Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
141.00000	35.37	251.63	234.04
142.00000	35.35	251.58	234.04
143.00000	35.32	251.53	234.04
144.00000	35.29	251.48	234.04
145.00000	35.26	251.43	234.04
146.00000	35.24	251.38	234.04
147.00000	35.21	251.33	234.04
148.00000	35.18	251.28	234.03
149.00000	35.16	251.24	234.03
150.00000	35.13	251.19	234.03
151.00000	35.10	251.14	234.02
152.00000	35.08	251.09	234.02
153.00000	35.05	251.05	234.01
154.00000	35.03	251.00	234.01
155.00000	35.00	250.95	234.00
156.00000	34.98	250.91	233.99
157.00000	34.95	250.86	233.99
158.00000	34.93	250.82	233.98
159.00000	34.90	250.77	233.97
160.00000	34.88	250.72	233.96
161.00000	34.85	250.68	233.96
162.00000	34.83	250.63	233.95
163.00000	34.80	250.59	233.94
164.00000	34.78	250.54	233.93
165.00000	34.75	250.50	233.92
166.00000	34.73	250.45	233.91
167.00000	34.70	250.41	233.90
168.00000	34.68	250.37	233.89
169.00000	34.66	250.32	233.87
170.00000	34.63	250.28	233.86
171.00000	34.61	250.24	233.85
172.00000	34.58	250.19	233.84
173.00000	34.56	250.15	233.82
174.00000	34.54	250.11	233.81
175.00000	34.51	250.06	233.80
176.00000	34.49	250.02	233.78

Table 6.4.3-14 (cont.)
Double-Ended Pump Suction Break
Byron Unit 1 & Braidwood Unit 1
(BWI SG) Maximum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
177.00000	34.47	249.98	233.77
178.00000	34.44	249.93	233.75
179.00000	34.42	249.89	233.74
180.00000	34.40	249.85	233.72
181.00000	34.38	249.81	233.71
182.00000	34.35	249.77	233.69
183.00000	34.33	249.72	233.67
184.00000	34.31	249.68	233.66
185.00000	34.29	249.64	233.64
186.00000	34.26	249.60	233.62
187.00000	34.24	249.56	233.60
188.00000	34.22	249.52	233.59
189.00000	34.20	249.48	233.57
190.00000	34.17	249.44	233.55
191.00000	34.15	249.40	233.53
192.00000	34.13	249.35	233.51
193.00000	34.11	249.31	233.49
194.00000	34.09	249.27	233.47
195.00000	34.07	249.23	233.45
196.00000	34.04	249.19	233.43
197.00000	34.02	249.15	233.41
198.00000	34.00	249.11	233.39
199.00000	33.99	249.08	233.33
209.00000	33.82	248.78	232.81
219.00000	33.67	248.48	232.32
229.00000	33.51	248.18	231.62
239.00000	33.36	247.91	231.10
249.00000	33.23	247.65	230.60
259.00000	33.10	247.39	230.12
269.00000	32.97	247.15	229.66
279.00000	32.85	246.92	229.22
289.00000	32.73	246.69	228.78
299.00000	32.62	246.47	228.37
309.00000	32.51	246.26	227.96
319.00000	32.41	246.05	227.57
329.00000	32.31	245.86	227.18

Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
339.00000	32.21	245.66	226.81
349.00000	32.11	245.47	226.44
359.00000	32.02	245.29	226.09
369.00000	31.93	245.11	225.75
379.00000	31.85	244.94	225.41
389.00000	31.76	244.77	225.08
399.00000	31.68	244.61	224.77
409.00000	31.60	244.44	224.45
419.00000	31.53	244.29	224.15
429.00000	31.45	244.13	223.84
439.00000	31.38	243.98	223.55
449.00000	31.30	243.83	223.26
459.00000	31.23	243.69	222.98
469.00000	31.17	243.55	222.70
479.00000	31.10	243.41	222.44
489.00000	31.03	243.28	222.17
499.00000	30.97	243.14	221.91
599.00000	30.40	241.94	219.40
699.00000	29.94	240.94	217.64
799.00000	28.76	238.46	221.56
899.00000	27.23	235.10	224.86
999.00000	25.90	232.04	227.61
1099.0000	24.77	229.32	229.72
1199.0000	23.75	226.77	231.50
1299.0000	22.82	224.37	233.03
1399.0000	21.98	222.12	234.33
1499.0000	21.21	220.00	235.44
1599.0000	20.51	218.02	236.39
1699.0000	19.56	215.22	234.20
1799.0000	18.71	212.61	232.27
1899.0000	17.94	210.14	230.48
1999.0000	17.23	207.80	228.82
2099.0000	16.58	205.57	227.27
2199.0000	15.99	203.46	225.83
2299.0000	15.44	201.45	224.49
2399.0000	14.93	199.55	223.23

Table 6.4.3-14 (cont.) Double-Ended Pump Suction Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
2499.0000	14.47	197.74	222.06
2599.0000	14.04	196.03	220.95
2699.0000	13.65	194.40	219.91
2799.0000	13.28	192.86	218.94
2899.0000	12.94	191.40	218.02
2999.0000	12.63	190.01	217.15
3099.0000	12.34	188.69	216.33
3199.0000	12.06	187.42	215.55
3299.0000	11.80	186.21	214.82
3399.0000	11.76	185.99	214.11
3499.0000	12.15	187.78	213.52
3599.0000	12.43	189.02	212.98
3699.0000	12.46	189.17	211.89
3799.0000	12.47	189.20	210.82
3899.0000	12.46	189.18	209.82
3999.0000	12.45	189.13	208.87
4999.0000	12.05	187.31	201.68
5999.0000	11.44	184.46	197.69
6999.0000	10.86	181.61	195.30
7999.0000	10.08	177.57	183.27
8999.0000	9.405	173.80	176.26
9999.0000	8.890	170.80	172.14
19999.000	7.193	159.73	150.84
29999.000	6.973	157.92	150.37
39999.000	6.768	156.12	149.96
49999.000	6.544	154.21	149.52
59999.000	6.347	152.33	149.09
69999.000	6.119	150.31	148.64
79999.000	5.891	148.30	148.16
89999.000	5.688	146.07	147.71
99999.000	5.475	144.01	147.24
199999.00	6.248	138.66	139.74
299999.00	6.315	137.19	139.38
399999.00	6.072	135.97	139.08
499999.00	5.803	134.35	138.76
599999.00	5.505	132.79	138.41

Table 6.4.3-14 (cont.)
Double-Ended Pump Suction Break
Byron Unit 1 & Braidwood Unit 1
(BWI SG) Maximum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
699999.00	5.220	131.45	138.09
799999.00	4.910	130.10	137.80
899999.00	4.647	128.57	137.48
999999.00	4.379	127.15	137.17
1000000.0	4.380	127.15	137.17

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Table 6.4.3-15 Double-Ended Hot Leg Break Byron Unit 1 & Braidwood Unit 1 (BWI SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
.0010	1.000	120.00	120.00
.5000	3.809	144.13	176.43
1.000	6.115	162.56	193.18
2.000	10.29	189.64	209.23
3.000	13.96	207.14	218.20
4.000	17.14	218.14	224.43
5.000	19.94	225.00	229.26
6.000	22.47	229.40	233.28
7.000	24.65	231.57	236.36
8.000	26.74	235.43	238.93
9.000	28.93	240.15	241.59
10.00	31.45	245.23	244.61
11.00	33.73	249.53	247.23
12.00	35.60	252.89	249.28
13.00	36.86	255.06	250.46
14.00	38.01	257.00	251.32
15.00	39.16	258.88	252.10
16.00	40.19	260.53	252.65
17.00	41.02	261.83	253.01
18.00	41.68	262.85	253.15
19.00	42.18	263.61	253.15
20.00	42.50	264.10	253.13
21.00	42.69	264.38	253.12
22.00	42.77	264.50	253.11
23.00	42.73	264.45	253.08
24.00	42.62	264.28	253.06
25.00	42.47	264.05	253.05
26.00	42.31	263.81	253.05

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Table 6.4.3-16
LOCA Containment Response Results For Byron Unit 2 and Braidwood Unit 2 (D5 SG)
Loss of Offsite Power Assumed

Case	Peak Press. (psig)	Peak Steam Temp. (°F)	Pressure (psig) @ 24 hours	Steam Temperature (°F) @ 24 hours
DEPS MINSI	37.71 @ 399 sec	255.65 @ 399 sec	8.68 @ 24 hrs	170.35 @ 24 hrs
DEPS MAXSI	36.77 @ 21.01 sec	254.89 @ 21.005 sec	7.314 @ 24 hrs	159.4 @ 24 hrs
DEHL MINSI	38.36 @ 21.079 sec	257.57 @ 21.079 sec	NA	NA

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Table 6.4.3-17
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
.001	1.000	120.0	120.0
.500	3.619	142.48	183.61
1.0	6.090	162.40	198.75
2.0	10.45	190.74	212.21
3.0	14.14	208.19	219.55
4.0	16.81	216.33	224.07
5.0	18.89	219.88	227.42
6.0	20.83	222.09	230.43
7.0	22.75	225.91	233.16
8.0	24.61	230.52	235.50
9.0	26.32	234.50	237.39
10.0	27.90	237.98	239.20
11.0	29.34	241.0	240.88
12.0	30.63	243.62	242.37
13.0	31.81	245.92	243.68
14.0	32.89	247.98	244.84
15.0	33.87	249.79	245.86
16.0	34.75	251.38	246.76
17.0	35.58	252.84	247.48
18.0	36.24	253.99	248.0
19.0	36.63	254.66	248.37
20.0	36.81	254.98	248.65
21.0	36.88	255.09	248.94
22.0	36.85	255.04	249.10
23.0	36.73	254.84	249.28
24.0	36.55	254.53	249.81
25.0	36.33	254.15	249.84
26.0	36.12	253.78	249.81
27.0	35.93	253.44	249.77
28.0	35.75	253.13	249.74
29.0	35.58	252.85	249.70
30.0	35.43	252.58	249.67

