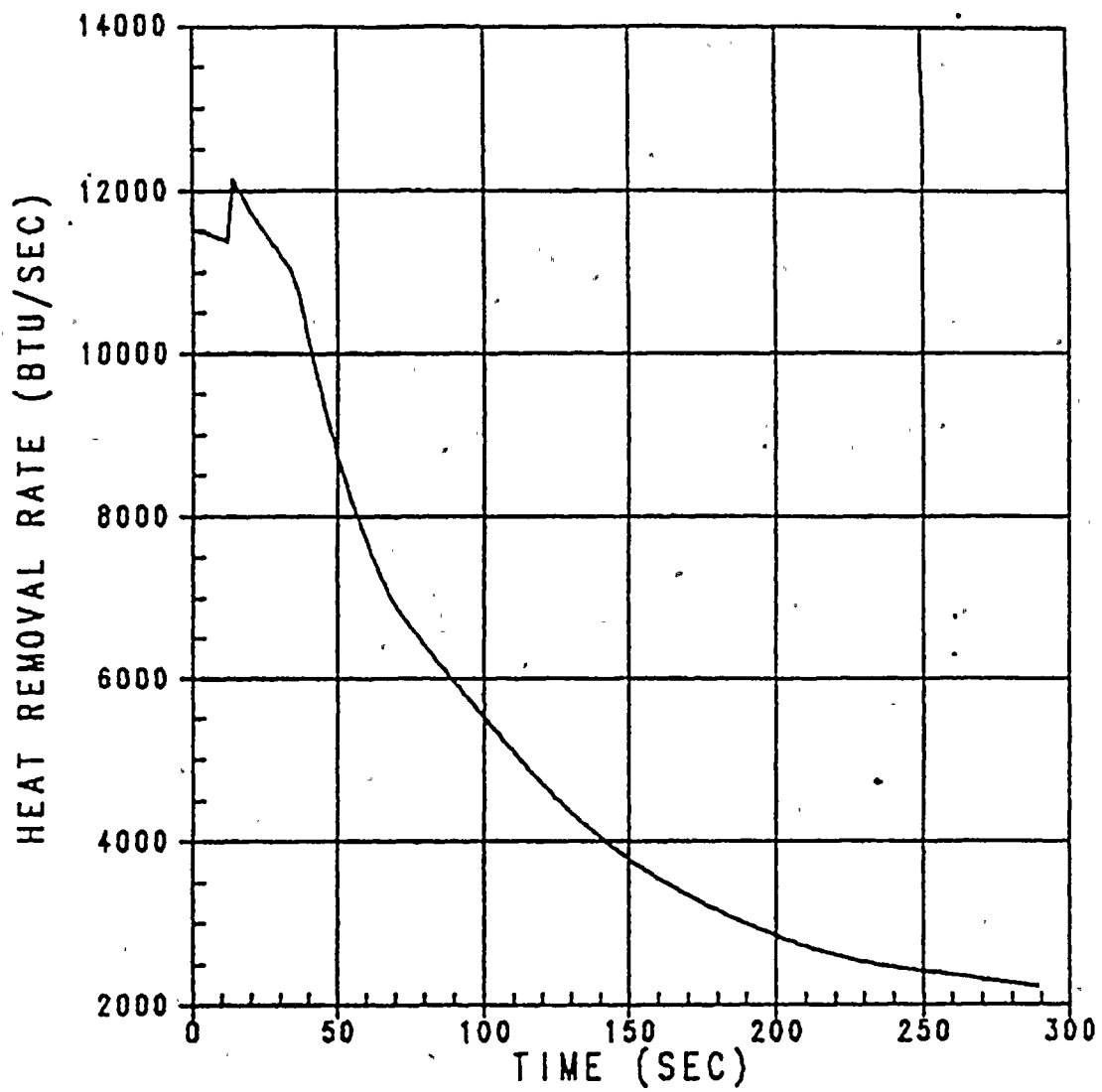


Figure 14.3.1-15 Upper Compartment Structural Heat Removal Rate
CD=0.4, Min SI
Donald C. Cook Unit 1

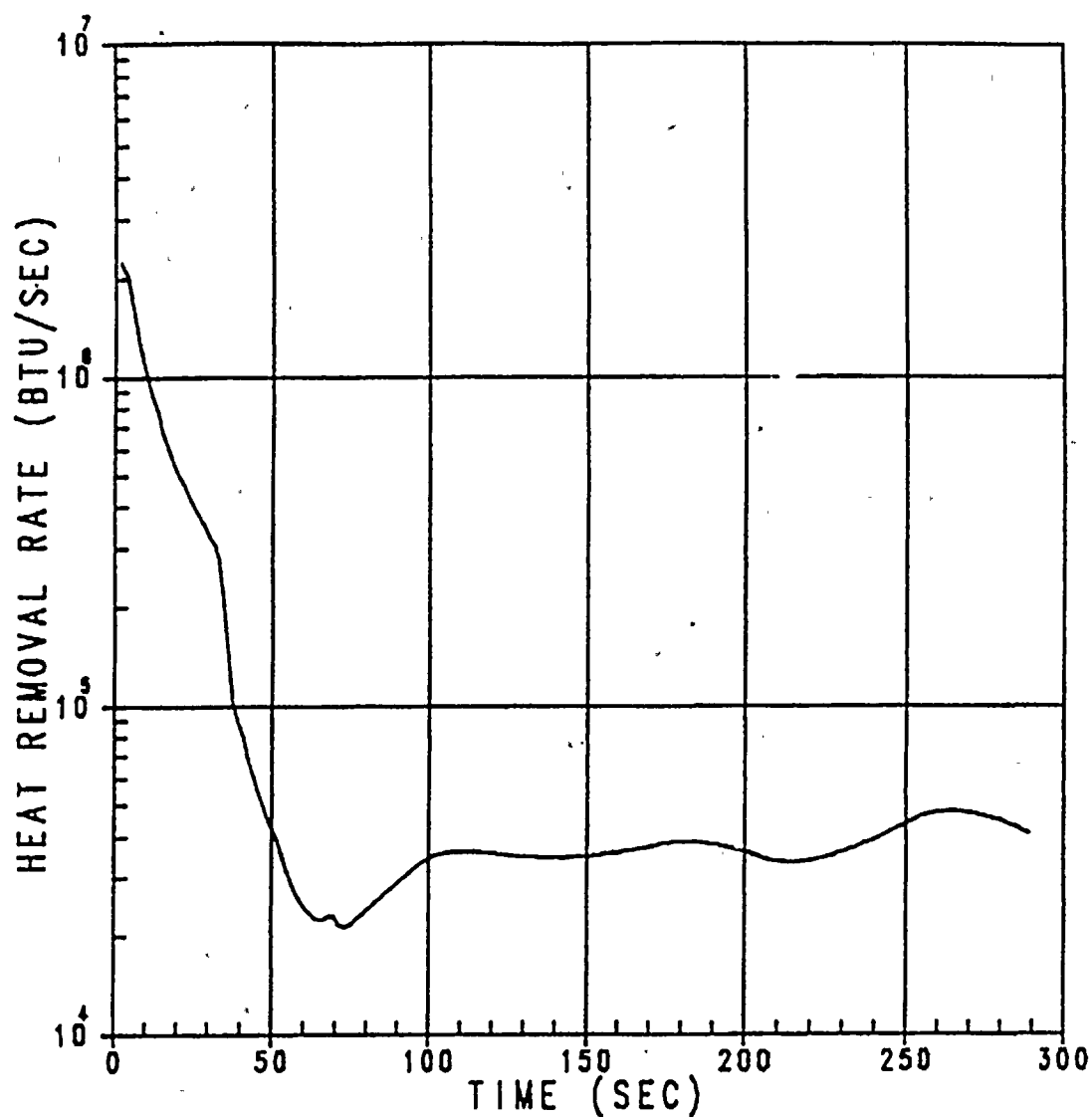


Figure 14.3.1-16 Lower Compartment Structural Heat Removal Rate
CD=0.4, Min SI
Donald C. Cook Unit 1

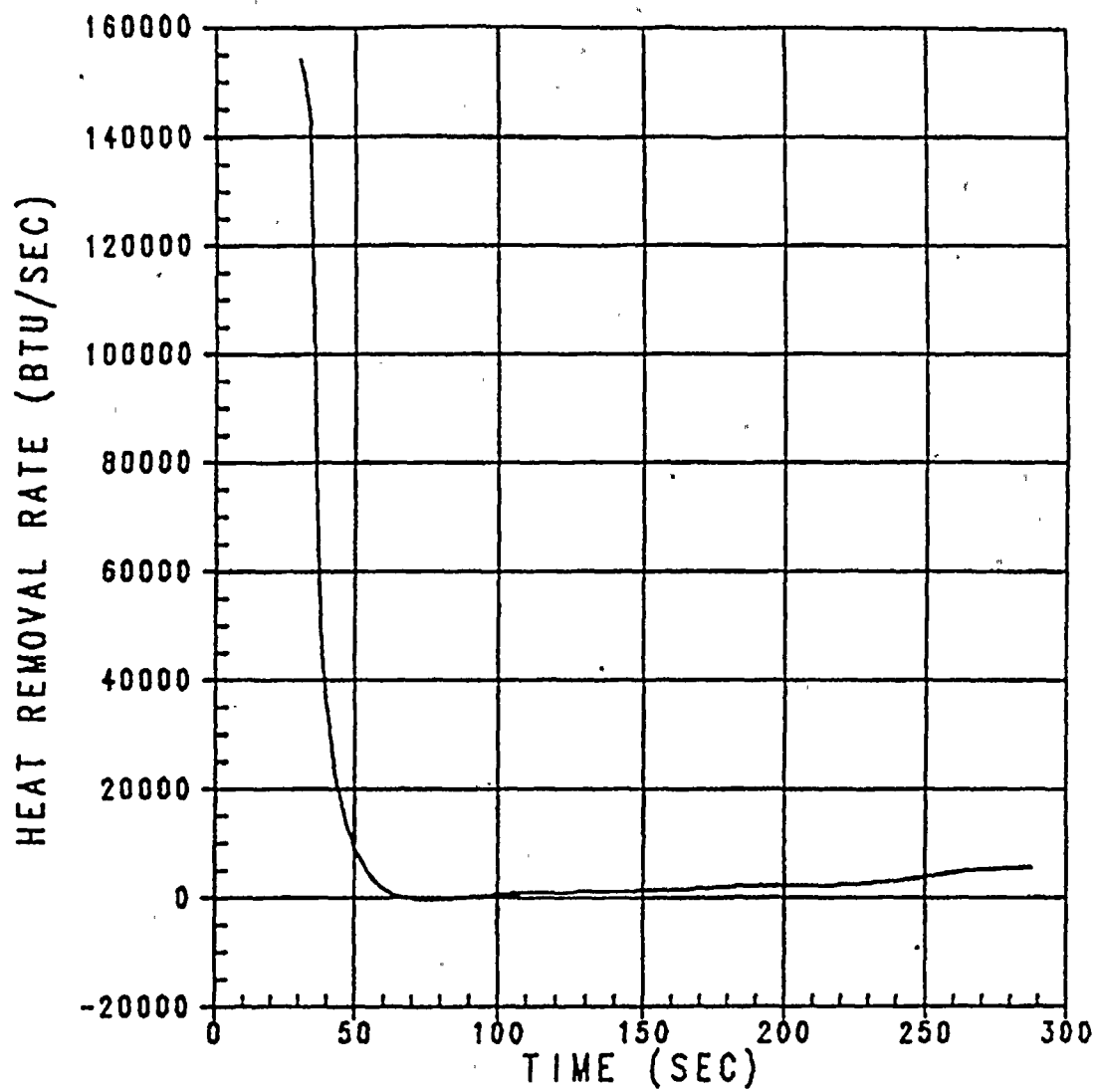


Figure 14.3.1-17 Heat Removal by Sump
CD=0.4, Min SI
Donald C. Cook Unit 1

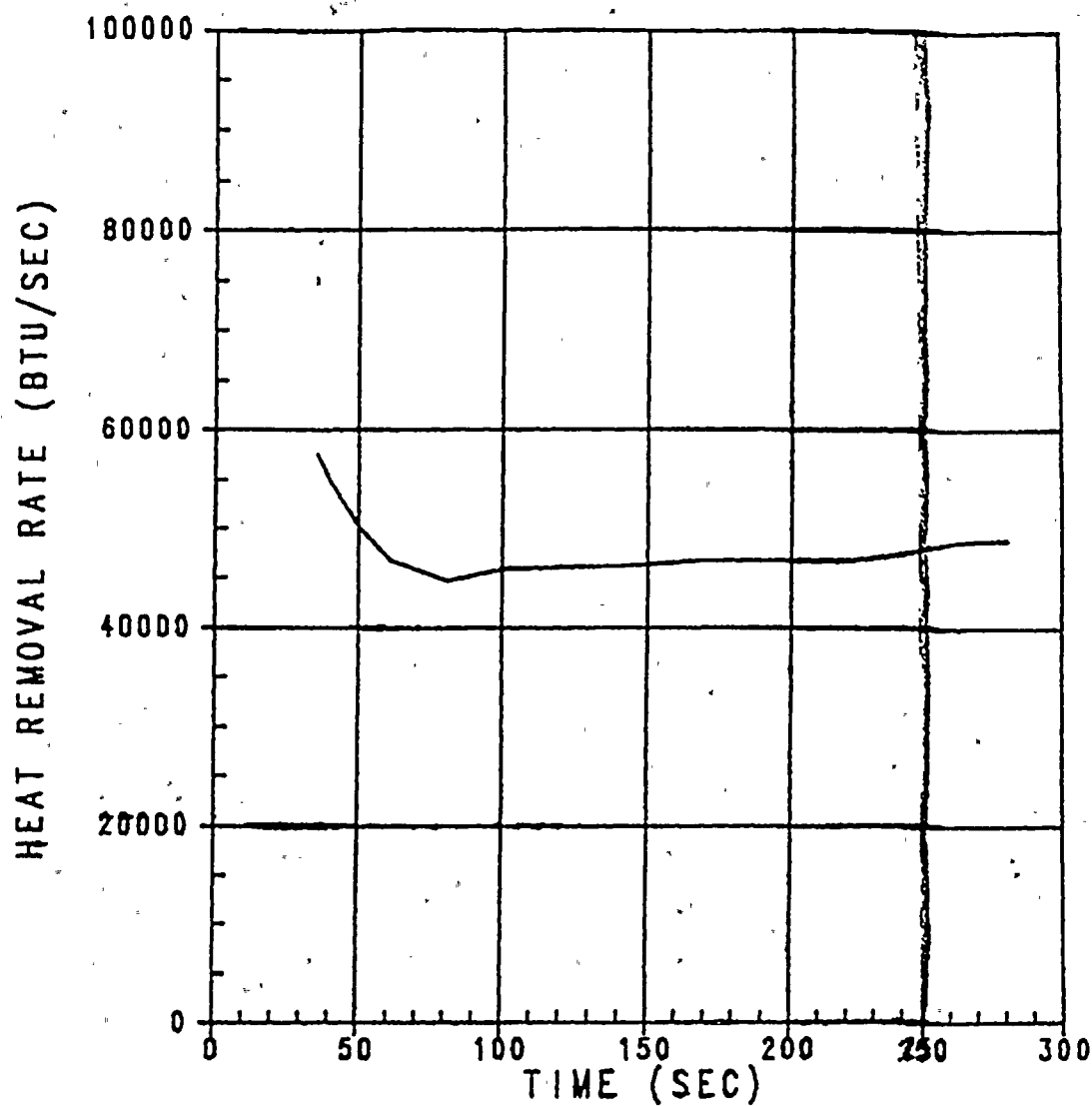


Figure 14.3.1-18 Heat Removal by Lower Compartment Spray
CD=0.4, Min SI
Donald C. Cook Unit 1

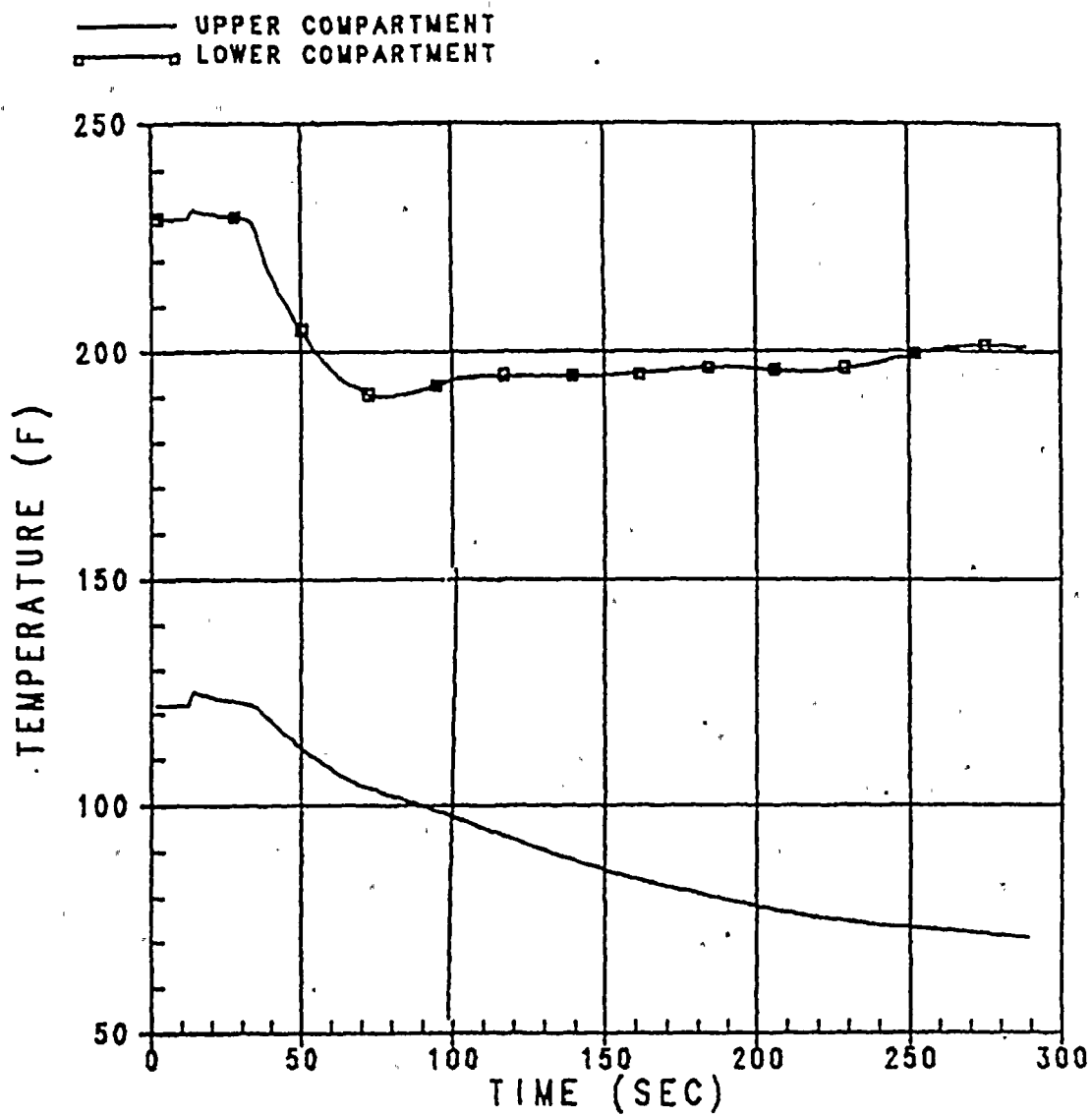


Figure 14.3.1-19 Containment Temperature
CD=0.4, Min SI
Donald C. Cook Unit 1

Table 14.3.2-7 shows the peak clad temperature obtained for the small break LOCA analysis performed using the high head safety injection system cross-tie valve closed. The table includes the application of various penalties as described under the heading small break LOCA model assessments.

Main Steam Safety Valve Setpoint Tolerance Relaxation

Additional small break LOCA analyses were performed to support an increase in the MSSV lift setpoint tolerance from $\pm 1\%$ to $\pm 3\%$. The limiting 3-inch HHSI cross tie valves closed analysis was performed for low pressure/low temperature (LPLT) operating conditions at a core power level of 3250 MWt, which has been previously demonstrated to result in the most limiting peak clad temperature. Since the basis for the limiting case determination remains valid, it was not necessary to perform the full break spectrum. An additional 3-inch break case initiated at low pressure/high temperature (LPHT) confirmed that the LPLT case remained bounding, and an additional 2-inch, cross-ties closed, break was run to provide further assurance that the limiting break size did not shift to a smaller break. Also included in all the analyses for unit 1 was a 25 gpm charging pump flow imbalance. The results of the limiting 3-inch break analysis are presented in the Sequence of Events Table 14.3.2-8 and the Results Table 14.3.2-9. Results of the non-limiting 2-inch case are provided in Tables 14.3.2-10 and 14.3.2-11. Both of these analyses were performed with the minimum AFW flow rate of 750 gpm.

Plots of the following parameters for the LPLT 3-inch break analysis are shown in Figures 14.3.2-41 through 14.3.2-48, and for the 2-inch break in Figures 14.3.2-50 through 14.3.2-57:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam flow
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate, and
- Safety injection mass flow rate

Figure 14.3.2-49 contains the power shape which is applicable to both cases.

The 3-inch cross ties closed case initiated at LPLT operating conditions is the most limiting of all the licensing basis small break analyses. Application of a burst and blockage penalty results in a peak clad temperature of 2068°F, which remains less than the 2200°F limit.

Main Steam Safety Valve Setpoint Tolerance Relaxation for 3588 MWt Operation

An additional small break LOCA analysis was performed to support an increase in the MSSV lift setpoint tolerance from $\pm 1\%$ to $\pm 3\%$ for operation at 3588 MWt with the HHSI cross-tie valves open. The analysis was performed for the LPLT operating condition which was previously demonstrated to be the most limiting condition. The analysis was also performed for the previously limiting 3-inch break, since the analysis for 3250 MWt indicated that the limiting break would not shift to a smaller size.

The $\pm 3\%$ MSSV setpoint tolerance analysis at 3588 MWt with the HHSI cross-ties open was performed using the same total core peaking factor and steam generator tube plugging level used in the analysis for 3250 MWt with the HHSI cross-ties closed. The minimum value for AFW flowrate, 750 gpm, was used for this case. However, some of the other conditions were changed for the analysis at 3588 MWt with the HHSI cross-tie valve open. The diesel generator start time was increased from 10 to 30 seconds. Since a loss of offsite power is assumed in the design basis LOCA analysis, this change results in a total delay of 47 seconds from the time of safety injection actuation until pumped safety injection flow to the RCS begins, and 80 seconds from the time that offsite power is lost until auxiliary feedwater flow begins. The safety injection flow rates with the HHSI cross-tie valves open have also been revised to reflect an increase from 10 to 15% degradation of the HHSI pump performance curve. The centrifugal charging pump assumptions of 10% pump head degradation and 25 gpm flow imbalance were maintained. The analyzed core power axial offset was reduced from +30% to +20% and the hot assembly average power peaking factor (\bar{P}_{HA}) was reduced from 1.433 to 1.38, which makes the analysis more consistent with anticipated core designs.

The small break LOCA analysis was performed with the Westinghouse Small Break LOCA ECCS Evaluation Model using the NOTRUMP computer code (References 1 and 2), including the recent changes in Reference 13 to incorporate modeling of safety injection into the broken loop and the COSI condensation model. Previously, safety injection into the broken loop was not modeled in the Westinghouse small

break LOCA analyses since it was assumed that the additional safety injection would be a benefit. Because recent studies have shown that the response to broken loop safety injection can result in an increase in the calculated PCT, modeling of safety injection into the broken loop has now been incorporated into the NOTRUMP small break evaluation model. The limiting break location was established to be the bottom of the cold leg, and the safety injection branch lines join the RCS at higher elevations of the cold legs. Therefore, the injection branch lines would deliver ECCS flow to the broken loop. A more realistic model for condensation of steam by pumped safety injection based on data from the COSI test facility has also been incorporated, which provides a benefit larger than the penalty for safety injection in the broken loop. The COSI condensation model was applied to the pumped safety injection to the broken loop and to the lumped intact loop in the NOTRUMP code.

The results of the analysis are presented in the Sequence of Events Table 14.3.2-12 and the Results Table 14.3.2-13.

Plots of the following parameters for the analysis are shown in Figures 14.3.2-58 through 14.3.2-65:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam flow
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate, and
- Safety injection mass flow rate

Figure 14.3.2-66 contains the power shape which was used for the analysis.

As shown in Table 14.3.2-13, the calculated peak clad temperature is 1047°F which is significantly less than the 2200°F limit. Because the beginning of life calculated peak clad temperature is low enough to preclude rod burst and a Zr/H₂O reaction temperature excursion following burst, no burst and blockage penalty is applied. The PCT of 1047°F for operation at 3588 MWt is also significantly less than the value of 2068°F calculated for operation at 3250 MWt. The reduction in the PCT for the 3588 MWt case relative to the 3250 MWt case is attributed to the higher SI flow at the increased power with the HHSI

cross-tie valves open, the COSI condensation model. The use of reduced axial offset and reduced \bar{P}_{HA} also contributed to the PCT reduction. These results demonstrate that the small break LOCA analysis for 3250 MWt with the HHSI cross-tie valves closed bounds the case for 3588 MWt with the cross-tie valves open.

30% Steam Generator Tube Plugging Analysis

An additional small break LOCA analysis was performed to support an increase in steam generator tube plugging level from 15% to a maximum of 30% in each steam generator. The analysis was performed for the limiting 3-inch break for reduced temperature and reduced pressure operating conditions with the HHSI cross-tie valves closed and a core power level of 3250 MWt, which was previously demonstrated to result in the most limiting clad temperature. An evaluation of the basis for the limiting case determination was performed and it was concluded that it was not necessary to perform a full break spectrum.

In addition to 30% steam generator tube plugging, some other changes were also included in the analysis. A diesel generator start time of 30 seconds was used, which results in a total delay of 47 seconds from the time of safety injection actuation until pumped safety injection flow to the RCS begins, and 80 seconds from the time that offsite power is lost until auxiliary feedwater flow begins. The safety injection flow rates with the HHSI cross-tie valves closed were revised to reflect an increase from 10 to 15% degradation of the HHSI pump performance curve. The centrifugal charging pump assumptions of 10% pump head degradation and 25 gpm flow imbalance were maintained. Also included in the analysis is a new core power shape based on an axial offset of +20% and a hot assembly average power factor (\bar{P}_{HA}) of 1.38, and an evaluation of up to 5% RCS loop flow asymmetry.

The analysis for 30% steam generator tube plugging modeled safety injection into the broken loop and used the more realistic COSI condensation model in the NOTRUMP Evaluation Model as described in Reference 13. The pumped safety injection flow and the accumulator flow to the broken loop were modeled, and the COSI condensation model was applied to the pumped safety injection to the broken loop and to the lumped intact loop.

The results of the 3-inch break analysis are presented in the Sequence of Events Table 14.3.2-14 and the Results Table 14.3.2-15.

Plots of the following parameters for the 3-inch break analysis are shown in Figures 14.3.2-67 through 14.3.2-75.

- RCS pressure
- Core mixture level
- Hot Spot Clad Temperature
- Core Outlet Steam Flow
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate
- Broken loop safety injection mass flow rate, and
- Lumped intact loop safety injection mass flow rate

Figure 14.3.2-76 contains the power shape used in the analysis.

Due to the modeling of safety injection in the broken loop with the COSI condensation model change, in conjunction with the reduced peaking factors, the PCT for the 30% steam generator tube plugging small break LOCA analysis is lower than for the previous small break analyses at 3250 MWt with the HHSI cross-tie valves closed. Because no rod burst was calculated to occur and the beginning of life calculated peak clad temperature is low enough to preclude a Zr/H₂O reaction temperature excursion following burst, no burst and blockage penalty is applied. The resulting total peak clad temperature of 1443°F is less than the 2200°F limit.

SBLOCA Model Assessments

10 CFR 50.46(a)(3) requires the record keeping and reporting of changes in LOCA evaluation models and of changes in the application of these models. Table 14.3.2-7 shows the peak clad temperature obtained for the small break LOCA analysis for both the high head safety injection cross-tie open and closed cases. The various changes to the evaluation models shown in the table are taken from references 15, and 16. The peak clad temperature result for all cases remains below 2200°F.

1. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.
2. Lee, N. et. al., "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," WCAP-10054-P-A, August 1985.
3. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8301, (Proprietary), June 1974.
4. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600, June 1979.
5. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
6. NRC Generic Letter 83-35 from D. G. Eisenhower, "Clarification of TMI Action Plan Item II.K.3.31," November 2, 1983.
7. Rupprecht, S. D., et. al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code," WCAP-11145-P-A, October 1986.
8. Reference deleted
9. Reference deleted
10. Reference deleted
11. Reference deleted
12. Reference deleted
13. Thompson, C. M., et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P Addendum 2 (Proprietary) and WCAP-10081-NP, Addendum 2 (Non-Proprietary), August 1994.
14. Reference deleted
15. Fitzpatrick, E. E. (I&M), letter to NRC Document Control Desk, March 22, 1996, AEP:NRC:1118K.
16. Fitzpatrick, E. E. (I&M), letter to NRC Document Control Desk, April 10, 1997, AEP:NRC:1118L.

Table 14.3.2-7

SAMLL-BREAK LOSS OF COOLANT ACCIDENT
10 CFR 50.46 ASSESSMENT RESULTS PEAK CLAD TEMPERATURE
WITH HHSI CROSS-TIE VALVES CLOSED
(Reference Tables 14.3.2-14 and 14.3.2-15)

<u>Parameter</u>	<u>Value¹</u>
Analysis PCT	1443°F
Prior LOCA Assessments Δ PCT	
1. NOTRUMP Specific Enthalpy Error, Δ PCT	+20°F
2. SALIBRARY Double Precision Errors, Δ PCT	-15°F
3. SBLOCTA Fuel Rod Initialization Error, Δ PCT	+10°F
LICENSING BASIS PCT + PERMANENT ASSESSMENTS, PCT	1458°F

¹ Evaluation Model: NOTRUMP, FQ=2.32, F_{AH}=1.55, SGTP=30%, 3250 MWt

TIME SEQUENCE OF EVENTS for CONDITION III EVENTSSmall-break Loss of Coolant Accident

(3" Break, LPLT*, HHSI Cross-ties Closed)

Main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

<u>Event</u>	<u>Time (sec)</u>
Break Occurs	0.0
Reactor trip signal	8.64
Safety injection signal	17.13
Start of safety injection	44.13
Start of auxiliary feedwater delivery	68.6
Loop seal venting	592
Loop seal core uncover	NA
Loop seal core recovery	NA
Boil-off core uncover	984
Accumulator injection begins	1680
Peak clad temperature occurs	1890
Top of core covered	NA
SI flow rate exceeds break flow rate	1890

*LPLT is low pressure, low temperature operating condition.

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS30% STEAM GENERATOR TUBE PLUGGING ANALYSISAT 3250 MWT WITH HHSI CROSS-TIE VALVES CLOSEDSmall-break Loss of Coolant Accident

<u>Event</u>	<u>Time (sec)</u>
	Reduced Temperature, Reduced Pressure <u>3-inch break</u>
Break occurs	0.0
Reactor trip signal	8.8
Safety injection signal	17.4
Start of safety injection	64.4
Start of auxiliary feedwater delivery	88.8
Loop seal venting	528
Loop seal core uncover	NA
Loop seal core recovery	NA
Boil-off core uncover	1054
Accumulator injection begins	1648
Peak clad temperature occurs	1748
Top of core recovered	2995
Combined pumped SI flow rate exceeds break flow rate	1856

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATIONS30% STEAM GENERATOR TUBE PLUGGING ANALYSISAT 3250 MWT WITH HHSL CROSS-TIE VALVES CLOSEDRESULTSReduced Temperature, Reduced Pressure
3-inch break

NOTRUMP Peak Clad Temperature (°F)	1443°F
Peak Clad Temperature Location (ft)	11.5
Peak Clad Temperature Time (sec)	1748
Local Zr/H ₂ O Reaction Maximum (%)	<1.0
Local Zr/H ₂ O Reaction Location (ft)	11.5
Total Zr/H ₂ O Reaction (%)	<1.0
Rod Burst	None
Burst and Blockage Penalty	None
Total Peak Clad Temperature (°F)	1443°F

CALCULATION:

NSS Power (MWt) 102% of	3250*
Peak Linear Power (kw/ft) 102% of	14.87
Hot Rod Power Distribution (kw/ft)	See Figure 14.3.2-76
Accumulator Water Volume (ft ³)	946

* Does not include pump heat

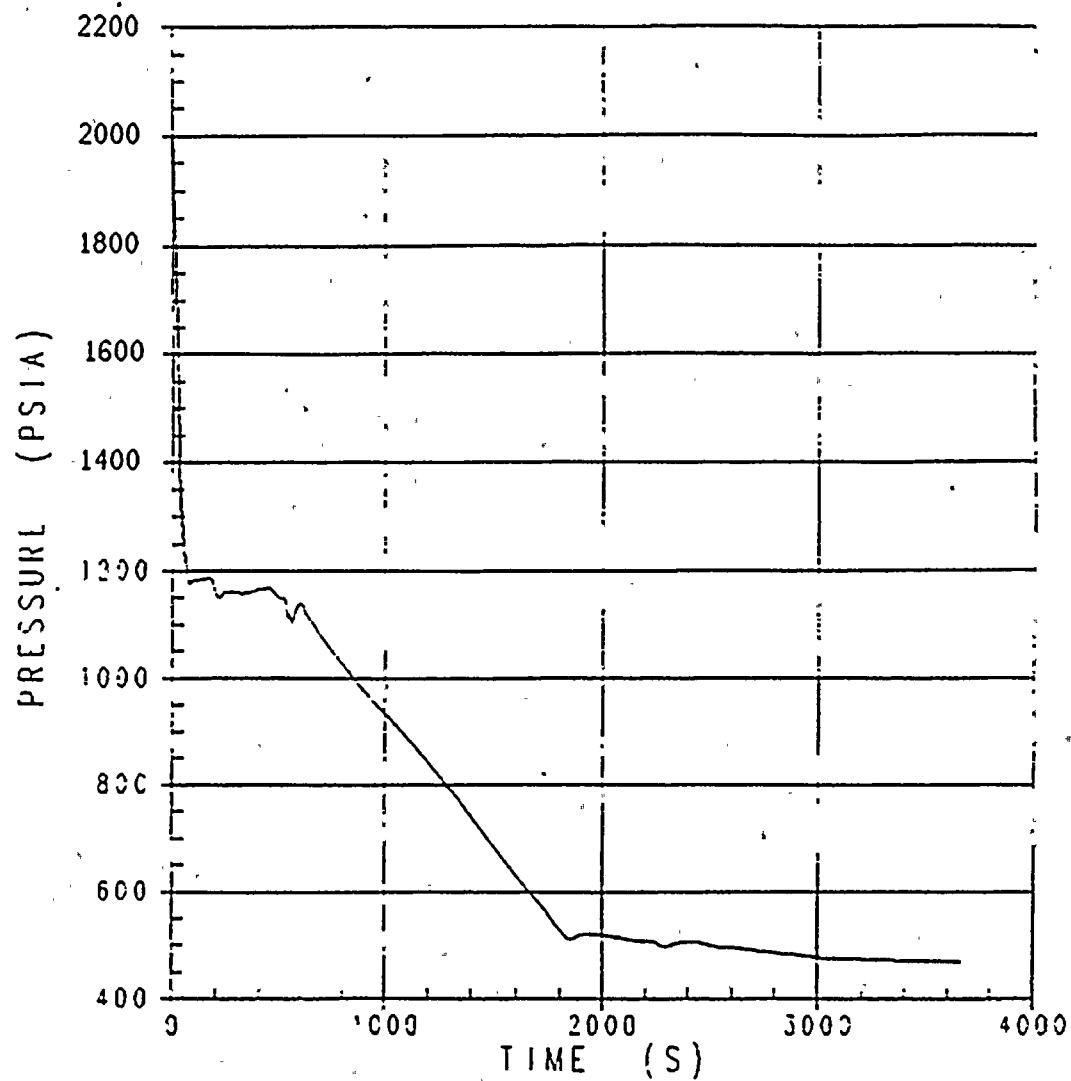


Figure 14.3.2-67
RCS Pressure (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

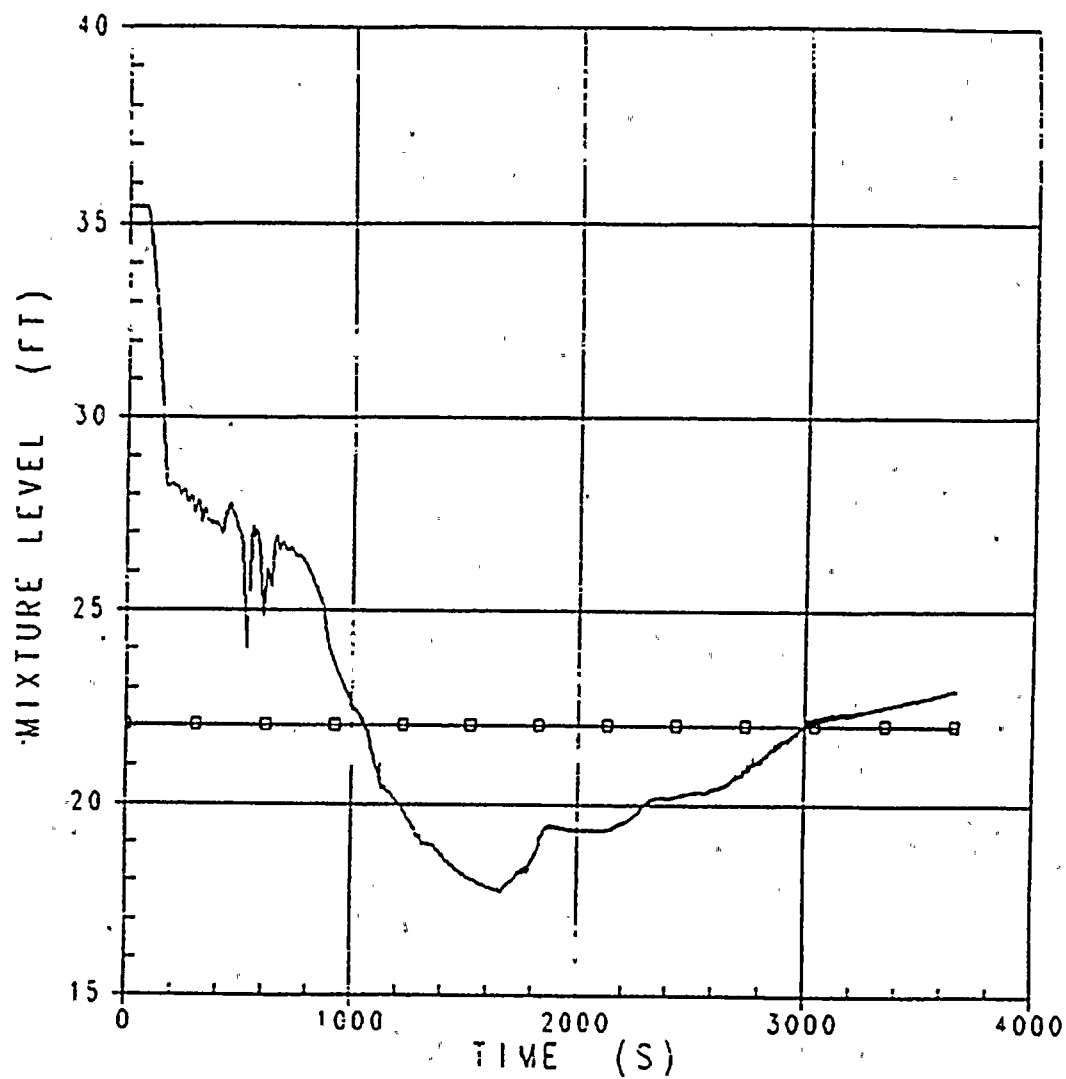


Figure 14.3.2-68
Core Mixture Level (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