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Table 6.4.3-17 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
31.0	35.29	252.33	249.63
32.0	35.17	252.11	249.57
33.0	35.06	251.93	249.16
34.0	34.99	251.79	248.62
35.0	34.92	251.67	248.11
36.0	34.86	251.56	247.63
37.0	34.80	251.45	247.19
38.0	34.74	251.35	246.77
39.0	34.69	251.26	246.37
40.0	34.64	251.17	246.0
41.0	34.60	251.08	245.65
42.0	34.55	251.01	245.33
43.0	34.51	250.93	245.02
44.0	34.48	250.86	244.73
45.0	34.44	250.81	244.37
46.0	34.42	250.76	243.98
47.0	34.39	250.71	243.61
48.0	34.37	250.67	243.26
49.0	34.35	250.63	242.92
50.0	34.33	250.60	242.60
51.0	34.31	250.56	242.30
52.0	34.30	250.53	242.0
53.0	34.28	250.51	241.73
54.0	34.27	250.48	241.46
55.0	34.26	250.46	241.20
56.0	34.24	250.44	240.96
57.0	34.23	250.42	240.73
58.0	34.23	250.40	240.50
59.0	34.22	250.38	240.29
60.0	34.21	250.37	240.09
61.0	34.20	250.36	239.89
62.0	34.20	250.34	239.70
63.0	34.19	250.33	239.53
64.0	34.19	250.32	239.36
65.0	34.18	250.32	239.19
66.0	34.17	250.29	239.04

Table 6.4.3-17 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Minimum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
67.0	34.15	250.26	238.89
68.0	34.14	250.24	238.75
69.0	34.13	250.21	238.62
70.0	34.12	250.20	238.63
71.0	34.12	250.21	238.68
72.0	34.13	250.23	238.71
73.0	34.15	250.25	238.74
74.0	34.21	250.34	238.75
75.0	34.29	250.44	238.75
76.0	34.37	250.55	238.76
77.0	34.44	250.64	238.76
78.0	34.51	250.73	238.77
79.0	34.58	250.82	238.78
80.0	34.65	250.90	238.78
81.0	34.71	250.97	238.79
82.0	34.77	251.05	238.79
83.0	34.83	251.11	238.80
84.0	34.89	251.18	238.81
85.0	34.95	251.24	238.82
86.0	35.0	251.29	238.82
87.0	35.05	251.35	238.83
88.0	35.10	251.40	238.84
89.0	35.14	251.44	238.86
90.0	35.18	251.46	238.88
91.0	35.21	251.49	238.90
92.0	35.25	251.51	238.92
93.0	35.28	251.53	238.94
94.0	35.30	251.54	238.97
95.0	35.30	251.55	238.99
96.0	35.31	251.56	239.01
97.0	35.31	251.57	239.03
98.0	35.31	251.57	239.06
99.0	35.32	251.58	239.08
100.0	35.32	251.58	239.10
101.0	35.32	251.58	239.13
102.0	35.32	251.58	239.15

Table 6.4.3-17 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Minimum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
103.0	35.32	251.58	239.17
104.0	35.32	251.58	239.19
105.0	35.32	251.58	239.21
106.0	35.32	251.57	239.23
107.0	35.31	251.57	239.25
108.0	35.31	251.57	239.28
109.0	35.31	251.56	239.30
110.0	35.31	251.56	239.32
111.0	35.30	251.55	239.34
112.0	35.30	251.54	239.36
113.0	35.30	251.54	239.39
114.0	35.29	251.53	239.41
115.0	35.29	251.52	239.43
116.0	35.28	251.51	239.45
117.0	35.28	251.50	239.47
118.0	35.27	251.49	239.49
119.0	35.27	251.49	239.51
120.0	35.26	251.48	239.53
121.0	35.26	251.47	239.56
122.0	35.25	251.46	239.58
123.0	35.25	251.45	239.60
124.0	35.24	251.44	239.62
125.0	35.24	251.43	239.64
126.0	35.23	251.42	239.66
127.0	35.23	251.41	239.68
128.0	35.22	251.40	239.70
129.0	35.22	251.39	239.72
130.0	35.21	251.38	239.74
131.0	35.20	251.37	239.76
132.0	35.20	251.36	239.78
133.0	35.19	251.35	239.80
134.0	35.19	251.34	239.82
135.0	35.18	251.33	239.84
136.0	35.18	251.32	239.86
137.0	35.17	251.32	239.88
138.0	35.17	251.31	239.90

Table 6.4.3-17 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
139.0	35.17	251.30	239.92
140.0	35.16	251.29	239.94
141.0	35.16	251.28	239.96
142.0	35.15	251.27	239.98
143.0	35.15	251.27	240.0
144.0	35.14	251.26	240.02
145.0	35.14	251.25	240.04
146.0	35.14	251.25	240.06
147.0	35.13	251.24	240.08
148.0	35.13	251.23	240.10
149.0	35.13	251.23	240.11
150.0	35.12	251.22	240.13
151.0	35.12	251.22	240.15
152.0	35.12	251.21	240.17
153.0	35.11	251.21	240.19
154.0	35.11	251.20	240.21
155.0	35.11	251.20	240.23
156.0	35.11	251.19	240.24
157.0	35.11	251.19	240.26
158.0	35.10	251.18	240.28
159.0	35.10	251.18	240.30
160.0	35.10	251.18	240.32
161.0	35.10	251.17	240.34
162.0	35.10	251.17	240.35
163.0	35.10	251.17	240.37
164.0	35.09	251.17	240.39
165.0	35.09	251.17	240.41
166.0	35.09	251.16	240.43
167.0	35.09	251.16	240.45
168.0	35.09	251.16	240.46
169.0	35.09	251.16	240.48
170.0	35.09	251.16	240.50
171.0	35.09	251.16	240.52
172.0	35.09	251.16	240.54
173.0	35.09	251.16	240.55
174.0	35.09	251.16	240.57

Table 6.4.3-17 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
175.0	35.09	251.16	240.59
176.0	35.09	251.16	240.61
177.0	35.10	251.17	240.62
178.0	35.10	251.17	240.64
179.0	35.10	251.17	240.66
180.0	35.10	251.17	240.68
181.0	35.10	251.17	240.69
182.0	35.10	251.18	240.71
183.0	35.10	251.18	240.73
184.0	35.11	251.18	240.74
185.0	35.11	251.19	240.76
186.0	35.11	251.19	240.78
187.0	35.11	251.20	240.79
188.0	35.12	251.20	240.81
189.0	35.12	251.21	240.83
190.0	35.12	251.22	240.85
191.0	35.13	251.22	240.86
192.0	35.13	251.23	240.88
193.0	35.14	251.24	240.90
194.0	35.15	251.25	240.92
195.0	35.16	251.27	240.95
196.0	35.17	251.29	240.98
197.0	35.18	251.31	241.01
198.0	35.19	251.34	241.04
199.0	35.20	251.36	241.07
209.0	35.33	251.57	241.35
219.0	35.45	251.79	241.64
229.0	35.58	252.01	241.93
239.0	35.70	252.24	242.21
249.0	35.83	252.46	242.49
259.0	35.96	252.69	242.77
269.0	36.09	252.91	243.04
279.0	36.22	253.14	243.31
289.0	36.35	253.36	243.57
299.0	36.48	253.58	243.83
309.0	36.61	253.80	244.09

Table 6.4.3-17 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
319.0	36.74	254.02	244.34
329.0	36.86	254.23	244.59
339.0	36.99	254.45	244.84
349.0	37.11	254.66	245.08
359.0	37.23	254.86	245.33
369.0	37.35	255.07	245.56
379.0	37.47	255.27	245.80
389.0	37.59	255.46	246.03
399.0	37.70	255.65	246.27
409.0	37.59	255.46	246.56
419.0	37.48	255.27	246.83
429.0	37.38	255.10	247.09
439.0	37.29	254.94	247.34
449.0	37.20	254.78	247.59
459.0	37.11	254.64	247.83
469.0	37.03	254.49	248.05
479.0	36.96	254.36	248.28
489.0	36.88	254.23	248.49
499.0	36.81	254.10	248.70
599.0	36.21	253.05	250.46
699.0	35.69	252.11	251.83
799.0	35.20	251.19	252.89
899.0	34.70	250.27	253.73
999.0	34.19	249.31	254.40
1099.0	33.66	248.30	254.93
1199.0	34.36	249.58	254.76
1299.0	35.33	251.33	254.51
1399.0	34.11	249.10	254.30
1499.0	33.07	247.14	254.06
1599.0	32.11	245.30	253.77
1699.0	31.21	243.52	253.45
1799.0	30.37	241.80	253.10
1899.0	29.56	240.12	252.72
1999.0	28.79	238.48	252.32
2099.0	28.05	236.88	251.90
2199.0	27.34	235.30	251.47

Table 6.4.3-17 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Minimum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
2299.0	26.67	233.76	251.02
2399.0	26.01	232.24	250.55
2499.0	25.39	230.75	250.08
2599.0	24.79	229.29	249.59
2699.0	24.21	227.85	249.09
2799.0	23.65	226.44	248.59
2899.0	23.11	225.05	248.08
2999.0	22.59	223.67	247.56
3099.0	22.09	222.33	247.04
3199.0	21.61	221.00	246.51
3299.0	21.14	219.69	245.98
3399.0	20.69	218.40	245.45
3499.0	20.25	217.13	244.91
3599.0	19.83	215.87	244.37
3699.0	19.24	214.09	243.54
3799.0	18.86	212.90	242.65
3899.0	19.17	213.85	241.83
3999.0	19.40	214.54	241.04
4999.0	20.28	217.16	234.16
5999.0	20.26	217.12	228.58
6999.0	19.88	215.99	223.75
7999.0	19.23	214.06	219.53
8999.0	18.45	211.64	215.74
9999.0	17.61	208.94	212.23
19999.0	13.62	194.30	188.65
29999.0	12.30	188.65	179.81
39999.0	11.53	185.07	175.58
49999.0	10.86	181.82	172.55
59999.0	10.26	178.73	169.81
69999.0	9.685	175.59	167.07
79999.0	9.072	172.11	164.22
89999.0	8.495	168.65	161.30
99999.0	7.945	165.15	158.36
199999.0	6.839	157.43	149.08
299999.0	6.410	154.12	146.38
399999.0	6.043	151.14	143.87

Table 6.4.3-17 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
499999.0	5.683	148.09	141.40
599999.0	5.343	145.04	138.91
699999.0	5.011	141.93	136.44
799999.0	4.700	138.87	133.97
899999.0	4.402	135.77	131.52
999999.0	4.123	132.76	129.07
1099999.0	3.965	131.03	126.75
1199999.0	3.927	130.64	126.41
1299999.0	3.892	130.29	126.09
1399999.0	3.844	129.78	125.81
1499999.0	3.796	129.24	125.40
1599999.0	3.764	128.89	125.01
1699999.0	3.720	128.40	124.62
1799999.0	3.676	127.90	124.23
1899999.0	3.634	127.42	123.84
1999999.0	3.590	126.91	123.45
2099999.0	3.550	126.42	123.07
2199999.0	3.488	125.67	122.70
2299999.0	3.473	125.48	122.26
2399999.0	3.428	124.95	121.91
2499999.0	3.387	124.42	121.52
2599999.0	3.350	123.98	121.13
2699999.0	3.311	123.48	120.75
2799999.0	3.267	122.91	120.36
2899999.0	3.238	122.52	119.96
2999999.0	3.204	122.02	119.57
3099999.0	3.170	121.54	118.97
3199999.0	3.163	121.41	118.85
3299999.0	3.156	121.28	118.77
3399999.0	3.143	121.12	118.67
3499999.0	3.139	121.07	118.58
3599999.0	3.126	120.91	118.48
3699999.0	3.117	120.77	118.38
3799999.0	3.105	120.65	118.29
3899999.0	3.095	120.52	118.19
3999999.0	3.083	120.41	118.10

Table 6.4.3-17 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
4099999.0	3.072	120.28	118.00
4199999.0	3.063	120.16	117.90
4299999.0	3.053	120.06	117.79
4399999.0	3.042	119.94	117.70
4499999.0	3.029	119.79	117.62
4599999.0	3.018	119.67	117.52
4699999.0	3.018	119.55	117.40
4799999.0	3.012	119.40	117.33
4899999.0	3.007	119.31	117.22
4999999.0	2.996	119.18	117.14
5099999.0	2.986	119.06	117.04
5199999.0	2.976	118.94	116.95
5299999.0	2.980	118.84	116.80
5399999.0	2.981	118.71	116.70
5499999.0	2.983	118.62	116.66
5599999.0	2.973	118.40	116.57
5699999.0	2.970	118.33	116.46
5799999.0	2.963	118.20	116.37
5899999.0	2.962	118.09	116.27
5999999.0	2.963	118.02	116.18
6099999.0	2.957	117.85	116.08
6199999.0	2.952	117.70	115.99
6299999.0	2.954	117.61	115.85
6399999.0	2.948	117.48	115.80
6499999.0	2.941	117.31	115.70
6599999.0	2.942	117.23	115.61
6699999.0	2.940	117.09	115.51
6799999.0	2.940	117.00	115.42
6899999.0	2.935	116.86	115.33
6999999.0	2.928	116.74	115.23
7099999.0	2.920	116.61	115.13
7199999.0	2.907	116.49	115.04
7299999.0	2.899	116.37	114.94
7399999.0	2.894	116.25	114.85
7499999.0	2.886	116.12	114.75
7599999.0	2.876	115.99	114.65

Table 6.4.3-17 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Minimum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
7699999.0	2.870	115.88	114.55
7799999.0	2.863	115.76	114.45
7899999.0	2.856	115.64	114.36
7999999.0	2.849	115.52	114.26
8099999.0	2.843	115.39	114.17
8199999.0	2.838	115.27	114.07
8299999.0	2.833	115.15	113.98
8399999.0	2.829	115.03	113.88
8499999.0	2.822	114.90	113.80
8599999.0	2.819	114.79	113.69
8699999.0	2.813	114.66	113.61
8799999.0	2.808	114.54	113.51
8899999.0	2.803	114.41	113.42
8999999.0	2.799	114.29	113.32
9099999.0	2.795	114.17	113.23
9199999.0	2.793	114.05	113.13
9299999.0	2.791	113.92	113.04
9399999.0	2.790	113.80	112.94
9499999.0	2.790	113.68	112.85
9599999.0	2.791	113.56	112.75
9699999.0	2.793	113.44	112.66
9799999.0	2.797	113.32	112.56
9899999.0	2.801	113.21	112.47
9999999.0	2.823	113.01	112.37
1000000.0	2.823	113.01	112.37

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Table 6.4.3-18
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Maximum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
.00100	1.0	120.00	120.00
.50000	3.619	142.47	183.61
1.00	6.089	162.36	198.75
2.00	10.44	190.58	212.20
3.00	14.13	207.89	219.54
4.00	16.79	215.88	224.05
5.00	18.86	219.30	227.40
6.00	20.79	221.38	230.41
7.00	22.73	225.86	233.09
8.00	24.58	230.45	235.44
9.00	26.29	234.43	237.34
10.0	27.86	237.89	239.17
11.0	29.29	240.90	240.87
12.0	30.58	243.51	242.37
13.0	31.74	245.80	243.70
14.0	32.81	247.83	244.87
15.0	33.79	249.63	245.90
16.0	34.66	251.21	246.81
17.0	35.47	252.66	247.55
18.0	36.13	253.81	248.06
19.0	36.51	254.46	248.43
20.0	36.69	254.77	248.72
21.0	36.76	254.89	249.03
22.0	36.72	254.81	249.14
23.0	36.60	254.60	249.71
24.0	36.41	254.29	250.07
25.0	36.19	253.90	250.04
26.0	35.99	253.55	250.00
27.0	35.80	253.23	249.96
28.0	35.63	252.93	249.93
29.0	35.46	252.62	249.89
30.0	35.29	252.33	249.86
31.0	35.14	252.05	249.66
32.0	35.03	251.86	248.90

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Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
33.0	34.95	251.72	247.94
34.0	34.88	251.59	247.03
35.0	34.81	251.47	246.18
36.0	34.75	251.36	245.36
37.0	34.69	251.26	244.58
38.0	34.64	251.16	243.84
39.0	34.59	251.07	243.14
40.0	34.54	250.98	242.46
41.0	34.50	250.90	241.82
42.0	34.45	250.82	241.20
43.0	34.41	250.75	240.62
44.0	34.38	250.68	240.05
45.0	34.34	250.61	239.51
46.0	34.31	250.55	238.99
47.0	34.28	250.49	238.50
48.0	34.25	250.44	238.02
49.0	34.22	250.39	237.56
50.0	34.19	250.34	237.11
51.0	34.17	250.29	236.69
52.0	34.14	250.25	236.28
53.0	34.12	250.21	235.89
54.0	34.10	250.17	235.50
55.0	34.08	250.13	235.14
56.0	34.06	250.10	234.78
57.0	34.04	250.06	234.44
58.0	34.03	250.03	234.10
59.0	34.01	250.00	233.78
60.0	34.00	249.97	233.47
61.0	33.98	249.95	233.17
62.0	33.96	249.91	232.89
63.0	33.94	249.87	232.63
64.0	33.92	249.82	232.37
65.0	33.89	249.78	232.13
66.0	33.87	249.74	231.89
67.0	33.85	249.70	231.66
68.0	33.83	249.66	231.44

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
69.0	33.81	249.62	231.22
70.0	33.79	249.58	231.01
71.0	33.76	249.54	230.81
72.0	33.74	249.50	230.62
73.0	33.73	249.47	230.43
74.0	33.71	249.42	230.30
75.0	33.70	249.35	230.30
76.0	33.68	249.29	230.29
77.0	33.67	249.22	230.30
78.0	33.65	249.14	230.35
79.0	33.63	249.07	230.40
80.0	33.61	249.00	230.44
81.0	33.60	248.93	230.49
82.0	33.58	248.86	230.53
83.0	33.56	248.79	230.57
84.0	33.55	248.72	230.61
85.0	33.54	248.65	230.65
86.0	33.52	248.58	230.69
87.0	33.51	248.52	230.73
88.0	33.49	248.45	230.76
89.0	33.48	248.39	230.80
90.0	33.47	248.33	230.83
91.0	33.46	248.27	230.87
92.0	33.45	248.20	230.90
93.0	33.44	248.14	230.94
94.0	33.41	248.08	230.97
95.0	33.38	248.02	231.00
96.0	33.35	247.95	231.03
97.0	33.31	247.89	231.06
98.0	33.28	247.83	231.09
99.0	33.25	247.77	231.12
100.00	33.22	247.71	231.14
101.00	33.18	247.65	231.17
102.00	33.15	247.59	231.19
103.00	33.12	247.53	231.22
104.00	33.09	247.47	231.24