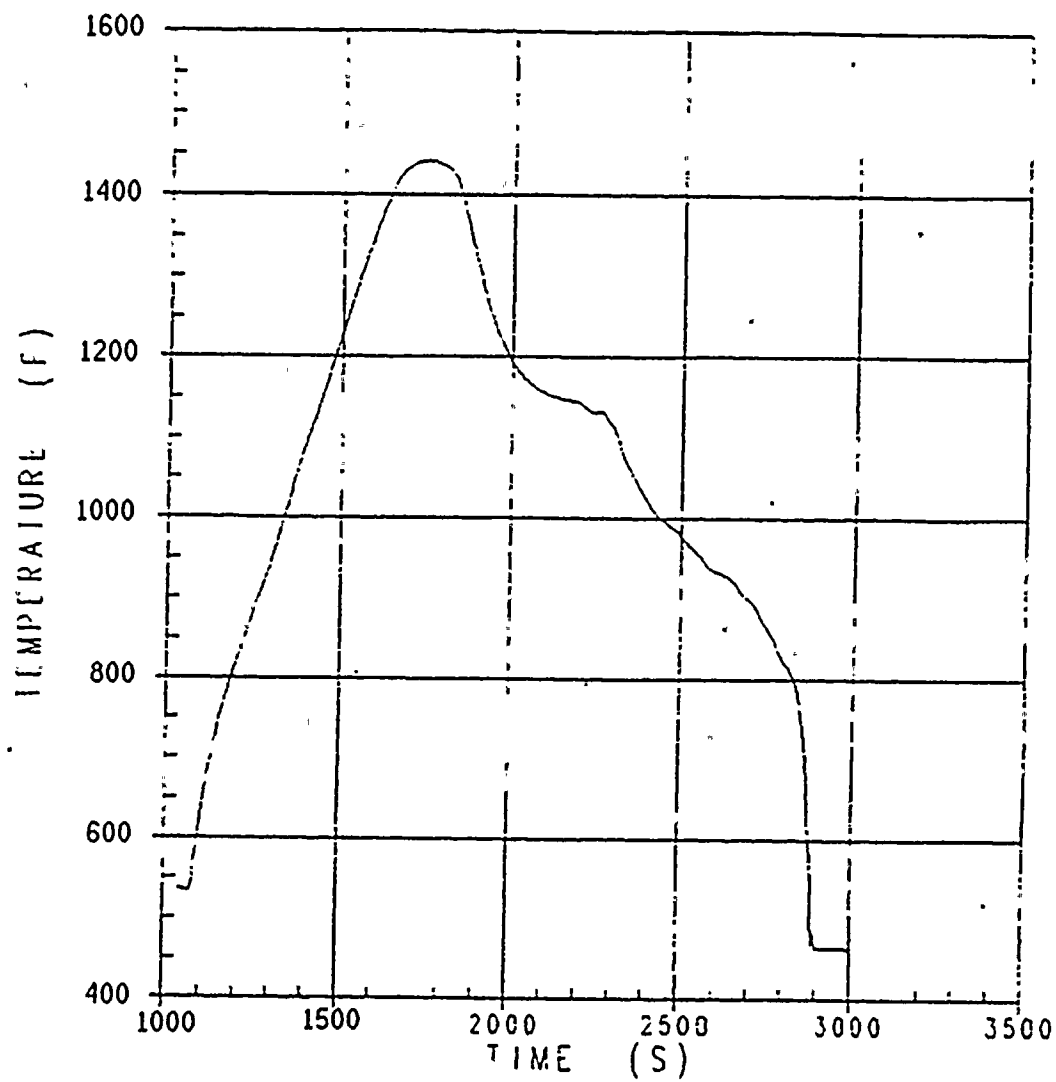


Figure 14.3.2-69
Hot Spot Clad Temperature (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

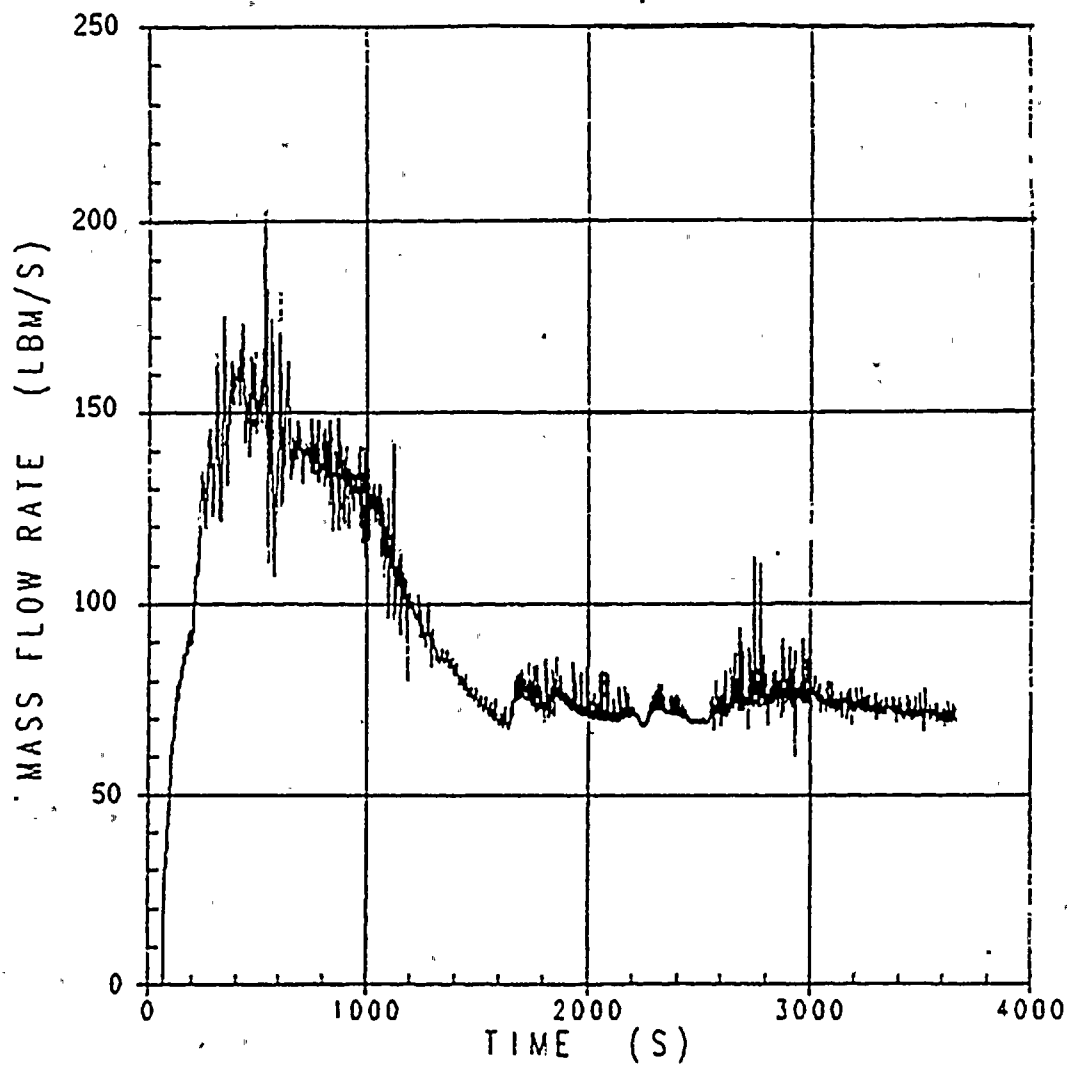


Figure 14.3.2-70
Core Outlet Steam Flow (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

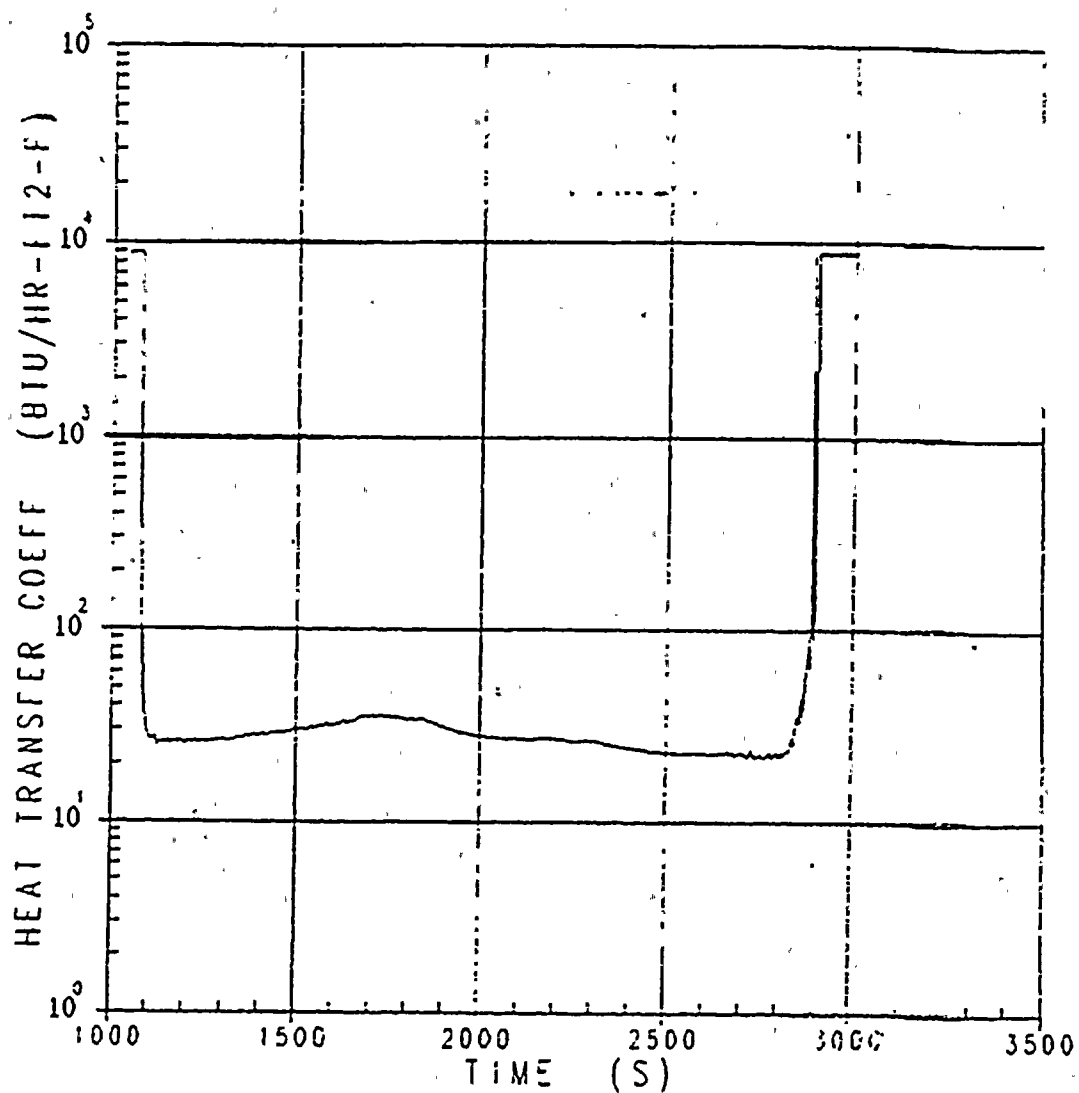


Figure 14.3.2-71
 Hot Spot Rod Surface Heat Transfer Coefficient (3 Inch, 30% SGTR)
 Reduced Temperature, Reduced Pressure
 Donald C. Cook Nuclear Plant Unit 1

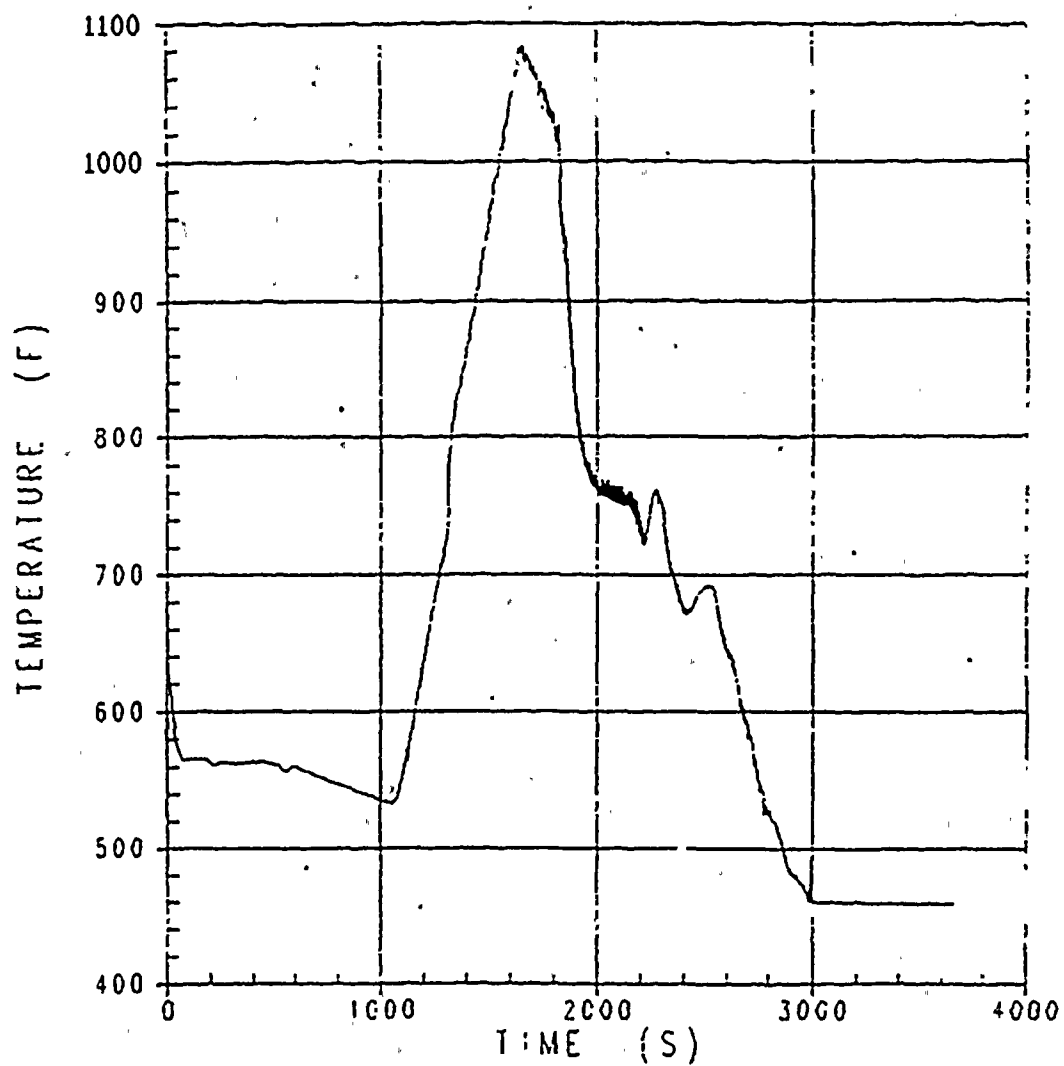


Figure 14.3.2-72
Hot Spot Fluid Temperature (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

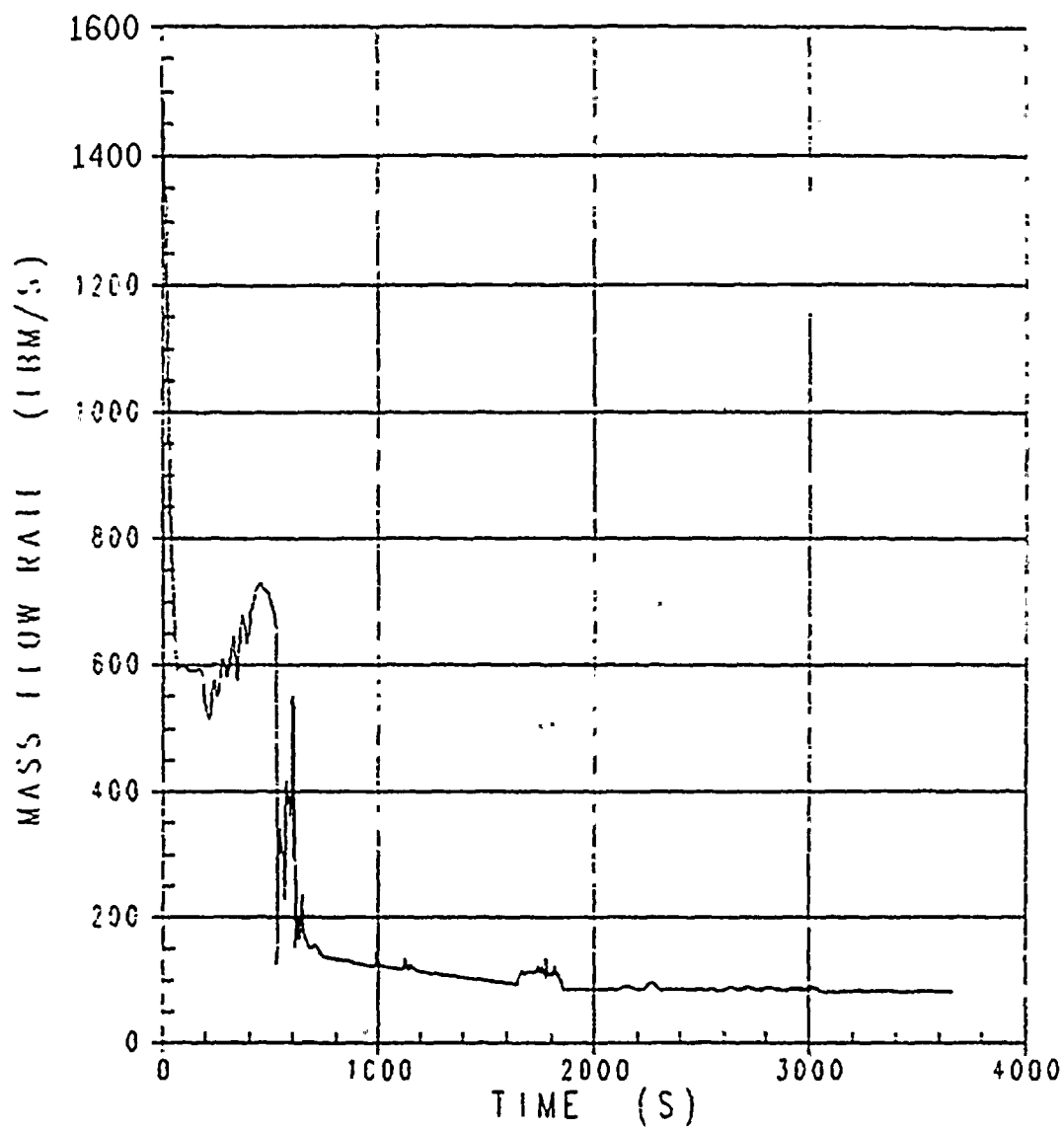


Figure 14.3.2-73
Cold Leg Break Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

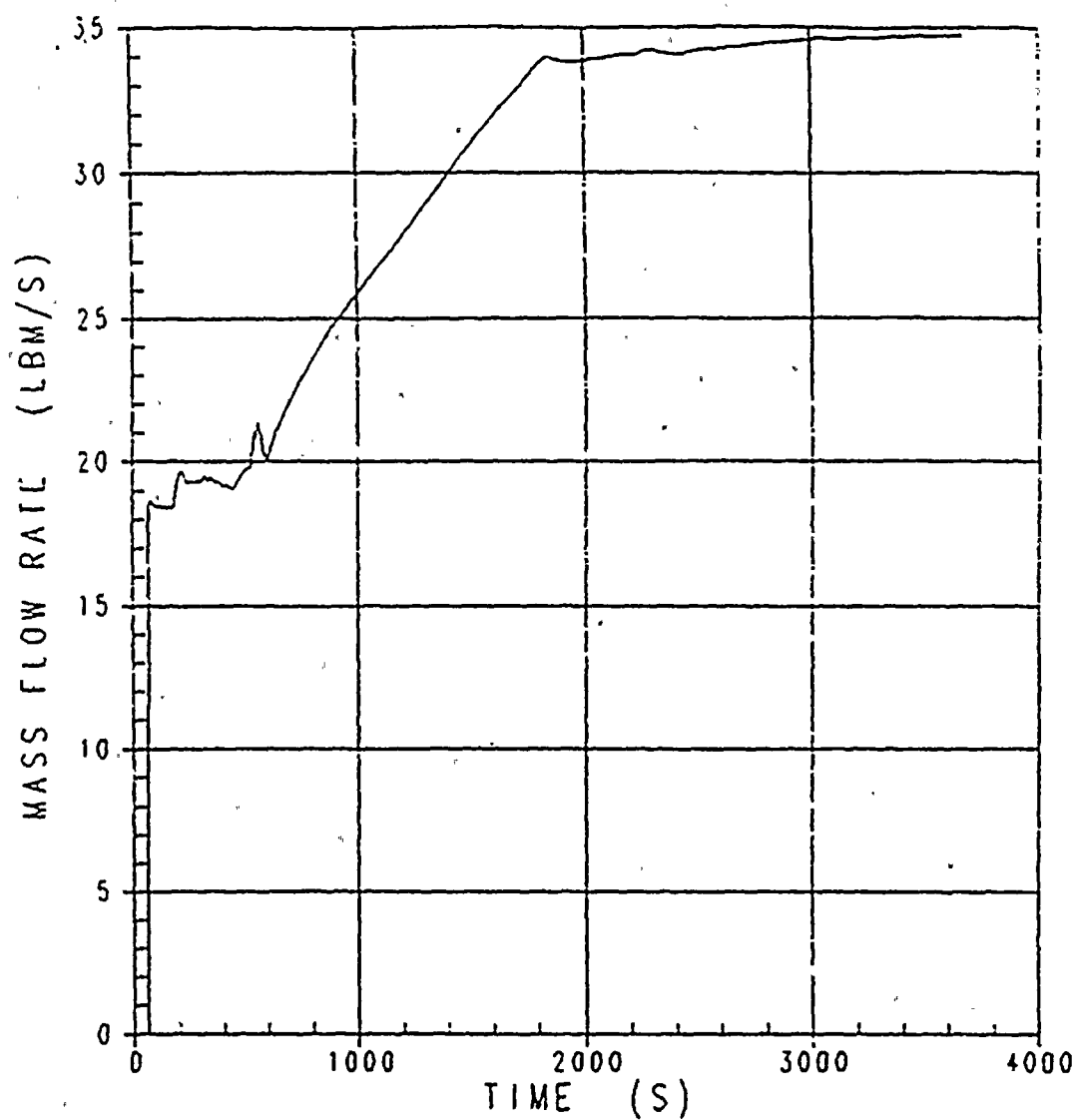


Figure 14.3.2-74
Broken Loop Safety Injection Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

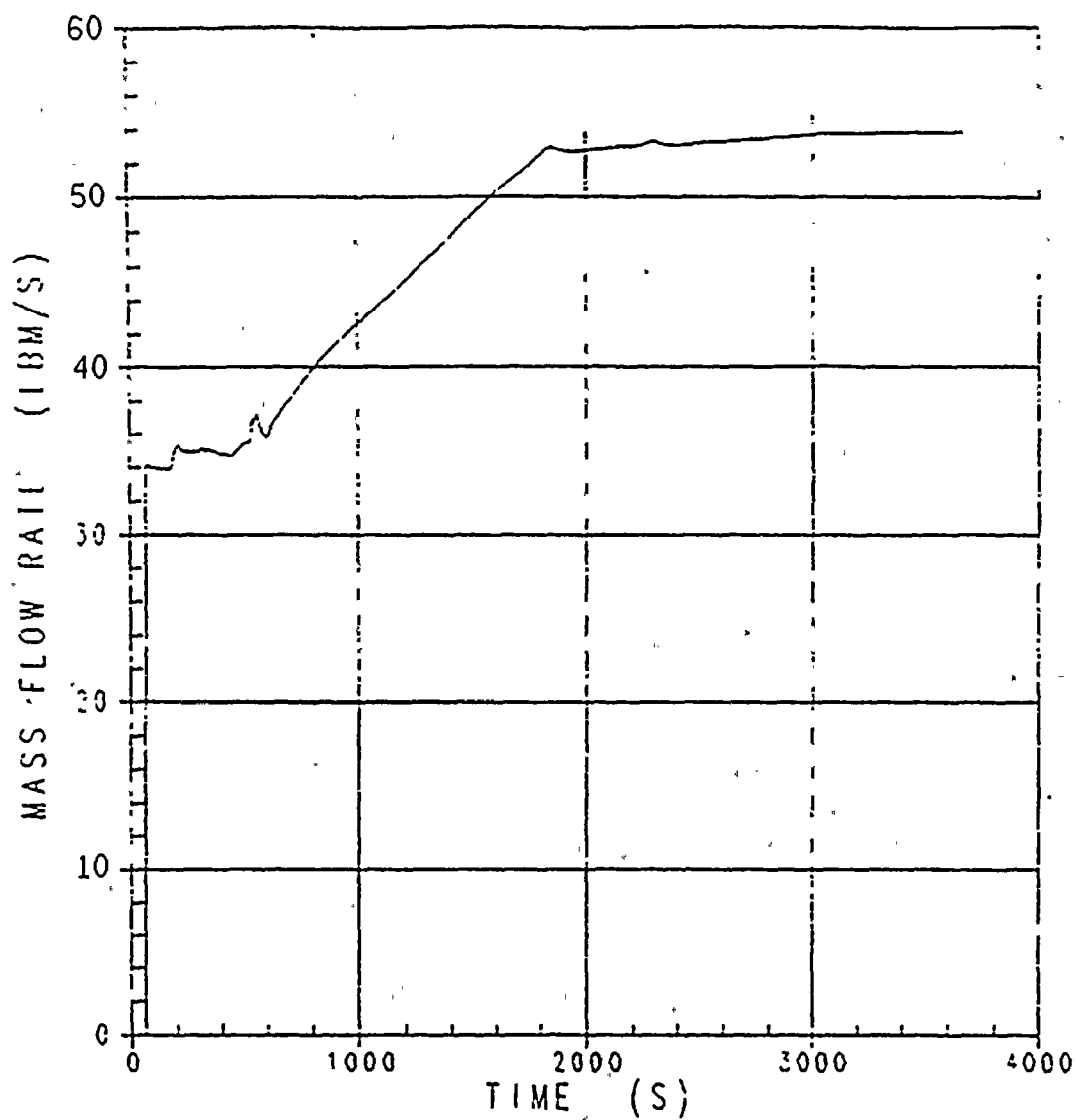


Figure 14.3.2-75
Lumped Intact Loop SI Mass Flow Rate (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

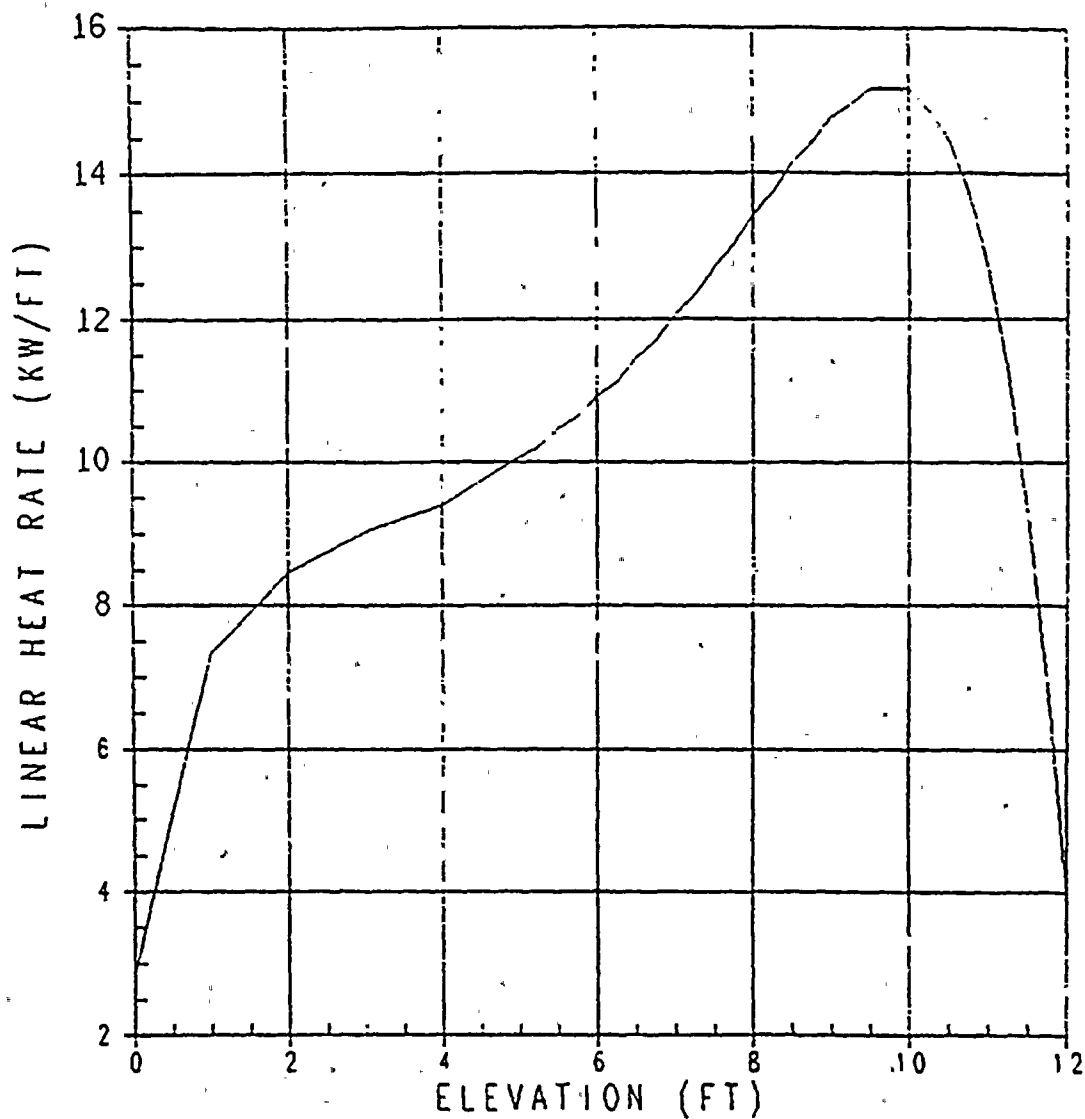


Figure 14.3.2-76
Hot Rod Power Distribution (3 Inch, 30% SGTP)
Reduced Temperature, Reduced Pressure
Donald C. Cook Nuclear Plant Unit 1

14.3.4 CONTAINMENT INTEGRITY ANALYSIS

14.3.4.1 CONTAINMENT STRUCTURE

14.3.4.1.1 Design Basis

The steel-lined, reinforced concrete containment structure, including foundations, access hatches, and penetrations is designed and constructed to maintain full containment integrity when subjected to accident temperatures and pressures, and the postulated earthquake conditions. Details of the Containment System, including General Design Criteria, are described in Chapter 5.

The containment design internal pressure is 12 psig. The effects of pipe rupture in the primary coolant system, up to and including a double-ended rupture of the largest pipe as well as a rupture of the main steam line, are considered in determining the peak accident pressure.

The internal structures of the containment vessel are also designed for subcompartment differential accident pressures. The accident pressures considered are due to the same postulated pipe ruptures as described for the containment vessel.

The other simultaneous loads in combination with the accident pressures, and the applicable load factors, are presented in detail in Chapter 5.

The functional design of the containment is based upon the following accident input source term assumptions and conditions:

1. The design basis accident blowdown mass and energy is put into the containment.
2. The hot metal energy is considered.
3. A reactor core power of 102% of 3413 MWt thermal power is used for decay heat generation.
4. Minimum Engineering Safety Features performance is assumed based upon the limiting single failure criterion.

The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended severance. Characterizing the performance of the ice condenser requires consideration of the rate of addition of mass and energy to the containment, as well as the total amounts of mass and energy added. Analyses have shown that the accident which produces the highest blowdown rate into the ice condenser containment results in the maximum containment pressure rise. That accident is the double-ended severance of a reactor coolant pipe.

Post-blowdown energy releases can also be accommodated without exceeding the containment design pressure.

14.3.4.1.2 Design Features

The reactor containment is a reinforced concrete structure consisting of a vertical cylinder, a hemispherical dome and a flat base. The interior is divided into three volumes, a lower volume which houses the reactor and Reactor Coolant System, an intermediate volume housing the energy absorbing ice bed in which steam is condensed and an upper volume which accommodates the air displaced from the other two volumes during a design basis pipe break accident.

The type of containment used for the Donald C. Cook Units 1 and 2 was selected for the following reasons:

1. The Ice Condenser Containment can accept large amounts of energy and mass inputs and maintain low internal pressures and leakage rates. A particular advantage of the ice condenser is its passive design not requiring an actuation signal.
2. The Ice Condenser Containment combines the required integrity, compact size, and carefully considered advanced design desirable for a nuclear station.

Consideration is given to subcompartment differential pressure resulting from a design basis accident. If an accident were to occur due to a pipe rupture in one of these relative small volumes, the pressure would build up at a faster rate than in the containment, thus imposing a differential pressure across the wall of the structure. Section 14.3.4.2, "Containment Subcompartments", presents the subcompartment differential pressure analyses.

The Ice Condenser Containment, incorporating forced circulation of the containment atmosphere together with the containment spray system, ensures the

During the blowdown transient, steam and air will flow through the ice condenser doors and also through the deck bypass area into the upper compartment. For the containment the bypass area is composed of two parts, a known leakage area of 2.2 ft² with a geometric loss coefficient of 1.5 through the deck drainage holes location at the bottom of the refueling cavity, and an undefined deck leakage area with a conservatively small loss coefficient of 2.5. Leakage through the backdraft damper of the air return fans was determined to be 0.18 sq. ft./damper and was considered in the known leakage area. A resistance network similar to that used in TMD is used to represent 6 lower compartment volumes, each with a representative portion of the deck leakage and the lower inlet door flow resistance adjacent to the lower compartment element. The inlet door flow resistance and flow area are calculated for small breaks that would only partially open these doors.

The coolant blowdown rate as a function of time is used with this flow network to calculate the differential pressures on the lower inlet doors and across the operating deck. The resultant deck leakage rate and integrated steam leakage into the upper compartment are then calculated. The lower inlet doors are initially held shut by the cold head of air behind the doors (approximately 1/2 - 1 pound per square foot). The initial blowdown from a small break opens the doors and removes the cold head on the doors. With the door differential pressure removed the door position is slightly open. An additional pressure differential of one pound per square foot is then sufficient to fully open the doors. The nominal door opening characteristic are based on test results.

One analysis conservatively assumed that flow through the postulated leakage paths is pure steam. During the actual blowdown transient, steam and air representative of the lower compartment mixture would leak through the holes; thus less steam would enter the upper compartment. If flow were considered to be a mixture of liquid and vapor, the total leakage mass would increase but the steam flow rate would decrease. The analysis also assumed that no condensing of the flow occurs due to structural heat sinks. The peak air compression in the upper compartment for the various break sizes is assumed with steam mass added to this value to obtain the total containment pressure. Air compression for the various break sizes is obtained from previous full scale section tests conducted at Waltz Mill.

The allowable leakage area for the following Reactor Coolant System break sizes was determined: DE, 0.6 DE, 3 ft², 8 inch diameter, 6 inch diameter, 2.5 inch diameter, and 0.5 inch diameter. For break sizes 3 ft² and above a series of deck leakage sensitivity studies was made to establish the total steam leakage to the upper compartment over the blowdown transient. This

steam was added to the air in the upper compartment to establish a peak pressure. Air and steam were assumed to be in thermal equilibrium, with the air partial pressure increased over the air compression value to account for heating effects. For these breaks, sprays were neglected. Reduction in compression ratio by return of air to the lower compartment was conservatively neglected. The results of this analysis are shown in Table 14.3.4-2. This analysis is confirmed by Waltz Mill tests conducted with various deck leaks equivalent to over 50 ft² of cack leakage for the double ended blowdown rate.

For breaks 8 inches in diameter and smaller, the effect of containment sprays was included. The method used is as follows: For each time step of the blowdown the amount of steam leaking into the upper compartment was calculated to obtain the steam mass in the upper compartment. This steam was mixed with the air in the upper compartment, assuming thermal equilibrium with air. The air partial pressure was increased to account for air heating effects. After sprays were initiated, the pressure was calculated based on the rate of accumulation of steam in the upper compartment. Reduction in pressure due to operation of the air recirculation fans has been conservatively neglected.

This analysis was conducted for the 8 inch, 6 inch and 2-1/2 inch break sizes assuming two spray pumps were operating (4000 gpm at 80°F). As shown in Table 14.3.4-2, the 8 inch break is the limiting case for this range of break sizes although the 0.6 DE is the limiting case for the entire spectrum of break sizes. With one spray pump operating (2000 gpm at 80°F) the limiting case for the entire spectrum of break sizes is the 8 inch case and results in an allowable deck leakage area of approximately 35 ft².

A second, more realistic, method was used to analyze this limiting case. This analysis assumed a 30 percent air, 70 percent steam mixture flowing through the deck leakage area. This is conservative considering the amount of air in the lower compartment during this portion of the transient. Operation of the deck fan would increase the air content of the lower compartment, thus increasing the allowable deck leakage area. Based on the LOTIC Code analysis a structural heat removal rate of over 8000 BTU/sec from the upper compartment is indicated. Therefore a steam condensation rate of 8 lb/sec was used for the upper compartment. The results indicated that with one spray pump operating and a deck leakage area of 36 ft², the peak containment pressure will be below design for the 8 inch case.

The 1/2 inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, either the lower compartment or the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area and that which can be tolerated without exceeding containment design pressure.

Effect of Dead-ended Volumes

There are several dead-ended compartments in the plant containment design which are connected to the lower compartment. The dead-ended volumes considered in the following analysis are the instrumentation room and the pipe trench. Additional study has shown that the fan accumulator rooms would also act as dead-ended volumes. Since the addition of dead-ended volume reduced peak compression pressure, the results presented for the following analysis are conservative.

In the preceding analysis of the containment compression ratio, it is conservatively assumed that only steam flows into the dead-ended volumes during the reactor coolant system blowdown. However, the results of certain full-scale section tests, which contained dead-ended volumes, showed that some air flowed into these volumes and remained there during the blowdown period, thus reducing the mass compression ratio for the containment. For example, one Waltz Mill test was run with the lower hemisphere of the receiver vessel vented to the lower compartment. From an air balance performed from pressure and temperature measurements at the time of peak compression pressure (9.6 psig), it was found that the ratio of the change in air mass to the initial mass in the dead-ended volume was:

$$\frac{\Delta M_a}{M_{a0}} = 0.18$$

This change in air mass is then corrected for the lower compression peak pressure of the plant design to give:

$$\frac{\Delta M_a}{M_{a0}} = 0.18 \times \frac{7.8 \text{ psig}}{9.6 \text{ psig}} = 0.15$$

The storage of air in the dead-ended volumes has the effect of reducing the mass of air stored in the downstream volumes at the time of the compression peak pressure.