Table 6.4.3-18 (cont.)
Double-Ended Pump Suction Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Maximum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
105.00	33.06	247.41	231.27
106.00	33.03	247.35	231.29
107.00	33.00	247.30	231.31
108.00	32.97	247.24	231.33
109.00	32.94	247.18	231.35
110.00	32.91	247.13	231.37
111.00	32.88	247.07	231.39
112.00	32.85	247.02	231.41
113.00	32.83	246.96	231.43
114.00	32.80	246.91	231.44
115.00	32.77	246.85	231.46
116.00	32.74	246.80	231.48
117.00	32.71	246.75	231.49
118.00	32.69	246.70	231.51
119.00	32.66	246.64	231.52
120.00	32.63	246.59	231.53
121.00	32.61	246.54	231.55
122.00	32.58	246.49	231.56
123.00	32.55	246.44	231.57
124.00	32.53	246.39	231.58
125.00	32.50	246.34	231.59
126.00	32.47	246.29	231.60
127.00	32.45	246.24	231.61
128.00	32.42	246.19	231.62
129.00	32.40	246.14	231.63
130.00	32.37	246.09	231.63
131.00	32.35	246.04	231.64
132.00	32.32	245.99	231.65
133.00	32.30	245.95	231.66
134.00	32.27	245.90	231.66
135.00	32.25	245.85	231.67
136.00	32.23	245.80	231.67
137.00	32.20	245.76	231.67
138.00	32.18	245.71	231.68
139.00	32.15	245.66	231.68
140.00	32.13	245.62	231.68

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
141.00	32.11	245.57	231.68
142.00	32.08	245.53	231.68
143.00	32.06	245.48	231.69
144.00	32.04	245.43	231.69
145.00	32.01	245.39	231.69
146.00	31.99	245.35	231.68
147.00	31.97	245.30	231.68
148.00	31.95	245.26	231.68
149.00	31.92	245.21	231.68
150.00	31.90	245.17	231.68
151.00	31.88	245.12	231.67
152.00	31.86	245.08	231.67
153.00	31.84	245.04	231.67
154.00	31.81	245.00	231.66
155.00	31.79	244.95	231.66
156.00	31.77	244.91	231.65
157.00	31.75	244.87	231.65
158.00	31.73	244.82	231.64
159.00	31.71	244.78	231.64
160.00	31.69	244.74	231.63
161.00	31.66	244.70	231.62
162.00	31.64	244.66	231.61
163.00	31.62	244.62	231.60
164.00	31.60	244.58	231.60
165.00	31.58	244.53	231.59
166.00	31.56	244.49	231.58
167.00	31.54	244.45	231.57
168.00	31.52	244.41	231.56
169.00	31.50	244.37	231.55
170.00	31.48	244.33	231.54
171.00	31.46	244.29	231.53
172.00	31.44	244.25	231.51
173.00	31.42	244.21	231.50
174.00	31.40	244.17	231.49
175.00	31.38	244.13	231.48
176.00	31.36	244.09	231.46

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
177.00	31.34	244.05	231.45
178.00	31.32	244.01	231.43
179.00	31.30	243.98	231.42
180.00	31.28	243.94	231.41
181.00	31.26	243.90	231.39
182.00	31.24	243.86	231.38
183.00	31.22	243.82	231.36
184.00	31.21	243.78	231.34
185.00	31.19	243.75	231.33
186.00	31.17	243.71	231.31
187.00	31.15	243.67	231.29
188.00	31.13	243.63	231.28
189.00	31.11	243.59	231.26
190.00	31.09	243.56	231.24
191.00	31.07	243.52	231.22
192.00	31.06	243.48	231.20
193.00	31.04	243.45	231.18
194.00	31.02	243.41	231.15
195.00	31.01	243.38	231.11
196.00	30.99	243.36	231.06
197.00	30.98	243.33	231.02
198.00	30.97	243.30	230.98
199.00	30.95	243.28	230.93
209.00	30.83	243.01	230.47
219.00	30.71	242.77	230.06
229.00	30.59	242.53	229.65
239.00	30.48	242.31	229.25
249.00	30.38	242.10	228.88
259.00	30.29	241.90	228.51
269.00	30.20	241.71	228.14
279.00	30.11	241.52	227.80
289.00	30.03	241.35	227.46
299.00	29.95	241.18	227.13
309.00	29.87	241.02	226.81
319.00	29.80	240.87	226.50
329.00	29.73	240.72	226.19

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
339.00	29.66	240.57	225.89
349.00	29.60	240.43	225.60
359.00	29.54	240.30	225.31
369.00	29.48	240.17	225.03
379.00	29.42	240.05	224.75
389.00	29.37	239.93	224.48
399.00	29.31	239.82	224.21
409.00	29.26	239.70	223.95
419.00	29.21	239.60	223.69
429.00	29.17	239.49	223.43
439.00	29.12	239.39	223.18
449.00	29.08	239.29	222.93
459.00	29.03	239.19	222.69
469.00	28.99	239.10	222.45
479.00	28.95	239.01	222.21
489.00	28.91	238.92	221.98
499.00	28.87	238.83	221.74
599.00	28.19	237.31	219.86
699.00	26.64	233.81	218.44
799.00	25.26	230.56	222.15
899.00	24.14	227.82	225.15
999.00	23.14	225.28	227.69
1099.0	22.25	222.92	229.87
1199.0	21.45	220.73	231.72
1299.0	20.61	218.35	231.31
1399.0	19.67	215.60	228.13
1499.0	18.82	213.01	225.25
1599.0	18.04	210.56	222.63
1699.0	17.34	208.23	220.23
1799.0	16.69	206.02	218.03
1899.0	16.09	203.92	216.01
1999.0	15.54	201.93	214.15
2099.0	15.04	200.03	212.43
2199.0	14.57	198.23	210.83
2299.0	14.14	196.52	209.35
2399.0	13.74	194.90	207.98

Table 6.4.3-18 (cont.)

Double-Ended Pump Suction Break

Byron Unit 2 & Braidwood Unit 2

(D5 SG) Maximum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
2499.0	13.37	193.35	206.69
2599.0	13.03	191.89	205.50
2699.0	12.71	190.49	204.38
2799.0	12.41	189.16	203.33
2899.0	12.14	187.88	202.35
2999.0	11.87	186.65	201.43
3099.0	11.63	185.47	200.57
3199.0	11.39	184.33	199.75
3299.0	11.17	183.23	198.98
3399.0	11.12	182.94	198.26
3499.0	11.43	184.44	197.60
3599.0	11.64	185.46	197.02
3699.0	11.63	185.43	196.07
3799.0	11.61	185.33	195.14
3899.0	11.59	185.21	194.28
3999.0	11.55	185.06	193.46
4999.0	11.10	182.88	187.28
5999.0	10.59	180.31	183.85
6999.0	10.14	177.95	181.80
7999.0	9.754	175.83	180.49
8999.0	9.405	173.88	179.58
9999.0	9.087	172.04	178.90
19999.0	8.556	168.57	177.29
29999.0	8.427	167.40	177.06
39999.0	8.278	166.24	176.86
49999.0	8.078	164.86	176.55
59999.0	7.891	163.46	176.28
69999.0	7.678	161.95	175.97
79999.0	7.470	160.38	175.63
89999.0	7.220	158.55	175.26
99999.0	6.940	156.45	174.81
199999.0	6.744	154.24	174.17
299999.0	6.624	152.77	173.85
399999.0	6.471	151.36	173.64
499999.0	6.312	149.88	173.28
599999.0	6.152	148.42	172.98

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
699999.0	5.988	146.94	172.67
799999.0	5.823	145.47	172.38
899999.0	5.652	143.94	171.99
999999.0	5.487	142.51	171.74
1099999.0	5.424	142.36	171.67
1199999.0	5.361	142.28	171.65
1299999.0	5.343	142.21	171.68
1399999.0	5.310	142.13	171.64
1499999.0	5.252	142.05	171.62
1599999.0	5.194	141.98	171.61
1699999.0	5.168	141.90	171.60
1799999.0	5.141	141.81	171.56
1899999.0	5.114	141.73	171.54
1999999.0	5.071	141.67	171.55
2099999.0	5.010	141.59	171.52
2199999.0	4.981	141.50	171.50
2299999.0	4.967	141.43	171.49
2399999.0	4.953	141.36	171.46
2499999.0	4.938	141.28	171.45
2599999.0	4.881	141.21	171.43
2699999.0	4.821	141.13	171.43
2799999.0	4.778	141.05	171.43
2899999.0	4.771	140.98	171.41
2999999.0	4.765	140.88	171.38
3099999.0	4.759	140.82	171.34
3199999.0	4.752	140.72	171.35
3299999.0	4.746	140.67	171.31
3399999.0	4.741	140.59	171.34
3499999.0	4.734	140.51	171.30
3599999.0	4.728	140.43	171.28
3699999.0	4.722	140.36	171.27
3799999.0	4.717	140.29	171.29
3899999.0	4.709	140.21	171.25
3999999.0	4.704	140.12	171.22
4099999.0	4.698	140.06	171.18
4199999.0	4.692	139.97	171.19