The compression ratio for the Cook Nuclear plant taking into account the dead-ended volumes is found from the following:

$$C_r = \frac{V_1 + V_2 + V_3 - 0.15 V_1}{V_3 + 0.645 V_2} \quad (5)$$

where:

V_4 = Dead-ended volumes (instrument room and pipe trench)

Substituting the original design basis compartment volumes, which preceded those shown in Table 14.3.4-1, the compression ratio calculated from Equation (5) is:

$$C_r = \frac{1,179,636 - 0.15 \times 61,702}{745,896 + 0.645 \times 126,940}$$

$$C_r = 1.41$$

The final peak pressure is:

$$P_3 = 15.0 (1.41)^{1.13} + 0.4$$

$$P_3 = 22.5 \text{ psia or } 7.8 \text{ psig}$$

Therefore, the effect of the dead-ended volume of 61,702 ft³ is to decrease the final peak compression pressure by 0.2 psig. The magnitude of this effect was further substantiated by a series of tests at Waltz Mill which were run at a mass compression ratio closely representative of the Cook plant design. Tests were run with and without a dead-ended volume equivalent to 155,000 ft³ for the containment design. In these tests, the effect of the dead-ended volume was measured to be 0.5 psig, which is equivalent to a 0.32 psi decrease in final peak pressure per 100,000 ft³ of dead-ended volume.

Long Term Containment Pressure Analysis

Early in the ice condenser development program it was recognized that there was a need for modeling of long term ice condenser containment performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO Code) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC Code.⁽¹⁾

The model of the containment consists of five distinct control volumes, as follows: the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartments. The ice condenser control volume with unmelted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the Waltz Mill ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the Reactor Coolant System, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas and the thermodynamic properties of steam are taken from the ASME steam tables.

Peak Containment Pressure Transient

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Donald C. Cook Units 1 and 2 containments:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two residual heat removal pumps and one of two residual heat removal heat exchangers with cross-tie valves closed providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.
2. 2.11×10^6 pounds of ice initially in the ice condenser which is at 14°F . This temperature assumption maximizes the air mass in the ice condenser and is conservative with respect to the 27°F Technical Specification limit.
3. The blowdown, reflood, and post reflood mass and energy releases described in Section 14.3.4.3.1 were used.

4. Blowdown and post blowdown ice condenser drain temperatures of 190°F and 130°F are used. (These numbers are based on Reference 2.)
5. Nitrogen from the accumulators in the amount of 4510 pounds is included in the calculations.
6. Essential service water temperature of 87.5°F is used for the spray heat exchanger and the component cooling heat exchanger.
7. The air return fan is effective 10 minutes after the transient is initiated.
8. No maldistribution of steam flow to the ice bed is assumed.
9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)
10. The initial conditions in the containment are temperatures of 57°F in the upper, 60°F in the lower, 60°F in the dead ended and 14°F in the ice bed volumes. All volumes are at a pressure of 0.3 psig and 15 percent relative humidity, with the exception of the ice bed, which is at 100 percent relative humidity.
11. During the injection phase when the containment spray pumps are taking suction from the RWST, spray pump flow of 2075 gpm is used for the upper compartment and 1006 gpm for the lower compartment. During the recirculation phase when the containment spray pumps are taking suction from the sump, containment spray flow to the upper compartment is 2136 gpm, containment spray flow to the lower compartment is 1025 gpm.
12. RHR spray initiation is assumed after switchover from injection to recirculation has been completed and containment pressure is greater than or equal to 8 psig. A residual containment spray flowrate of 2000 gpm is used.
13. Containment structural heat sink data are assumed with conservatively low heat transfer rates, and may be found in Table 14.3.4-4.
14. The operation of one containment spray heat exchanger ($UA = 3.107 \times 10^6$ BTU/hr-°F) for containment cooling and the operation of one

residual heat removal heat exchanger ($UA = 2.22 \times 10^6$ BTU/hr-°F) for core cooling. The component cooling heat exchanger was modeled at 3.58×10^6 BTU/hr-°F.

15. The air return fan returns air at a rate of 39,000 cfm from the upper to lower compartment.
16. An active sump volume of 40,600 ft³ is used.
17. The refueling water storage tank is at a temperature of 105°F.
18. 102% of 3413 MWt power is used in the calculation.
19. Credit is taken for subcooling of the ECC water from the RHR heat exchanger.
20. Essential service water flow to the containment spray heat exchanger was modeled as 2000 gpm. The essential service water flow to the component cooling heat exchanger was modeled as 5000 gpm. The component cooling flow to the RHR heat exchanger was modeled as 5000 gpm.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 14.3.4-6, Containment pressure transient.

Figure 14.3.4-7, Upper compartment temperature transients.

Figure 14.3.4-8, Lower compartment temperature transients.

Figure 14.3.4-9, Active and inactive sump temperature transient.

Figure 14.3.4-10, Ice melt transient.

In addition, Table 14.3.4-5 gives energy accountings at various points in the transient.

The analysis results show that the maximum calculated containment pressure is 11.49 psig, for the double-ended pump suction minimum safeguards case. This

pressure peak occurs at approximately 7752 seconds, with ice bed meltout at approximately 5423 seconds.

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a 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. Table 14.3.4-4 is a summary of the containment structural heat sinks used in the analysis. The material property data used are found in Table 14.3.4-6.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami. An explanation of the manner of application is given in Reference (4).

When applying the Tagami correlation, a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to 72 BTU/hr-ft². This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

Relevant Acceptance Criteria

The LOCA mass and energy analysis has been performed in accordance with the criteria shown in the Standard Review Plan (SRP) section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K have been included by confirmation that the calculated pressure is less than the design pressure, and because all available sources of energy have been included, which is more restrictive than the old GDC criteria, appendix H of the original FSAR, to which the Donald C. Cook Plants are licensed. These sources include: reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, metal-water reaction energy, and stored energy in the secondary system.

Although the Donald C. Cook Nuclear Plant is not a Standard Review Plan plant, the containment integrity peak pressure analysis has been performed in accordance with the criteria shown in the SRP section 6.2.1.1.b, for ice condenser containments. Conformance to GDC's 16, 38, and 50 is demonstrated by showing that the containment design pressure is not exceeded at any time in the transient. This analysis also demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA.

Conclusions

Based upon the information presented for the Steam Generator Tube Plugging Program, it may be concluded that operation with the revised plant conditions and increased operating margins for the Donald C. Cook Nuclear Plant is acceptable. Operation with the RHR cross-tie valve closed was also shown to be

more limiting than operation with the valve open since there is less safety injection water available for steam condensation. Operation with the revised plant conditions, increased operating margins and the RHR cross-tie valve closed results in a calculated peak containment pressure of 11.49 psig, as compared to the design pressure of 12.0 psig. Thus, the most limiting case has been considered, and has been demonstrated to yield acceptable results.

14.3.4.1.3.2 Steam Line Break

Following a steam line break in the lower compartment of an ice condenser plant, two distinct analyses must be performed. The first analysis, the short term pressure analysis, has been performed with the TMD Code. The second analysis, the long term analysis, does not require the large number of nodes which the TMD analysis requires. The computer code which performs this analysis is the LOTIC⁽⁵⁾ Code.

The LOTIC Code includes the capability to calculate the superheat conditions, and has the ability to begin calculations from time zero.^(6, 7, 8) The major thermodynamic assumption which is used in the steam break analysis is complete re-evaporation of the condensate under superheated conditions for large breaks. For the most limiting small breaks, no re-evaporation is assumed; however, convective heat transfer as detailed in Reference (7) is used. The version of the LOTIC Code which incorporates the above is the LOTIC3 Code.⁽⁹⁾

This code was used to perform the steam line break analyses and is the version which has been accepted for this use. (10,11)

Peak Containment Temperature Transients

The following are the major input assumptions used in the LOTIC3 steam break analysis:

1. Minimum safeguards are employed, e.g., one of two spray pumps and one of two air return fans.
2. The air return fan is effective 10 minutes after the high-high containment pressure signal is read.
3. A uniform distribution of steam flow into the ice bed is assumed.
4. The total initial ice mass used water 2.11×10^6 lbs.
5. The initial conditions in the containment are a temperature of 120°F in the lower and dead ended volumes, a temperature of 57°F in the upper volume, and a temperature of 27°F in the ice condenser. All volumes are at a pressure of 0.3 psig and a relative humidity of 15%.
6. A spray pump flow of 2075 gpm is used in the upper compartment and 1006 gpm in the lower compartment. The spray initiation time assumed was 115 sec. after reaching the high-high setpoint.
7. The refueling water storage tank temperature is assumed to be 105°F.
8. The essential service water used on the spray heat exchanger and the component cooling water heat exchanger is modeled at a temperature of 87.5°F.
9. Containment structural heat sinks as presented in Table 14.3.4-4 were used.
10. The air return fan empties air at a rate of 39,000 cfm from the upper to the lower compartments.
11. The material property data given in Table 14.3.4-6 were used.

12. The mass and energy releases given in Tables 14.3.4-7 and 14.3.4-8 were used. Since these rates are considerably less than the RCS double ended breaks, and their total integrated energy is not sufficient to cause ice bed meltout, the containment pressure transients generated for the previously presented double ended pump suction RCS break is considerably more severe.
13. The heat transfer coefficients to the containment structures are based on the work of Tagami. An explanation of their manner of application is given in References (4, 6 and 7).

Results

The results of the analysis are presented in Table 14.3.4-9. The worst case of the double ended steam line breaks was a 1.4 ft² break, occurring at 102% power with main steam line isolation valve failure (MSIV). This temperature transient is shown in Figures 14.3.4-11-A and 14.3.4-11-B.

The results from the steam line split ruptures (or small breaks) are presented in Table 14.3.4-10. The worst case for these cases was a 0.942 ft² small break, occurring at 30% power, with a MSIV failure. A temperature transient of this case is presented in Figures 14.3.4-12-A and 14.3.4-12-B.

Parameter studies have been performed as part of previous analyses, varying the ice mass between 2.0 and 2.45 million pounds. These previous ice mass parameter studies have shown that the maximum containment calculated temperatures are not sensitive (less than 1°F change) to these ice mass changes.

Sensitivity of the Results

The previous section pertains to the steam line break analysis and its subsequent response in identifying the limiting small break. The following evaluation describes additional sensitivity studies of a generic nature, done for breaks smaller and up to 0.942 ft² at 30% power (12).

The LOTIC-3 computer code was employed in the generic analysis. The LOTIC-3 computer code⁽⁹⁾ was found to be acceptable for the analysis of steam line breaks with the following restrictions:

- a. Mass and energy release rates are calculated with an approved model.

ITEM	PEAK DIFFER. PRESSURE DP[1-25] DP[6-25]	PEAK DIFFER. PRESSURE DP[2-25] DP[5-25]	PEAK DIFFER PRESSURE DP(7,8,9 TO 25)	PEAK PRESSURE SHELL P40, P45
Structural design	16.6 psi	12.0 psi	12.0 psi	12.0 psi
Original base	14.1 psi	10.6 psi	8.2 psi	10.8 psi
New base	16.8 psi	12.2 psi	10.7 psi	13.1 psi
New total	18.7 psi	13.0 psi	11.2 psi	14.0 psi

Additionally, the peak calculated pressure for the internal shell elements 41-44 and the peak calculated differential pressure across the operating deck for elements 3 and 4 were below the 12.0 psi structural design value.

The previously mentioned calculated pressures and differential pressures exceed the original structural design basis. The structural adequacy of this compartment was evaluated using acceptance criteria found in Section 5.2.2.3 of the FSAR and was confirmed.

Early sensitivity studies, illustrated in Table 14.3 --21 (see Section 14.3.4.2.3, "Sensitivity Studies"), demonstrated the effects of changes in certain variables on the operating deck differential pressure and the shell pressure. The purpose of that study was to illustrate the sensitivity of the TMD code results to different input and assumption conditions and to illustrate the inherent analysis conservatism. The purpose of the tables was not to supply an extrapolation tool for all subcompartments since the work was done for a specific subcompartment and trends may be different for other compartments. For example, the effect of initial compartment pressure on the peak differential pressure can be either a benefit or a penalty depending upon the flow regime before and during the peak. Additionally, if the peak occurs later in time the trend will be geometry dependent. That is, the pertinent downstream element would pressurize differently based upon specific key variables, such as flow areas and resistance into and out of the element. A combination of both sonic and subsonic flow regime periods could occur over the total transient. Since the new analysis is sufficiently different when compared to the original sensitivity basis, Table 14.3.4-21 should only be used for guidance.

14.3.4.2.3.5 Short Term Containment Analysis Conclusions

The results of the short-term containment analyses and evaluations for the Cook Nuclear Power Plants demonstrate that, for the pressurizer enclosure, the fan accumulator room and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. For the loop compartments, the peak calculated pressures are higher than the FSAR design allowables; for these areas, structural evaluations were performed for these compartments for the revised peak pressures. The structural adequacy was confirmed through evaluations using Section 5.2.2.3 of the FSAR as acceptance criteria.

14.3.4.2.3.6 Reactor Cavity Evaluation

The design of the concrete structure surrounding the reactor vessel is designed for the following criteria.

1. Provide support for the reactor vessel under the dead weight, seismic, and reactor coolant pipe rupture loading conditions.
2. Attenuate the neutron flux sufficiently to prevent excessive activation of plant compartments.
3. Reduce the residual radiation from the core, reactor internals, and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.

As a result of criterion 1, the reactor support concrete structure will withstand the pressure that builds up within the annulus defined by the concrete cavity and the reactor vessel, following rupture of a reactor coolant pipe, without losing its structural integrity.

The reactor cavity pressure analysis was performed for Cook Nuclear Plant Units 1 and 2 for a NSSS power level of 3600 MWt. The purpose of this analysis is to calculate the initial pressure response in the reactor cavity to a loss of coolant accident. The reactor cavity pressure analysis was performed for the upper and lower reactor cavities, the reactor vessel annulus and the reactor pipe annulus.

The SGTP Program parameters affect the Reactor Cavity Pressure Analysis through the mass and energy releases provided as input into the analysis. There is no direct impact of SGTP level on short-term mass and energy release rate calculations. The major impact results from changes to RCS temperature. For short-term effects, higher release rates typically result from cooler RCS conditions. The mass and energy releases used as input for the Reactor Cavity Pressure Analysis reflected limiting conditions and therefore, the NSSS performance parameters for the SGTP Program did not impact the results.

As shown in Table 14.3.4-30, vent areas from the upper and lower reactor cavities were 175 and 70 square feet, respectively. The LOCA break flow split is such that 75% of the break flow discharges to the upper reactor cavity and loop compartments, with the remaining 25% entering the reactor annulus. Of

The calculated values are well below the design values. Therefore, structural integrity is ensured for the pipe annuli and reactor vessel annulus.

14.3.4.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

This analysis presents the mass and energy releases to the containment subsequent to a hypothetical loss-of-coolant (LOCA).

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS reaches initial equilibration with containment.
2. 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.
3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
4. 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.

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 froth 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.

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

For long-term considerations the break location analyzed is the double-ended pump suction guillotine break (10.48 ft²). Pump suction break mass and energy releases 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 is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting directly 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, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the releases would continually decrease). The mass and energy releases for the hot leg break have not been included in the scope of this containment integrity analysis because for the hot leg break only the blowdown phase of the transient is of any significance. Since there are no reflood and post-reflood phases to consider, the limiting peak pressure calculated would be the compression peak pressure and not the peak pressure following ice bed meltout.

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 is, in general, less limiting than 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 included in the scope of this analysis.

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 flow rates 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 all ice condenser plants.

In summary, the analysis of the limiting break location for an ice condenser containment has been performed. The double-ended pump suction guillotine break has historically been considered to be the limiting break location, by virtue of its consideration of all energy sources present in the RCS. This break location provides a mechanism for the release of the available energy in the RCS, including both the broken and intact loop steam generators. Inclusion of these energy sources conservatively results in the maximum amount of ice being melted in the event of a LOCA.

14.3.4.3.1 Mass and Energy Release Data

14.3.4.3.1.1 Short Term Mass and Energy Release Data

Early Design Analyses

The Mass and energy release rate transients for all the design cases are given in Figures 14.3.4-71 thru 14.3.4-78. All cases are generated with the SATAN-V break model consisting of Moody-Modified Zaloudek critical flow correlations applied at the break element. Since no mechanistic constraints have been established for full guillotine rupture, an instantaneous pipe severance and disconnection is assumed for all transients. Assumptions specific to the early design transients are as follows:

For the hot leg mass and energy release rate transient to loop compartments:

Figures 14.3.4-71, -72

1. A double ended guillotine type break.
2. A break located just outside the biological shield.
3. A break located in the worst loop.
4. A six node upper plenum model.
5. A 16 node broken hot leg pipe model.

6. A discharge coefficient (C_D) equal to 1.
7. A 100% power condition with $T_{hot} = 606.4^\circ\text{F}$ and $T_{cold} = 540.4^\circ\text{F}$.

For the cold leg mass and energy release rate transient to loop compartments:

Figures 14.3.4-73, -74

1. A double ended guillotine type break.
2. A break located just outside the biological shield.
3. A break located in the worst loop.
4. A seven node downcomer model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. A full power condition with $T_{hot} = 606.4^\circ\text{F}$ and $T_{cold} = 540.4^\circ\text{F}$.

For hot leg mass and energy release rate transients to subcompartments:

Figures 14.3.4-75, -76

1. A single ended split type break.
2. A break just outside the hot leg nozzle.
3. A break in the pressurizer loop.
4. A six node upper plenum model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. Full power condition $T_{hot} = 606.4^\circ\text{F}$ and $T_{cold} = 540.4^\circ\text{F}$.

For the cold leg mass and energy release rate transient to subcompartments:

Figures 14.3.4-77, -78

1. A single ended split type break.
2. A break just outside the cold leg nozzle.
3. A break in the pressurizer loop.
4. A seven node downcomer model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. A full power condition $T_{hot} = 606.4^\circ\text{F}$ and $T_{cold} = 540.4^\circ\text{F}$.

For the mass and energy release rate transient to the pressurizer enclosure, a 6 inch spray line pipe break was considered (Figures 14.3.4-79, -80):

1. A guillotine type break modeled as a 0.147 ft² split in the cold leg at the pump discharge (area of the six inch pressurizer spray feed line) and a 0.087 ft² split in the top of the pressurizer (area of 4 inch spray nozzle).
2. Valves in spray lines are assumed to be open.
3. No pipe resistance for the feed line considered.
4. A full power condition $T_{hot} = 606.4^{\circ}\text{F}$ and $T_{cold} = 540.4^{\circ}\text{F}$.
5. A discharge coefficient (C_D) equal to 1.

The mass and energy release rate transients for all the generated cases are supported by an extensive investigation of short term phenomena. Section 14.3.4.5 includes detailed discussion of the phenomena and the results.

Current Design Basis Analyses

Analyses were conducted to support changes in Reactor Power and revised RCS parameters, such as enthalpy, on the mass and energy releases. Details of the subcompartment evaluation are presented in Section 14.3.4.2.3.2 for the Pressurizer Enclosure Evaluation, Section 14.3.4.2.3.4 for the Loop Compartments Evaluation and, Section 14.3.4.2.3.6, for the Reactor Cavity Evaluation.

14.3.4.3.1.2 Long Term Mass and Energy Release Data

Application of Single Failure Analysis

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the pump suction (DEPS) break. 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 compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. 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, thereby minimizing the safety injection flow. As additional conservatism has been included in this analysis in that the closure of the RHR crosstie valve has been considered because it results in a further reduction in safety injection flow. The analysis further considers the RHR and SI pump head curves to be degraded by 15% and the charging pump head curve to be degraded by 10%. This results in the greatest SI flow reduction for the minimum safeguards case.

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 February 1978 ECCS calculation (Reference 32). The methodology for the use of this model is described in Reference 22.

Table 14.3.4-31 present the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

The mass and energy releases for the double-ended pump suction break, given in Table 14.3.4-31 terminate 28.0 seconds after the postulated accident.

Reflood Mass and Energy Release Data

The WREFLOOD code used for computing the reflood transient, is a modified version of that used in the ECCS calculation (Reference 32). The methodology for the use of this model is described in Reference 22.

The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system.

The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving ECCS water. Even though the Reference 22 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 33). This assumption is justified and supported by test data, and is 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 ECCS water. The second is a single phase mixing of condensate and ECCS water. Since the steam release 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 4), which are the largest scale data available and thus most clearly simulates 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 flowrates 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 22. 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 post-blowdown, limiting break for the containment integrity peak pressure analysis is the pump suction double ended rupture break. For this break, there are two flowpaths 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 team 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 22 and 23.

Table 14.3.4-33 presents the calculated mass and energy release for the reflood phase of the pump suction double ended rupture with minimum safety injection.

The transients of the principal parameters during reflood are provided in Table 14.3.4-35.

Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 21) is used for computing the post-reflood transient.

The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. The methodology for the use of this model is described in Reference 22.

After containment depressurization, the mass and energy release available to containment is generated directly from core boiloff/decay heat.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference 24 and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
3. Fission rate is constant over the operating history of maximum power level.
4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).
5. Operation time before shutdown is 3 years.
6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Tables 14.3.4-37 presents the two phase (froth) mass and energy release data for the double-ended pump suction break with minimum safety injection. Data for these tables are terminated at the end of froth time, after which the LOTIC code performs its own core boiloff calculation.

Sources of Mass and Energy

The sources of mass and energy considered in the LOCA mass and energy release analysis are given in Tables 14.3.4-39 and 14.3.4-40 for the double-ended pump suction break with minimum safety injection.

The mass sources are the reactor coolant system, accumulators, and pumped safety injection. The energy sources include:

1. Reactor coolant system water
2. Accumulator water
3. Pumped injection water
4. Decay Heat
5. Core stored energy
6. Reactor coolant system metal
7. Steam generator metal
8. Steam generator secondary energy
9. 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.

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 the analysis. Although Cook Nuclear Plant Unit 1 is not a Standard Review Plan Plant, 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:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time

4. End of reflood time
5. Time of broken loop steam generator equilibration
6. Time of intact loop steam generator equilibration

The methods and assumptions used to release the various energy sources are given in Reference 22 except as noted in the reflood mass and energy section, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

Significant Modeling Assumptions

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperatures of the reactor coolant system (100% full power conditions)
2. An allowance in temperature for instrument error and dead band (+5.1°F)
3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)
4. Core rated power of 3413 MWt
5. Allowance for calorimetric error (+2 percent of power)
6. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)
7. Allowance in core store energy for effect of fuel densification
8. A margin in core stored energy (+15 percent included to account for manufacturing tolerances)
9. An allowance for RCS initial pressure uncertainty (+67 psi)

10. Steam generator tube plugging leveling (0% uniform)

Maximizes reactor coolant volume and fluid release

Maximizes heat transfer area across the SG tubes

Reduces coolant loop resistance, which reduces delta-p upstream of break and increases break flow

Thus based on the above conditions and assumptions, a bounding analysis of Cook Nuclear Plant Units 1 and 2 is made for the release of mass and energy from the RCS in the event of a LOCA to support the SGTP Program.

14.3.4.4 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES INSIDE CONTAINMENT

A series of steamline breaks were analyzed to determine the most severe break condition for the containment temperature and pressure response. The assumptions on the initial conditions are taken to maximize the mass and energy released. The range of possible operating conditions for the Donald C. Cook Nuclear Plants are presented in Table 14.1-1 for Unit 1 and Table 14.1.0-1 for Unit 2. The subsections that follow discuss; the short-term mass and energy releases, which addresses steamline break effects in the steam generator enclosure and the fan accumulator room, followed by the long-term mass and energy releases.

in their final position and the pump is assumed to be at full speed and to draw suction from the RWSI. The volume containing the low concentration borated water is swept into the core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

- l. For the at-power cases, reactor trip is available by safety injection signal, overpower protection signal (high neutron flux reactor trip or OPAT reactor trip), and low pressurizer pressure reactor trip signal.
- m. Offsite power is assumed available. Continued operation of the reactor coolant pumps maximizes the energy transferred from the reactor coolant system to the steam generators.
- n. No steam generator tube plugging is assumed to maximize the heat transfer characteristics.

Break Flow Calculations

a. Steam Generator Blowdown

The LOFTRAN computer code (Reference 26) was used to calculate the break flows and enthalpies of the release through the steam line break. Blowdown mass/energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant thick metal heat storage, and reverse steam generator heat transfer.

b. Steam Plant Piping Blowdown

The calculated mass and energy releases include the contribution from the secondary steam piping. For all ruptures, the steam piping volume blowdown begins at the time of the break and continues until the entire piping inventory is released. The flow rate is determined using the Moody correlation and the pipe cross sectional area.

Single Failure Effects

- a. Failure of a main steam isolation valve (MSIV) increases the volume of steam piping which is not isolated from the break. When all

valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and the isolation valves in the other steam lines, including safety and relief valve headers and other connecting lines, will feed the break. For the cases which modeled a failure of a MSIV, the steam line volumes associated with Unit 2 were assumed since the volume available for blowdown for this scenario is greater than Unit 1. For the cases which did not model a failure of a MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than Unit 2.

- b. Failure of a diesel generator would result in the loss of one containment safeguards train, resulting in minimum heat removal capability.
- c. Failure of a feedwater isolation valve would result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flash into steam and exit through the break. For consistency with the FSAR steamline break mass/energy release analysis, all cases conservatively assumed failure of the feedwater isolation valve, which resulted in the additional inventory available for release through the steambreak and in higher than normal main feedwater flows.
- d. Failure of the auxiliary feedwater runout control equipment could result in higher auxiliary feedwater flows entering the steam generator prior to realignment of the auxiliary feedwater system. For cases where the runout control operates properly, a constant auxiliary feedwater flow of 775 gpm to the faulted steam generator was assumed. This value was increased to 1375 gpm to simulate a failure of the runout control.

The calculated mass and energy rates for the long term steamline break analysis at full power are presented as Tables 14.3.4-7 and 14.3.4-8.

25. Moody, F.J., "Maximum flow Rate of Single Component, Two-Phase Mixture," ASME publication, Paper NO. 64-HT-35.
26. Burnett, T. W. T., et. al., "LOFTRAN Code Description", WCAP-7907-A, April 1, 1984.
27. Zaloudek, F.R., "Steam-Water Critical Flow From High Pressure Systems," Interim Report, HW-68936, Hanford Works, 1964.
28. Henry, R.E., "A Study of One- and Two-Component, Two-Phase Critical Flows at Low Qualities," ANL-7430.
29. Henry, R.E., "An Experimental Study of Low-Quality, Steam-Water Critical Flow at Moderate Pressures," ANL-7740.
30. Kramer, F.W., "FLASH: A Program for Digital Simulation of the Loss-of-Coolant Accident," Westinghouse Atomic Power Division, WCAP-1678, January, 1961.
31. Zaloudek, F.R., "The Critical Flow of Hot Water Through Short Tubes," HW-77594, Hanford Works, 1963.
32. "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev. 1, (Non-Proprietary).
33. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1", June 9, 1989.

TABLE 14.3.4-1

DONALD C. COOK ICE CONDENSER ANALYSIS PARAMETERS

Reactor Containment Volume (net free volume)	
Upper Compartment, ft ³	774,481
Ice Condenser, ft ³	110,520
Lower Compartment (active), ft ³	301,583
Total Active Volume, ft ³	1,186,584
Lower Compartment (dead ended), ft ³	60,727
Total Containment Volume, ft ³	Not Applicable
Reactor Containment Air Compression Ratio	1.403
NSSS Power, MWt	3413
Design Energy Release to Containment	
Initial blowdown mass release, lb	542,360
Initial blowdown energy release, BTU	334.4 x 10 ⁶
Ice Condenser Parameters	
Weight of ice in condenser, lb	2.11 x 10 ⁶
<u>Additional System Parameters</u>	
Core Inlet Temperature (+5.1°F), °F	552.5*
Initial Steam Generator Steam Pressure, psia	836.3
Assumed Maximum Containment Back Pressure, psia	26.7

This is information utilized in the current containment pressure analysis discussed in Section 14.3.4.1.3.1.

*Includes +4.1°F allowance for instrument error, deadband, and +1°F for cold leg streaming.

TABLE 14.3.4-2

DECK LEAKAGE SENSITIVITY

<u>Break Size</u>	<u>5 ft² Deck Leak Air Compression Peak (psig)</u>	<u>Deck Leakage Area (ft²)</u>	<u>Spray Flow Rate (gpm)</u>	<u>Resultant Peak Contain- ment Pressure (psig)</u>
Double ended	7.8	54	0	12.0
0.6 double ended	6.6	46	0	12.0
3 ft ²	6.25	50	0	12.0
8-inch diameter	5.5	56	4000	12.2
8-inch diameter	5.5	35	2000	12.0
8-inch diameter*	5.5	56	2000	11.3
6-inch diameter	5.0	56	4000	10.4
2 1/2-inch diameter	4.0	56	4000	8.5
1/2-inch diameter	3.0	50	4000	3.0

*This case assumes upper compartment structural heat sink steam condensation of 8 lb/sec and 30 percent of deck leakage is air.

TABLE 14.3.4-3

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TABLE 14.3.4-4

STRUCTURAL HEAT SINK TABLE

<u>SURFACES</u>		<u>AREA (FT²)</u>	<u>THICKNESS (FT)</u>
<u>Upper Compartment Material</u>			
1.	Paint	32500.	0.001083
	Carbon Steel	32500.	0.0469
	Concrete	32500.	2.0
2.	Paint	10086.	0.001083
	Concrete	10086.	2.0
3.	Paint	5380.	0.001250
	Concrete	5380.	1.5
4.	Paint	11970.	0.00125
	Concrete	11970	1.0
<u>Lower Compartment Material</u>			
5.	Paint	5069.	0.00125
	Concrete	5069.	2.0
6.	Paint	13660.	0.00125
	Concrete	13660	1.5
7.	Paint	16730.	0.00125
	Concrete	16730.	1.0
8.	Paint	8665.	0.00125
	Concrete	8665.	2.0
<u>Ice Condenser</u>			
9.	Steel	180600.	0.00663
10.	Steel	76650.	0.0217
11.	Steel	28670.	0.0267

TABLE 14.3.4-4 (Cont'd)

STRUCTURAL HEAT SINK TABLE

	<u>SURFACES</u>	<u>AREA (FT²)</u>	<u>THICKNESS (FT)</u>
12.	Paint	3336.	0.000833
	Concrete	3336.	0.333
13.	Steel and Insulation	19100.	1.0
	Steel	19100.	0.0625
14.	Steel and Insulation	13055.	1.0
	Concrete	13055.	1.0

TABLE 14.3.4-5

ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. End of Blowdown (<u>t=10.0 sec</u>)	Approx. End of Reflood (<u>t=249.7 sec</u>)
* Ice Heat Removal	207.7	250.3
* Structural Heat Sinks	17.37	44.73
* RHR Heat Exchanger Heat Removal	0	0
* Spray Heat Exchanger Heat Removal	0	0
Energy Content of Sump	188.94	250.0
Ice Melted (Pounds)	0.67 (10^6)	0.84 (10^6)

* Integrated Energies

TABLE 14.3.4-5 (Cont'd)

ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. Time of Ice Melt Out <u>(t=5423 sec)</u>	Approx. Time of Peak Pressure <u>(t=7752 sec)</u>
* Ice Heat Removal	567.21	567.21
* Structural Heat Sinks	82.52	112.68
* RHR Heat Exchanger Heat Removal	49.0	77.31
* Spray Heat Exchanger Heat Removal	58.31	92.3
Energy Content of Sump	583.6	599.3
Ice Melted (Pounds)	2.11	2.11

* Integrated Energies

TABLE 14.3.4-6

MATERIAL PROPERTY DATA

<u>Material</u>	Thermal Conductivity <u>BTU/hr-ft-°F</u>	Volumetric Heat Capacity <u>BTU/ft³-°F</u>
Paint	0.0833	28.4
Concrete	0.8	22.6
Steel	26.0	56.4
Steel and Insulation	0.2	3.663

TABLE 14.3.4-7

UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT
 102% POWER, 1.4 ft₂ DOUBLE ENDED RUPTURE
 FAILURE - MSIV

<u>TIME (sec)</u>	<u>MASS (lbm/sec)</u>	<u>ENERGY (MBtu/sec)</u>
.0000	.0000	.0000
.2000	9753.	11.68
1.400	8708.	10.45
3.800	7436.	8.940
6.000	7228.	8.693
8.000	7069.	8.504
10.00	6882.	8.281
11.60	6658.	8.014
12.00	6441.	7.752
12.20	6224.	7.490
13.00	5353.	6.443
14.20	4047.	4.871
15.20	2959.	3.562
16.00	2090.	2.516
16.40	1657.	1.995
17.00	1482.	1.784
22.00	1306.	1.572
24.00	1253.	1.509
26.00	1208.	1.455
28.00	1169.	1.408
30.00	1137.	1.369
32.00	1110.	1.337
34.00	1087.	1.309
36.00	1068.	1.285
45.00	1006.	1.211
75.00	878.0	1.056
100.0	851.6	1.024
200.0	831.4	.9998
280.0	825.3	.9924
282.5	789.4	.9485
285.0	695.6	.8349
287.5	619.9	.7431
292.5	455.2	.5429
300.0	241.3	.2850
320.0	112.0	.1308
350.0	106.5	.1244
610.0	3.231	.0038

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TABLE 14.3.4-8

UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT
 30% POWER, 0.942 FT₂ SPLIT BREAK
 FAILURE - MSIV

<u>TIME (sec)</u>	<u>MASS (lbm/sec)</u>	<u>ENERGY (MBtu/sec)</u>
.0000	.0000	.0000
.2000	1873.	2.234
5.600	1744.	2.085
7.000	1734.	2.073
10.00	1718.	2.054
13.00	1703.	2.036
13.60	1698.	2.031
14.80	1688.	2.020
15.60	1681.	2.011
16.00	1677.	2.007
18.00	1629.	1.950
20.00	1522.	1.825
26.00	1284.	1.544
35.00	1061.	1.277
40.00	974.2	1.173
45.00	905.9	1.091
50.00	853.1	1.028
60.00	782.0	.9418
70.00	741.1	.8925
80.00	719.1	.8659
90.00	707.4	.8517
100.0	701.3	.8444
110.0	698.3	.8408
120.0	696.7	.8388
150.0	695.1	.8369
200.0	694.1	.8357
270.0	692.9	.8342
290.0	691.3	.8323
292.5	667.1	.8028
295.0	607.8	.7312
297.5	554.0	.6658
320.0	476.8	.5711
337.5	403.4	.4820
352.5	344.5	.4106
367.5	296.7	.3531
395.0	183.5	.2174
410.0	136.6	.1609
432.5	114.3	.1345
495.0	106.6	.1252
605.0	109.5	.1282

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TABLE 14.3.4-9

1.4 FT2 DOUBLE-ENDED STEAMLINE BREAKS

Operating Power, %	102	70
Aux. Feed Failure	w/o	w/o
MSIV Failure	w	w
$T_{\max}, ^\circ\text{F}$	322.71	321.72
Time of T_{\max} , sec	7.03	7.92

4.6 FT2 DOUBLE-ENDED STEAMLINE BREAKS

Operating Power, %	102	70
Aux. Feed Failure	w/o	w/o
MSIV Failure	w	w
$T_{\max}, ^\circ\text{F}$	322.68	321.9
Time of T_{\max} , sec	6.11	6.13

TABLE 14.3.4-10

STEAMLINE RUPTURES

Size of Break, ft ²	0.86	0.942	0.942
Hot Operating Power %	102	30	30
Aux. Feed Failure	w	w/o	w
MSIV Failure	w/o	w	w/o
T _{max} , °F	325.9	326.0	325.6
Time of T _{max} , sec	83.4	73.5	86.1

TABLE 14.3.4-30

<u>Compartment</u>	<u>Free Volume</u> (ft ³)	<u>Vent Area</u> (ft ²)
Upper Reactor Cavity	19,731	175
Lower Reactor Cavity	14,335	70
Steam Generator	7,956	264
Pressurizer	3,537	42
Fan Room	26,423	226

TABLE 14.3.4-31
DOUBLE-ENDED PUMP SUCTION
BLOWDOWN MASS AND ENERGY RELEASE

TIME SECONDS	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
.000	.0	.0	.0	.0
.101	40932.2	22363.5	21842.5	11892.7
.201	41809.9	22984.5	23698.7	12911.5
.301	46791.4	23944.9	23386.2	12864.8
.401	47304.8	26505.7	23083.2	12605.8
.601	44920.5	25777.3	21588.6	11803.0
.800	45062.1	26395.2	19910.9	10889.6
1.10	41715.9	25037.9	18871.6	10327.0
1.40	38993.8	23888.1	18466.4	10106.3
2.30	31603.5	20634.8	17983.4	9642.5
2.80	26248.9	17695.2	17008.4	9311.5
3.00	21790.0	14847.8	16632.6	9107.5
3.40	19301.6	13347.2	15908.2	8714.5
3.90	18078.7	12537.0	15034.4	8239.8
4.60	15511.6	10751.8	13993.7	7674.4
5.20	13955.8	9663.8	13348.7	7323.9
6.20	12521.1	8669.3	12570.9	6900.4
6.60	12409.8	8475.8	13180.5	7237.4
8.00	12926.3	9608.2	12562.2	6907.8
8.40	12289.7	8355.6	12442.3	6843.7
8.80	10124.1	7526.0	12205.8	6713.4
9.40	9779.3	7313.0	11841.3	6514.6
11.0	9523.6	6842.3	10932.1	5959.4
13.8	7137.6	5509.2	9219.1	5074.5
16.4	5492.8	4593.4	7813.7	4311.0
18.4	4381.2	3819.6	6705.5	3474.5
18.8	4172.2	3729.5	9594.3	4972.0
19.0	4012.7	3668.6	5136.1	2661.2
19.2	3961.1	3645.6	8361.9	4245.4
19.4	3820.3	3688.4	8618.3	4403.8
19.6	3734.8	3730.0	5453.2	2703.5
19.8	3554.0	3671.6	9394.0	4148.0
20.0	3322.7	3582.1	4907.9	2449.3
20.2	3066.4	3451.8	6907.8	3152.7
20.6	2642.1	3188.3	4697.0	2086.8
20.8	2411.6	2951.7	5832.6	2567.9
21.0	2237.3	2755.6	3815.9	1683.9
21.4	1909.0	2365.5	4740.7	1974.2
21.8	1705.5	2122.4	2922.4	1187.0
22.0	1562.8	1947.9	6501.6	2457.3
22.2	1445.6	1805.1	4567.9	1744.3
22.4	1337.0	1672.2	5030.5	1926.8
22.8	1176.0	1475.1	3300.6	1231.8
23.6	914.5	1151.5	1329.9	459.4
23.8	833.4	1050.2	2964.0	796.5
24.4	475.4	600.7	2402.3	633.4
24.6	413.0	523.1	2443.3	754.2
25.8	140.8	179.4	2309.9	648.0
27.0	66.5	85.2	299.7	93.9
28.0	19.0	24.5	126.0	45.3

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TABLE 14.3.4-33
DOUBLE-ENDED PUMP SUCTION
MINIMUM SI REFLOOD MASS AND ENERGY RELEASE

TIME SECONDS	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
28.0	.0	.0	.0	.0
28.3	143.2	168.1	1818.3	162.7
28.5	122.8	144.3	1791.4	160.3
28.7	110.9	130.3	1778.9	159.2
29.7	108.0	126.8	1717.9	153.8
30.0	119.6	140.4	1696.2	151.8
30.7	123.7	147.6	1638.1	148.4
34.1	148.1	174.1	1508.0	135.0
36.2	160.6	188.8	1431.7	128.1
38.1	170.9	200.9	1371.5	122.8
39.1	213.4	231.1	2017.3	223.3
40.1	351.3	414.5	3809.9	331.6
41.2	371.3	438.4	4023.2	341.6
42.2	368.0	434.6	3989.7	341.3
43.2	362.3	427.2	3929.7	344.9
43.2	350.6	413.5	3807.4	360.8
47.2	339.7	400.8	3691.6	347.3
48.2	335.6	419.7	3602.6	363.8
49.2	335.3	419.3	3683.2	361.6
51.2	346.0	409.2	3747.0	349.9
53.2	337.3	397.9	3694.5	338.9
55.2	329.2	388.3	3607.1	328.5
57.2	321.6	379.3	3524.8	318.7
59.2	314.5	370.8	3446.8	309.4
61.2	307.8	362.9	3372.4	300.5
62.3	313.7	369.5	338.7	190.2
63.3	456.3	539.8	301.8	244.3
64.3	448.9	531.0	299.4	240.1
68.3	416.7	490.1	282.9	219.4
70.3	399.6	472.1	276.3	210.4
74.3	372.0	439.2	263.7	194.1
78.3	347.1	409.5	252.5	179.5
81.3	329.9	389.1	244.9	169.7
82.3	324.5	382.6	242.6	166.5
86.3	304.0	358.3	213.4	154.9
93.3	272.4	321.1	220.1	137.4
97.3	257.3	302.8	211.5	128.9
101.3	243.3	286.3	207.4	121.3
105.3	230.9	271.5	202.4	114.6
107.3	225.1	264.7	200.0	111.6
117.3	201.1	236.3	190.1	96.9
127.3	183.5	215.4	183.0	89.8
135.3	173.0	203.0	178.8	84.4
143.3	165.1	193.7	175.8	80.4
153.3	158.1	185.5	173.1	76.9
163.3	153.5	180.0	171.4	74.5
173.3	150.1	176.0	170.1	72.7
193.3	148.0	173.5	169.4	71.4
223.3	148.4	173.4	169.6	70.9
249.7	150.3	175.9	170.5	71.4

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TABLE 14.3.4-35
DOUBLE-ENDED PUMP SUCTION
MINIMUM SI PRINCIPAL PARAMETERS DURING REFLOOD

TIME SECONDS	FLOODING		CARRYOVER	CORE	DOWNCOMER	FLOW	TOTAL	INJECTION		ENTHALPY BTU/LBM
	TEMP DEGREE F	RATE IN/SEC	FRACTION	HEIGHT FT	HEIGHT FT	FRACTION		ACCUMULATOR (POUNDS MASS PER SECOND)	SPILL	
28.0	219.9	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.5	217.1	19.962	.000	.56	.35	1.000	7168.8	7168.8	.0	89.50
28.9	215.1	8.452	.000	1.05	.55	1.000	707.4	707.4	.0	89.50
29.2	214.8	3.412	.018	1.18	.97	1.000	6999.3	6999.3	.0	89.50
29.4	214.8	4.303	.041	1.24	1.27	1.000	6953.1	6953.1	.0	89.50
30.5	215.1	2.570	.314	1.50	3.0	.598	6672.9	6672.9	.0	89.50
33.1	216.7	2.141	.534	1.77	7.00	.485	6194.7	6194.7	.0	89.50
36.2	218.7	2.220	.635	2.00	11.55	.445	5728.8	5728.8	.0	89.50
41.2	221.6	3.530	.710	2.36	15.99	.578	4713.3	4713.3	.0	89.50
43.0	222.6	3.401	.724	2.50	16.00	.575	4549.3	4549.3	.0	89.5
44.2	223.4	3.319	.731	2.60	16.00	.573	4455.0	4455.0	.0	89.5
47.2	225.3	3.149	.742	2.81	16.00	.566	4240.3	4240.3	.0	89.50
48.2	226.0	3.256	.745	2.87	16.00	.576	4472.3	4049.6	.0	87.94
50.1	227.2	3.177	.750	3.00	16.00	.575	4397.4	3971.2	.0	87.90
58.2	232.9	2.899	.761	3.50	16.00	.562	3981.5	3548.8	.0	87.71
61.2	235.1	2.817	.764	3.67	16.00	.558	3851.3	3419.7	.0	87.64
62.3	235.8	3.092	.765	3.73	15.98	.612	437.7	.0	.0	72.99
63.3	236.5	3.715	.763	3.80	15.79	.628	403.2	.0	.0	72.99
66.2	238.4	3.515	.766	4.01	15.23	.626	408.8	.0	.0	72.99
74.3	242.8	3.035	.770	4.52	13.95	.621	421.8	.0	.0	72.99
83.3	244.3	2.628	.711	5.00	12.92	.614	431.8	.0	.0	72.99
95.3	242.3	2.224	.769	5.56	12.02	.604	441.3	.0	.0	72.99
106.3	243.4	1.949	.768	6.00	11.55	.594	447.3	.0	.0	72.99
121.3	244.0	1.694	.766	6.53	11.31	.581	452.4	.0	.0	72.99
136.3	243.2	1.537	.764	7.00	11.36	.571	455.8	.0	.0	72.99
155.3	244.3	1.423	.765	7.55	11.68	.563	458.8	.0	.0	72.99
171.8	243.5	1.375	.765	8.00	12.07	.560	460.6	.0	.0	72.99
191.3	244.3	1.347	.768	8.52	12.61	.559	462.4	.0	.0	72.99
210.0	243.8	1.338	.770	9.00	13.17	.581	463.9	.0	.0	72.99
229.3	244.3	1.334	.774	9.49	13.75	.563	465.3	.0	.0	72.99
231.3	244.3	1.334	.774	9.54	13.81	.563	465.5	.0	.0	72.99
249.7	244.1	1.337	.776	10.00	14.37	.565	466.7	.0	.0	72.99

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TABLE 14.3.4-37
DOUBLE-ENDED PUMP SUCTION
MINIMUM SI POST REFLOOD MASS AND ENERGY RELEASE

TIME SECONDS	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
249.7	200.8	251.0	285.3	103.3
254.7	201.2	251.5	284.9	103.2
259.7	200.4	250.6	285.7	103.2
264.7	200.8	251.0	285.3	103.0
269.7	200.0	250.0	286.1	103.1
274.7	200.3	250.5	285.8	102.9
279.7	199.6	249.5	286.6	103.0
284.7	199.9	249.8	286.3	102.8
289.7	199.0	248.8	287.1	102.9
299.7	199.6	249.5	286.6	102.5
304.7	198.7	248.5	287.4	102.6
309.7	199.0	248.7	287.2	102.5
314.7	198.1	247.7	288.0	102.5
324.7	198.5	248.1	287.6	102.2
329.7	197.6	247.0	288.5	102.3
339.7	197.9	247.4	288.2	102.0
344.7	197.0	246.2	289.2	102.1
359.7	197.2	246.5	289.0	101.7
364.7	196.2	245.3	289.9	101.8
399.7	195.8	244.7	290.4	101.2
404.7	194.8	243.5	291.4	101.3
424.7	194.4	243.1	291.7	100.9
449.7	193.3	241.6	292.9	100.7
454.7	193.8	242.3	292.3	100.4
469.7	192.5	240.6	293.7	100.4
474.7	192.8	241.0	293.3	100.2
489.7	191.7	239.7	294.4	100.2
499.7	191.7	239.7	294.4	99.9
519.7	190.7	238.4	295.5	99.7
529.7	190.9	238.7	295.2	99.5
534.7	190.1	237.7	296.0	99.5
544.7	190.1	237.7	296.0	99.3
549.7	189.5	236.9	296.7	99.4
554.7	189.9	237.4	296.2	99.2
574.7	188.6	235.8	297.5	99.0
579.7	188.6	235.8	297.5	98.9
609.7	187.3	234.1	298.8	98.5
614.7	82.0	102.5	404.1	120.8
870.8	82.0	102.5	404.1	120.8
870.9	79.9	94.4	406.3	88.7
874.7	79.8	99.4	406.3	155.7
1979.7	64.8	75.2	421.4	84.8
1982.3	64.7	80.5	103.3	96.5
2222.2	64.7	80.5	103.3	96.5
2222.3	63.4	78.8	332.2	158.0
2316.8	63.4	78.8	332.2	158.0

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TABLE 14.3.4-39
DOUBLE-ENDED PUMP SUCTION
MINIMUM SI

		MASS BALANCE					
TIME (SECONDS)		.00	28.00	28.00	249.66	870.94	2316.82
		MASS (THOUSAND LBM)					
INITIAL	IN RCS AND ACC	771.32	771.32	771.32	771.32	771.32	771.32
ADDED MASS	PUMPED INJECTION	.00	.00	.00	91.01	393.02	1087.36
	TOTAL ADDED	.00	.00	.00	91.01	393.02	1087.36
TOTAL AVAILABLE		771.32	771.32	771.32	862.33	1164.34	1858.68
-DISTRIBUTION	REACTOR COOLANT	537.32	57.74	67.87	135.93	135.93	135.93
	ACCUMULATOR	234.00	171.20	161.07	.00	.00	.00
	TOTAL CONTENTS	771.32	228.94	228.94	135.93	135.93	135.93
EFLUENT	BREAK FLOW	.00	542.36	542.36	726.39	1028.39	1722.73
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	542.36	542.36	726.39	1028.39	1722.73
TOTAL ACCOUNTABLE		771.32	771.30	771.30	862.31	1164.32	1858.66

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TABLE 14.3.4-40
DOUBLE-ENDED PUMP SUCTION
MINIMUM SI

		ENERGY BALANCE					
TIME (SECONDS)	.00	.00	28.00	28.00	249.66	870.94	2316.82
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS, ACC, S GEN	901.43	901.43	901.43	901.43	901.43	901.43
ADDED ENERGY	PUMPED INJECTION	.00	.00	.00	6.64	28.69	84.57
	DECAY HEAT	.00	8.98	8.96	34.20	87.00	181.86
	HEAT FROM SECONDARY	.00	-5.10	-5.10	-5.10	-2.19	4.03
	TOTAL ADDED	.00	3.87	3.87	35.75	113.49	270.46
TOTAL AVAILABLE		901.43	905.30	905.30	937.18	1014.92	1171.89
DISTRIBUTION	REACTOR COOLANT	318.00	12.74	13.64	30.54	30.54	30.54
	ACCUMULATOR	20.94	15.32	14.42	.00	.00	.00
	CORE STORED	28.06	13.71	13.71	3.19	3.16	2.92
	PRIMARY METAL	178.97	168.74	168.74	143.60	92.94	63.41
	SECONDARY METAL	84.19	84.08	84.08	77.83	59.46	35.57
	STEAM GENERATOR	271.26	275.82	275.82	252.03	189.66	116.14
	TOTAL CONTENTS	901.43	570.41	570.41	507.19	375.76	248.57
EFFLUENT	BREAK FLOW	.00	334.41	334.41	421.73	630.90	916.99
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	334.41	334.41	421.73	630.90	916.99
TOTAL ACCOUNTABLE		901.43	904.82	904.82	928.92	1006.66	1165.56

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TABLE 14.3.4-41

PARAMETERS USED IN BOUNDING STEAMLINE BREAK
MASS/ENERGY RELEASES FOR UNIT 1 AND UNIT 2

<u>Parameter</u>	<u>Parameter Value</u>
NSSS Power, MWt	3608
Core Power, MWt	3588
RCS Flow, gpm/loop	88500
Minimum Measure Flow, gpm/loop	91600
RCS Temperature, °F	618.0
Core Outlet	615.2
Core Average	584.6
Vessel Average	581.3
Vessel/Core Inlet	547.3
Steam Generator Outlet	547.1
Zero Load	547.0
RCS Pressure, psia	2250
	or
	2100
Steam Pressure, psia	820
Steam Flow (10 ⁶ lb/hr total)	16.00
Feedwater Temp., °F	449.0
SG Tube Plugging, %	0

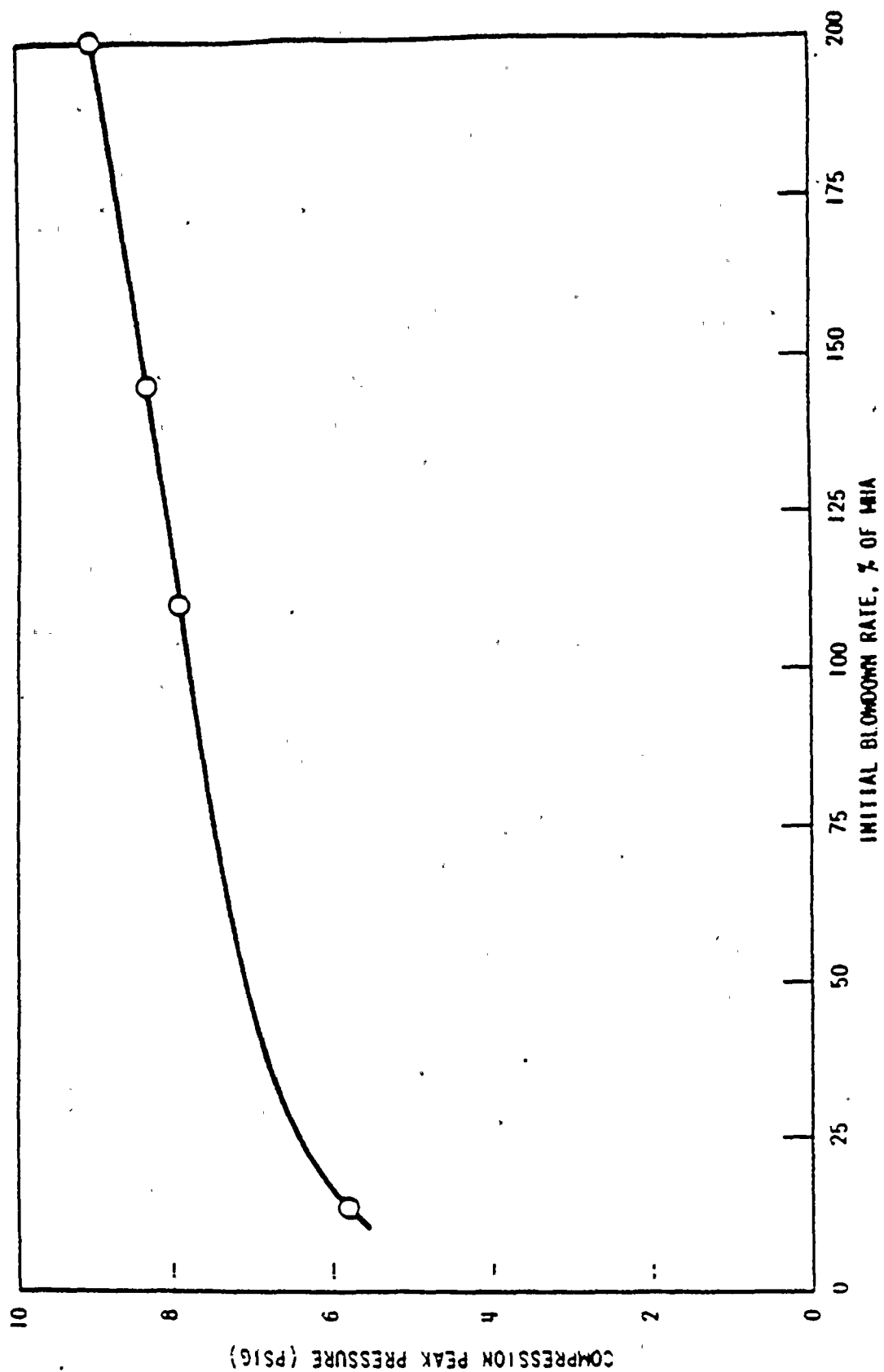
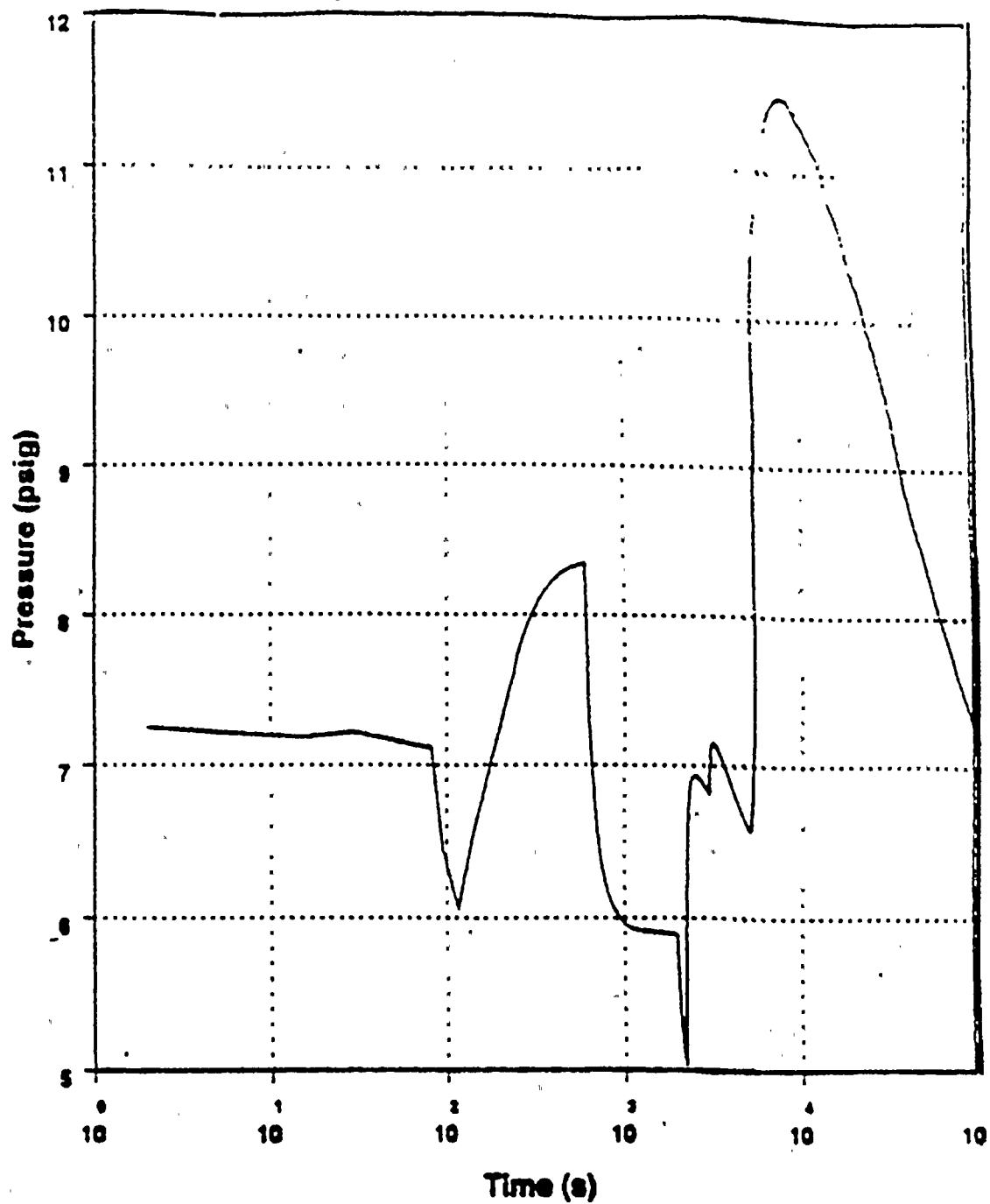


Figure 14.3.4-5 Upper Compartment Peak Compression Pressure versus Blowdown Rate for Tests With 175% Energy Release



Containment Pressure

Figure 14.3.4-6 LOCA Mass Energy Release Containment Integrity
Containment Pressure

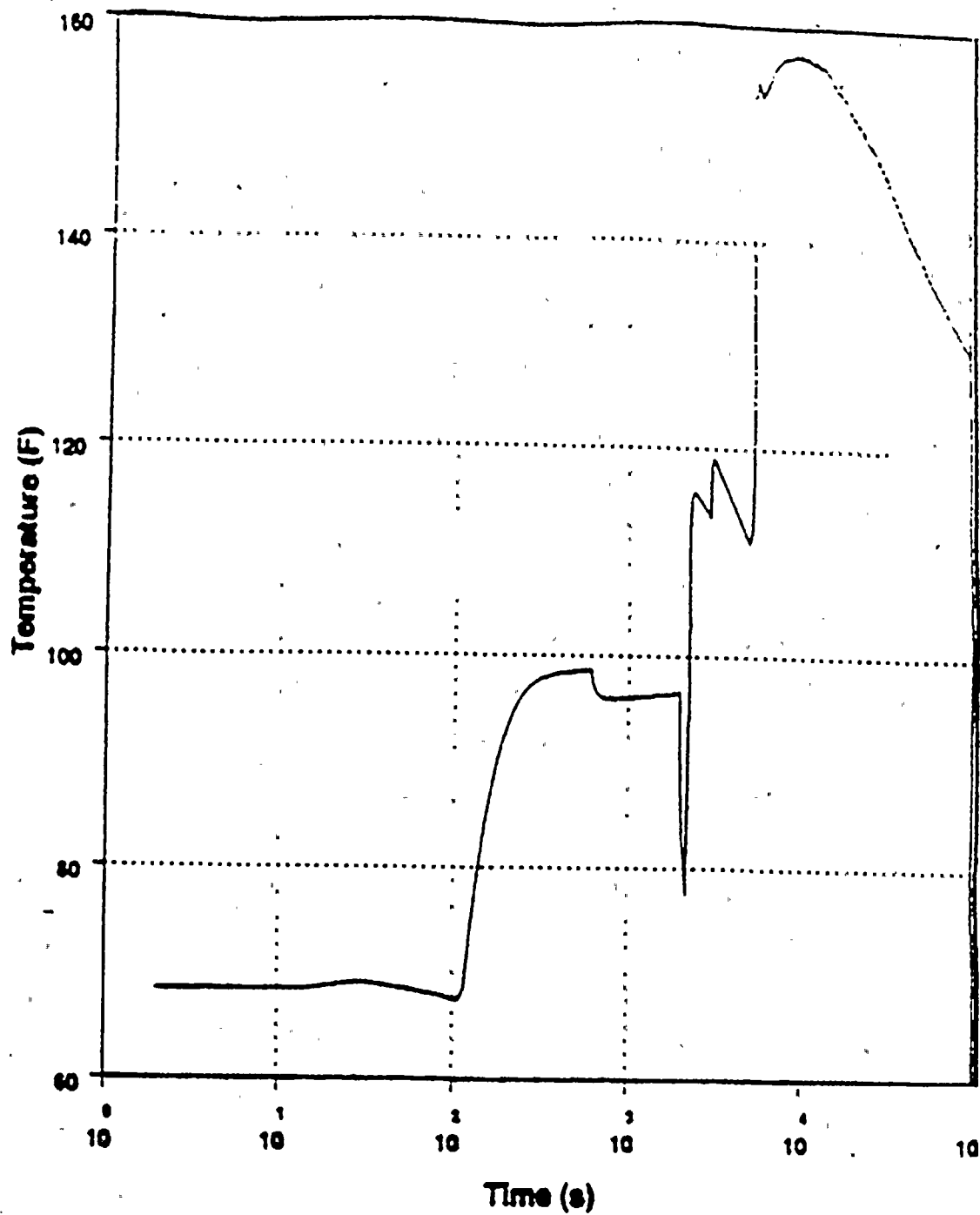
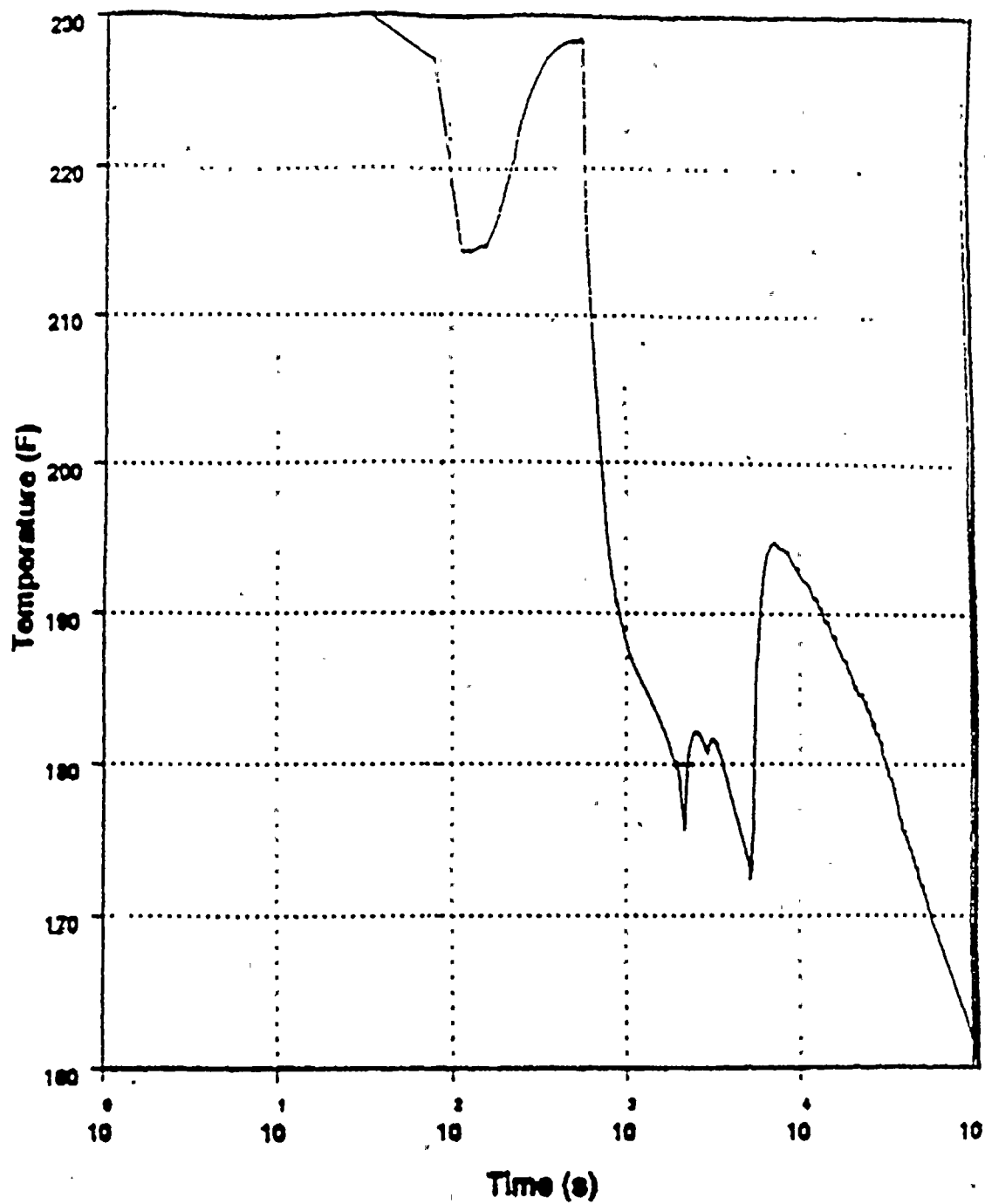


Figure 14.3.4-7 LOCA Mass Energy Release Containment Integrity
Upper Containment Temperature



Lower Cont. Temp.

Figure 14.3.4-8 LOCA Mass Energy Release Containment Integrity
Lower Containment Temperature

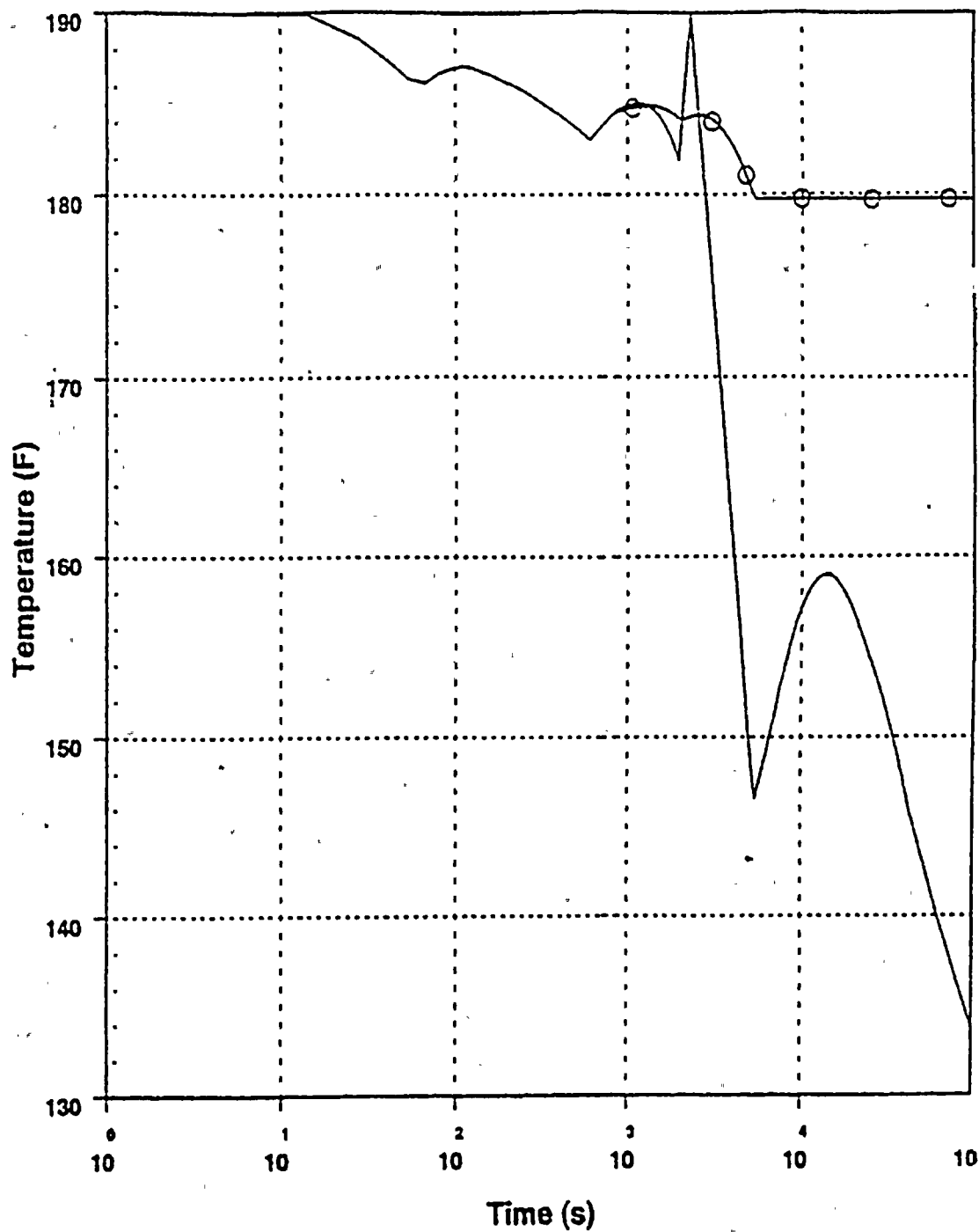
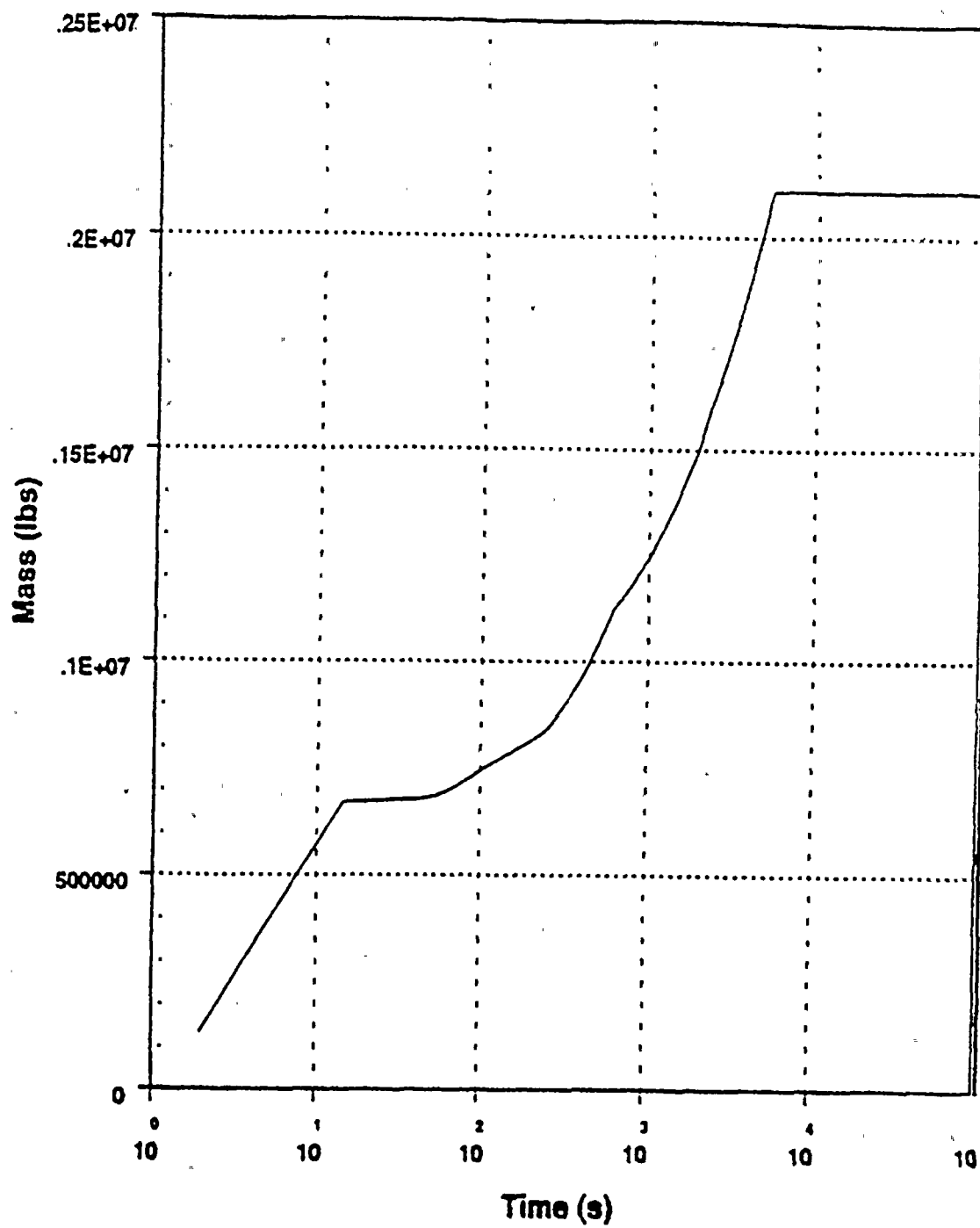
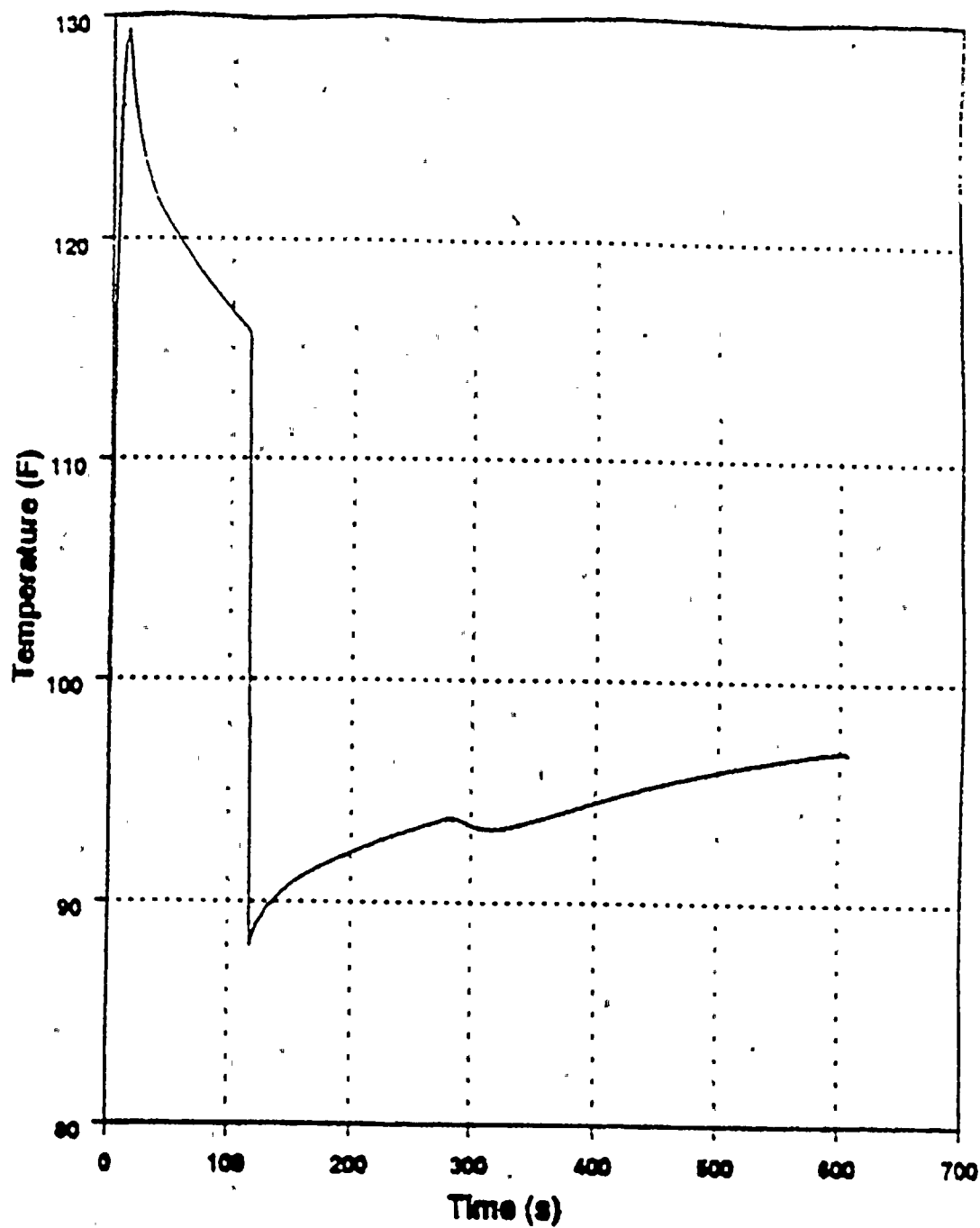


Figure 14.3.4-9 LOCA Mass and Energy Release Containment Integrity
Active and Inactive Sump Temperature Transient



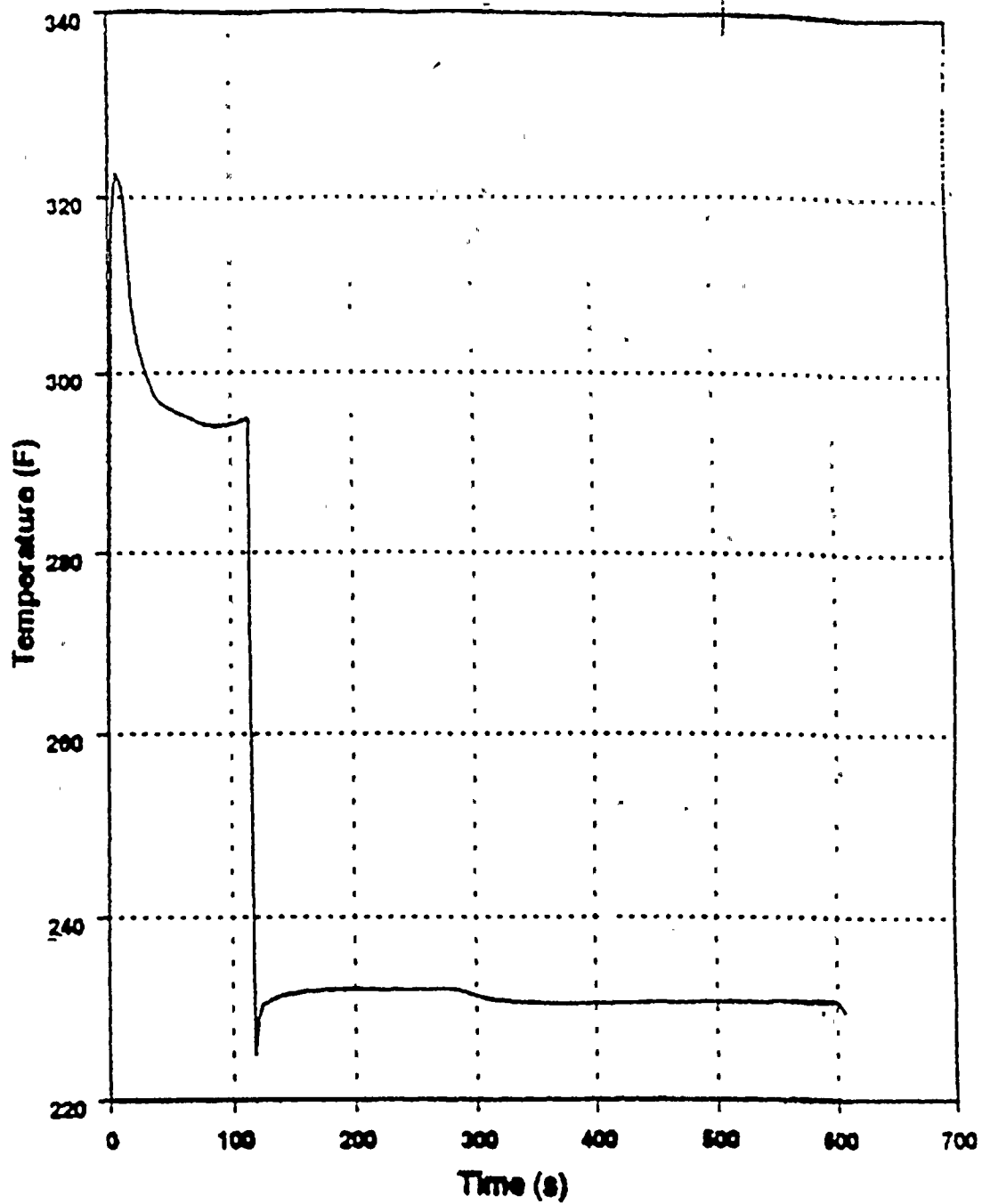
Melted Ice Mass

Figure 14.3.4-10 LOCA Mass and Energy Release Containment Integrity
Ice Melt Transient