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
4299999.0	4.684	139.87	171.19
4399999.0	4.679	139.83	171.17
4499999.0	4.674	139.75	171.17
4599999.0	4.668	139.67	171.14
4699999.0	4.662	139.60	171.13
4799999.0	4.655	139.50	171.10
4899999.0	4.650	139.44	171.09
4999999.0	4.644	139.37	171.04
5099999.0	4.639	139.29	171.04
5199999.0	4.631	139.18	171.05
5299999.0	4.626	139.13	171.01
5399999.0	4.621	139.06	171.01
5499999.0	4.615	138.98	170.99
5599999.0	4.603	138.81	171.00
5699999.0	4.604	138.81	170.95
5799999.0	4.599	138.75	170.95
5899999.0	4.593	138.68	170.90
5999999.0	4.587	138.59	170.91
6099999.0	4.558	138.26	170.96
6199999.0	4.576	138.45	170.90
6299999.0	4.570	138.36	170.86
6399999.0	4.565	138.28	170.84
6499999.0	4.559	138.21	170.83
6599999.0	4.554	138.14	170.84
6699999.0	4.536	137.92	170.85
6799999.0	4.543	137.98	170.79
6899999.0	4.526	137.77	170.81
6999999.0	4.531	137.83	170.76
7099999.0	4.527	137.75	170.73
7199999.0	4.521	137.67	170.72
7299999.0	4.515	137.59	170.71
7399999.0	4.510	137.52	170.71
7499999.0	4.485	137.22	170.74
7599999.0	4.500	137.36	170.66
7699999.0	4.494	137.29	170.64
7799999.0	4.489	137.21	170.63

Table 6.4.3-18 (cont.) Double-Ended Pump Suction Break Byron Unit 2 & Braidwood Unit 2 (D5 SG) Maximum Safeguards			
Time (sec)	**Pressure (psia)**	**Steam Temp (°F)**	**Sump Temp (°F)**
7899999.0	4.483	137.12	170.62
7999999.0	4.478	137.05	170.60
8099999.0	4.474	136.98	170.59
8199999.0	4.468	136.90	170.57
8299999.0	4.463	136.83	170.54
8399999.0	4.458	136.75	170.54
8499999.0	4.452	136.68	170.54
8599999.0	4.447	136.60	170.51
8699999.0	4.443	136.52	170.50
8799999.0	4.438	136.44	170.48
8899999.0	4.432	136.36	170.47
8999999.0	4.427	136.29	170.44
9099999.0	4.423	136.20	170.43
9199999.0	4.418	136.14	170.43
9299999.0	4.412	136.05	170.40
9399999.0	4.406	135.95	170.41
9499999.0	4.403	135.91	170.38
9599999.0	4.395	135.78	170.38
9699999.0	4.392	135.72	170.36
9799999.0	4.388	135.67	170.32
9899999.0	4.379	135.52	170.35
9999999.0	4.379	135.51	170.29
10000000.	4.379	135.51	170.29

81.

Table 6.4.3-19
Double-Ended Hot Leg Break
Byron Unit 2 & Braidwood Unit 2
(D5 SG) Minimum Safeguards

Time (sec)	Pressure (psia)	Steam Temp (°F)	Sump Temp (°F)
.0010	1.000	120.00	120.00
.5000	3.789	143.92	176.27
1.000	6.047	161.91	192.89
2.000	10.16	188.41	209.02
3.000	13.77	205.39	217.94
4.000	16.86	215.66	224.03
5.000	19.56	221.95	228.76
6.000	22.00	225.83	232.68
7.000	24.24	229.63	235.79
8.000	26.26	234.36	238.27
9.000	28.14	238.48	240.55
10.00	29.86	242.07	242.61
11.00	31.40	245.14	244.40
12.00	32.76	247.73	245.91
13.00	33.96	249.95	247.17
14.00	35.02	251.86	248.16
15.00	35.97	253.53	248.85
16.00	36.75	254.88	249.27
17.00	37.35	255.89	249.43
18.00	37.79	256.63	249.45
19.00	38.10	257.15	249.42
20.00	38.29	257.46	249.39
21.00	38.36	257.57	249.36
22.00	38.33	257.52	249.32
23.00	38.12	257.18	249.29
24.00	37.89	256.79	249.27
25.00	37.67	256.42	249.24
26.00	37.46	256.08	249.22
27.00	37.27	255.75	249.19
28.00	37.10	255.46	249.17
29.00	36.93	255.18	249.14
30.00	36.79	254.93	249.11

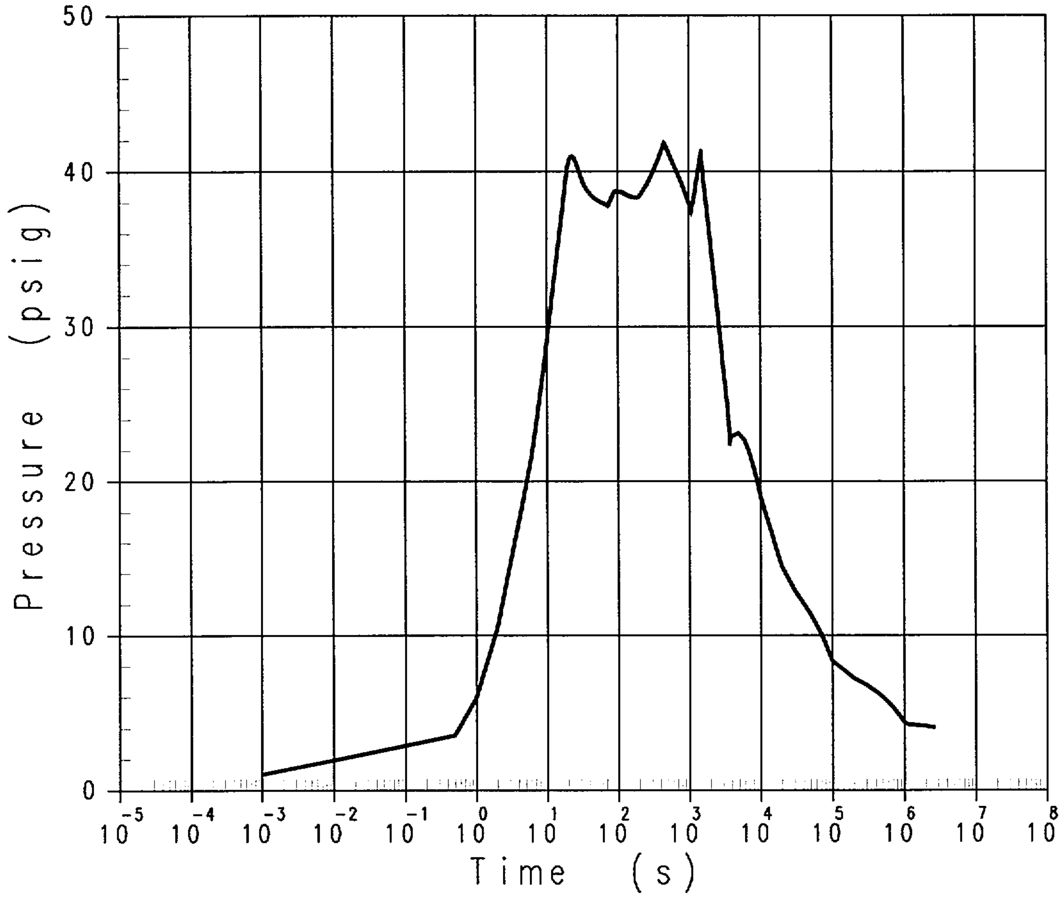

Figure 6.4.3-1

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Pump Suction Break Loss of Offsite Power/Minimum

Safeguards Assumptions Containment Pressure vs. Time

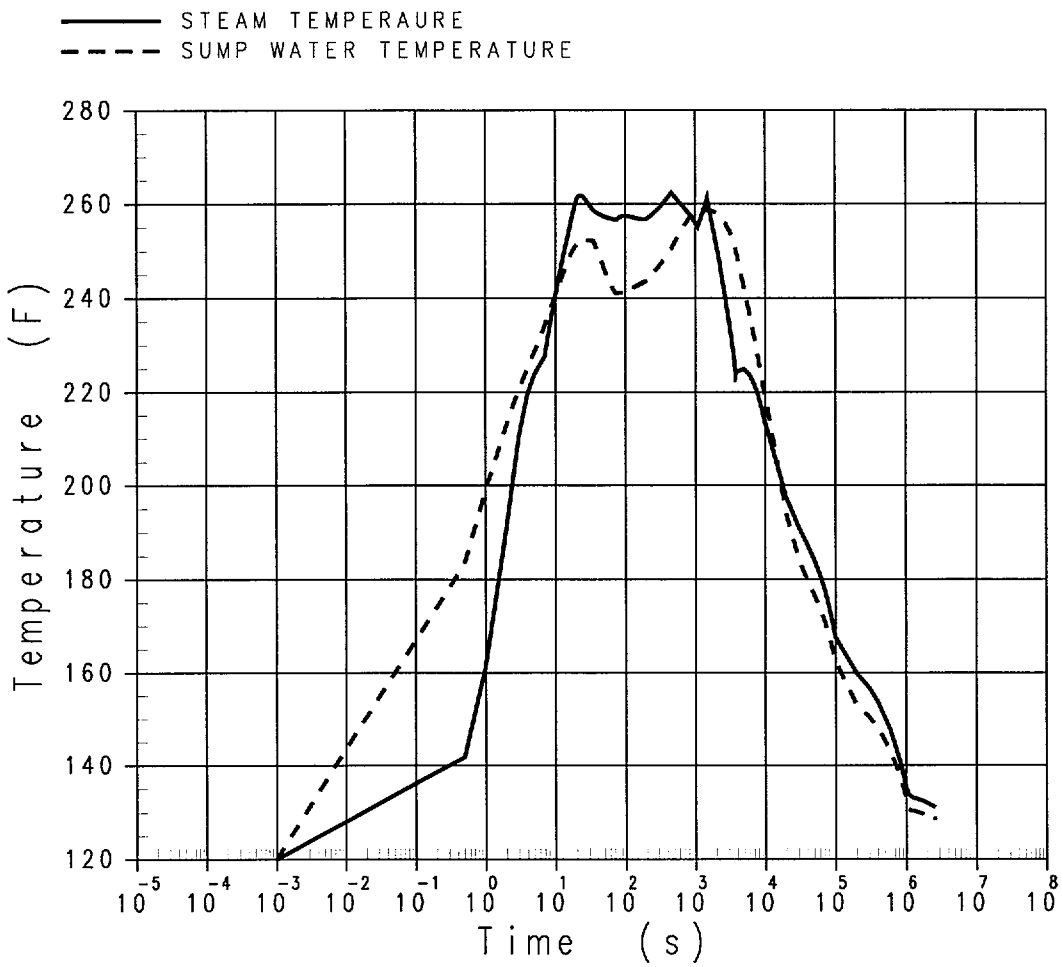

Figure 6.4.3-2

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Pump Suction Break Loss of Offsite Power/Minimum