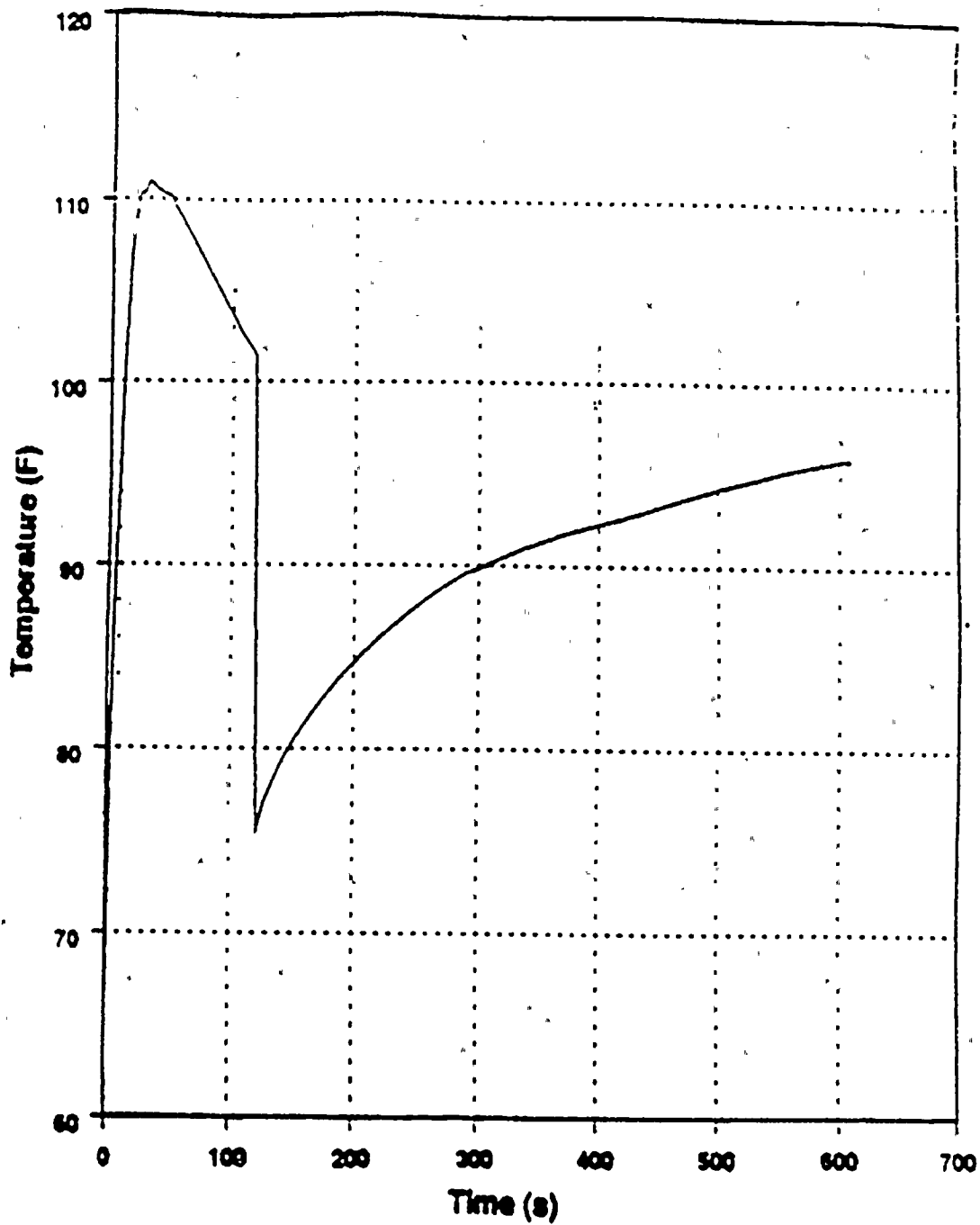
TEMPERATURE (F)

Figure 14.3.4-11A 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure
Upper Compartment Temperature



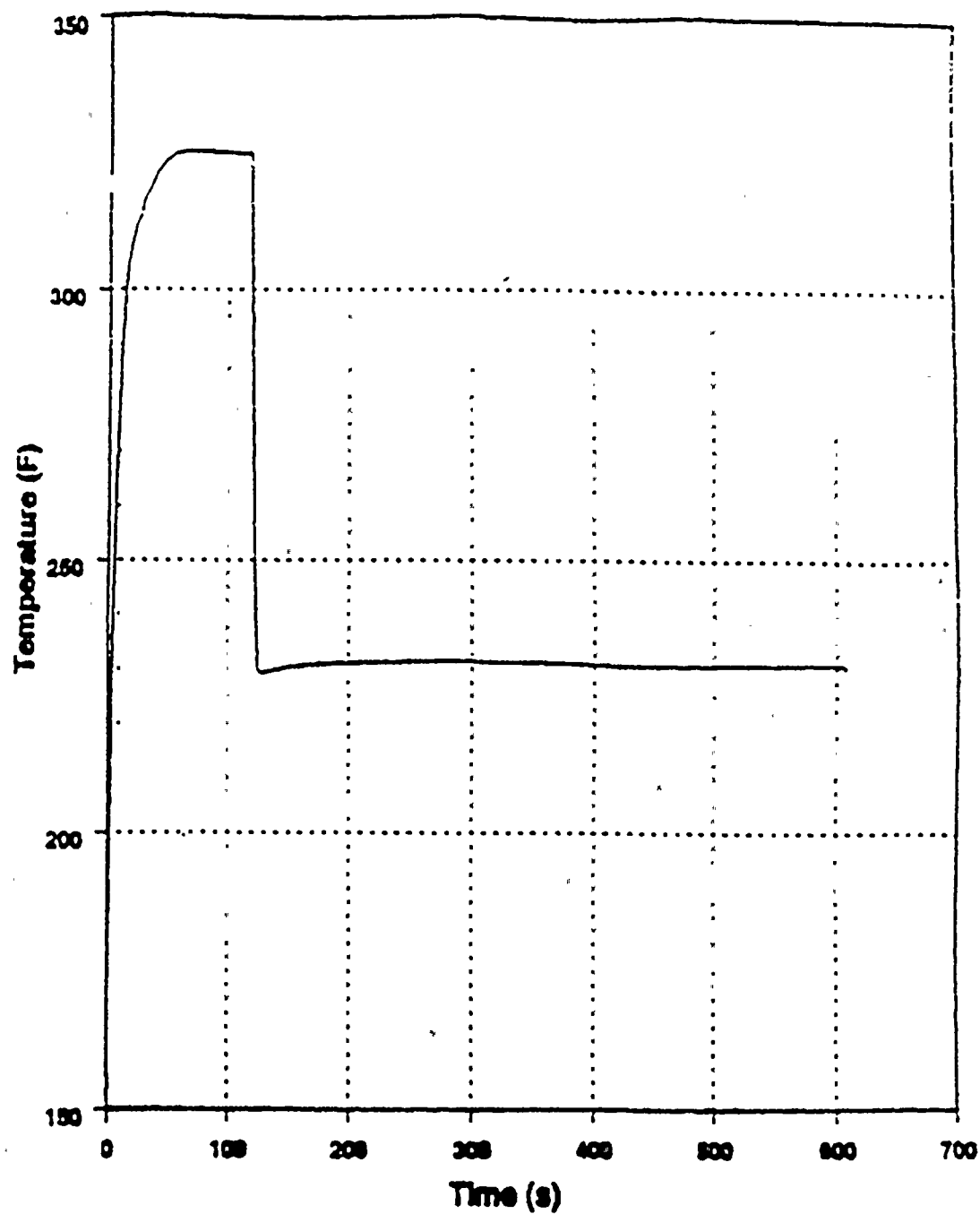
TEMPERATURE (F)

Figure 14.3.4-11-8 102% Power, 1.4 sq. ft. Double Ended Rupture - MSIV Failure
Lower Compartment Temperature



TEMPERATURE (F)

Figure 14.3.4-12-A 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Upper Compartment Temperature



TEMPERATURE (F)

Figure 14.3.4-12-8 30% Power, 0.942 sq. ft. Split Break - MSIV Failure
Lower Compartment Temperature

14.3.5 ENVIRONMENTAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT *

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident do not result in radiation exposures which exceed the limits specified in 10 CFR 100. The calculated radiation doses are summarized in Table 14.3.5-7 and Figure 14.3.5-2.

In calculating the above doses it is assumed that following the LOCA, personnel access to vital areas will not be unduly limited nor will vital safety equipment be unduly degraded by the post-accident radiation fields. This concern was specifically raised following issuance of the license, and the analyses, special equipment, etc., required to meet this concern are addressed in References 16 through 29.

Chapters 5 and 6 describe the protective systems and features of the unit which are specifically designed to limit the consequences of a major loss-of-coolant accident. The capability of the Emergency Core Cooling System for preventing melting of the fuel clad and the ability of the passive Ice Condenser Containment to absorb the blowdown resulting from a major loss of coolant are discussed in Subchapters 14.3.2 and 14.3.4, respectively. The capability of the engineered safety features in meeting dose limits set in 10 CFR 100 is demonstrated in this section.

Basic Radioactivity Removal Features

Removal of iodine from the Donald C. Cook Nuclear Plant containment atmosphere following a loss-of-coolant accident is affected by the containment spray system, the ice condenser, and several natural processes.

* The analyses to demonstrate that the offsite dose consequence results of radioactivity released to the environment in the event of a loss of coolant accident analysis are presented in the Unit 2 section 14.3.5. The information presented in this Unit 1 section of 14.3.5 on that topic is for historical information only. However, the information concerning the Control Room Habitability Analysis in this Unit 1 section of 14.3.5 is applicable to both Units 1 and 2.

The effectiveness of a spray system in quickly and efficiently removing radioactive iodine has been repeatedly demonstrated in tests conducted at Oak Ridge National Laboratory (ORNL), Containment Systems Experiment (CSE), and by reactor manufacturers. For Cook Nuclear Plant, American Electric Power Service Corporation (AEPSC) has performed jointly with Battelle-Columbus Laboratories a detailed analysis to further study the removal of gaseous elemental iodine by sprays and to calculate the efficiency when certain non-ideal factors are considered. From this work we can state that the spray system in Cook Nuclear Plant will by itself reduce the radioactive iodine leakage from the containment to values which result in doses below 10 CFR 100 from the maximum hypothetical accident, before taking credit for the ice condenser removing any gaseous elemental iodine.

Westinghouse has demonstrated the removal of gaseous elemental iodine through the process of steam condensation in an ice column, and they have evaluated the efficiency of this iodine removal. This work is reported in WCAP-7426 Topical Report Iodine Removal in the Ice Condenser System, dated March 1970⁽¹⁾. The data from the Westinghouse tests shows efficient iodine removal from steam and from mixtures of steam and noncondensibles such as air. These tests and the iodine removal model applied to the results show that the ice condenser in a large containment building reduces the gaseous elemental iodine concentration after an accident. The treatment in the Westinghouse report ignored any other iodine removal system (that is, a containment spray system, except for the assumptions implicit in the steam production predictions).

Because the ice condenser is a substantial additional iodine removal feature, and although the Cook Nuclear Plant meets the requirements of 10 CFR 100 before taking credit for its iodine removal capability, the effect of a range of ice condenser iodine removal efficiencies on the

model. The effect of the Ice Condenser as an iodine removal mechanism has been included by the addition of an ice condenser removal efficiency to the recirculation fan input term in the continuity equation for the upper volume.

The removal of a soluble component by a reactive spray in a three volume system under conditions of a constant mass transfer rate coefficient in each compartment, and with recirculation flow between the compartments, assumption of complete mixing, and ice condenser removal of iodine, is given by the following equations:

$$\frac{dC_B}{dt} = -\lambda_B C_B + \frac{Q_R}{V_B} C_F - \frac{Q_R}{V_B} C_B$$

$$\frac{dC_T}{dt} = -\lambda_T C_T + \frac{\xi Q_R}{V_T} C_B - \frac{Q_R}{V_T} C_T \quad \text{Where: } \xi = 1 - \zeta$$

$$\frac{dC_F}{dt} = -\lambda_F C_F + \frac{Q_R}{V_F} C_T - \frac{Q_R}{V_F} C_F$$

Where:

- ζ = ice condenser iodine removal efficiency
- C_B = concentration in sprayed portion of lower volume
- C_T = concentration in upper volume
- C_F = concentration in fan-accumulator room
- λ_B = removal coefficient in sprayed portion of lower volume
- λ_F = removal coefficient in sprayed portion of fan-accumulator room
- λ_T = removal coefficient in sprayed portion of upper volume
- Q_R = recirculation fan flow rate
- V_B = total volume of lower compartment
- V_T = total volume upper compartment
- V_F = total volume of fan-accumulator room

with the initial conditions:

$$t = 0, C_B = C_{B_0} \text{ and } C_T = 0, C_F = 0, \text{ which yields}$$

$$\text{at } t = 0, \frac{dC_B}{dt} = (\lambda_B + \frac{Q_R}{V_B}) C_{B_0} \text{ and } \frac{dC_T}{dt} = \frac{Q_R}{V_T} C_{B_0}$$

$$\text{and } \frac{dC_F}{dt} = 0$$

While these simultaneous differential equations can be solved analytically under the assumption that the removal coefficients in each region (i.e., λ_B , λ_T , and λ_F) are constant, the spray code mentioned previously was used to calculate these removal coefficients as a function of time; that is, taking into account the conservative assumption that liquid phase and gas phase mass transfer resistances reduce the removal coefficients as a function of time. The result of this analysis is shown in Figure 14.3.5-1, which shows the decrease in overall airborne iodine concentration as a function of time for ice condenser iodine removal efficiencies of zero, 20% and 40% although significantly higher efficiencies are expected for iodine removal in the calculation of doses. The figure shows that the containment spray system by itself is a very effective iodine removal mechanism. In 20 minutes, the total amount of radioactive iodine airborne in the containment has been reduced by a factor of 100 (i.e., only 1% of the initial airborne iodine concentration is left) with the above-mentioned non-ideal phenomena taken into account (Curve 1). The figure also shows that our conservative analytical representation of spray coalescence causes a significant reduction in the efficiency of spray performance.

For example, at about 700 seconds after the accident, without spray droplet coalescence, the initial airborne iodine concentration would be reduced by a factor of 1000 (Curve 2). With our coalescence model, the factor is about 16 (i.e., only 6% of the initial airborne concentration is left, see Curve 1).

Recirculation Leakage

Subsequent to the emptying of the refueling water storage tank during the initial phase of emergency core cooling, water from the containment sump is recirculated by the residual heat removal pumps and spray pumps and cooled via the residual heat exchangers and spray heat exchangers and then returned to the Reactor Coolant System and containment. Because the sump water contains the radioactivity of the spilled reactor coolant, the potential off-site exposure due to operation of these external recirculation paths is evaluated.

The recirculation loop leakage to the auxiliary building from the components of the Emergency Core Cooling System during recirculation is approximately 1770 cubic centimeters per hour. The leakage from the Containment Spray System amounts to 2806 cubic centimeters per hour.*

During the recirculation phase of post-accident cooling, the sump water maximum temperature is calculated to be about 140°F at the initiation of recirculation so that essentially no leakage is flashed to vapor. The volatility of iodine from a simulated recirculation loop solution has been experimentally investigated.

A solution including boric acid and sodium hydroxide was "spiked" with molecular iodine and then evaporated to dryness at 200°F in a flowing air stream. The vapor generated by the evaporation process included iodine entrainment measured to be less than 10^{-4} of the iodine inventory in the original solution. For conservatism in the analysis, it is assumed that approximately 4576 cc/hr leak to the auxiliary building for a period of two hours after initiation of recirculation. The auxiliary building ventilation system or Engineered Safety Features Ventilation System equipped with charcoal filter then discharges the volatile iodine to the atmosphere.

*These values were used in the original offsite dose analysis. Subsequent analyses assumed approximately 10 gpm of leakage, either in the auxiliary building or back to the RWST. License amendments 49 (Unit 1) and 34 (Unit 2) require implementation of a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids to as low as practical levels.

It is also assumed that all the released iodine activity inventory is in the water in the sump which has a total volume of about 2.2×10^9 cc including reactor coolant and emergency cooling water, plus the water from the melting of approximately 50% of the total ice. Under these assumptions, the combined leakage from the recirculation system results in a dose of less than 0.05 mrem to the thyroid in two hours at the site boundary.* This dose is negligible and would be even less since the temperature of the recirculated water is substantially reduced so that little or no vaporization occurs.

The potential off-site consequences from the design basis leakage in the circulating system discussed here was reevaluated. The results of the surface dose rates outside the concrete wall to each of the post-accident recirculating systems equipment compartments are in Table 14.3.5-8.

Further reanalyses were performed in this area following issuance of the license. The references containing this reanalysis are discussed at the beginning of this Section 14.3.5.

Control Room Habitability Analysis

Introduction

Analyses were performed to set operating limits on the control room emergency ventilation system in order to ensure the requirements of General Design Criterion 19 were met. (GDC 19 limits 30 day doses to control room personnel to 5 rem whole body, or its equivalent. The NRC has defined "or equivalent" as meaning 30 rem to the thyroid and 30 rem to the skin.)

The key assumptions of the analysis are contained in Table 14.3.5-9. The analyses included determination of a plant-specific 95th percentile atmospheric dispersion factor (χ/Q).

*Subsequent offsite dose and control room dose analyses assumed approximately 10 gpm of leakage. This leakage could be from any combination of recirculation fluid leaking in the auxiliary building or from ECCS fluid leaking into the RWST. No credit was taken for engineered safeguards feature ventilation system iodine removal. (References 39 & 40)

Failure of the toilet damper in the open position would represent a slight breach in the Unit 2 control room pressure envelope, reducing the positive pressure developed by the pressurization fan. Since the control room would still remain at a positive pressure with this damper open, this failure would have a negligible impact on control room dose.

Failure of the normal intake damper to close would result in unfiltered air being drawn into the control room. This flow through the normal intake damper is administratively limited such that it is less than 200 cfm. Thus, failure of this damper to isolate would result in an additional 200 cfm of unfiltered in-leakage being admitted to the control room until such time as the damper can be isolated. The control room dose analyses assume failure of this damper occurs at time zero. The damper is assumed to be manually closed two hours later.

Results

The evaluation determined that dose to control room operators will be within the 5 rem whole body, 30 rem thyroid, and 30 rem skin limits (including the effects of failure of the normal intake damper) if inleakage is limited by the following equation:

$$y = -0.048X + 153, \text{ where:}$$

y = unfiltered inleakage (cfm)

X = filtered intake (cfm)

The inleakage limits established by the thyroid dose analyses were determined to bound the inleakages based on the whole body and skin dose limits.

REFERENCES. SECTION 14.3.5

1. WCAP-7426 Westinghouse Topical Report. Iodine Removal in the Ice Condenser System, March 1970.
2. TID-14844: Calculation of Distance Factors for Power and Test Reactor Sites, March, 1962, Division of Licensing and Regulation, AEC.
3. TID-14844: Ibid Table IV.
4. Toner, D. F. and Scott, J. L., "Fission Product Release from UO_2 " Nuclear Safety, No. 3 (2), December 1961.
5. Falle, J. (Editor), "Uranium Dioxide: Properties and Applications, Naval Procedure, Division of Reactor Development, USAEC," U. S. Government Printing Office, 1961.
6. Croft, J. F., et al., "Experiments on the Deposition of Airborne Iodine of High Concentration," AEEW-R265, June 1963.
7. McCormack, J. D. and Hilliard, R. K., "Natural Removal of Fission Products Released from UO_2 Fuel in Condensing Steam Environments," International Symposium on Fission Product and Transport Under Accident Conditions, CCNF-650407, April 1965.
8. Parker, G. W., et al., "Behavior of Fission Products Released from Simulated Fuels in the Containment Mockup Facility," p. 83, Nuclear Safety Program Semiannual Progress Report for Period Ending June 30, 1965, ORNL-3843.
9. Griffiths, V., "The Removal of Iodine from the Atmosphere by Sprays," AMCS (S), p. 45.

REFERENCES, SECTION 14.3.5 (cont'd.)

39. Calculation RD-97-01.

40. Calculation RD-94-01.

TABLE 14.3.5-9

KEY ASSUMPTIONS USED IN EVALUATING
THE CONTROL ROOM DOSES DUE TO A LOCA
FOR THE DONALD C. COOK NUCLEAR PLANT
UNITS 1 AND 2

Source Term

The core iodine and noble gas inventories are based on a 3588 MWt core. This bounds the licensed power for both Unit 1 (3250 MWt) and Unit 2 (3411 MWt).

100% of the core noble gases are released to the containment.

<u>Isotope</u>	<u>Curies</u>
Kr 85m	2.6E7
Kr 85	8.3E5
Kr 87	4.8E7
Kr 88	6.8E7
Xe 131m	7.1E5
Xe 133m	2.9E7
Xe 133	2.0E8
Xe 135m	4.1E7
Xe 135	4.2E7
Xe 138	1.6E8

50% of the core iodine is released to the containment.

<u>Isotope</u>	<u>Curies</u>
I-131	5.0E7
I-132	7.3E7
I-133	1.0E8
I-134	1.1E8
I-135	9.5E7

Iodine Plateout Factor 0.5

Iodine Species Fraction (%)

Elemental	0.955
Organic	0.020
Particulate	0.025

TABLE 14.3.5-9 (cont'd)
(2 of 3)

Containment Leak Rate, %/day

0-24 hr	0.25
24 hr to 30 days	0.125

Containment Iodine Removal

Organic Removal	no credit taken
Elemental Removal	
Spray	per UFSAR Figure 14.3.5-1
Ice Cond.	30% efficiency
Part. Removal (hr-1)	
DF \leq 100	6.7
DF > 100	0.67

Atmospheric Dispersion Factor (Chi/Q)

0-8 hour (sec/m ³)	7.85E-4
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The 0-8 hour Chi/Q is adjusted for wind speed, wind direction, and occupancy according to Table 1 of Murphy-Campe.

Control Room HVAC Filter Efficiency, %

Particulate	99
Elemental, Organic	95

Miscellaneous

Control Room Volume (ft ³)	62,356
Breathing Rate (m ³ /sec)	3.47E-4

TABLE 14.3.6-1

Corrosion of Aluminum Alloys in Alkaline Sodium Borate Solution

Temperature F	Alloy Type	Test Duration	Corrosion Rate		pH	Exposure Condition	Reference
			mg/dm ² /hr	mils/year			
275	5052	3 hrs.	96.2	1230	9	Solution	WCAP-7153, Table 9
275	5005	3 hrs.	840	10770	9	Solution	WCAP-7153, Table 9
200	6061	320 hrs.	15.4	197	9.3	Solution	WCAP-7153, Table 8 WCAP-7153, Figure 9
210	5052	7 days	53.0	678	9	Solution	WCAP-7153, Table 7 WCAP-7153, Figure 8
210	5052	2 days	14.0	179	9	Solution	WCAP-7153, Table 5
210	5005	2 days	27.1	347	9	Solution	WCAP-7153, Table 5
284	5052	1 day	54	692	9.3	Spray	ORNL-TM-2425, Table 3.13
284	5052	1 day	31.5	403	9.3	Solution	ORNL-TM-2425, Table 3.13
212	6061	3 days	126	1610	9.3	Spray	ORNL-TM-2368, Table 3.6
212	6061	3 days	110	1410	9.3	Solution	ORNL-TM-2368, Table 3.6
150	6061	7 days	2.9	37.1	9.3	Solution	PWRD recent data
150	5052	7 days	4.2	53.8	9.3	Solution	PWRD recent data

TABLE 14.3.6-2

Post-Accident Containment Temperature Transient
Used in the Calculation of Aluminum Corrosion

<u>Time Interval (sec)</u>	<u>Temperature (°F)</u>
0 - 1000	240
1000 - 3600	170
3600 - 20400	187
20400 - 1 Day	161
>1 Day	147

- b. Circumferential breaks were examined in piping runs and branch runs exceeding a nominal 1-inch diameter. A circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the cross-sectional flow area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to cause pipe movements perpendicular to the plane of the break.

Design Basis Crack

A design basis crack is defined as a single open crack of a size of one-half the pipe inside diameter in length and one-half the pipe wall thickness in width. The location of this break can be anywhere along the length of the pipe.

Crack Location

Where high-energy pipes are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, a single postulated crack in the pipe system has been postulated at the most adverse location. The criteria for evaluating the effects of jet impingement and resulting steam-air environment are discussed below.

14.4.2.3 Criteria For Pipe Rupture Induced Loads

Pipe Whip

The reaction load resulting from pipe rupture has the duration and initial conditions to adequately represent the jet stream dynamics and the system pressure characteristics.

The piping systems in which pipe ruptures were considered are defined in Section 14.4.2.1. The loads induced by pipe rupture include the effects of any line restrictions, for example, flow limiters, between the pressure source and the break location.

If a whipping pipe impacts an adjacent pipe of equal or greater nominal pipe size and equal or heavier wall thickness, the impacted pipe will be considered to be free from rupture. Protection from pipe whip is not required if pipe rupture occurs in such a manner that the unrestrained movement of either end of the ruptured pipe about a plastic hinge, formed at the nearest restraint or anchorage, cannot impact any structure, system, or component required for that incident.

Jet Impingement

Jet impingement loads on safety-related equipment, components, and structures have been considered for the design basis cases defined in Section 14.4.2.2. The magnitude and area of influence of the jet was determined for each break according to the break location, size and orientation criteria given in Section 14.4.2.2 and the procedures described in Section 14.4.7. The jet forces or loads at the point of rupture are consistent with those used in the pipe whip analysis, and were based on the most severe fluid pressure and temperature conditions occurring during normal operating modes.

Jet Erosion of Concrete

The erosion of concrete by steam jets was evaluated in WCAP-7391, "Pressurized Water and Steam Jets Effects On Concrete" by Westinghouse Atomic Power Division.

In summary, five reinforced concrete beams were subjected to steam jets with nozzle diameters of 1, 2, and 4 inches. The distances investigated between nozzles and beams were 1 foot and 4 feet; the initial system pressure was 2250 psi. The results are as follows:

the main steam accessway. The main steam piping from steam generators Nos. 2 and 3 exit the containment at the west main steam enclosure, enter the main steam stop valves and pipe rupture restraints. They then enter the main steam accessway where they run horizontally to the turbine building. The above routing is shown isometrically in Figure 14.4.2-17.

Design Bases Breaks

Description of Break Locations

Potential design basis pipe break locations are as shown on the Main Steam Isometric, Figure 14.4.2-17 and described below. The stresses for the main steam piping system were calculated with the aid of a computer program using general flexibility and response spectra model analysis techniques. The combined stress values due to thermal expansion, pressure, weight, and seismic loading conditions have been computed. The results of these calculations are presented in Tables 14.4.4-1 and 14.4.4-2.

Postulated design basis break locations outside the containment have been determined on the basis of ANSI B31.1-0-1967 calculated stress values and the criteria given in Section 14.4.2.2. These consist of:

- a. The terminal points of the main steam lines at the turbine stop valves in the Turbine Building and at the containment wall.
- b. The branch point connection in the main steam line for the turbine bypass headers and intermediate points and the terminal point on this branch line.
- c. The branch point connection in the main steam line for the steam supply to the auxiliary feedwater turbine-driven pumps.

- d. The branch point connection in the main steam line for the safety valve and power relief valve header.
- e. Two additional intermediate points chosen on the basis of relatively highest stress level.

Of the total number of breaks on all steam lines, none exceeded $0.8 S_A$ or $0.8 (S_h + S_A)$. See Table 14.4.4-4 for the stress values at the postulated break locations.

Required Equipment

The equipment required for shutdown in the event of a design basis break in the main steam line is given in Table 14.4.2-1. Operability of this equipment provides for reactor trip and the capability to maintain the reactor at hot shutdown after the break, as well as ultimately achieving cold shutdown. Required equipment includes associated piping, cables, and structures required for the equipment to perform its function.

Protection from Potential Pipe-Whip Damage

No additional pipe rupture restraints are required to protect the required equipment listed in Table 14.4.2-1 following the postulated breaks of the main steam line.

Protection from Jet Impingement

The locations that require protection from postulated jets have been determined. These locations are the intersections between cabling required to support operation of equipment listed in Table 14.4.2-1 and the high energy lines. Protection is provided at these locations by either 1) installation of impingement barriers, 2) moving the cable to non-critical locations, or 3) other appropriate measures.

14.4.3 ANALYSIS OF EMERGENCY CONDITIONS

The analyses of emergency conditions listed below are general in nature since it is deemed appropriate to allow for assessment of the incident prior to ultimately bringing the reactor to cold shutdown.

For all of the postulated high energy break sources, appropriate detection and shutdown can be brought about solely through use of the instrumentation listed in Table 14.4.2-1. However, the following analysis presents a consideration of the expected method of operation associated with the high energy line break outside the containment.

14.4.3.1 Main Steam Line Rupture

The following systems provide for the necessary safeguards system response to a steam pipe rupture outside the containment.

1. Safety injection system actuation from any of the following:
 - a. Two out of three low pressurizer pressure signals.
 - b. Low main steam line pressure (two out of four lines).
 - c. Two out of three high differential pressure signals between a steam line and the remaining steam lines.
2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

* This "HYBRID" Steamline Break Protection was installed in Unit 1 during the refueling outage of 1997. The previous "OLD" Steamline Break Protection required high steam line flow in two out of four main steam lines, in coincidence with either low-low reactor coolant system average temperature (two out of four loops) or low main steam line pressure (two out of four lines).

3. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, and trip the main feedwater pumps.
4. Trip of the fast acting main steam isolation valves (analyzed for a closure time of 8 seconds) occurs on any of the following:
 - a. Low steam line pressure (two out of four lines).*
 - b. High steam flow in any two steam lines in coincidence with low-low reactor coolant system average temperature in any two loops.

Each steam line has a fast-closing stop valve capable of stopping flow in either direction. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. In addition each main steam line incorporates a 16 inch diameter venturi type flow restrictor which is located inside the containment. These components limit the rate of release of steam for an outside break.

5. Safety injection actuation will also initiate automatic start of the two motor-driven feed pumps. The low-low-level signal in any two steam generators will start the turbine-driven feed pump.

The plant is designed to accept the steam line rupture outside the containment with concurrent loss of offsite power (diesel power available only) and a single active failure in a required system.

For small steam line breaks at power which do not cause the reactor power to reach a point at which an immediate reactor trip would occur, no reactor core safety limit will be violated. The small break will result in a continued loss of water from the secondary side of the plant and will eventually result in condenser hotwell low level. This low

* This "HYBRID" Steamline Break Protection was installed in Unit 1 during the refueling outage of 1997. The previous "OLD" Steamline Break Protection required high steam flow in coincidence with low steam line pressure.

level will result in a loss of main feedwater, and the reactor will be tripped on low-low steam generator level or feed/steam flow mismatch. After the trip, steam release through the break will cause reactor coolant system cooldown. The cooldown would occur until the steam generator feeding the break empties. The cooldown will automatically initiate safety injection on low pressurizer pressure. Initiation of safety injection will isolate all main feedwater by tripping closed the main feedwater control valves, tripping closed the main feedwater pump discharge valves, and tripping the main feedwater pumps.

Should the plant be at hot standby or subcritical at the time of a small steam line break, the plant will be cooled down by the operator who would have other systems available following the incident to facilitate an orderly shutdown of the reactor. Hence the method and procedure to be used for shutdown will be determined by the operator based on the equipment available.

14.4.3.2 Feedwater Line Rupture

The following systems provide necessary protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator.
2. Reactor trip on steam/feedwater flow mismatch coincident with low water level in any steam generator.
3. Two motor driven auxiliary feedwater pumps (450 gpm nominal each) which are started on:
 - a. Low-low level in any steam generator
 - b. Trip of all main feed pumps

- c. Any safety injection signal
- d. Blackout signal
- e. Manually

Each of these pumps feeds two steam generators in its unit.

- 4. One turbine driven auxiliary feedwater pump (900 gpm) which is started on:
 - a. Low-low level in any two steam generators
 - b. Reactor coolant pump bus undervoltage
 - c. Manually

The turbine driven auxiliary feedwater pump feeds the four steam generators on its unit.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Following the initiation of the low-low level trip, the auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease. The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the reactor coolant system relief or safety valves. The plant is designed to accept this failure (feedwater line rupture) with concurrent loss of off-site power (diesel generator power available only) and a single active failure in a required system. In addition, all required systems are operable from the control room or accessible for manual operation.

TABLE 14.4.4-3 intentionally deleted;

Table text moved to Table 14.4.10-2

TABLE 14.4.4-4

STRESS VALUES AT POSTULATED BREAK LOCATIONS
ALLOWABLE STRESS = 30,000 PSI

Main Steam Leads 1 & 4

<u>Node</u>	<u>Failure No.</u>	<u>Total Stress</u>
1	1A, 4A	12,500
2	1B, 4B	28,120
3	1I, 4I	20,640
4	1C, 4C	32,060
6	1D, 4D	27,820
9	1E, 4E	18,810
26	1F, 4F	8,280
27	1G, 4G	8,280
5	1H, 4H	25,790

Main Steam Leads 2 & 3

1	2A, 3A	7,446
3	2B, 3B	22,095
4	2H, 3H	21,142
19	2D, 3D	8,413
5	2F, 3F	22,464
6	2G, 3G	21,981
8	2C, 3C	25,105
20	2E, 3E	7,755
21	2I, 3I	7,406

Main Steam to Auxiliary Feed Pump Turbine

37	A5B	18,530
40	A5C	17,750
39	B5B	17,390
38	B5C	17,890
45	5D	19,790
46	5E	23,420
63	5F	25,790
	5G	8,740

(See Table 14.4.2-5 for Failure Descriptions)

Instruments

The instruments required for the high energy line incident as identified in Table 14.4.10-1 were enclosed and/or modified by their respective manufacturers to withstand the anticipated adverse environment.

The instruments and transmitters identified in Table 14.4.10-2 are located in areas unaffected by the adverse environment. The instrumentation supply and signal lines and cable routings were reviewed to determine if they are routed in areas of possible adverse environment. Any items found to be routed in these areas were either rerouted or adequately shielded from these adverse conditions.

14.4.10.3 Seals

The control room and electrical switchgear room are provided with seals on doors and penetrations which adequately protect these areas from the adverse environment associated with the high-energy line incident. For those areas containing equipment which is not qualified to perform its function under this adverse environment, seals are provided on doors and penetrations.

The only door from the above areas to the auxiliary building is an emergency fire exit from the back of the control room panel. This door is under strict administrative control, sealed to control room isolation criteria, and exits to an area containing no high energy lines.

14.4.10.4 Ventilation

Protection From Auxiliary Building Environment

No structural modifications were required to prevent the adverse steam environment from entering the electrical switchgear room or the control rod drive equipment room. Seals on the doors adequately control the steam input from a line rupture.

Protection from Turbine Building Environment

The Seismic Class I auxiliary feedwater pump rooms, battery rooms, and the 4160 volt switchgear rooms are similarly isolated from any adverse environment resulting from postulated high-energy pipe ruptures in the turbine building by the inclusion of back-draft or fire curtain dampers in each ventilation duct penetrating its boundary and by sealed doors.

TABLE 14.4.10-1

TRANSMITTERS TO BE PROTECTED
WITH SUPPLY AND SIGNAL LINES

Feedwater Flow

FFC - 210, 211, 220, 221, 230, 231, 240, 241

1st Stage H.P. Turbine Pressure

MPC - 253, 254

S.G. Main Steam Pressure

MPP - 211, 221, 231, 241

MPP - 210, 220, 230, 240

TABLE 14.4.10-2
SUPPLY & SIGNAL LINES TO BE PROTECTED

Pressurizer

NLP - 151, 152, 153

NPP - 151, 152, 153

NPS - 153

NRV - 151, 152, 153

Steam Generator

BLP - 110, 111, 112, 120, 121, 122, 130, 131, 132, 140, 141, 142

MRV - 213, 223, 233, 243, 211, 221, 231, 241, 212, 222, 232, 242

Refueling Water Storage Tank (RWST)

ILA - 950, 951

ITA - 900

Boron Injection Tank

ITC - 251

ITA - 250

Residual Heat Removal

IRV - 311, 310, 320

IFC - 315, 325

Essential Service Water

WFA - 701, 705, 702, 706

WPA - 707, 705, 706, 708

WPS - 701, 705, 702, 706

WPI - 707, 705, 706, 708

WRV - 766, 767, 768, 769, 776, 777, 778, 779, 761, 762, 763, 764, 771,
772, 773, 774

WDS - 701, 702, 703, 704

Component Cooling Water

CRA - 415, 425

CLA - 410, 411, 412, 413

CRV - 412, 410, 411

For radiation considerations, a mild environment is one in which the integrated dose is less than 10^4 rads. For organic materials, radiation qualification may be readily justified by existing test data or operating experience for radiation exposures below 10^4 rads. For electronic components, however, failures in metal oxide semiconductor devices occur at somewhat lower doses. For this reason, radiation qualification for electronic components may have a lower exposure threshold.

14.4.11.2 HELB Inside Containment

The LOCA and the MSLB are considered inside containment. The LOCA will result in maximum radiation doses, and elevated temperatures and pressures. The LOCA may also activate the containment spray system, producing an environment of chemical spray for some portion of the accident.

The MSLB will usually produce higher temperatures and pressures, but will release less radiation, although it may also activate the containment spray system. Radiation doses from the MSLB are essentially nil when steam generator tube integrity is maintained.

The LOCA and the MSLB provide bounding conditions for temperature, pressure, and radiation.

Environment conditions are further discussed below.

14.4.11.2.1 Temperature and Pressure

The long-term temperature and pressure profiles for the LOCA in Units 1 and 2 are shown in Figures 14.3.4-6 and 14.3.4-7. Temperature and pressure profiles for the MSLB inside lower containment are shown in Figures 14.3.4-11 through 14.3.4-16. Table 14.4.11-2 tabulates the peak calculated temperatures and pressures for the LOCA and MSLB and feedwater line breaks inside containment.

14.4.11.2.2 Chemical Spray

Following a LOCA, the containment sump water will consist of spray water, melted ice impregnated with sodium tetraborate, and primary system water.

The Technical Specifications limits for capacity and boron concentration for the various contributors to the containment spray were used to evaluate the range of boron in the containment spray⁽⁶⁾. During the injection phase, the boron concentration range of the spray is approximately 2400-2600 ppm, resulting in a pH of 6.8 to 7.0. During the recirculation phase, the initial solution pH is 12.9 for approximately two hours⁽⁷⁾. Following this, the boron concentration and pH would be approximately 2400 ppm and 9.3, respectively⁽⁸⁾.

14.4.11.2.3 Flooding Elevation

The flood level for the containment sump is 613'-2"⁽⁹⁾. Any safety-related equipment located below the flood level will actuate before it becomes submerged. No equipment is presently required to be qualified for submergence.

14.4.11.2.4 Humidity

It is assumed that the containment atmosphere will be pure steam or a mixture of steam and noncondensibles at 100% relative humidity.

14.4.11.2.5 Radiation

Radiation doses inside containment are calculated by using the integrated gamma and beta radiation dose tables for either the upper or lower volume compartments of the containment. For devices above elevation 613'-2", the radiation doses in Table 14.4.11-3 are used. For devices that have been submerged below elevation 613'-2", the radiation doses in Table 14.4.11-4 are used.

TABLE 14.4.11-1 DELETED

(For pagination purposes, this page represents pages 14.4.11-8 through 14.4.11-18.)

Table 14.4.11-2
PEAK ENVIRONMENTAL QUALIFICATION CONDITIONS FOR
LOCA, MSLB, AND FEEDWATER LINE BREAK INSIDE CONTAINMENT

LOCA		
Location	<u>Temp(°F)</u>	<u>Press(psia)</u>
- Upper Comp.	160 ^(a)	27.2 ^(c)
- Lower Comp.	230 ^(b)	27.2 ^(c)
- Inst. and F/A Room	230	28.6 ^(d)

MSLB AND FEEDWATER LINE BREAK

<u>Location</u>	<u>Temp(°F)</u>	<u>Press(psia)</u>
- Upper Comp.	130 ^(e)	35.5 ^(g)
- Lower Comp.	326 ^(f)	35.5 ^(g)

(a) USFAR Figure 14.3.4-7

(b) UFSAR Figure 14.3.4-8

(c) USFAR Figure 14.3.4-6

(d) USFAR Figure 14.3.4-46

(e) USFAR Figure 14.3.4-11A

(f) UFSAR Table 14.3.4-10

(g) UFSAR Table 14.3.4-46 reports this value calculated for the Steamline Break in the Steam Generator Doghouse. The Steam Generator Doghouse peak pressure is a bounding value for Upper and Lower Containment.

Table 14.4.11-9

Peak Environmental Qualification Conditions for HELB Outside Containment

<u>Compartment</u>	<u>Temperature (°F)</u>	<u>Pressure (psia)</u>
East Main Steam Enclosure	488 (a)	2.62 (k)
West Main Steam Enclosure	449 (b)	2.62 (k)
Main Steam Accessway	398 (c)	26.2 (k)
Diesel Generator Pipe Tunnel	350 (d)	26.2 (k)
Turbine Driven Pump Room	313 (e)	16.0 (f)
Vestibule	225 (f)	16.0 (f)
ESW Tunnel	282 (d)	16.0 (f), (l)
Feedwater Tunnel	212 (g)	26.2 (f), (m)
Turbine Room (Turbine Bldg. 609' Elev.)	298 (h)	26.2 (f), (n)
Diesel Generator Room	217 (d)	14.9 (d)
Startup Blowdown Flashtank Room	295 (i), (j)	14.9 (i), (j)

- (a) Calculation TH-93-01, Superheated Steam - East Enclosure Temperature Profile During Cold Weather, March 3, 1993.
- (b) Calculation TH-90-07, Steamline Break Outside Containment, November 30, 1990.
- (c) Specification DCC-NOSS-106-QCN, Analytical Basis for Environmental Qualification of Equipment, Rev. 2, November 27, 1995, value attributed to calculation TH-90-07.
- (d) Calculation TH-96-01, Diesel Generator Room Temperatures and Pressures Following an HELB in the West Steam Enclosure, January 4, 1996.
- (e) Specification DCC-NOSS-106-QCN, Analytical Basis for Environmental Qualification of Equipment, Rev. 2, November 27, 1995, value attributed to Letter, R. G. Vasey to B. J. Gerwe, "Temperature of Turbine Driven Auxiliary Pump (TDAFP) Room Following a High Energy Line Break," June 11, 1993.
- (f) Letter, J. F. Etzweiler to L. F. Caso, February 27, 1980.
- (g) Calculation TH-96-04, Feedwater Line Break in Accessway between East Enclosure and Auxiliary Building, February 6, 1996.

- (h) Calculation TH-95-01, Donald C. Cook Nuclear Plant Analysis of Main Steam Line Break in Turbine Building, November 6, 1995.
- (i) Calculation TH-95-16, Start-up Blowdown Flashtank Room Post-HELB Conditions With Door #369 Open to AES Ventilation Shaft Room, February 6, 1996.
- (j) Calculation TH-96-05, HELB Evaluation for Various Door Configurations in the AES Shaft Room, April 22, 1996.
- (k) Letter, R. G. Vasey to K. J. Munson, August 7, 1986.
- (l) A peak pressure of 14.9 psia was calculated in Reference (d).
- (m) A peak pressure of 16.0 psia was calculated in Reference (g).
- (n) A peak pressure of 15.2 psia was calculated in Reference (h).

UNIT 1

14.4.11-27

July, 1997

14.0 SAFETY ANALYSIS

This chapter presents an evaluation of the safety aspects of Unit 2 of Cook Nuclear Plant and demonstrates that Unit 2 can be operated safely even if highly unlikely occurrences are postulated. It also shows that radiation exposures to the public as a result of these highly unlikely occurrences do not exceed the guidelines of 10 CFR 100.

Unit 2 of Cook Nuclear Plant was initially loaded with fuel fabricated by Westinghouse Electric Corporation for the first three cycles. From Cycle 4 through Cycle 7, reload fresh fuel was fabricated by Siemens Power Corporation previously known as Advanced Nuclear Fuel and Exxon Nuclear Company. Starting with Cycle 8, the fabrication of fresh reload fuel is again furnished by Westinghouse, this time using the 17 x 17 Vantage 5 fuel assembly design. The transition to a reactor core completely composed of Westinghouse Vantage 5 fuel assemblies was completed at the beginning of Cycle 10 (i.e., the 1994 refueling outage). To the extent that the safety analyses in this chapter involve a particular fuel design, it is the Westinghouse Vantage 5 fuel that is considered.

This chapter is divided into the three sections described below, each section dealing with a different (licensing basis) category of fault conditions.* The ANS Conditions II, III, and IV are based on the anticipated frequency of their occurrence and are related to the licensing basis categories as described below. There are four ANS fault conditions: Condition I, Condition II, Condition III and Condition IV. ANS Condition I occurrences do not require a safety analysis because they represent normal operational transients.

ANS Condition II occurrences are faults that may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no ANS Condition II occurrence shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

* The three categories of fault conditions analyzed in this chapter do not have a one to one correspondence with the ANS Conditions II, III, and IV, but each fault condition in each category is also identified as either ANS Condition II, III or IV.

ANS Condition III occurrences are faults that may occur very infrequently during the life of the plant. They may be accompanied by the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. An ANS Condition III occurrence will not, by itself, generate an ANS Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers.

ANS Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. ANS Condition IV occurrences shall not cause a fission product release to the environment resulting in radiation exposure to the public in excess of the guidelines in 10 CFR 100. A single ANS Condition IV occurrence shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment.

Core and Coolant Boundary Protection Analysis. Section 14.1

The majority of the fault conditions discussed in this section are ANS Condition II occurrences. Section 14.1 also includes an ANS Condition III occurrence, complete loss of forced reactor coolant (Section 14.1.6.1), and an ANS Condition IV occurrence, locked rotor (Section 14.1.6.2).

Standby Safeguards Analysis. Section 14.2

The fault conditions listed in this section are very infrequent and may lead to a breach of fission product barriers. Section 14.2 includes events other than ANS Condition III occurrences, such as rupture of a control rod drive mechanism housing (Section 14.2.6), major rupture of a main feedwater pipe (Section 14.2.8), and rupture of a steam line (Section 14.2.5), which are ANS Condition IV occurrences.

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

The reactor control and protection system is relied upon to protect the core and reactor coolant boundary against the following fault conditions:

1. Uncontrolled RCCA bank withdrawal from a subcritical condition.
2. Uncontrolled RCCA bank withdrawal at power.
3. RCCA misalignment (this encompasses 14.1.3 RCCA misoperation and 14.1.4 RCCA drop).
4. Uncontrolled boron dilution.
5. Loss of forced reactor coolant flow (including locked rotor).
6. Startup of an inactive reactor coolant loop.
7. Loss of external electrical load or turbine trip.
8. Loss of normal feedwater.
9. Excessive heat removal due to feedwater system malfunction.
10. Excessive load increase
11. Loss of offsite power (LOOP) to the station auxiliaries.
12. Turbine-generator overspeed.

A reactor trip is defined for analytical purposes as the insertion of all full length RCCAs except the most reactive one which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCCA condition at a time when shutdown is required.

Instrumentation is provided for continuously monitoring all individual RCCAs together with their respective bank position. This is done in the form of a deviation alarm system. Procedures are established to correct deviations. In the worst case the plant will be shutdown in an orderly manner and the condition corrected. Such occurrences are expected to be extremely rare based on operation and test experience to date. In summary, reactor protection is designed to prevent cladding damage in all fault conditions listed above.

The simulation of the fault conditions listed above was based upon a number of conservative assumptions summarized in the following sections. Parameters and assumptions that are common to various safety analyses are described below to avoid repetition in subsequent sections.

This material applies to most of the safety analyses described in sections 14.1 and 14.2 and the steam mass and energy release portions of sections 14.3.4 and 14.4. There is also some information related to LOCAs. Most of the information related to LOCA and containment analyses can be found in section 14.3.

14.1.0 Plant Characteristics and Initial Conditions Used in Safety Analyses

14.1.0.1 Plant Conditions

The "full window" (cases 1 through 6) of the range of plant nominal operating conditions assumed in the safety analyses are presented in Table 14.1.0-1. The Non-LOCA safety analyses and evaluations presented in the following sections (Sections 14.1 and 14.2) provide support for a "full window" (cases 3 through 6) of the range of plant nominal operating conditions when a full Westinghouse VANTAGE 5 core is in place at Cook Nuclear Plant Unit 2. Cases 1 and 2 were used for safety analyses for two fuel transition cycles, cycles 8 and 9. Brief descriptions of cases 1 through 6 follows Table 14.1.0-1.

14.1.0.2 Initial Conditions

For most occurrences which are DNB limited, nominal values of initial conditions and the RCS minimum measured flow (366,400 gpm) are assumed. The allowances on core thermal power, RCS temperature, pressure and flow are determined on a statistical basis and are included in the design limit DNBR as described in WCAP-11397 (Reference 1). This procedure is known as the Revised Thermal Design Procedure (RTDP).

For occurrences that are not DNB limited or in which RTDP is not employed, the initial conditions are obtained by adding the maximum steady-state errors to nominal values. In addition, the RCS thermal design flow (354,000 gpm) is used. The following maximum steady-state errors are considered:

- | | |
|----------------------------|--|
| A. Core Power | + 2% calorimetric error allowance |
| B. RCS Average Temperature | + 4.1°F/-5.6°F controller and measurement error allowance |
| C. RCS Pressure | ± 62.6 psi steady-state fluctuations and measurement error allowance |

Tables 14.1.0-2 and 14.1.0-3 summarize initial conditions and computer codes used in the safety analysis of occurrences in sections 14.1.1 through 14.1.12 and sections 14.2.5, 14.2.6, and 14.2.8, and shows which occurrences employed a DNB analysis using the RTDP.

14.1.0.3 Core Thermal Power Distribution

The transient response of the reactor system is dependent on the initial core thermal power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of RCCAs and through operation instructions. The power distribution may be characterized by the radial peaking factor, $F_{\Delta H}$, and the total peaking factor, F_Q . The peaking factor limits are given in the Technical Specifications.

For occurrences which may be DNB limited the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to RCCA insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figures 14.1.0-5 and 14.1.0-6. All occurrences that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shapes used in the DNB calculation are discussed in Chapter 3.

The radial and axial power distributions described above are input to the THINC Code as described in Chapter 3.

For occurrences which may be overpower limited the total peaking factor, F_Q is of importance. All occurrences that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

For overpower occurrences which are slow with respect to the fuel rod thermal time constant, for example the uncontrolled boron dilution occurrence which lasts many minutes, and the excessive load increase occurrence which reaches equilibrium without causing a reactor trip, fuel temperature limits are discussed in Chapter 3. For overpower occurrences which are fast with respect to the fuel rod thermal time constant, for example the uncontrolled RCCA bank withdrawal from a subcritical condition and RCCA ejection occurrences which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation is performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and fuel rod power, a typical value at beginning-of-life (BOL) for high power fuel rods is approximately 7 seconds.

14.1.0.4 Reactivity Coefficients Assumed in the Safety Analyses

The transient response of the reactor coolant system is dependent on reactivity feedback effects, in particular the moderator temperature

coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 3.

In the safety analyses of certain occurrences, conservatism requires the use of large reactivity coefficients, whereas in the safety analyses of the other occurrences, conservatism requires the use of small reactivity coefficients. Some analyses, such as loss of reactor coolant from cracks or ruptures in the RCS, do not depend on reactivity feedback effects. The values used are given in Tables 14.1.0-2 and 14.1.0-3. Figure 14.1.0-1 shows the upper and lower Doppler power coefficients, as a function of core thermal power, used in the safety analyses. The justification for use of conservatively large versus small reactivity coefficients is treated on a case-by-case basis. In some cases this implies that conservative parameters from both beginning and end-of-life (EOL) are used for a given occurrence to bound the effects of core life. For example, in a load increase occurrence it is conservative to use a small Doppler defect typical of end-of-life (EOL) and a small moderator coefficient typical of beginning-of-life (BOL).

14.1.0.5 Rod Cluster Control Assembly (RCCA) Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCA and the variation in RCCA worth as a function of RCCA position.

With respect to safety analyses, the critical parameter is from the start of insertion up to the dashpot entry or approximately 85% of the RCCA travel. For safety analyses, the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The RCCA position versus time assumed in the safety analyses is shown on Figure 14.1.0-2.

Figure 14.1.0-3 shows the fraction of total negative reactivity insertion versus normalized RCCA insertion for a core where the axial power distribution is skewed to the lower region of the core. This curve is used as input to all safety analyses point kinetics core models. There

is inherent conservatism in the use of this curve in that it is based on a bottom skewed axial power distribution. For cases other than those associated with axial xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial power distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown on Figure 14.1.0-4. The curve shown in this figure was obtained by combining Figures 14.1.0-2 and 14.1.0-3. Except where specifically noted otherwise, the safety analyses assume a total negative reactivity insertion of 4.0% $\Delta k/k$ following a reactor trip. This assumption is consistent with the core design.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 14.1.0-4) is used in the safety analyses.

For safety analyses requiring the use of a dimensional diffusion theory code (TWINKLE, Reference 6), the negative reactivity insertion resulting from a reactor trip is calculated directly by the code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 14.1.0-2 is used as a code input.

14.1.0.6 Reactor Trip Points and Time Delays to Reactor Trip Assumed in the Safety Analyses

14.1.0.6.a Reactor Protection System (RPS) Setpoints and Time Delays

A reactor trip signal acts to open the two reactor trip breakers connected in series feeding power to the RCCA control drive mechanisms (CDMs). The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the reactor trip breakers, and in the release of the RCCAs by the CDMs. The total delay to a reactor trip

is defined as the time delay from the time that reactor trip conditions are reached to the time the RCCAs are free and begin to fall. Limiting reactor trip setpoints assumed in the safety analyses and the time delay assumed for each reactor trip function are given in Table 14.1.0-4. It should be noted that the high pressurizer water level reactor trip was assumed in the safety analyses.

The safety analyses presented in the following sections assume that the reference average temperatures (T' and T'') used in the OTDT and OPDT setpoint equations are rescaled to the full power RCS average temperature each time the cycle RCS average temperature is changed. It is also assumed that the reference pressure (P') in the OTDT equation is set equal to the appropriate nominal RCS pressure (2250 psia or 2100 psia). The safety analyses also assume recalibration of the NIS excore detectors to compensate for the changes in coolant density each time the cycle operating conditions are changed.

Reference is made in Table 14.1.0-4 to the overtemperature (OT) and overpower (OP) ΔT reactor trips shown in Figures 14.1.0-5 and 14.1.0-6. These revised OTAT and OPAT setpoints were calculated based on the new core thermal safety limits using the methodology described in Reference 2. Because of the use of the W-3 correlation for ANF fuel in the transition cycles, the core thermal safety limits for transition cycles are limited by the ANF fuel. For a full VANTAGE 5 fuel, these core thermal safety limits are less restrictive. Two sets of OTAT and OPAT setpoints were calculated. The first set of these setpoints is calculated based on the most restrictive core thermal safety limits in the transition cycles (Cycles 8 and 9) and the second set is calculated for a full core of VANTAGE 5 fuel. The following DNB-related safety analyses are performed twice to include the variation in the core thermal safety limits and the OTAT and OPAT reactor trip setpoints between a mixed core and a full VANTAGE 5 core:

(a) Uncontrolled RCCA Withdrawal at Power

(b) Excessive Load Increase Incident

(c) Excessive Heat Removal due to Feedwater System Malfunctions

(d) Loss of External Electric Load or Turbine Trip

Figure 14.1.0-5 presents the allowable RCS loop average temperature and vessel ΔT as a function of RCS pressure for the transition cycles (Cycles 8 and 9). This figure presents the most limiting operating configuration (nominal core thermal power = 3588 MWt, nominal RCS T-avg = 576°F, nominal RCS pressure = 2250 psia) of the potential future rerating range of conditions described in Table 14.1.0-1 (case 1) for the calculation of the OTAT and OPAT protection setpoints. A RCS flow rate of 366,400 gpm was assumed for generating these setpoints.

The OTAT and OPAT setpoints calculated for the transition cycles (cycles 8 and 9) are being used in cycles 10 and 11. This is conservative.

Figure 14.1.0-6 presents the allowable RCS loop average temperature and vessel ΔT as a function of RCS pressure for the cycles (Cycle 10 and beyond) with a full VANTAGE 5 core. This figure presents the most limiting operating configuration (nominal core thermal power = 3588 MWt, nominal RCS T-avg = 581.3°F, nominal RCS pressure = 2100 psia) of the potential rerating range of conditions described in Table 14.1.0-1 (case 4) for the calculation of the OTAT and OPAT protection setpoints. A RCS flow rate of 366,400 gpm was assumed for generating these setpoints.

The boundaries of operation defined by the OPAT and OTAT trip setpoints are represented as "protection lines" on these diagrams. The protection lines include all adverse instrumentation and setpoint errors so that under nominal conditions a reactor trip would occur within the area bounded by these lines. The utility of these diagrams is the fact that the limit imposed by any given DNBR can be represented as a line. The DNBR lines represent the locus of conditions for which DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given RCS pressure have a DNBR greater than the safety analysis limit value. These diagrams show that DNB is prevented for all cases if

the area enclosed within the maximum protection lines is not traversed by the applicable DNBR limit line at any point for a given pressurizer pressure.

The area of permissible operation (power, pressure, and temperature) is bounded by a combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); Overpower and Overtemperature ΔT (variable setpoints).

The differences between the limiting trip setpoint assumed for the safety analyses and the nominal reactor trip setpoint in Table 14.1.0-4 represents an allowance for instrumentation channel error and setpoint error. Nominal reactor trip setpoints are specified in the plant Technical Specifications and are shown in Table 14.1.0-4 for completeness. The reactor protection system (RPS) channels are calibrated and instrument response times determined periodically in accordance with the Technical Specifications.

14.1.0.6.b Engineered Safety Features (ESF) Actuation Setpoints and Time Delays

Table 14.1.0-5 presents the limiting ESF setpoints assumed in the safety analyses and the time delay assumed for each ESF actuation function. The nominal value of the low steamline pressure setpoint assumed was 500 psig. The revised low steamline pressure setpoint value provides operating margin for the potential reduced temperature operating conditions of Table 14.1.0-1 (cases 2, 5, and 6). The difference between the limiting ESF actuation setpoint assumed for the safety analyses and the nominal ESF actuation setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal ESF actuation setpoints are specified in plant Technical Specifications and are shown in Table 14.1.0-5 for completeness.

14.1.0.7 Plant Systems and Components Available for Mitigation of Occurrence Effects

Table 14.1.0-6 is a summary of reactor trip functions, engineered safety features actuation functions, and other equipment available for mitigation of accident effects. The trips and actuations in the Table 14.1.0-6 include some that are anticipatory and/or backup functions. These trips and actuations are not necessarily taken credit for the safety analyses.

In the safety analyses of the Chapter 14.1 occurrences, control system action is considered only if that action results in more severe occurrence results. No credit is taken for control system operations if that operation mitigates the results of an occurrence. For some occurrences, the analysis is performed both with and without control system operation to determine the worst case.

14.1.0.8 Residual Decay Heat

For the non-LOCA safety analyses, conservative core residual decay heat generation based on long-term operation at the initial power level preceding the reactor trip is assumed. The 1979 ANS residual decay heat standard (Reference 3) plus uncertainty was used for calculation of residual decay heat levels. Figure 14.1.0-7 presents this curve as a function of time after shutdown.

14.1.0.8.1 Distribution of Residual Decay Heat Following a LOCA

During a LOCA, the core is rapidly shutdown by void formation or RCCA insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4% which represents the fraction of heat generated within the clad and pellet drops to 95% for the hot fuel rod in a LOCA.

*Credit is not taken for RCCA insertion for the large break LOCA.

For example, 0.5 seconds after the initiation of a postulated double-ended large break LOCA (LBLOCA) about 30% of the energy generated in the fuel rods results from gamma ray absorption. Part of the gamma ray from the hot fuel rod is absorbed in the fuel rods surrounding the hot fuel rod. A conservative estimate of this effect is that 10% of the gamma ray (or 3% of the total energy) from the hot fuel rod is deposited in the fuel rod surrounding the hot fuel rod. Since the water density is considerably lower at this time, an average of 98% of the available energy is deposited in the fuel rods. The remaining 2% energy is absorbed by water, thimbles, sleeves, and grids. The net effect is that a factor of 0.95 (98% - 3%) rather than 0.974 should be applied to the residual decay heat production in the hot fuel rod.

14.1.0.9 Other Assumptions

Those analyses that model the mitigative effects of Protection and/or Engineer Safeguards Features have used the response times provided in Tables 7.2-6 and 7.2-7.

Some input assumptions differ somewhat from values that may be found elsewhere in the UFSAR. In particular, Tables 14.1.0-7, 14.1.0-8, and 14.1.0-9 display RCS volumes, steam generator mass, RCS pressure drops used in the current analyses. These tables can be found in reference 9.

14.1.0.10 Computer Codes Utilized

Summaries of some of the principal computer codes used in the safety analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given occurrence, such as those used in the analysis of the reactor coolant system pipe rupture (Section 14.3.1), are summarized in their respective safety analyses sections. The codes used in the analysis of each occurrence have been listed in Tables 14.1.0-2 and 14.1.0-3.

1. FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, density). The code uses a fuel model which simultaneously exhibits the following features:

- a. A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accidents.
- b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.
- c. The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

FACTRAN is further discussed in Reference 4.

LOFTRAN

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. All 4 (four) reactor coolant loops are modeled in LOFTRAN program. This code simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and RCCAs are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients. The reactor protection system (RPS) is simulated to include reactor trips on high neutron flux, overtemperature ΔT , overpower ΔT , high and low RCS pressure, low RCS flow, and high pressurizer level. Control systems are also simulated including RCCA, steam dump, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core thermal safety limits.

LOFTRAN is further discussed in Reference 5.

3. TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, RCCA motion, and others. Various edits are provided; e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 6.

4. THINC

The THINC-IV computer program is used to perform thermal-hydraulic calculations. The THINC-IV code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions. The THINC-IV code is described in detail in References 7 and 8, including models and correlations used.

14.1.0.11 REFERENCES

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4. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO₂ Fuel Rod," WCAP-7908, June 1972.
5. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.
6. Risher, D. H., Jr., and Barry, R. F., "TWINKLE - a Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028-A, January 1975.
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9. McFetridge, R. H., "American Electric Power Service Corporation, Donald C. Cook Nuclear Plant Unit 2, 3600 MWT Upgrading Program Engineering Report," Draft WCAP -14488, January 1996.

TABLE 14.1.0-1

RANGE OF PLANT NOMINAL CONDITIONS
USED IN SAFETY ANALYSES*

<u>Parameter</u>	<u>Case 1</u>	<u>Case 2</u>
NSSS Power, Mwt	3600	3600
Core Power, Mwt	3588 (1)	3588 (1)
RCS Flow, (gpm/loop)	88,500	88,500
Minimum Measured Flow, (total gpm)	366,400	366,400
<u>RCS Temperatures, °F</u>		
Core Outlet	613.5	585.8
Vessel Outlet	610.2	582.3
Core Average	579.5	550.1
Vessel Average	576.0	547.0
Vessel/Core Inlet	541.8	511.7
Steam Generator Outlet	541.6	511.4
Zero Load	547.0	547.0
RCS Pressure, psia	2250	2250
Steam Pressure, psia	780.4	587.0
Steam Flow, (10 ⁶ lb/hr total)	15.98	15.90
Feedwater Temp., °F	449.0	449.0
% SG Tube Plugging	10	10

* A brief description of each case follows Table 14.1.0-1

- (1) LOCA analysis with residual heat removal (RHR) or high head safety injection (HHSI) Crosstie valves closed based on 3411 Mwt.

TABLE 14.1.0-1 (continued)

RANGE OF PLANT NOMINAL CONDITIONS
USED IN SAFETY ANALYSES*

<u>Parameter</u>	<u>Case 3</u>	<u>Case 4</u>	<u>Case 5</u>	<u>Case 6</u>
NSSS Power, Mwt	3600	3600	3600	3600
Core Power, MWt	3588 (1)	3588 (1)	3588 (1)	3588 (1)
RCS Flow, (gpm/loop)	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)	366,400	366,400	366,400	366,400
<u>RCS Temperatures, °F</u>				
Core Outlet	618.4	618.2	585.8	585.7
Vessel Outlet	615.2	615.0	582.3	582.2
Core Average	584.8	584.9	550.1	550.1
Vessel Average	581.3	581.3	547.0	547.0
Vessel/Core Inlet	547.3	547.6	511.7	511.8
Steam Generator Outlet	547.1	547.4	511.4	511.5
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250	2100	2250	2100
Steam Pressure, psia	820.0	820.0	587.0	587.0
Steam Flow, (10 ⁶ lb/hr total)	16.0	16.0	15.9	15.9
Feedwater Temp., °F	449.0	449.0	449.0	449.0
% SG Tube Plugging	10	10	10	10

* A brief description of each case follows Table 14.1.0-1

(1) LOCA analysis with residual heat removal (RHR) or high head safety injection (HHSI) crosstie valves closed based on 3411 MWt.

A brief description of various cases listed in Table 14.1.0-1

Case 1 and 2: These parameters cases were used to support operation during mixed core cycles (Cycles 8 and 9).

Case 3: These parameters incorporate a core power level of 3588 MWt, an NSSS power level of 3600 MWt (which includes 12 MWt for reactor coolant pump heat), an average steam generator tube plugging level of 10%, RCS pressure of 2250 psia, and an upper bound vessel average temperature of 581.3°F. This parameter case was used to support high RCS temperature and high RCS pressure operation for a full VANTAGE 5 core (Cycle 10 and beyond).

Case 4: These parameters incorporate the same features as case 3, except the RCS pressure is 2100 psia. This parameter case was used to support high RCS temperature and low RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

Case 5: These parameters incorporate the same features as case 3, except the lower bound vessel average temperature is 547°F. This parameter case was used to support low RCS temperature and high RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

Case 6: These parameters incorporate the same features as case 5, except the RCS pressure is 2100 psia. This parameter case was used to support low RCS temperature and low RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

TABLE 14.1.0-2
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed				Revised Thermal Design Procedure	Initial NSSS Thermal Power Output ⁽⁸⁾ (MWt)	Reactor Vessel Coolant Flow (GPH)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density ($\Delta K/\text{gm/cc}$)	Doppler	DNB Correlation					
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	See Section 14.1.1.2	NA	(11)	W-3 ANF WRB-2 and W-3 V-5	No	0	162,840	547.0	2037.0(6)
RCCA Misalignment	LOFTRAN THINC	NA	NA	NA	W-3 ANF WRB-2 V-5	Yes	3600	366,400	581.3	2100.0(10)
Uncontrolled Boron Dilution	NA NA	NA NA	NA NA	NA NA	NA NA	NA NA	3600 0	NA NA	NA NA	NA NA
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	NA	Max(4)	W-3 ANF WRB-2 V-5	Yes	3608	366,400	581.3(12)	2100.0(10)
Locked Rotor (Peak Pressure)	-LOFTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	585.4	2312.6

*NA - Not Applicable

- (1) Minimum Doppler power coefficient (pcm/%power) = $-9.55 + 0.3732Q$, where Q is in % power (see Figure 14.1.0-1)
- (2) Multiple power levels, Tav_g, and reactivity feedback cases were examined.
- (3) Intentionally omitted
- (4) Maximum Doppler power coefficient (pcm/%power) = $-19.4 + 0.7176Q$, where Q is in % power (see Figure 14.1.0-1)
- (5) Minimum and maximum reactivity feedback cases were examined.
- (6) Core Pressure.
- (7) Full Power Doppler Power defect at BOL and EOL assumed to be -966 pcm and -893 pcm respectively.
- (8) Core thermal power.
- (9) Includes reactor coolant pump heat, if applicable.
- (10) For transition cycles, pressurizer pressure is 2250 psia.
- (11) Zero Power Doppler Power Defect at BOL assumed to be -1081 pcm.
- (12) For Transition Cycles, Vessel Average Temperature is 576°F.
- (13) Zero Power Doppler only Power defect at BOL and EOL assumed to be -965 pcm and 849 pcm, respectively.

TABLE 14.1.0-2 (Continued)
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed				Revised Thermal Design Procedure	Initial NSSS Thermal Power Output ⁽⁹⁾ (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation					
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	585.4	2037.4
Loss of Normal Feedwater	LOFTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	585.4	2312.6
Loss of Offsite Power (LOOP) to the Station Auxiliaries	LOFTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	541.4	2312.6
Rupture of a Steam Pipe	LOFTRAN THINC	See Figure 14.2.5-1	NA	See Figure 14.2.5-2	W-3 ANF W-3 V-5	NO	0	354,000	547.0	2100.0

*NA - Not Applicable

- (1) Minimum Doppler power coefficient (pcm/%power) = $-9.55 + 0.03732Q$, where Q is in % power (see Figure 14.1.0-1)
- (2) Multiple power levels, Tav_g, and reactivity feedback cases were examined.
- (3) Intentionally omitted.
- (4) Maximum Doppler power coefficient (pcm/%power) = $-19.4 + 0.07176Q$, where Q is in % power (see Figure 14.1.0-1)
- (5) Minimum and maximum reactivity feedback cases were examined.
- (6) Core Pressure.
- (7) Full Power Doppler Power defect at BOL and EOL assumed to be -966 pcm and -893 pcm respectively.
- (8) Core thermal power.
- (9) Includes reactor coolant pump heat, if applicable.
- (10) For transition cycles, pressurizer pressure is 2250 psia.
- (11) Zero Power Doppler Power Defect at BOL assumed to be -1081 pcm.
- (12) For Transition Cycles, Vessel Average Temperature is 576°F.
- (13) Zero Power Doppler only Power defect at BOL and EOL assumed to be -965 pcm and -849 pcm, respectively.

TABLE 14.1.0-2 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed				Revised Thermal Design Procedure	Initial NSSS Thermal Power Output ^m (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation					
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 14.2.6	NA	(7), (13)	NA	NA	3660(8) 0	354,000 162,840	585.4 547.0	2037.4(6)
Rupture of Feedwater Pipe	LOFTRAN	NA	.54	Max(4)	NA	NA	3680	354,000	585.4	2162.6

*NA - Not Applicable

- (1) Minimum Doppler power coefficient (pcm/%power) = $-9.55 + 0.03732Q$, where Q is in % power (see Figure 14.1.0-1)
- (2) Multiple power levels, T_{avg} , and reactivity feedback cases were examined.
- (3) Intentionally omitted.
- (4) Maximum Doppler power coefficient (pcm/%power) = $-19.4 + 0.07176Q$, where Q is in % power (see Figure 14.1.0-1)
- (5) Minimum and maximum reactivity feedback cases were examined.
- (6) Core Pressure.
- (7) Full Power Doppler Power defect at BOL and EOL assumed to be -966 pcm and -893 pcm respectively.
- (8) Core thermal power.
- (9) Includes reactor coolant pump heat, if applicable.
- (10) For transition cycles, pressurizer pressure is 2250 psia.
- (11) Zero Power Doppler Power Defect at BOL assumed to be -1081 pcm.
- (12) For Transition Cycles, Vessel Average Temperature is 576°F.
- (13) Zero Power Doppler only Power defect at BOL and EOL assumed to be -965 pcm and -849 pcm, respectively.

TABLE 14.1.0-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED: SEPARATE FULL VANTAGE 5 CORE ANALYSES

Fault Conditions	Computer Codes Utilized	Reactivity Coefficients Assumed				Revised Thermal Design Procedure	Initial NSSS Thermal Power Output ⁽¹⁾ (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
		Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation					
Uncontrolled Rod Cluster Assembly Bank Withdrawal At power (2),	LOFTRAN	NA*	.54	Max(3)	WRB-2	Yes	3680	366,400	581.3	2100.0
		+5	NA	Min(1)			2165 361		567.6 550.4	
Loss of Electrical Load or Turbine Trip (4)	LOFTRAN	NA	.54	Max(3)	WRB-2	Yes	3600	366,400	581.3	2100.0
		+5	NA	Min(1)						
Excessive Heat Removal Due to Feedwater System Malfunction	LOFTRAN	NA	.54	Min(1)	WRB-2	Yes	3600	366,400	581.3	2100.0
		NA	.54	Min(1)	WRB-2	Yes	0	366,400	547.0	2100.0
Excess Load Increase	LOFTRAN	NA	0	Min(1)	WRB-2	Yes	3600	366,400	581.3	2100.0
		NA	.54	Max(3)						

*NA - Not Applicable

- (1) Minimum Doppler power coefficient (pcm/%power) = $-9.55 + 0.03732Q$, where Q is in % power (see Figure 14.1.0-1)
 (2) Multiple power levels, Tav_g, and reactivity feedback cases were examined.
 (3) Maximum Doppler power coefficient (pcm/%power) = $-19.4 + 0.07176Q$, where Q is in % power (see Figure 14.1.0-1)
 (4) Minimum and maximum reactivity feedback cases were examined.
 (5) Includes reactor coolant pump heat, if applicable.

TABLE 14.1.0-4

RPS TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN NON-LOCA SAFETY ANALYSES

<u>Trip Function</u>	<u>Nominal Setpoint</u>	<u>Point Assumed In Analysis</u>	<u>Limiting Trip Time Delay (Seconds)</u>
Power range high neutron flux, high setting /	109%	118%	0.5
Power range high neutron flux, low setting	25%	35%	0.5
Overtemperature ΔT	See Table 2.2-1	Variable, see Figures 14.1.0-5,6	8.0 ^a
Overpower ΔT	in Tech Spec	Variable, see Figures 14.1.0-5,6	8.0 ^a
High pressurizer pressure	2385 psig	2428 psig	2.0
Low pressurizer pressure	1950 psig	1907 psig	2.0
High pressurizer water level	92% of span	100% span	2.0
Low reactor coolant flow (From loop flow detectors)	90% loop flow	87% loop flow	1.0
Undervoltage trip	2905 volts each bus	NA ^b	1.5
Underfrequency trip	57.5 Hz	57 Hz	0.6
Low-low steam generator level	21% of narrow range span	0.0% of narrow range span	2.0

^a Time delay (including RTD bypass loop fluid transport delay, bypass loop piping thermal capacity, RTD time response, and reactor trip circuit including channel electronics delay) from the time the temperature difference in the coolant loops exceeds the reactor trip setpoint until the RCCAs are free to fall. The time delay assumed in the analysis supports a total 6 second response time of the combined RTD time response, reactor trip circuit delay, and channel electronics delay presented in the updated Technical Specifications.

^b No explicit value assumed in the analysis. Undervoltage reactor trip setpoint assumed reached at initiation of analysis.

NA Not Applicable

TABLE 14.1.0-5

ESF ACTUATION SETPOINTS AND TIME DELAYS TO ACTUATION
ASSUMED IN NON-LOCA SAFETY ANALYSES

<u>ESF Actuation Function</u>	<u>Nominal Setpoint</u>	<u>Limiting Actuation Setpoint Assumed In Analyses</u>	<u>Time Delay (Seconds)</u>
Safety Injection (SI)			
- Low pressurizer pressure	1900 psig	1800 psig	27 w/ offsite power (Note 1)
			37 w/o offsite power (Note 2)
- Low steamline pressure	600 psig	344 psig	27 w/ offsite power (Note 1)
			37 w/o offsite power (Note 2)
Auxiliary Feedwater (AFW)			
- Low-low steam generator water level	21% of narrow range span	0.0% of narrow range span	60 ^a
High steam generator Level Turbine Trip	67% of narrow range span	82% of narrow range span	60 ^a
Steamline Isolation on low steam line pressure	NA	NA	11 ^b
Feedwater Line Isolation on high steam generator water level	NA	NA	11 ^c
Feedwater Line Isolation on low steam line pressure	NA	NA	8 ^c

TABLE 14.1.0-5 (continued)

ESF ACTUATION SETPOINT AND TIME DELAYS TO ACTUATION
ASSUMED IN NON-LOCA SAFETY ANALYSES

- a For Loss of Normal Feedwater and Loss of offsite power to Station Auxiliaries occurrences, the delay time assumed is 60 seconds from the initiation of the signals.
- For Feedwater Line Break event, the delay time assumed is 600 seconds (10 minute operator action delay) from the initiation of the break.
- b Steamline isolation total delay time includes valve closure time, and electronics and sensor delay. Technical Specifications require 8.0 second valve closure time.
- c Feedwater Line isolation total delay time includes valve closure time and electronics and sensor delay time.

Note 1: Emergency diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish safety injection (SI) path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.

Note 2: Emergency diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valve close) is included.