Safeguards Assumptions Containment Steam and Sump Water Temperatures vs. Time

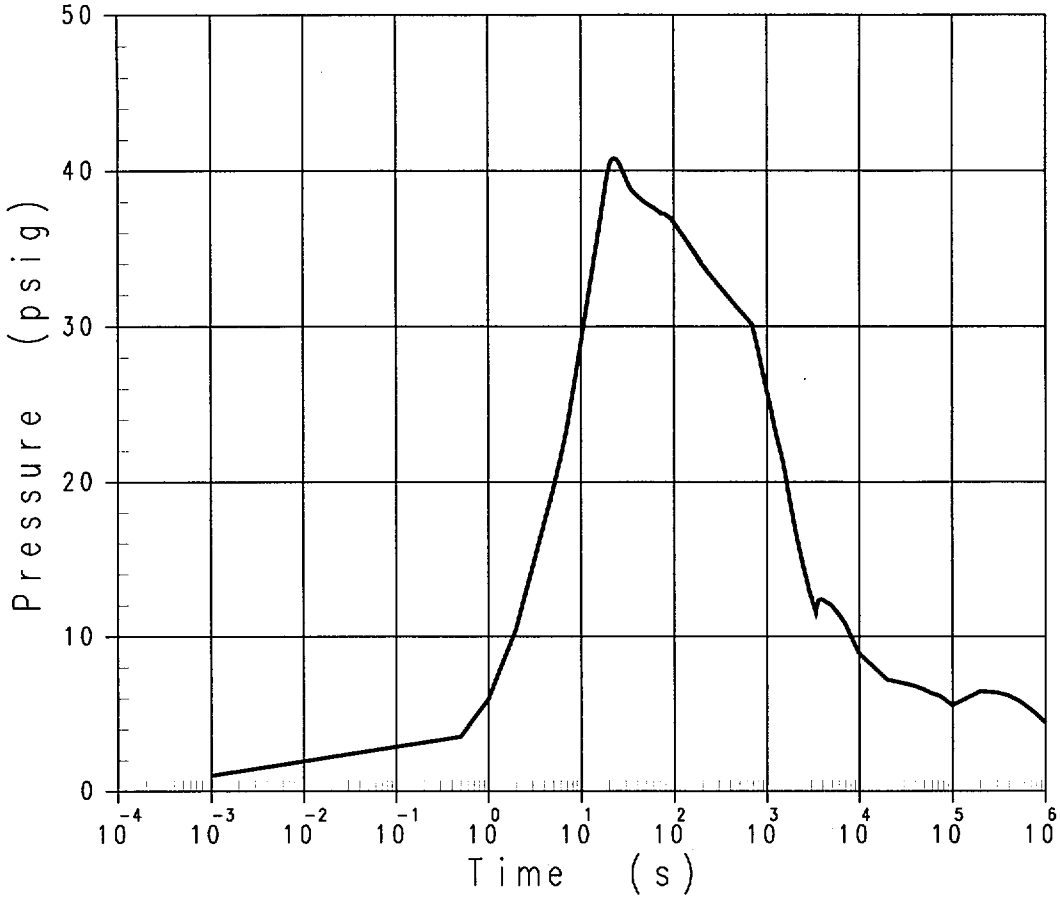

Figure 6.4.3-3

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Pump Suction Break Offsite Power Available/Maximum

Safeguards Assumptions Containment Pressure vs. Time

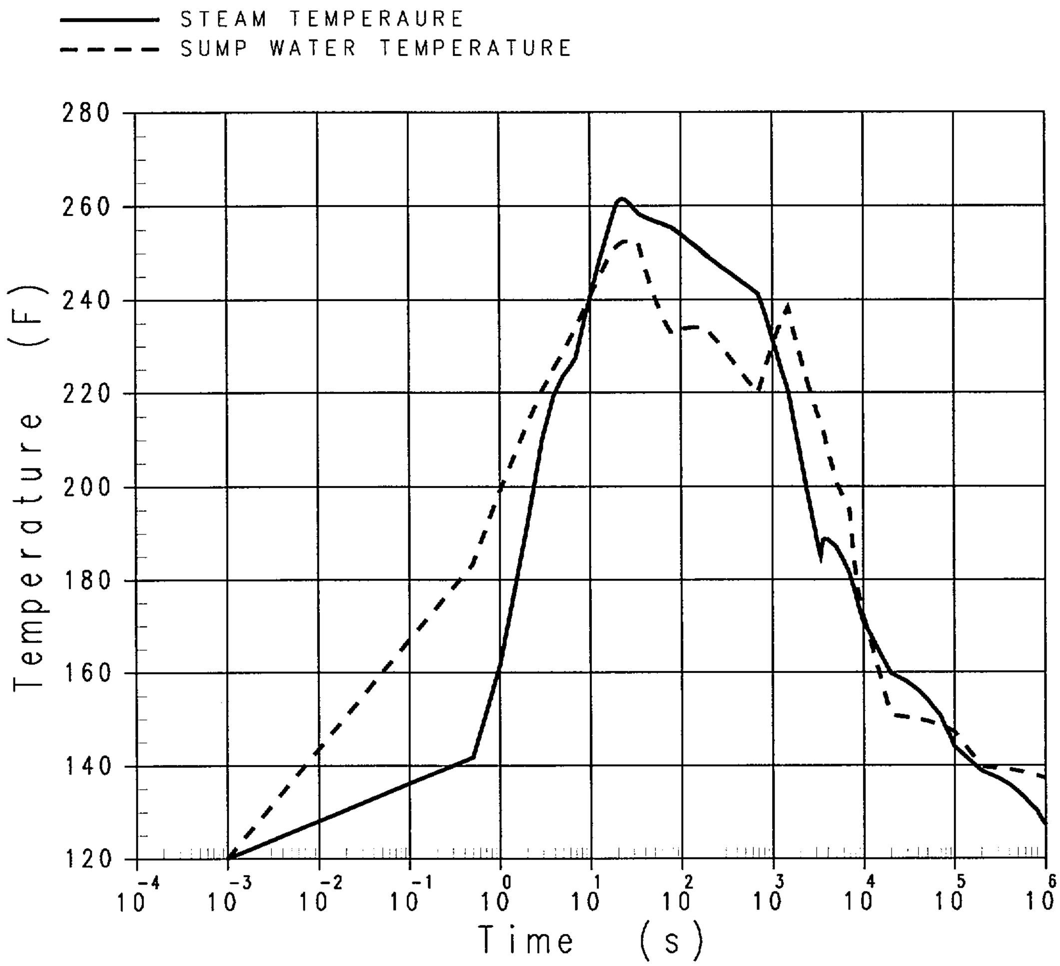

Figure 6.4.3-4

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Pump Suction Break Offsite Power Available/Maximum