NA: Not Applicable

TABLE 14.1.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	<u>Fault Conditions</u>	<u>Reactor Trip Functions</u>	<u>ESF Actuation Functions</u>	<u>Other Equipment</u>	<u>ESF Equipment</u>
14.1.1	Uncontrolled RCCA bank withdrawal from a subcritical condition	Power range high flux (low setpoint)	NA	NA	NA
14.1.2	Uncontrolled RCCA bank withdrawal at power	Power range high flux, overtemperature delta-T, high pressurizer pressure, high pressurizer level	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.3	RCCA misalignment	Power range negative flux rate			
14.1.4	(including rod drop)				
14.1.5	Uncontrolled Boron Dilution	Source range high flux power range high flux overtemperature delta-T	NA	Low insertion limit annunciators for boration	NA
14.1.6.1	Partial and complete loss of forced reactor coolant flow	Low flow, undervoltage underfrequency	NA	Steam generator safety valves	NA
14.1.6.2	Reactor coolant pump shaft seizure (locked rotor)	Low flow	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.7	Startup of an inactive reactor coolant loop (Note 1)	-	-	-	-

TABLE 14.1.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	<u>Fault Conditions</u>	<u>Reactor Trip Functions</u>	<u>ESF Actuation Functions</u>	<u>Other Equipment</u>	<u>ESF Equipment</u>
14.1.8	Loss of external elec- trip load or turbine trip	High pressurizer pres- sure overtemperature delta-T, lo-lo steam generator level	Steam generator lo-lo level	Pressurizer safety valves, steam gen- erator safety valves	Auxiliary Feedwater System
14.1.19	Loss of normal feedwater	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator safety valves, pressurizer safety valves	Auxiliary Feedwater System
14.1.10	Feedwater system mal- functions that result in an increase in feed- water flow	Power range high flux, (low and high set- points), steam gener- ator lo-lo level (Intact steam generators)	High steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	NA
14.1.11	Excessive load increase	Power range high flux, overtemperature delta-T, overpower delta-T	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.12	Loss of offsite power to the station auxiliaries	Steam generator lo-lo level	Steam generator lo-lo level	Steam generator valves, pressurizer safety valves	Auxiliary Feedwater System
14.2.4	Steam generator tube failure	Reactor Trip System	Engineered Safety Features Actuation System	Steam generator safety and/or relief valves, steamline stop valves	Emergency Core Cool- ing System, Auxiliary Feedwater System, Emergency Power System

TABLE 14.1.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	<u>Fault Conditions</u>	<u>Reactor Trip Functions</u>	<u>ESF Actuation Functions</u>	<u>Other Equipment</u>	<u>ESF Equipment</u>
14.2.5	Rupture of a Steam Line	SIS, low pressurizer pressure, manual	Low pressurizer pressure, low compensated steamline pressure, high containment pressure, manual	Feedwater isolation valves, steamline stop valves	Auxiliary Feedwater System, Safety Injection System
	Inadvertent opening of a steam generator relief or safety valve	SIS	Low pressurizer pressure, low compensated steamline pressure	Feedwater isolation valves, steamline stop valves	Auxiliary Feedwater System, Safety Injection System
14.2.6	Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate	NA	NA	NA
14.2.8	Feedwater system pipe break	Steam generator lo-lo level, high pressurizer pressure, SIS	High containment pressure, steam generator lo-lo water level, low compensated steamline pressure	Steamline isolation valves, feedline isolation, pressurizer self-actuated safety valves, steam generator safety valves	Auxiliary Feedwater System, Safety Injection system
14.3	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System Component Cooling Water System steam generator safety and/or relief valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

NOTE 1: This cannot occur in Modes 1 and 2 as restricted by the Cook Nuclear Plant Unit 2 Technical Specifications.

TABLE 14.1.0-7

DONALD C. COOK UNIT 2 3600 MWT UPRATING PROGRAM

INPUT ASSUMPTIONS FOR RCS VOLUMES

<u>Input Assumption</u>	<u>Initial Condition</u>	
	<u>0% SGTP</u>	<u>10% SGTP</u>
Reactor Vessel (ft ³)	4764	4764
Steam Generators (ft ³ - total)	4308 (1)	4003 (1)
Reactor Coolant Pumps (ft ³ - total)	314	314
Loop Piping (ft ³ - total)	1175	1175
Surge Line Piping (ft ³)	43	43
Pressurizer (ft ³)	1800	1800
	-----	-----
Total RCS Volume (ft ³) (Ambient Conditions)	12,404	12,099
Total RCS Volume (ft ³) (Hot Conditions includes 3% for thermal expansion)	12,776	12,462

Notes:

- (1) The SG tube volume is assumed to be 762 ft³/SG (3048 ft³ total). The increase in SG plugging from 0% to 10% results in a total reduction in SG tube volume of 305 ft³.

TABLE 14.1.0-8

DONALD C. COOK UNIT 2 3600 MWT UPRATING PROGRAM
INPUT ASSUMPTIONS FOR STEAM GENERATOR SECONDARY MASS

<u>Input Assumption</u>	<u>Initial Condition</u>		
	0% SGTP (Original Design) Case	<u>10%/15% SGTP Cases</u>	
		<u>Low Temp</u>	<u>High Temp</u>
Steam generator secondary side mass (Total lbs/SG)	99,000	98,000 (2)	105,000 (3)

NOTES:

- (1) Initial conditions are presented for SGTP levels of 0% (Original Design) and 10%/15% to bound the range of SGTP levels at 3600 MWt.
- (2) For Tavg of 547°F
- (3) For Tavg of 579°F

TABLE 14.1.0-9
DONALD C. COOK UNIT 2 3600 MWT UPRATING PROGRAM
REACTOR COOLANT SYSTEM PRESSURE DROP ⁽¹⁾

	<u>0% SGTP</u>	<u>15% SGTP</u>
Reactor Vessel, including nozzles (psi)	51.79	49.07
Loop Piping (psi)	5.53	5.23
Steam Generator (psi)	<u>31.98</u>	<u>39.58</u>
Total (psi)	89.30(1)	93.88(1)

NOTES:

- (1) Pressure drops calculated at Best Estimate Flow.

feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup incident since it limits the power to a tolerable level prior to protective action. After the initial power burst, the neutron flux is momentarily reduced and then, if the incident is not terminated by a reactor trip, the neutron flux increases again, but at a much slower rate.

Termination of the startup incident by the previously discussed protection channels prevents core damage. In addition, the reactor trip from pressurizer high pressure serves as a backup to terminate the incident before an overpressure condition could occur.

14.1.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation, then, an average core heat transfer calculation, and finally, the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear power calculation is performed using spatial neutron kinetics methods (TWINKLE)⁽¹⁾ to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN⁽²⁾. The average heat flux is next used in THINC^(3,4) for transient DNBR calculations.

Analysis of this transient incorporates the neutron kinetics, including six delayed neutron groups and the core thermal and hydraulic equations. In addition to the neutron flux response, the average fuel, clad and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup incident, the following additional assumptions are made concerning the initial reactor conditions:

1. Since the magnitude of the neutron flux peak reached during the initial part of the transient is strongly dependent on the Doppler power reactivity coefficient, a conservatively low value for Doppler power defect (-1081 pcm) is used for any given rate of reactivity insertion.
2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator temperature reactivity coefficient. Although during normal operation (100% rated power), the moderator coefficient will not be positive at any time in core life, a highly conservative value has been used in the analysis to yield the maximum peak core heat flux. The analysis is based on a moderator coefficient which was at least +5 pcm/ $^{\circ}$ F at the zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code used in the analysis is a diffusion theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature.
3. The reactor is assumed to be at hot zero power (547 $^{\circ}$ F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-to-water heat transfer, a larger fuel thermal capacity, and a less-negative (smaller absolute magnitude) Doppler coefficient. The less-negative Doppler coefficient reduces the Doppler feedback effect thereby

RCS temperature of 581.3°F along with a nominal pressure of 2250 psia was found to produce the most conservative results.

Assumptions made in the analysis are:

- A. The plant is initially operating at 102 percent of the Cook Nuclear Plant Unit 2 core power level of 3588 MWt, plus 20 MWt for reactor coolant pump heat.
- B. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip. The ANS 1979 Decay Heat Model plus two sigma uncertainty was assumed.
- C. Reactor trip occurs on steam generator low-low level at 0.0% of narrow range span.
- D. The worst single failure in the auxiliary feedwater system occurs (e.g., failure of turbine driven auxiliary feedwater pump).
- E. Auxiliary feedwater is delivered to four steam generators at a rate of 450 gpm. The 450 gpm is assumed evenly split among the four steam generators and is delivered by two motor driven pumps at a steam generator pressure of 1123 psia.* Automatic initiation of the auxiliary feedwater is assumed 60 seconds after a low-low steam generator signal is actuated.
- F. Secondary system steam relief is achieved through the steam generator safety valves. First four safety valves at an actuation pressure of 1123 psia were assumed in the analysis.*

*An evaluation has been performed to justify an increase in the as-found tolerance of the main steam safety valves (MSSVs) from $\pm 1\%$ to $\pm 3\%$. The evaluation took credit for the staggered actuation of the MSSVs. The evaluation assumed that the MSSVs opened at 3% above the nominal lift pressure for each valve. The evaluation demonstrated that the secondary side pressure (assuming the staggered actuation of the MSSVs) would not exceed 1123 psia during the time when AFW is being supplied. The secondary side pressure transient would not preclude the AFW flow rate assumed in the analysis from being supplied to the steam generators.

- G. The initial reactor coolant average temperature is 4.1°F higher than the nominal value of 581.3°F, and initial pressurizer pressure is 62.6 psi higher than the nominal pressure of 2250 psia.
- H. The initial pressurizer water level is assumed to be at the maximum nominal setpoint (61.1% NRS) plus uncertainties (5% NRS).
- I. Pressurizer power operated relief valves (PORVs) are assumed operable to maximize pressurizer water volume.
- J. The maximum pressurizer spray flow rate is assumed to maximize pressurizer water volume.
- K. An auxiliary feedwater line purge volume of 100 ft³ per loop was assumed. This is the volume that needs to be purged before the relatively cold auxiliary feedwater reaches the steam generators.

Plant characteristics and initial conditions are shown in Table 14.1.0-2.

Results

Figures 14.1.9-1 through 14.1.9-3 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the collapse of voids and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the motor driven auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease.

The plot of pressurizer water volume clearly shows that the pressurizer does not fill. For comparison purposes, the pressurizer fills at 1889 ft³ (which includes the pressurizer surge volume).

The calculated sequence of events for this transient are shown in Table 14.1.9-1.

14.2.2 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

The inadvertent release of radioactive liquid to the environment is not considered a credible accident. Any radioactive liquids must ultimately be diverted to the monitor tanks, and any tritium from the CVCS to the monitor tanks also, prior to discharge. (Liquids from these tanks are sampled and monitored for acceptable radioactive levels before being released to the lake.) Erroneous sampling and malfunction of the radiation monitor would have to occur sequentially to discharge radioactive liquid inadvertently, and this series of events is not considered credible.

Waste Evaporator Condensate and Monitor Tanks

Any spillage of radioactive fluid due to equipment leaks or ruptures would drain directly to either the sump tank or waste holdup tanks, or would accumulate in the area sumps prior to being pumped to the waste holdup tanks. Radioactive liquids to be processed by the waste disposal system are ultimately stored in the waste holdup tanks.

Periodically the contents of the waste holdup tanks and the laundry tanks are analyzed and if the radioactive level is within discharge limits, the liquid is transferred to the waste evaporator condensate tanks and then to the monitor tanks for release.

Effluents from the waste disposal system and monitor tanks 3 and 4 are released, not recycled. Distillate from the CVCS boric acid evaporator is discharged to monitor tanks. The contents of monitor tanks 1 and 2 are analyzed before being pumped to the primary water storage tanks. Occasionally it may be necessary to dispose of some of the boric acid distillate for tritium control. (If analysis of the contents of the monitor tank is within prescribed limits for discharge to the environment, the liquid is pumped directly to the waste liquid discharge line after the normally locked-closed valve in this line is opened.) The radiation monitor downstream prevents discharge of fluids above prescribed levels as explained in the preceding paragraph.

A representative sample is obtained from the monitor tank to determine appropriate release setpoints. Administrative clearance must be granted to open a locked-closed valve. In the highly unlikely event that the locked-closed valve is opened and the tank contents are inadvertently pumped to the discharge tunnel for release to the lake without being previously analyzed for activity, the radiation monitors setpoint is set such that the release will not exceed release limits. If it did, the radiation monitor would trip the second valve downstream of the monitor and terminate the release. Therefore, a pumping accident having radiological consequences is not considered credible.

Condensate Storage Tank, Primary Water Storage Tank, and Refueling
Water Storage Tank

The condensate storage tank and the primary water storage tank are essentially free from radionuclides. The refueling water storage tank contains a relatively low level of radioactivity. These tanks are not connected to the radwaste system. In the unlikely event of loss of water from any of these tanks the water will percolate down the underground water table, which is estimated to be at elevation 590', that is, about 20 feet below ground level. The hydraulic gradient of the ground is very low; less than 4%. Our studies show a minimum of 50 years would be required for the water to reach the nearest ground water well. The spilled water would preferentially follow the very small natural ground gradient toward the lake and would be eventually diluted in the lake water. By the time any radioactive materials reach the nearest drinking water intake from the lake,

Bridgman is 2.5 miles away from the plant discharge, resultant dilution, dispersion, and radioactive decay will have reduced the radiological consequences to insignificance.*

- * The information presented here refers to the original Unit 1 studies. Later results of studies on this subject are included in Section 14.2 of the Unit 2 Updated FSAR.

Auxiliary Building Liquid Waste Storage Tanks

The inadvertent release of radioactive liquid waste to the environment is not considered a credible accident. Any spillage of radioactive fluid due to equipment leaks or ruptures would drain directly to either sumps or waste holdup tanks. Radioactive liquid wastes are diverted to tanks to be processed for release. Tanks are sampled and analyzed to determine that the concentration of radioactive nuclides can be released within discharge limits. The release must pass through a normally locked closed valve, a radiation monitor and another valve in series prior to reaching the discharge tunnels for release to the lake. Administrative clearance must be granted to open the locked closed valve. In the highly unlikely event that the locked-closed valve is opened and the tank contents are inadvertently pumped to the discharge tunnel for release to the lake without being previously analyzed for activity, the radiation monitors setpoint is set such that the release will not exceed release limits. If it did, the radiation monitor would trip the second valve downstream of the monitor and terminate the release. Therefore, a pumping accident involving radioactive waste releases having radiological consequences is not considered credible.

Piping

The pipes running from the refueling water storage tank, the primary water storage tank, and the condensate tank to the auxiliary building are installed in a pipe tunnel. In case of a break in any of these pipes, the water will enter the auxiliary building sump, from where it will be processed as described in the Auxiliary Building liquid waste tanks. No pipes from these tanks are directed toward the containment building.

CVCS Holdup Tanks

To ensure that the CVCS holdup tank(s) failure will not result in concentrations in excess of 10 CFR Part 20 at the nearest potable water supply, an analysis, based on the assumptions given in Standard Review Plan Section 15.7.3, has been performed for the rupture of a CVCS holdup tank in the auxiliary building. No credit has been taken for liquid retention by the foundation of the auxiliary building. The tank was assumed to be 80% full at the time of rupture and the liquid spilled is reactor coolant at one percent failed fuel. The capacity of these interconnected twin tanks is 128,000 gallons. Parameters used for this analysis are shown in Table 14.2.2-1. An Instantaneous Plane Source model with $y=0$ and $E_y=0$ is used. For a rectangular angular plane source of width f , parallel to the x - y plane and centered at the origin, this model is represented by the equation:

$$C = \left[\frac{m'}{4R_d n (\pi E_x t)^{\frac{1}{2}}} \right] \exp \left[-\frac{(x-Ut)^2}{4E_x t} + \lambda t \right] \left[\operatorname{erf} \left(\frac{y+f/2}{2(E_y t)^{\frac{1}{2}}} \right) - \operatorname{erf} \left(\frac{y-f/2}{2(E_y t)^{\frac{1}{2}}} \right) \right]$$

where:

- C = concentration of dissolved constituent
- m' = instantaneous release per unit area
- R_d = retardation factor
- n = effective porosity
- E_x = coefficient of dispersion in the x -direction
- t = time
- U = approximate rate of radionuclide movement
- λ = radioactive decay constant
- E_y = coefficient of dispersion in the y -direction



General

The accident examined is the complete severance of a single steam generator tube (SGTR). The accident is assumed to take place at a reactor power level of 3588 MWt with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power operated relief valves (and safety valves if their setpoint is reached).

The steam generator tube material is Inconel 600 and as the material is highly ductile, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance and an accumulation of minor leaks which exceed the Technical Specification limits is not permitted during unit operation.

The operator is expected to determine that a steam generator tube rupture (SGTR) has occurred, to identify and isolate the ruptured steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize the contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured steam generator. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale that ensures that break flow to the ruptured steam generator is terminated before the water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Description of Accident

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

1. Pressurizer low pressure and low level alarms are actuated, and prior to plant trip, charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip, as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that steam generator.
2. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer until a reactor trip signal is generated by low pressurizer pressure or overtemperature ΔT . A safety injection signal, initiated by low pressurizer pressure follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.
3. The steam generator blowdown liquid monitor and/or the air ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.
4. The reactor trip automatically trips the turbine, and if outside power is available, the steam dump valves open, permitting steam dump to the condenser. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator power operated relief valves (and the steam generator safety valves if their setpoint is reached).
5. Following plant trip, the continued action of auxiliary feedwater supply and borated safety injection flow [supplied from the refueling water storage tank (RWST)] provide a heat sink. Thus, steam bypass to the condenser, or in the case of loss of outside

power, steam relief to atmosphere, is attenuated during the time in which the recovery procedure leading to isolation is being carried out.

6. Safety injection flow results in restoration of pressurizer water level.

Results

In estimating the mass transfer from the reactor coolant system through the broken tube, the following assumptions were made:

- a. Plant trip occurs automatically as a result of low pressurizer pressure.
- b. Following the initiation of the safety injection signal, both centrifugal charging pumps are actuated and continue to deliver flow.
- c. After reactor trip the break flow equilibrates to the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.2.4-1 of Unit 1. In the original accident analysis, the resultant break flow is assumed to persist from plant trip until 30 minutes after the accident initiation. An assessment has been made of the impact on the original analysis of allowing the operator longer than 30 minutes to terminate break flow to the faulted steam generator. That assessment has shown that the break flow termination could be increased up to two hours (provided the steam generator does not over fill) without exceeding the offsite dose radiological guidelines discussed in the "Conclusions" portion of this section (Reference 2).
- d. The steam generators are controlled at the safety valve setting minus 3% tolerance rather than the power operated relief valve setting.

- e. The original analysis assumed that the operator identifies the accident type and terminates break flow to the ruptured steam generator within 30 minutes of accident initiation. An assessment has been made of the impact on the original analysis of allowing the operator longer than 30 minutes to terminate break flow to the faulted steam generator. That assessment has shown that the break flow termination could be increased up to two hours (provided the steam generator does not over fill) without exceeding the offsite dose radiological guidelines discussed in the "Conclusions" portion of this section. (Reference 2)

The above assumptions lead to a conservative upper bound of 140,264 pounds for the total amount of reactor coolant transferred to the ruptured steam generator and 56,525 pounds for the total amount of steam released to the atmosphere via the ruptured steam generator as a result of the steam generator tube rupture accident.

Recovery Procedure

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the Emergency Operating Procedures (EOPs). The EOPs are based on guidance in the Westinghouse Owner's Group Emergency Response Guidelines (Reference 1) which addresses the recovery from a SGTR with and without offsite power available.

The major operator actions include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

High secondary side activity, as indicated by the secondary side radiation monitors will typically provide the initial indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator level, a high radiation indication on the main air ejector monitor, or from the steam generator blowdown liquid monitor. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease off-scale on the narrow range for all of the steam generators. The auxiliary feedwater flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will return to the narrow range earlier in that steam generator and will continue to increase more rapidly. This response, as indicated by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage.

3. Cool down the RCS using the intact steam generators.

After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power-operated relief valves (PORVs) on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate

primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

Conclusion

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming a simultaneous loss of offsite power such that liquid does not enter the steam piping space. The doses to the public as a result of a steam generator tube rupture have been shown to be less than the permissible limits of 10 CFR Part 100. These limits are: for pre-accident iodine spike, the thyroid dose in 10 CFR 100 or 300 rem, for accident-initiated iodine spike, 10% (small fraction) of 10 CFR 100 or 300 rem thyroid and 2.5 rem gamma body.

References

1. Westinghouse Owners Group; Emergency Response Guidelines; Published by Westinghouse Electric Corporation for the Westinghouse Owners Group.
2. Westinghouse Letter to AEP NSD-NT-ESI-97-388 (AEP-97-102); American Electric Power Donald C. Cook Nuclear Plant Units 1 and 2, Revised SGTR FSAR Section 14.2.4; dated June 26, 1997.

effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and are thus different for each case studied.

The analyses assumed initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load conditions at time zero.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than for steam line breaks occurring at power.

- G. In computing the steam flow during a steam line break, the Moody Curve (Reference 4) for $fL/D = 0$ is used.
- H. The total delay time assumed for the steamline isolation is 11 seconds from receipt of actuation signal. The 11 second steamline isolation time includes valve closure time, and electronics and sensor delay. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location following steamline isolation, no more than one steam generator would

experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Plant characteristics and initial conditions are shown in Table 14.1.0-2.

Results

The limiting case for Cases a through e was shown to be the double-ended rupture located upstream of the flow restrictor with offsite power available (case b). Table 14.2.5-1 lists the limiting statepoints for this worst case. The results presented are a conservative indication of the events which would occur assuming a steam line rupture.

Figures 14.2.5-4 through 14.2.5-6 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) upstream of the flow restrictor at initial no-load conditions.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steamline and the remaining steamlines, or by low steam line pressure in two steamlines will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high-high containment pressure signals or low steamline pressure or high steam flow coincident with low-low T-avg. Even with the failure of one valve, release from the other steam generators is terminated by steamline isolation while the one generator blows down. The steam line stop valves are assumed to be fully closed in less than 11 seconds from receipt of a closure signal.

As shown in Figure 14.2.5-6, the core attains criticality with the RCCAs inserted (with the design shutdown margin assuming one stuck RCCA) before boron solution (2400 ppm from RWST) enters the RCS. A peak core power less than the nominal full power value is attained.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC^(10,11) calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the LOFTRAN computer code⁽¹²⁾. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 14.2.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation to provide worst case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distribution before and after ejection for a worst case can be found in Reference (4). During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis Reference (4).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is

accumulators. The accumulators are conservatively modelled at 2300 ppm for the post-LOCA subcriticality requirement.

An evaluation has been performed to determine the effect of a 3 minute SI interruption during the switchover to sump recirculation on the LBLOCA analysis. This scenario could occur if the RHR pump which was first switched over to recirculation fails at the time the other RHR pump is secured for switchover. Using a conservatively short estimate of the RWST draindown time and a bounding scenario for the availability of pumped injection, it was shown (Reference 28) that the short-term peak clad temperature results are not challenged by a three minute interruption of all ECCS flow.

14.3.1.1.3 Core and System Performance

14.3.1.1.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50⁽¹⁾.

14.3.1.1.3.2 Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974)⁽⁶⁾. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI,

WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)⁽⁵⁾; Kelly et al. (1974)⁽⁹⁾; Young et al. (1987)⁽⁴⁾; and Bordelon et al. (1974)⁽⁶⁾. Code modifications are specified in References 2, 7, 13, and 17. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill phase of the accident.

The BASH code is used to determine the RCS response during the reflood phase of the transient. The LOTIC computer code, described by Hsieh and Raymund in WCAP-8355 (1975) and WCAP-8345 (1974)⁽³⁾, calculates the containment backpressure transient. The containment backpressure transient is input to BASH for the purpose of calculating the reflood transient. The LOCBART computer code calculates the thermal transient of the hottest fuel rod in the three phases. The improved fuel performance model, described in Reference 15, generates the initial fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the break mass and energy flow rates that are assumed to be vented to the containment during blowdown.

At the end of the blowdown, information on the state of the system is transferred to the WREFLOOD code which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The large break analysis was performed with the December 1981 version of the evaluation model modified to incorporate the BASH⁽⁴⁾ computer code.

- Figures 14.3.1-15a-g The clad temperature at the hot spot is shown for the hot rod.
- Figures 14.3.1-16-18 The containment backpressure transient used in the analysis is provided for Cases A, F and G (the minimum and maximum SI flow cases, and the 3413 Mwt cross tie valve closed case).
- Figures 14.3.1-19-27 These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray for Cases A, F and G.
- Figures 14.3.1-28-30 These figures show the temperature transients in both the lower and upper compartments of containment and flow from the upper to lower compartments for Cases A, F and G.

The peak clad temperature calculated for a large break is 2140°F, which is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 6.80 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, corresponding to less than 0.3 percent hydrogen generation, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Assessment for Changes in Models and Applications

10 CFR 50.46(a)(3) requires the record keeping and reporting of changes in LOCA evaluation models and of changes in the application of these models. Reference 21 reports the following permanent changes that apply to Unit 2 of Cook Nuclear Plant:

1. An update of the fuel rod model in the LOCA evaluation model to maintain consistency with the latest approved version of the Westinghouse fuel rod design code, and
2. An estimation of the change in results is included to account for combining most severe LOCA and plant-specific seismic forces upon the steam generator tubes in determining the steam generator flow reduction during a LOCA event.

Additionally, the potential impact of assuming only the average rod in the hottest assembly for computing fuel rod burst and related channel blockage effects was evaluated as not requiring any adjustment in the results. Reference 21 describes these changes in models and applications in greater detail. Tabulations of the assessments against peak clad temperature due to these changes for the limiting cases are shown in Table 14.3.1-6. In each case, the peak clad temperature result remains less than 2200 degrees F.

The current licensing basis peak clad temperatures given in Table 14.3.1-6 are based on large break LOCA analysis contained in Reference 24. In these analyses, the peak clad temperatures for structural metal heat modeling and power margin allocation have been specified. The current licensing basis peak clad temperatures given in Table 14.3.1-6 includes these two additional margins.

For both cases A and G in Table 14.3.1-6, a correction to account for errors found in LUCIFER of -6°F was applied based on the information given in Reference 25. A correction of 253°F was also applied to account for changes to the power shape in the LBLOCA analysis. In addition, a credit of 237°F was taken for the effect of the hot leg nozzle gap effect. These corrections are described in reference 26. For both cases, the licensing basis peak clad temperature is below 2200°F.

Reference 27 reports an error in the translation of fluid conditions from the SATAN to LOCTA codes. The error results in a peak cladding temperature penalty of 15°F. The peak cladding temperature for Cases A and G remain below the 2200°F limit.

An evaluation has been performed which demonstrates that the large break LOCA PCTs for cases A and G in Table 14.3.1-6 will remain below 2200°F for a reduction in safety injection flow rates resulting from an increase in the RHR and high head safety injection pump head degradation from 10% to 15%.

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28. AEP-97-004/NSD-SAE-ESI-97-020; "American Electric Power, Donald C. Cook Nuclear Plant Units 1 and 2; LBLOCA Evaluation for 3 Minute SI Interruption; January 17, 1997 (Westinghouse Letter to American Electric Power).

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 14.3.1-1

LARGE BREAK LOCA - CASES ANALYZED

- CASE A - $C_D=0.6$, 3588 Mwt Core Power, High Temperature ($T_{HOT}=615.2^{\circ}F$), High Pressure ($P_{RCS}=2313$ psia), $F_Q=2.220$, $F_{AH}^N=1.620$, Minimum SI with cross-tie valves open. Limiting break case, i.e., this case had highest PCT for all cases analyzed.
- CASE B - $C_D=0.4$, 3588 Mwt Core Power, High Temperature ($T_{HOT}=615.2^{\circ}F$), High Pressure ($P_{RCS}=2313$ psia), $F_Q=2.240$, $F_{AH}^N=1.620$, Minimum SI with cross-tie valves open.
- CASE C - $C_D=0.8$, 3588 Mwt Core Power, High Temperature ($T_{HOT}=615.2^{\circ}F$), High Pressure ($P_{RCS}=2313$ psia), $F_Q=2.240$, $F_{AH}^N=1.620$, Minimum SI with cross-tie valves open.
- CASE D - $C_D=0.6$, 3588 Mwt Core Power, Low Temperature ($T_{HOT}=582.3^{\circ}F$), High Pressure ($P_{RCS}=2313$ psia), $F_Q=2.220$, $F_{AH}^N=1.620$, Minimum SI with cross-tie valves open.
- CASE E - $C_D=0.6$, 3588 Mwt Core Power, High Temperature ($T_{HOT}=615.0^{\circ}F$), Low Pressure ($P_{RCS}=2037$ psia), $F_Q=2.220$, $F_{AH}^N=1.620$, Minimum SI with cross-tie valves open.
- CASE F - $C_D=0.6$, 3588 Mwt Core Power, High Temperature ($T_{HOT}=615.2^{\circ}F$), High Pressure ($P_{RCS}=2313$ psia), $F_Q=2.220$, $F_{AH}^N=1.620$, Maximum SI with cross-tie valves open.

TABLE 14.3.1-6
LARGE BREAK LOCA RESULTS FUEL CLADDING DATA

	Case A C _b =0.6 Min SI 3588 Mwt 615.2°F <u>2313 psia</u>	Case B C _b =0.4 Min SI 3588 Mwt 615.2°F <u>2313 psia</u>	Case C C _b =0.8 Min SI 3588 Mwt 615.2°F <u>2313 psia</u>	Case D C _b =0.6 Min SI 3588 Mwt 582.3°F <u>2313 psia</u>	Case E C _b =0.6 Min SI 3588 Mwt 615.0°F <u>2037 psia</u>	Case F C _b =0.6 Max SI 3588 Mwt 615.2°F <u>2313 psia</u>	Case G C _b =0.6 RHR X-Tie 3413 Mwt 611.2°F <u>2313 psia</u>
T _{hot} =							
P _{RCS} =							
Peak Clad Temperature (°F) Computed in Analysis	2140.0	1848.2	1766.0	1878.4	2074.7	2102.7	2090.0
Model Assessments:							
Fuel Rod Model Update	+10.0						+10.0
Seismic and LOCA Forces on Steam Generator Tubes	+20.0						+20.0
Structural Metal Heat Modeling	-25.0						-25.0
Lucifer Error Correction	-6.0						-6.0
Power Margin	-98.0						
Skewed Power Shape Penalty	+253.0						+253.0
Hot Leg Nozzle Gap Benefit	-237.0						-237.0
Translation From Satan to LOCTA	+15						+15
Current Licensing Basis	2072.0*						2120.0*
Peak Clad Temperature Location (ft)	9.75	8.75	6.25	9.75	9.75	9.75	9.75
Peak Clad Temperature Time (sec)	258.9	250.1	57.9	239.9	255.4	253.1	244.4
Local Zr/H ₂ O Reaction Maximum (%)	6.80	3.56	2.97	3.30	5.71	6.18	6.08
Local Zr/H ₂ O Reaction Location (ft)	9.75	6.25	5.25	9.75	9.75	9.75	9.75
Total Zr/H ₂ O Reaction (%)	<0.3	<0.3	<0.3	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time (sec)	45.79	60.93	50.66	50.11	46.05	46.04	46.10
Hot Rod Burst Location (ft)	6.00	6.25	5.25	6.00	6.00	6.00	6.00

CALCULATION ASSUMPTIONS:

Peak Linear Power (KW/ft), 102% 12.714 (12.721 for Case G)
 Peaking Factor (at License Rating) 2.220 (2.335 for Case G)
 Accumulator Water Volume (ft³) per accumulator 946
 Cycle Analyzed All

* See the LOCA evaluation log maintained by the Nuclear Safety & Analysis Section for temporary margin allocations.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 14.3.1-7
CASE A - LARGE BREAK LOCA $C_D=0.6$ MINIMUM SAFEGUARDS
MASS AND ENERGY RELEASE RATES

<u>Time (sec)</u>	<u>Mass Flow Rate (lbm/sec)</u>	<u>Energy Flow Rate (BTU/sec)</u>
0.0	70562.	37960066.
1.0	66324.	34809386.
2.0	58446.	30872684.
3.0	47776.	25444113.
4.0	40310.	21684814.
5.0	32388.	17815357.
6.0	30679.	17103044.
7.0	29057.	16373321.
8.0	27299.	15517895.
9.0	25547.	14706648.
10.0	22446.	13347482.
11.0	19737.	11937678.
12.0	17525.	10722934.
12.4	16806.	10271911.
13.0	15567.	9618894.
14.0	13863.	8692759.
15.0	12346.	7910304.
16.0	10803.	7134722.
17.0	9785.	6598154.
18.0	8687.	5904895.
19.0	7013.	5001042.
20.0	4975.	3600314.
21.0	5361.	3099603.
22.0	7165.	3249819.
23.0	7503.	2958259.
24.0	7368.	2506588.
25.0	6741.	1964716.
26.0	5803.	1452731.
27.0	5513.	1313192.
28.0	4940.	1064918.
29.0	4386.	833363.
30.0	3459.	548032.
31.0	2581.	354346.
32.0	1419.	80449.
33.0	1406.	79650.
34.0	1393.	78887.
35.8	1381.	78166.
40.0	193.2	7361.
46.0	193.2	7361.
46.6	193.2	7408.
66.0	608.1	208262.
86.5	623.2	208283.
109.9	631.6	204153.
135.2	637.8	199042.
171.5	676.6	204636.
257.9	691.8	200029.

SBLOCA Model Assessments

10CFR50.46(a)(3) requires the record keeping and reporting of changes in LOCA evaluation models and of changes in the application of these models. Table 14.3.2-14 shows the peak clad temperature obtained for the small break LOCA analysis for both the high head safety injection cross-tie open and closed cases. The various changes to the evaluation models shown in the table are taken from references 9, 12, 14, 15, 16, and 17. The peak clad temperature result for all cases remains below 2200°F.

An evaluation has been performed which demonstrates that the small break LOCA PCTs for cases 1 and 3 in Table 14.3.2-14 will remain below 2200°F for a reduction in safety injection flow rates resulting from an increase in the RHR and high head safety injection pump head degradation from 10% to 15%.

References, Section 14.3.2

1. "Acceptance Criteria for Emergency Core Cooling Systems for Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.
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3. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600, June 1979.
4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January 1980.
5. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
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9. Rupprecht, S. D., et al, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A, October 1986.
10. Fitzpatrick, E. E. (I&M) letter to T. E. Murley (NRC), July 18, 1991, AEP:NRC:1118B.
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16. Fitzpatrick, E. E. (I&M), letter to NRC Document Control Desk, March 22, 1996, AEP:NRC:1118K.
17. Fitzpatrick, E. E. (I&M), letter to NRC Document Control Desk, April 10, 1997, AEP:NRC:1118L.

TABLE 14.3.2-13

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATIONS (4" Break)RESULTS

	LPHT
	<u>w/ MSSV</u>
NOTRUMP Peak Clad Temperature (°F)	1531
Peak Clad Temperature Location (ft)	11.25
Peak Clad Temperature Time (sec)	846.1
Local Zr/H ₂ O Reaction Maximum (%)	0.459
Local Zr/H ₂ O Reaction Location (ft)	11.25
Total Zr/H ₂ O Reaction (%)	<1.0
Rod Burst	None
Artificial Leak-By Penalty (°F)	12
Burst and Blockage Penalty (°F)	None
Total Peak Clad Temperature (°F)	1543
CALCULATION:	
NSSS Power MWt 102% of	3588*
Peak Linear Power kw/ft 102% of	12.764
Hot Rod Power Distribution (kw/ft)	See Figure 14.3.2-68
Accumulator Water Volume, cu. ft.	946

LPHT is low pressure, high temperature operating condition.

W/ MSSV is main steam safety valve setpoint tolerance increase case at 3588 MWt core power with HHSI crossties open.

* Does not include pump heat

TABLE 14.3.2-14

SMALL BREAK LOSS OF COOLANT ACCIDENT CALCULATION
PEAK CLAD TEMPERATURE (°F) ASSESSMENTS SINCE LAST ANALYSIS

<u>Parameter</u>	<u>Case 1*</u>	<u>Case 2*</u>	<u>Case 3*</u>
Value computed in analysis	1956	1947	1531
Drift Flux Flow Regime Errors	-13	-13	-13
Lucifer Error Corrections	-16	-16	-16
Boiling Heat Transfer Correlation Error	-6	-6	-6
Steam Line Isolation Logic Error	+18	+18	+18
Containment Spray during SBLOCA	N/A	N/A	N/A
Axial Nodalization, RIP Model Revision, and SBLOCA Error Corrections Analysis	+57	-45	+3
NOTRUMP Specific Enthalpy Error	+20	+20	+20
Burst and Blockage/Time in Life	+0	+74	+0
SBLOCA Fuel Rod Initialization Error	+10	+10	+10
Loop Seal Evaluation Error	-38	-38	-38
Licensing Basis & Permanent Assessments	1988	1929	1529

* Case 1 = Power level of 3250 MWt**, HHSI Cross-tie Closed

Case 2 = Power level of 3413 MWt**, HHSI Cross-tie Closed

Case 3 = Power level of 3588 MWt**, HHSI Cross-tie Open

**Note that the power level used in the analyses is 102% of the value given and does not include pump heat.

14.3.5.3.2 Fuel Handling Accident

A discussion on the bounding case for a fuel handling accident inside of the auxiliary building can be found in Section 14.2.1 of the Unit 1 FSAR for the potential power uprate of Unit 2 to 3588 MWt. An analysis of the fuel handling accident both inside the auxiliary building and inside containment was performed by Westinghouse. The results are presented in Table 14.3.5-6.

14.3.5.3.3 Locked Rotor (Westinghouse)

See Unit 2 UFSAR Section 14.1.6.

14.3.5.3.4 Steam Generator Tube and Main Steamline Ruptures

The results of a steam generator tube rupture radiological analysis at a power level of 3588 MWt is found in Table 14.3.5-6. (The analysis is also discussed in Unit 1 FSAR Chapter 14.2.7.)

For steam line break, the Cook Nuclear Plant Unit 2 FSAR analysis (Section 14.2.5) indicated that there would be no fuel failures for this transient. Thus, the Technical Specification limits on coolant activity will be the controlling factor for offsite doses. This limit has not been changed so the FSAR dose results (Unit 1 UFSAR Section 14.2.7) remain bounding.

As with the locked rotor transient, if fuel failures were projected to occur, they would be bounded by the assumptions in the loss of coolant analysis presented in Section 14.3.5.3.1 and therefore only a small fraction of the 10 CFR 100 limits.

14.3.5.3.5 RCC Assembly Ejection Incident

As discussed in Unit 2 UFSAR Section 14.2.6, analyses indicate that fuel and clad limits are not exceeded. From this it is concluded that there is no likelihood of a sudden fuel dispersal into the coolant. Since the peak pressure does not cause stress to exceed stress limits, no further consequence to the RCS is likely. The analyses also show that less than 10% of the fuel rods in the core will enter DNB and release fission products.

14.3.5.3.6 LOCA Event

The salient parameters used in the LOCA analysis are presented in Table 14.3.5-5. The resulting offsite doses are presented in Table 14.3.5-6. These doses are within the 10 CFR 100 guidelines. This information is applicable to both Unit 1 and Unit 2 of the UFSAR.

14.3.5.3.7 Control Room Habitability

Analyses associated with control room habitability are presented in Section 14.3.5 of the Unit 1 FSAR. These analyses bound both Units 1 and 2.

14.3.5 REFERENCES

1. "ORIGEN - The ORNL Isotope Generation and Depletion Code," M. J. Bell, Oak Ridge National Laboratory, ORNL-4628, May 1973.
2. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay from ENDF/B," ORNL RSIC, ORNL DLC-38/ORYX-E, September 1975.
3. WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985.
4. NUREG/CR-5008, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.
5. "FIPCO, A Computer Code for Calculating the Distribution of Fission Products in Reactor Systems," WCAP-7949 (proprietary), August 1972.

PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT

General

Core power level, Mwt 3588

Source Term

Fifty percent of the core iodine is assumed to be uniformly distributed in the lower containment at time zero (TID-14844/Regulatory Guide 1.4)

I-131	5.0×10^7 curies
I-132	7.3×10^7
I-133	1.0×10^8
I-134	1.1×10^8
I-135	9.5×10^7

Iodine Plate-out Factor 0.5

Iodine species

Elemental	0.91
Organic	0.04
Particulate	0.05

100 percent of the core noble gas is released to containment.