Safeguards Assumptions Steam and Sump Water Temperatures vs. Time

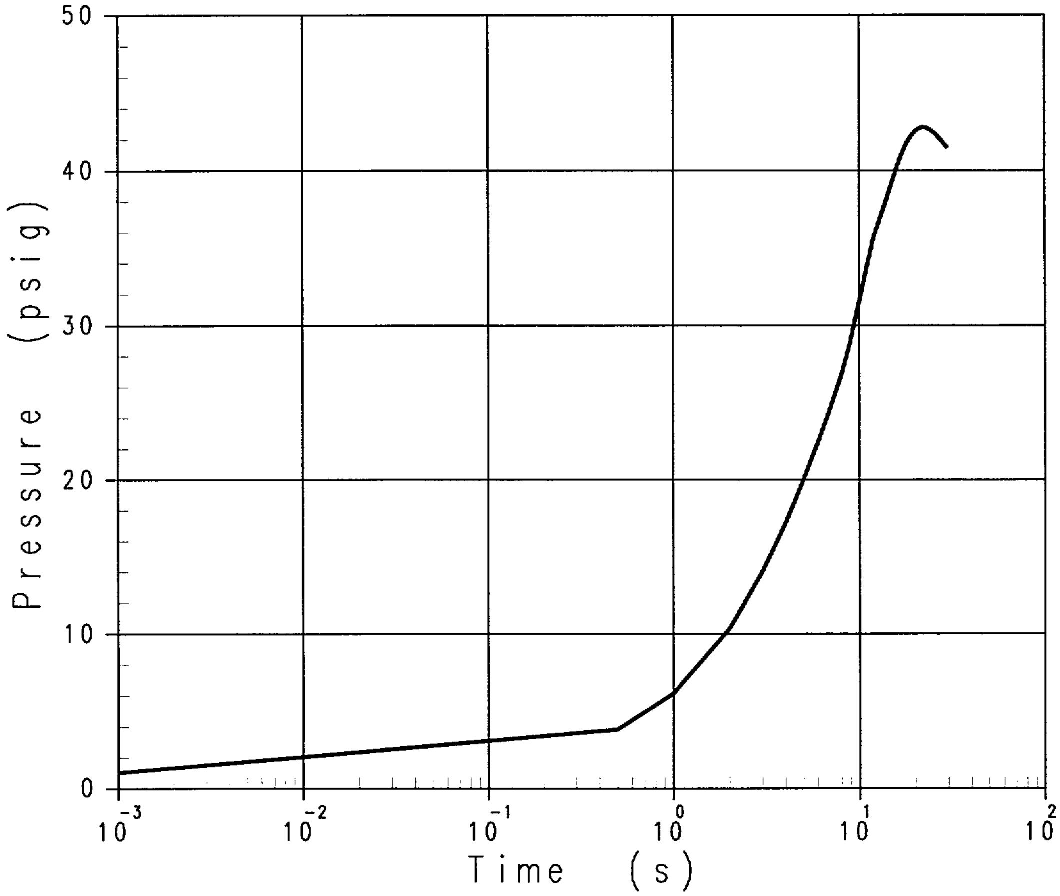

Figure 6.4.3-5

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Hot Leg Break Loss of Offsite Power

Assumptions Containment Pressure vs. Time

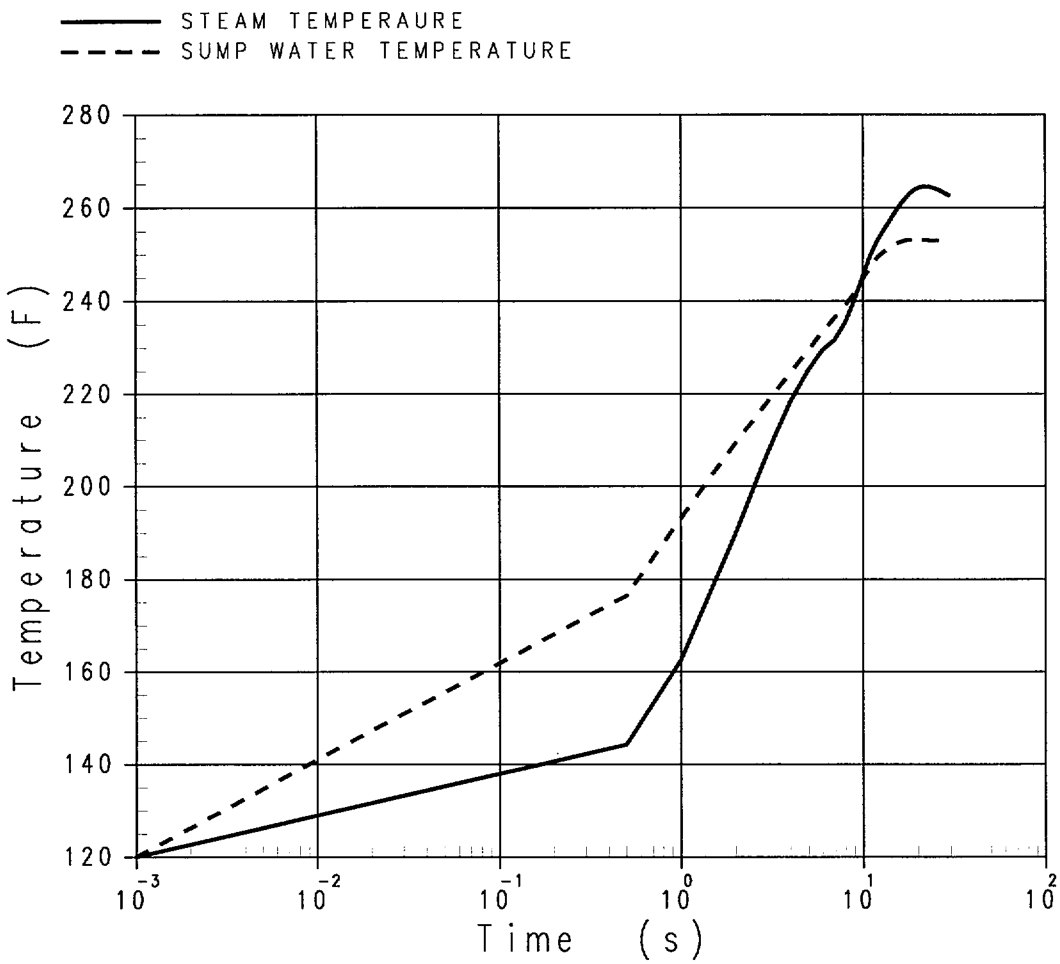

Figure 6.4.3-6

Byron Unit 1 and Braidwood Unit 1 Uprate Project (BWI SG)

Double Ended Hot Leg Break Loss of Offsite Power

Assumptions Steam and Sump Water Temperatures vs. Time

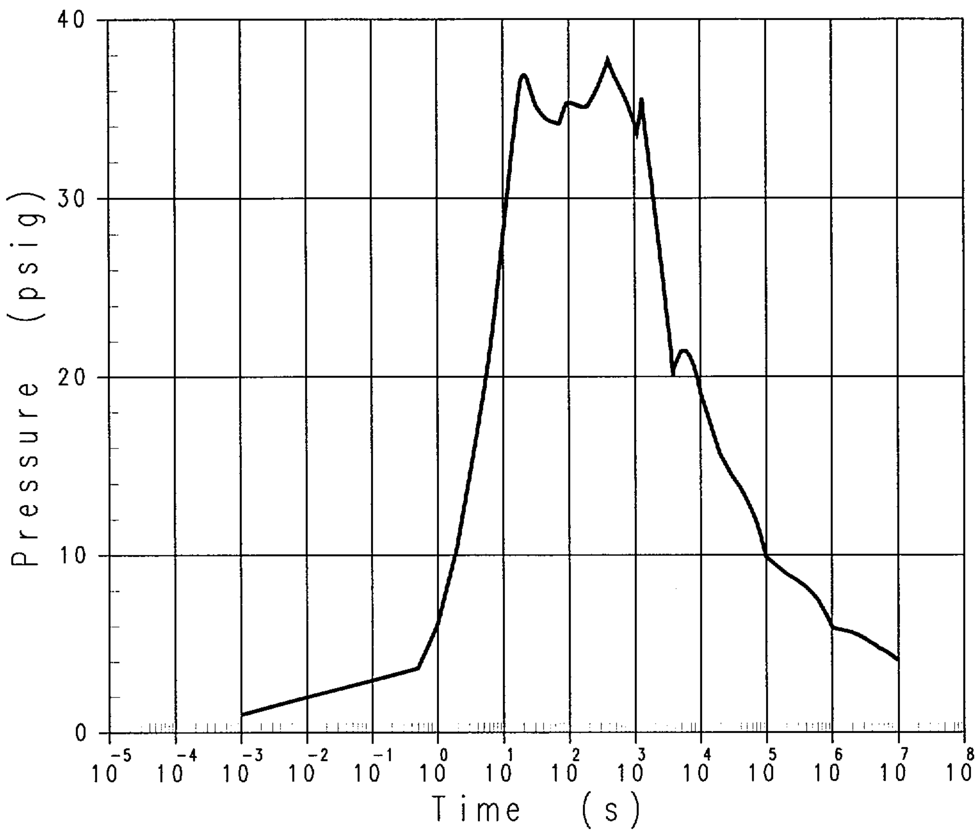

Figure 6.4.3-7

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Pump Suction Break Loss of Offsite Power/Minimum

Safeguards Assumptions Containment Pressure vs. Time

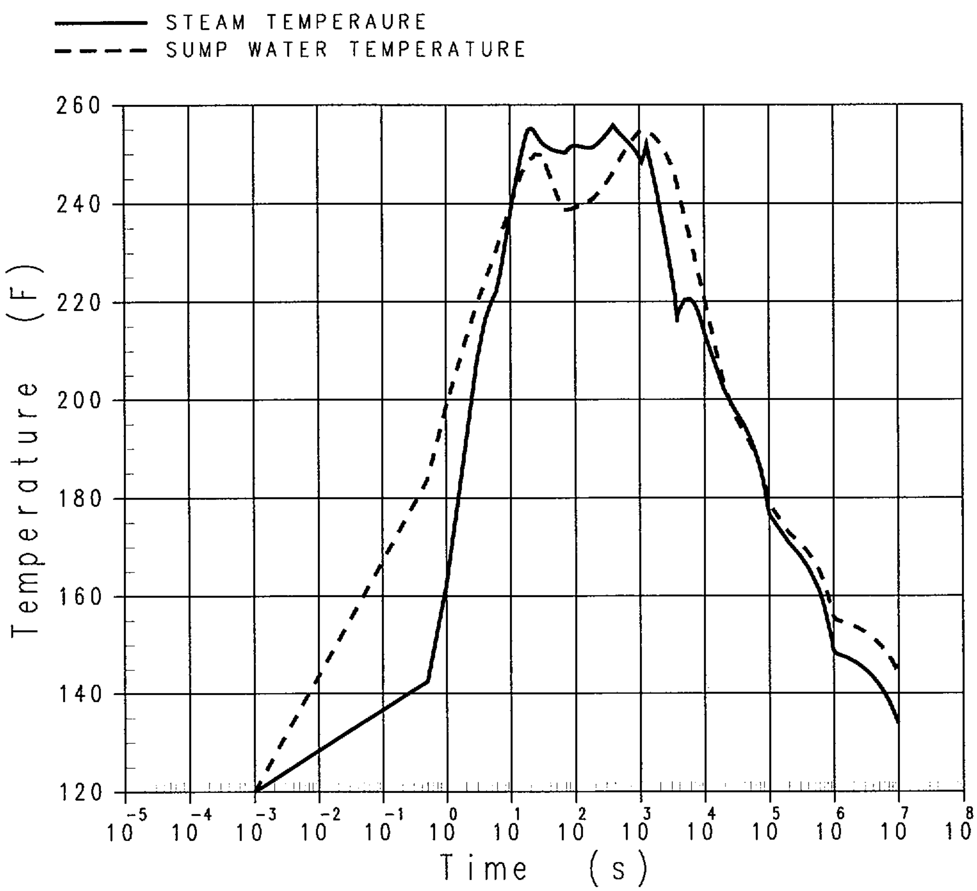

Figure 6.4.3-8

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Pump Suction Break Loss of Offsite Power/Minimum