Kr-85m	2.6×10^7 curies
Kr-85	8.3×10^5
Kr-87	4.8×10^7
Kr-88	6.8×10^7
Xe-131m	7.1×10^5
Xe-133m	2.9×10^7
Xe-133	2.0×10^8
Xe-135m	4.1×10^7
Xe-135	4.2×10^7
Xe-138	1.6×10^8

Containment Parameters

Volume of upper containment, ft ³	7.74×10^5
Volume of lower containment (Includes dead ended volumes)	3.62×10^5
Volume of ice beds	1.11×10^5

Containment leak rate

0-24 hr, percent/day	0.25
> 24 hr.	0.125

Table 14.3.5-5
PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT

(PAGE 2 OF 3)

Containment Parameters (cont'd)

Containment Spray System *

Upper Containment	
Spray flow rate, gpm	2075
Spray fall height, ft	85
Lower Containment	
Spray flow rate, gpm	1006
Spray fall height, ft	50
Spray Injection Starts, Sec	115
Air Steam Flow Rates, cfm	
0-10 min.	416,000 (average flow rate from lower to upper containment)
>10 min.	39,000 (recirculated between lower and upper containment through the ice beds)

Iodine Removal Parameters

Upper Containment	
Elemental iodine removal by spray, hr^{-1}	
injection spray (115 sec. to 16 min.)	10
recirculation spray (>20 min.)	3.2
Particulate iodine removal by spray, hr^{-1}	7.3
Lower Containment	
Elemental iodine removal by spray, hr^{-1}	
injection spray (115 sec to 16 min)	10
recirculation spray (>20 min)	2.8
Particulate iodine removal by spray, hr^{-1}	4.6
Ice Condenser Iodine removal efficiency	
0-10 min.	0
10-40 min.	0.3
>40 min.	0
Elemental iodine DF (includes the combined effects of sprays and the ice condenser)	100
Particulate iodine DF	100

* A four minute interruption in the operation of the containment spray system during the transfer to cold leg recirculation is assumed in the offsite dose evaluation.

Table 14.3.5-5
PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT

(PAGE 3 OF 3)

Miscellaneous Parameters

Atmospheric dispersion factors at the site boundary and at the outer boundary of the low population zone (LPZ), sec/m^3 , and breathing rates, m^3/sec :

	<u>Site Boundary</u>	<u>LPZ</u>
0-2 hr.	3.15×10^{-4}	7.5×10^{-5}
2-24 hr.	0	7.5×10^{-5}
1-5 days	0	2.6×10^{-6}
5-30 days	0	7.9×10^{-7}
	<u>Breathing Rate</u>	
0-8 hr.	3.47×10^{-4}	
8-24 hr.	1.75×10^{-4}	
1-30 days	2.32×10^{-4}	

Table 14.3.5-6

ESTIMATED DOSES FOR 3588 MWT POWER OPERATION

<u>Accident Description</u>	<u>Doses in rem</u>
Fuel Handling Accident in the Auxiliary Building	
0-2 hour thyroid at SB	2.6
0-2 hour whole body at SB	0.6
Fuel Handling Accident in Containment	
0-2 hour thyroid at SB	100
0-2 hour whole body at SB	1.4
Steam Generator Tube Rupture	
0-2 hour site boundary	
thyroid	1.7
whole body	0.2
0-8 hour Low Population Zone (LPZ)	
thyroid	0.4
whole body	0.05
Large-Break LOCA *	
0-2 hour	
thyroid	154
whole body	2.4
0-30 day LPZ	
thyroid	134
whole body gamma	1.8
* Calculated doses	

14.3.7 LONG TERM COOLING

This section has been revised to incorporate updated analytical material⁽⁵⁾ regarding pressurized thermal shock (PTS) and the prevention of Reactor Coolant System (RCS) overpressurization conditions during, or subsequent to, periods of rapid and/or prolonged system cooldown. This material indicates that certain small LOCAs and double-ended (or equivalent) SGTRs with high break flow rates have replaced large steamline breaks, certain small break LOCAs and feedwater line breaks as the dominant PTS related accidents. General discussions of the PTS issues, including stagnant loop concerns and Emergency Response Guideline (ERG) actions are presented. The methodology followed in applying this generic information specifically to the Donald C. Cook Nuclear Plant is summarized. The PTS screening criteria and limiting approved calculated PTS values projected to the end of vessel life determined in accordance with 10 CFR 50.61 are also provided for Donald C. Cook Units 1 and 2. Finally, a brief general discussion of long term cooling and the use of WOG-ERG based plant emergency operating procedures to prevent excessive cooldown and reduce any PTS related risk is presented.

1. Generic Background and Description

A combination of severe cooling (thermal shock) and high pressure produces the condition that is called pressurized thermal shock (PTS). Within the thick walls of the reactor pressure vessel, a substantial temperature gradient can be produced by rapid cooling of the inner surface. This gradient results in thermal stresses that are tensile in nature and that are a maximum at the inner surface of the vessel. If the system is pressurized during or after the cooldown occurs, an additional pressure stress is imposed on the vessel wall, again being tensile in nature and having a maximum at the inner surface. It is this combined pressure-temperature stress that is of primary concern for PTS.

A limiting PTS condition that may challenge the reactor vessel integrity can occur during a severe transient such as a loss of coolant accident (LOCA), a secondary side depressurization (steamline or feedline break), or a steam generator tube rupture (SGTR). Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside of the vessel wall followed by high repressurization,
- significant degradation of vessel material toughness caused by radiation embrittlement, and
- the presence of a critical-size defect in the vessel wall.

A PTS concern arises if one of these transients acts on the highly irradiated beltline region of a reactor vessel where a reduced fracture resistance exists because of the neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Westinghouse Owners Group (WOG) completed a generic evaluation program of the severity of the thermal shock transient and submitted a report to the NRC, WCAP-10019, dated December 1981⁽¹⁾. This generic report concluded that, based on a conservative assessment, all of the Westinghouse Pressurized Water Reactors, including Donald C. Cook Units 1 and 2, could continue to operate for a considerable number of years before the reactor vessel integrity acceptance criteria would be violated. The results of the WOG program demonstrated that no immediate reactor vessel integrity concerns exist.

As part of this continuing effort another report was submitted to the NRC in May 1982 by the Westinghouse Owners Group⁽²⁾. This report provided additional information on WCAP-10019 and responded to the NRC's short term action needs as the Staff perceived them⁽³⁾. Several methodological differences exist between References (1) and (2), particularly on the subject of crack arresting as the basis on which to predicate vessel integrity.

In the above studies, and also a probabilistic transient evaluation by the NRC⁽⁴⁾, it was recognized that transients leading to stagnation of flow in the reactor coolant loops while safety injection flow continues may be an additional candidate contributing to PTS. The stagnant loop considerations are discussed below.

2. General Discussion of PTS Risk Including Stagnant Reactor Coolant Loop Conditions

During a transient or emergency event, all reactor coolant pumps (RCPs) may be stopped due to loss of support conditions (e.g., offsite power supply, cooling water to motors or seals) or due to meeting the RCP trip criteria. In this latter situation, the operator would be directed to trip the RCPs if a safety injection pump is running and the RCP trip parameter is reached (e.g., low RCS subcooling or low RCS pressure). After RCP trip, unless the residual heat removal (RHR) system is in service and is removing decay heat, natural circulation flow will be needed to remove core decay heat through the steam generators.

If natural circulation flow decreases or is stopped in one or more loops and safety injection (SI) flow is maintained to the cold legs of the affected loops, the relatively cold SI water will mix with the water in the cold legs and vessel downcomer. This can cause a PTS concern for the reactor vessel if the RCS pressure remains or becomes high.

For a rigorous determination of the PTS risk for the plant, literally thousands of postulated cooldown scenarios could be considered. However, by applying appropriately conservative approximations, it was possible to focus on the limiting cases and also analyze this problem on a generic basis. A generic study of PTS risk, including stagnant loop considerations, was performed by the Westinghouse Owners Group and is provided in WCAP-10319⁽⁵⁾.

The PTS study including stagnant loop conditions consisted of three main efforts:

- an event tree analysis of the seven transient families believed to include all the potential stagnant loop transients that contribute to the overall PTS risk.
- a thermal hydraulic analysis to determine a characteristic pressure, final temperature (reflecting the depth of the cooldown),

and time constant (reflecting the rate of cooldown) for each of the unique sequences or "bins" identified in the event tree analysis, and

- application of the NRC probabilistic fracture mechanics (PFM) model from Reference (4) to determine the PTS risk associated with each of the identified sequences or bins.

In the WCAP-10319 study, the PTS risk is defined based on the frequency of significant flaw extension for longitudinal flaws that may exist at the inner surface of the reactor vessel. As noted below, welds oriented in the circumferential direction have a less stringent PTS screening criterion (i.e., 300 versus 270°F), so selection of flaws oriented in the axial or longitudinal direction for a measure of the PTS risk is appropriate and conservative. This selected PTS risk parameter is evaluated as a function of the mean surface RT_{NDT} (reference temperature nil-ductility transition). This vessel surface RT_{NDT} parameter, now referred to as the vessel RT_{PTS} (reference temperature for pressurized thermal shock), is provided in Figure 14.3.7-1. This figure can be applied to most of the WOG member plants including the Donald C. Cook Nuclear Plant.

In Figure 14.3.7-1, the PTS contributions associated with each of the seven categories or transient types investigated in the WCAP-10319 WOG study are provided. The corresponding "WOG TOTAL" is also shown and compares closely with the "NRC TOTAL" from Reference (4). The results of this WOG study show that the overall PTS risk for a typical Westinghouse plant is dominated by LOCAs and SGTRs, specifically small LOCAs with equivalent diameters in a certain range (about 2" to 6" for a 4-loop plant) and double-ended (or equivalent) SGTRs that occur simultaneously with a loss of offsite power or other conditions resulting in a trip of all RCPs.

Hot leg LOCAs were specifically analyzed for the generic study to maximize the amount of cold safety injection and accumulator water added to the RCS cold legs and vessel downcomer regions. Note that for smaller LOCA cases, SI flow would be able to keep up with break flow; if

the RCPs are tripped for these cases, natural circulation flow would be maintained and there would be no uncontrolled cooldown of the cold legs and vessel downcomer regions. For larger break sizes, the cooldown would be uncontrolled but the RCS depressurizes quickly. Thus, for these extremes in break sizes, either the temperature or pressure stress contributions are minimized, so these events become less severe from the PTS perspective than LOCAs in the 2" to 6" range analyzed for the PTS studies.

Some of the scenarios used for the SGTR cases also account for extended delays in termination of safety injection following the initiation of operator actions to cooldown and depressurize the RCS (Note: In addition to identifying and isolating the ruptured steam generator, these operator actions are credited for the SGTR event). Manual SI termination delays following depressurization of the RCS to the ruptured S/G pressure were assumed for these cases. This delay tends to maximize the time that flow in the ruptured loop is slowed down or stagnates prior to SI termination. The maximum delay cases correspond to a total transient time of more than 60 minutes for operator action to terminate SI flow. This category of SGTR is calculated to contribute less than 1% to the estimated total SGTR initiating event frequency of 3.9×10^{-2} occurrences per reactor-year.

It is also important to point out that in recent years, there has been an increased awareness of the need for performing any required operator SGTR accident response actions in a timely manner. This increased operator awareness, combined with a high probability that the RCPs would be left operating for a design basis SGTR event, tends to make the SGTR contribution to overall PTS risk in WCAP-10319 conservatively high when applied to most WOG plants including the Donald C. Cook Nuclear Plant.

The WOG study indicated that the cold legs and vessel downcomer do not, on the average, cool down as much for the SGTR cases as for small LOCAs. It is because of this that below an RT_{PTS} of 290°F, small LOCAs

are the dominant contributors to PTS risk, despite the greater initiating event frequency of the SGTR cases. At RT_{PTS} values above 290°F, SGTRs become the dominant contributor to PTS risk. A similar trend could be expected for the Donald C. Cook Nuclear Plant since the initiating event frequencies used in the generic PTS studies are comparable to or bounding when compared to those expected at the Cook plant.

Besides small LOCA and SGTR, the next most limiting event identified in Figure 14.3.7-1 is the loss of heat sink transient. This event is actually treated as a small hot leg LOCA since the operator would, based on the ERGs, initiate high pressure SI and open the pressurizer PORVs. This bleed and feed mode of recovery is used for the unlikely situation in which AFW is not available and other modes of secondary cooling (e.g., recovery of main feedwater) cannot be performed. The characteristic pressure associated with this scenario is conservatively assumed to be 2000 psig in WCAP-10319. With capability to open all three pressurizer PORVs at Donald C. Cook Units 1 and 2, the RCS pressure would be expected to be less than 2000 psig for this bleed and feed mode of recovery. Thus, application of the generic curve for this specific contribution is considered to be appropriate.

The other PTS transient scenarios, including those involving loop stagnation (such as secondary depressurization, anticipated transient without scram, and feedline break), do not contribute significantly to the total frequency. This supports the WOG position that the overall risk from PTS is dominated by small LOCA and SGTR events and is not affected by other candidate sequences, including severe cooldown transients that result in stagnant loop conditions. Note that small steamline breaks are no longer significant contributors to the total frequency of flaw extension for a typical Westinghouse PWR as previously suggested in an earlier WOG PTS risk study⁽²⁾.

Based on the assessment summarized above, results of the stagnant loop study are considered applicable and conservative for the Donald C. Cook Nuclear Plant. Emergency operating procedures based on the ERGs are available and the operators are trained to follow the procedures. This includes significant actions such as tripping the RCPs based on specified criteria, throttling AFW flow when called for, terminating SI flow when required, and only cooling the RCS within specified limits. Sensitivities to various operator action times and credible failures are also considered, as was noted above for the delayed SI termination SGTR cases.

Based on the results from the stagnant loop evaluation and limiting PTS and screening values as described below, it can be concluded that, as long as plant emergency operating procedures based on the ERGs are followed, stagnant loop transients do not significantly increase the PTS risk for a typical Westinghouse-designed plant such as the Donald C. Cook Nuclear Plant.

3. Limiting RT-PTS Values and Screening Criteria

In July 1985, the NRC published a new rule under 10 CFR 50.61 entitled "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS) Events." This new rule established screening limits for the calculated reference temperature for pressurized thermal shock (RT_{PTS}) as follows: a) 270°F for plates, forgings, and axial weld materials, and b) 300°F for circumferential weld materials. The RT_{PTS} must be calculated as per paragraph (b) (2) of 10 CFR 50.61.

In May 1991, the NRC issued a revision to 10 CFR 50.61. This revision incorporated the calculational methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Material," U.S. Nuclear Regulatory Commission, May 1988. In addition, this revision required licensees to submit projected values of RT_{PTS} for the reactor beltline materials for the time of the submittal and for the projected expiration date of the operating license. Per this requirement, the Cook Nuclear Plant submitted a plant-specific RT_{PTS} calculation for both

units to the NRC⁽⁶⁾. The controlling material and the calculated RT_{PTS} values at the end of the operating licenses are as follows:

Unit 1

Intermediate to lower shell
circumferential weld (9-442)
Calculated RT_{PTS} = 216°F
Screening Limit = 300°F

Unit 2

Intermediate shell plate (C5556-2)
Calculated RT_{PTS} = 217°F
Screening Limit = 270°F

The plant-specific RT_{PTS} calculation for Cook Unit 1 is based on the July 1985 of 10 CFR 50.61 and the plant-specific RT_{PTS} calculation for Cook Unit 2 is based on the May 1991 version of 10 CFR 50.61.

The NRC has approved the submittal and issued a Safety Evaluation report dated October 1, 1991⁽⁷⁾. The RT_{PTS} will be recalculated and submitted to the NRC whenever changes in core loadings, surveillance measurements, or other information indicates a significant change in projected values per the requirements of 10 CFR 50.61.

4. Emergency Response Guidelines and Their Application for Long Term Cooling

The generic Emergency Response Guidelines (ERGs) developed by the Westinghouse Owners Group contain appropriate steps to prevent or mitigate the effects of PTS events. These generic guidelines were developed as part of the program for implementation of item I.C.1 of NUREG-0737 and were submitted to the NRC⁽⁸⁾. As the ERGs were developed, a PTS review of the ERGs was performed to ensure that the actions taken were appropriate⁽⁹⁾. Revision 1 of the ERGs included the results of this PTS review and this version of the ERGs has been transmitted to the NRC⁽¹⁰⁾. As reported above, both Donald C. Cook units have limiting RT_{PTS} projected values that are greater than 200°F but less than 250°F. With these results, both units are considered to be in Category II with respect to PTS mitigation actions based on the ERGs⁽¹¹⁾. The Donald C. Cook Units 1 and 2 emergency operating procedures are based on the WOG ERGs.

For initial cooldown following either of the dominant accident scenarios, the operator is instructed to use the intact steam generators (S/Gs) to remove plant decay and stored heat. This is done by feeding the S/Gs with auxiliary feedwater to maintain an indicated S/G water level within the narrow-range level instrument span, and relieving S/G pressure by means of the steam dump valves (if offsite power is available or can be quickly restored). The main steam system atmospheric safety or relief valves can be used if the steam dump system is not available. Long term cooling continues using the intact S/Gs, auxiliary feedwater and the atmospheric safety or relief valves.

In order to assure effective long-term cooling for a LOCA certain additional operator actions are assumed. These actions are principally (1) to switch the ECCS from the injection phase to the recirculation phase, (2) to place the reactor coolant pumps in a condition where they can most effectively aid core cooling, and (3) to switch the ECCS from cold leg recirculation to hot leg recirculation at the appropriate time to prevent boron precipitation. All of these items and other appropriate actions that need to be taken to ensure PTS risk is minimized are specified in the plant emergency operating procedures.

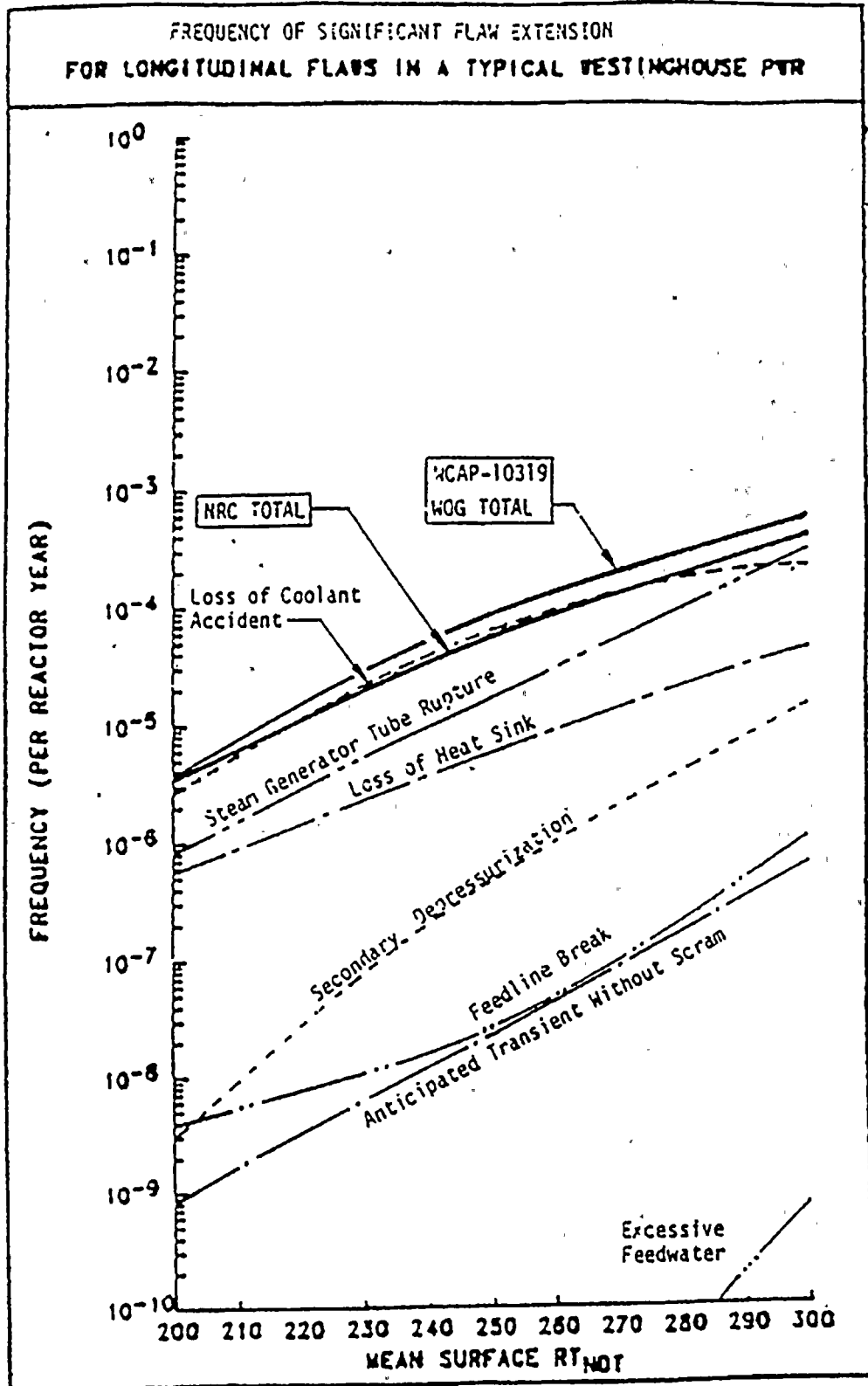
Control of long term cooling for these and other accident scenarios requires specific actions and responses unique to the type of accident that has occurred. The plant emergency operating procedures provide the operator instructions for controlling long term cooling and minimizing PTS risk during anticipated accidents. The emergency operating procedures cover the range of activities required to take the plant to a safe shutdown condition, identify the parameters that must be monitored, the equipment that must be available, and establish the sequence for performing the required activities. Procedural compliance ensures that activities are completed within time frames required to prevent challenging PTS risk concerns and minimize operator errors. The complete set of emergency operating procedures also account for other unusual conditions such as loss of offsite power, some multiple failures (such as LOCA plus SGTR) and equipment failures/unavailability.

References

1. Letter No. OG-58 from Mr. R. W. Jurgensen, Chairman, WOG, to Mr. D. G. Eisenhower, Director, NRC, entitled, "Thermal Shock to Reactor Pressure Vessel," dated May 14, 1981. (Attachment "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants" (WCAP-10019)).
2. Letter No. OG-70 from Mr. O. D. Kingsley, Chairman, WOG, to Mr. Harold R. Denton, Director, NRC, entitled, "Supplemental Information on Reactor Vessel Integrity," dated May 18, 1982. (Attachment "Summary of Evaluations Related to Reactor Vessel Integrity" (WCAP-10019-S1)).
3. Letter from Mr. T. M. Novak, NRC, Assistant Director for Operating Reactors, to Mr. O. D. Kingsley, Chairman, WOG, dated March 16, 1982.
4. NRC Policy Issue, Enclosure A, "NRC Staff Evaluation of Pressurized Thermal Shock," SECY-82-465, November 23, 1982.
5. Westinghouse Electric Corporation, "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, From Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants, WCAP-10319, December 1983.
6. Letter No. AEP:NRC:0561D, "Donald C. Cook Nuclear Plant, Units 1 and 2, Updated Reference Temperature, Pressurized Thermal Shock Analyses," dated August 7, 1990, from Mr. P. Alexich to T. E. Murley.
7. NRC SER, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 157 to Facility Operating License No. DPR-58, and Amendment No. 141 to Facility Operating License No. DPR-74, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Units Nos. 1 and 2, Docket Nos. 50-315 and 50-316," dated October 1, 1991.
8. Letter No. OG-64 from Mr. R. W. Jurgensen, Chairman, WOG, to Mr. D. G. Eisenhower, Director, Division of Licensing, entitled, "Emergency Response Guideline Program," dated November 30, 1981.
9. Letter No. OG-72 from Mr. O. D. Kingsley, Chairman, WOG, to Mr. Harold R. Denton, Director, NRC, entitled, "PTS Review of the ERGs," dated June 22, 1982.
10. Letter No. OG-111 from Mr. J. J. Sheppard, Chairman, WOG, to Mr. H. L. Thompson, Director, Division of Human Factors Safety, entitled "Transmittal of Revision 1 of Emergency Response Guidelines Revision 1," dated November 30, 1983.

11. Emergency Response Guidelines - Revision 1B (High-Pressure Version),
Westinghouse Owners Group, February 28, 1992.

Figure 14.3.7-1



UNIT 2

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14.3.8 NITROGEN BLANKETING

This section was a portion of the response to question 212.34 of Appendix Q of the original FSAR which addressed the issue of nitrogen blanketing from the accumulators in SBLOCA.

Subsequent to 1977, extensive analyses have been performed to study the general behavior of SBLOCA. Credible sources of non-condensables in SBLOCA have been considered in both references 1 and 2. Accumulator nitrogen was not identified as a potential source of non-condensables. Credible non-condensables were specifically addressed in the development of the NOTRUMP code which is currently the analysis of record for both units.

References:

1. WCAP 9600, Report on Small Break Accidents for Westinghouse NSSS System, Approved T. M. Anderson, June 1979.
2. WCAP 10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, N. Lee, S. D. Rupprecht, W. R. Schwarz, W. D. Tauche, August 1985.
3. Letter, Steven A. Varga, NRC Staff to John Dolan, May 23, 1985.

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TABLE 1.2-1

COMPARISON OF DESIGN PARAMETERS **

REFERENCE LINE NO.	EDWARD C. COOK MILLER PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BLANCH UNITS 1 AND 2 FINAL REPORT	H. M. ROBINSON #2 FINAL REPORT
THERMAL AND HYDRAULIC DESIGN PARAMETERS					
1	Total Primary Heat Output, MWt	2250	2250	2258	2200
2	Total Core Heat Output, Btu/hr	$11,090 \times 10^6$	$11,090 \times 10^6$	9413×10^6	7470×10^6
3	Heat Generated in Fuel, %	97.4	97.4	97.4	97.4
4	Maximum Thermal Overpower	122	122	122	122
5	System Pressure, Nominal, psia	2250	2250	2250	2250
6	System Pressure, Minimum Steady State, psia	2220	2220	2220	2220
7	Hot Channel Factor				
8	Heat Flux, q_h	1.79	1.79	1.23	3.23
8	Enthalpy Rise, Δh	1.60	1.60	1.77	1.77
9	DNB Ratio at Nominal Operating Conditions	1.97	2.02	2.00	2.11
10	Minimum DNB for Design Transients	1.10	1.30	1.30	1.30
Coolant Flow					
11	Total Flow Rate, lb/hr	175.4×10^6	175.0×10^6	116.3×10^6	101.5×10^6
12	Effective Flow Rate for Heat Transfer, lb/hr	129.5×10^6	128.9×10^6	130×10^6	97.0×10^6
13	Effective Flow Area for Heat Transfer, ft ²	51.4×10^3	51.4×10^3	51.4×10^3	51.4×10^3
14	Average Velocity Along Fuel Rods, ft/sec	12.5	12.3	15.4	14.3
15	Average Mass Velocity, lb/hr-ft ²	2.53×10^6	2.52×10^6	2.53×10^6	2.32×10^6
Coolant Temperature, °F					
16	Design Nominal Inlet	516.1	530.2	543	552.5
	Maximum Inlet Due to Instrumentation Error and Deadband, °F	540.3	576.2	567	556.5
17	Average Rise in Vessel, °F	63.0	44.1	53.0	57.6
18	Average Rise in Core	65.7	66.8	55.5	60.0
19	Average in Core	570.1	566.8	571.0	582.5
20	Average in Vessel	567.8	563.2	569.5	581.3
21	Nominal Outlet of Hot Channel	662.5	631.3	633.5	662
22	Average Film Coefficient, Btu/hr-ft ² -°F	5850	5800	5790	5600

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** This table is retained for historical purposes only. It compares original Cook Plant parameters to other similar nuclear plants.

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	M. B. ROBINSON #2 FINAL REPORT
24	Average Film Temperature Difference, °F	35.4	35.6	30.3	31.0	31.0
	Heat Transfer at 100% Power					
25	Active Heat Transfer Surface Area, ft ²	32,200	32,200	32,200	28,715	42,440
26	Average Heat Flux, Btu/hr-ft ²	207,900	207,900	175,600	175,600	171,600
27	Maximum Heat Flux, Btu/hr-ft ²	379,600	379,600	567,300	491,000	334,200
28	Average Thermal Output, kw/ft	6.7	6.7	5.7	5.7	5.3
29	Maximum Thermal Output, kw/ft	18.8	18.8	18.4	16.0	17.0
30	Maximum Clad Surface Temperature at Nominal Pressure, °F	657	657	657	657	657
	Fuel Central Temperature, °F					
31	Maximum at 100% Power	4250	4250	4090	3750	4030
32	Maximum at Overpower	4500	4500	4380	4000	4300
33	Thermal Output, kw/ft at Maximum Overpower	21.1	21.1	20.6	17.9	20.0
	COKE MECHANICAL DESIGN PARAMETERS					
	Fuel Assembly					
34	Design	BCC Canless 15x15	BCC Canless 15x15	BCC Canless 15x15	BCC Canless 14x14	BCC Canless 15x15
35	Rod Pitch, in.	0.563	0.563	0.563	0.556	0.563
36	Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426	7.763 x 7.763	8.426 x 8.426
37	Fuel Weight (as UO ₂), pounds	216,600	216,600	216,000	120,130	176,200
38	Total Weight, pounds	276,000	276,000	276,000	154,510	226,200
39	Number of Grids per Assembly	7	7	9	7	7
	Fuel Rods					
40	Number	19,372	19,372	19,372	21,659	32,028
41	Outside Diameter, in.	0.422	0.422	0.422	0.422	0.422
42	Diametral Gap, in. (Region 1, 2) (Region 3)	0.0075 0.0085	0.0075 0.0085	0.0065	0.0065	0.0065
43	Clad Thickness, in.	0.0263	0.0263	0.0263	0.0263	0.0263
44	Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
	Fuel Pellets					
45	Material	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered	UO ₂ Sintered
46	Density (% of Theoretical)	94.93-92	94.93-92	94.92-91	94.92-91	94.92-91
47	Diameter, in. (Region 1, 2) (Region 3)	0.1659 0.1649	0.1659 0.1649	0.1669	0.1669	0.1669
48	Length, in.	0.6000	0.6000	0.6000	0.6000	0.6000
	Rod Cluster Control Assemblies					
49	Neutron Absorber Cladding Material	3% Cd-15% In-80% Ag Type 304 SS-Cold Worked	3% Cd-15% In-80% Ag Type 304 SS-Cold Worked	3% Cd-15% In-80% Ag Type 304 SS-Cold Worked	3% Cd-15% In-80% Ag Type 304 SS-Cold Worked	3% Cd-15% In-80% Ag Type 304 SS-Cold Worked
50	Clad Thickness, in.	0.019	0.019	0.019	0.019	0.019
51	Number of Clusters	53	53	53	37	53
52	Number of Control Rods per Cluster		20	20	14	

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		RONALD C. GINK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	M. B. ROBINSON #2 FINAL REPORT
	Core Structure					
54	Core Barrel I.D./O.D., in.	148.0/152.5	148.0/152.5	148.0/152.5	109.0/112.5	133.075/137.075
55	Thermal Shield I.D./O.D., in.	158.5/166.0	158.5/166.0	158.5/166.0	115.3/122.5	
	FINAL NUCLEAR DESIGN DATA					
	Structural Characteristics					
56	Fuel Weight (As UO_2), lbs.	216,600	216,600	216,000	120,130	176,260
57	Clad Weight, lbs.	44,547	44,547	44,600	24,260	36,300
58	Core Diameter, in. (Equivalent)	132.7	132.7	132.5	96.5	119.5
59	Core Height, in. (Active Fuel)	144	143.4	144	144	144
	Reflector Thickness and Composition					
60	Top - Water plus Steel, in.	10	10	10	10	10
61	Bottom - Water plus Steel, in.	10	10	10	10	10
62	Side - Water plus Steel, in.	15	15	15	15	15
63	H ₂ O/U ₂ (Cold Volume Ratio)	4.09	4.09	4.18	4.20	4.18
64	Number of Fuel Assemblies	193	193	193	121	157
65	UO_2 Rods per Assembly	206	206	206	179	204
	Performance Characteristics					
66	Loading Technique	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform
	Fuel Discharge Burnup, MWD/MTU					
67	Average First Cycle	14,000	14,000	14,700	15,100	14,500
68	Equilibrium Core Average	21,800	21,800	24,700	33,000	33,000
	Feed Enrichments, weight %					
69	Region 1	2.25	2.25	2.2	2.27	1.85
70	Region 2	2.80	2.80	2.7	3.03	2.55
71	Region 3	3.30	3.30	3.2	3.40	3.10
	Equilibrium	3.2	3.2	-	3.40	3.10
	Control Characteristics					
	Effective Multiplication (Beginning of Life)					
72	Cold, No Power, Clean	1.183	1.183	1.257	1.211	1.180
73	Hot, No Power, Clean	1.154	1.154	1.999	1.167	1.38
74	Hot, Full Power, Xe and Sm Equilibrium	1.092	1.092	1.152	1.113	1.077

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.	DAVID C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	H. B. ROBINSON #2 FINAL REPORT
	Mod Cluster Control Assemblies	5X C4-15X 1a-80X Ag	5X C4-15X 1a-80X Ag	5X C4-15X 1a-80X Ag	5X C4-15X 1a-80X Ag
75	Material	53	53	53	53
76	Number of MCC Assemblies	20	20	16	20
77	Number of Absorber Rods per MCC Assembly	See Table	See Table	See Table	See Table
78	Total Rod Worth	3.2.1-3	3.2.1-3	3.2.1-3	3.2.1-3
	Boron Concentrations				
	To shut reactor down with no rods inserted, clean (k eff = .99)				
79	Cold/hot, ppm/ppm	1408/1265	1408/1265	1460/1370	1598/1676
	To control at power with no rods inserted, clean/equilibrium xenon and samarium, ppm/ppm	1168/850	1168/850	1200/780	1465/1007
80	Boron worth, Hot	12.4 k/k / 85 ppm	12.4 k/k / 85 ppm	12.4 k/k / 89 ppm	12.4 k/k / 130 ppm
81	Boron worth, Cold	12.4 k/k / 70 ppm	12.4 k/k / 70 ppm	12.4 k/k / 72 ppm	12.4 k/k / 98 ppm
82					5.6 k/k
	Kinetic Characteristics				
83	Moderator Temperature Coefficient, k/k/°F	-0.3×10^{-4} -3.2×10^{-4} to	-0.3×10^{-4} -3.2×10^{-4} to	-0.3×10^{-4} -3.0×10^{-4} to	-0.3×10^{-4} -3.5×10^{-4} to
84	Moderator Pressure Coefficient, k/k/psi	10.1×10^{-6} $+4.0 \times 10^{-6}$ to	10.3×10^{-6} $+4.0 \times 10^{-6}$ to	-0.3×10^{-6} $+3.0 \times 10^{-6}$ to	-0.3×10^{-6} 3.5×10^{-6} to
85	Moderator Density Coefficient, k/k/g/cm ³	-0.1×10^{-3} -0.8×10^{-3} to	-0.1×10^{-3} -0.8×10^{-3} to	$+0.03$ to -0.30	-0.10 to -0.30
86	Doppler Coefficient, k/k/°F	-1.0×10^{-3} -1.7×10^{-3} to	-1.0×10^{-3} -1.7×10^{-3} to	-1.1×10^{-3} -1.8×10^{-3} to	-1×10^{-3} -1.6×10^{-3} to
	REACTOR COOLANT SYSTEM - CODE REQUIREMENTS				
	Component				
87	Reactor Vessel	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		DONALD C. CIRM NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	M. B. ROBINSON #3 FINAL REPORT
88	Steam Generator Tube Side	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
89	Shell Side	ASME III Class C ^a	ASME III Class C ^a	ASME III Class C ^a	ASME III Class C ^a	ASME III Class C ^a
90	Pressurizer	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
91	Pressurizer Heated Tank	ASME III Class C	ASME III Class C	ASME III Class C	ASME Class C	ASME Class C
92	Pressurizer Safety Valves	ASME III	ASME III	ASME III	ASME III	
93	Reactor Coolant Piping	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM						
94	Reactor Primary Heat Output, MW	3250	3250	2758	1518.5	2200
95	Reactor Primary Heat Output, Btu/hr	11,090 × 10 ⁶	11,090 × 10 ⁶	9413 × 10 ⁶	5181 × 10 ⁶	7508 × 10 ⁶
96	Operating Pressure, psig	2235	2235	2235	2235	2235
97	Reactor Inlet Temperature	536.3	530.2	563	552.5	546.2
98	Reactor Outlet Temperature	599.3	594.3	596.0	610.1	602.1
99	Number of Loops	4	4	4	2	3
100	Design Pressure, psig	2485	2485	2485	2485	
101	Design Temperature, °F	650	650	650	650	650
102	Hydrostatic Test Pressure (Cold), psig	3107	3107	3110	3110	3110
103	Coolant Volume, including pressurizer, cu. ft.	12,612	12,710	12,600	6450	9088
104	Total Reactor Flow, gpm	150,000	150,000	178,000	268,500	
104A	Total Reactor Flow lb/sec	17,765	17,765			
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL						
105	Material	Same as others See Table 4.2-1	Same as others See Table 4.2-1	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with austenitic stainless steel
106	Design Pressure, psig	2485	2485	2485	2485	2485
107	Design Temperature, °F	650	650	650	650	650
108	Operating Pressure, psig	2235	2235	2235	2235	2235
109	Inside Diameter of Shell, in.	173	173	173	132.0	155.5
110	Outside Diameter Across Nozzles, in.	262-7/16	262-7/16	262-7/16	264 1/16	236
111	Overall Height of Vessel & Enclosure Head, ft-in.	43-9 11/16	43-9 23/32 (Unit 1) 43-9 15/16 (Unit 2)	43 9-11/16	39-0	41-6
112	Minimum Clad Thickness, in.	5/32	5/32	5/32	5/32	5/32

^a The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

TABLE 1.2-1 (Continued)

REFERENCE LINE NO.		WALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 FINAL REPORT	ZION STATION UNITS 1 AND 2 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	POINT BEACH UNITS 1 AND 2 FINAL REPORT	M. B. BURNHAM #2 FINAL REPORT
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS						
113	Number of Units	4	4	4	2	3
114	Type	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator	Vertical U-Tube with integral- moisture separator
115	Tube Material	Inconel	Inconel	Inconel	Inconel	Inconel
116	Shell Material	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel
117	Tube Side Design Pressure, psig	2485	2485	2485	2485	2485
118	Tube Side Design Temperature, °F	650	650	650	650	650
119	Tube Side Design Flow, lb/hr	33.9 x 10 ⁶	33.8 x 10 ⁶	36.1 x 10 ⁶	33.4 x 10 ⁶	33.9 x 10 ⁶
120	Shell Side Design Pressure, psig	1085	1085	1085	1085	1085
121	Shell Side Design Temperature, °F	600	600	556	556	556
122	Operating Pressure, Tube Side, Nominal, psig	2235	2235	2235	2235	2235
123	Operating Pressure, Shell Side, Maximum, psig	1085 (design)	1085 (design)	1105.3	1070	1070
124	Maximum Moisture at Outlet at Full Load, %	1/4	1/4	1/4	1/4	1/4
125	Hydrostatic Test Pressure, Tube Side (cold), psig	3107	3107	3110	3110	3110
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS						
126	Number of Units	4	4	4	4	4
127	Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge
128	Design Pressure, psig	2485	2485	2485	2485	2485
129	Design Temperature, °F	650	650	650	650	650
130	Operating Pressure, Nominal, psig	2235	2235	2235	2235	2235
131	Suction Temperature, °F	539	539	556	551.3	546.3
132	Design Capacity, gpm	88,500	87,500	80,000	80,000	88,500
133	Design Head, ft.	277	277	252	259	261
134	Hydrostatic Test Pressure (cold), psig	3107	3107	3110	3110	3110
135	Motor Type	AC Induction single speed	AC Induction single speed air cooled	AC Induction single speed	AC Induction single speed air cooled	AC Induction single speed air cooled
136	Motor Rating (nameplate)	6000 HP	6000 HP	6000 HP	6000 HP	6000 HP
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING						
137	Material	See Table 4.2-1	See Table 4.2-1	Austenitic SS	Austenitic SS	Austenitic SS
138	Hot Leg - I.D., in.	29	29	29	29	29
139	Cold Leg - I.D., in.	27-1/2	27-1/2	27-1/2	27-1/2	27-1/2
140	Between Pump and Steam Generator - I.D., in.	31	31	31	31	31
141	Design Pressure, psig	2485	2485	2485	2485	2485