Safeguards Assumptions Steam and Sump Water Temperature vs. Time

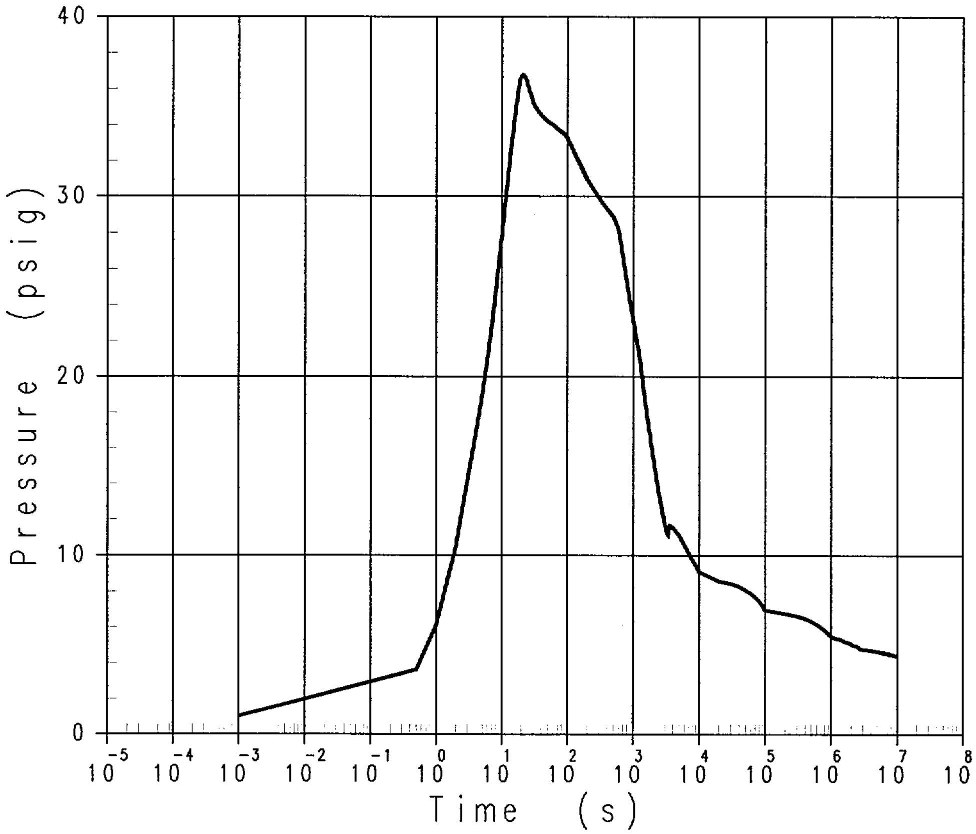

Figure 6.4.3-9

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Pump Suction Break Offsite Power Available/Maximum

Safeguards Assumptions Containment Pressure vs. Time

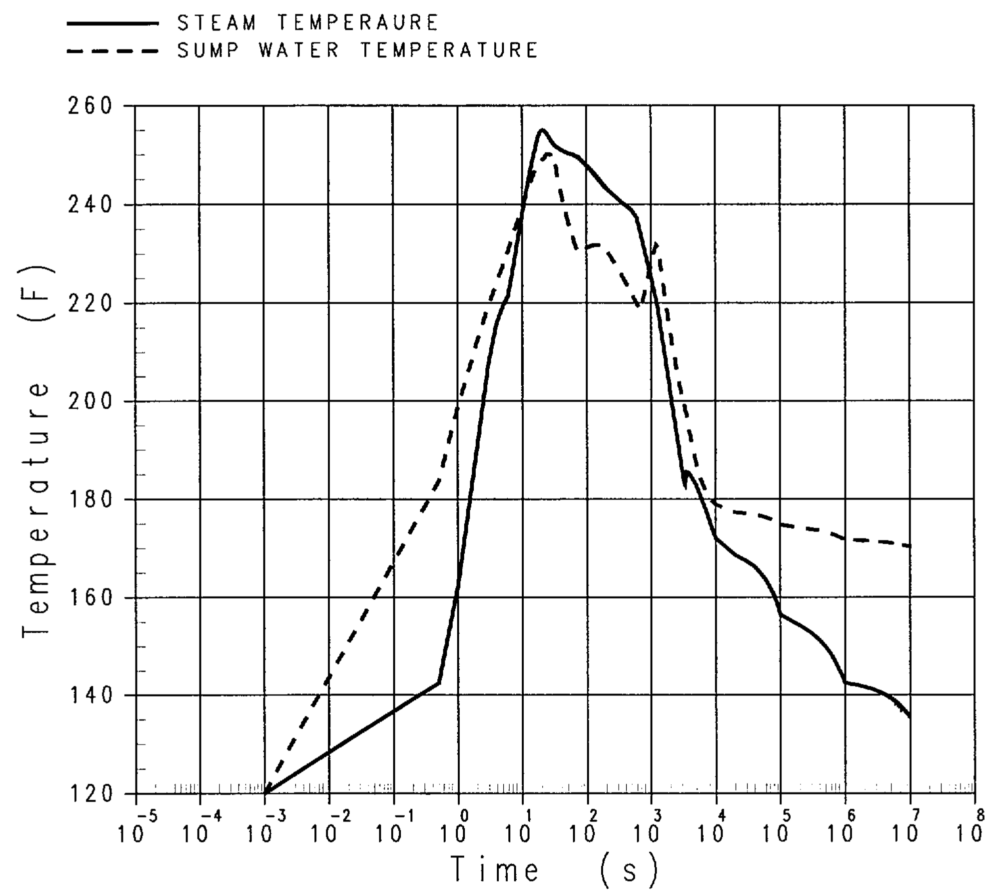

Figure 6.4.3-10

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Pump Suction Break Offsite Power Available/Maximum

Safeguards Assumptions Steam and Sump Water Temperatures vs. Time

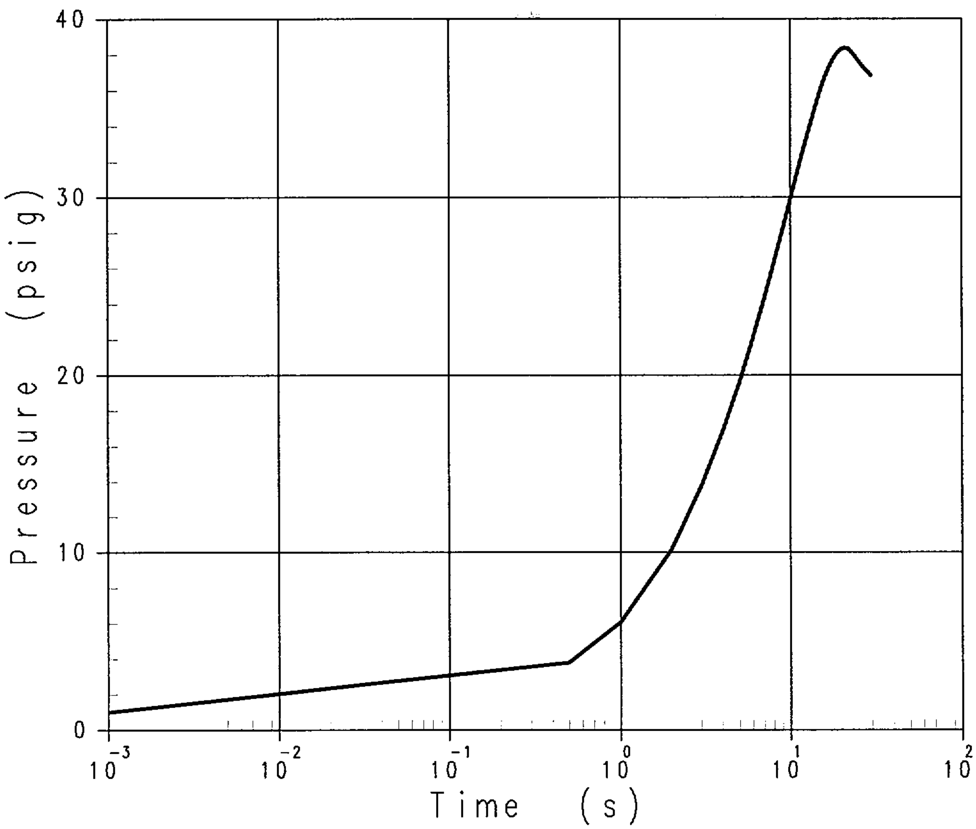

Figure 6.4.3-11

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Hot Leg Break, Loss of Offsite Power

Assumptions Containment Pressure vs. Time

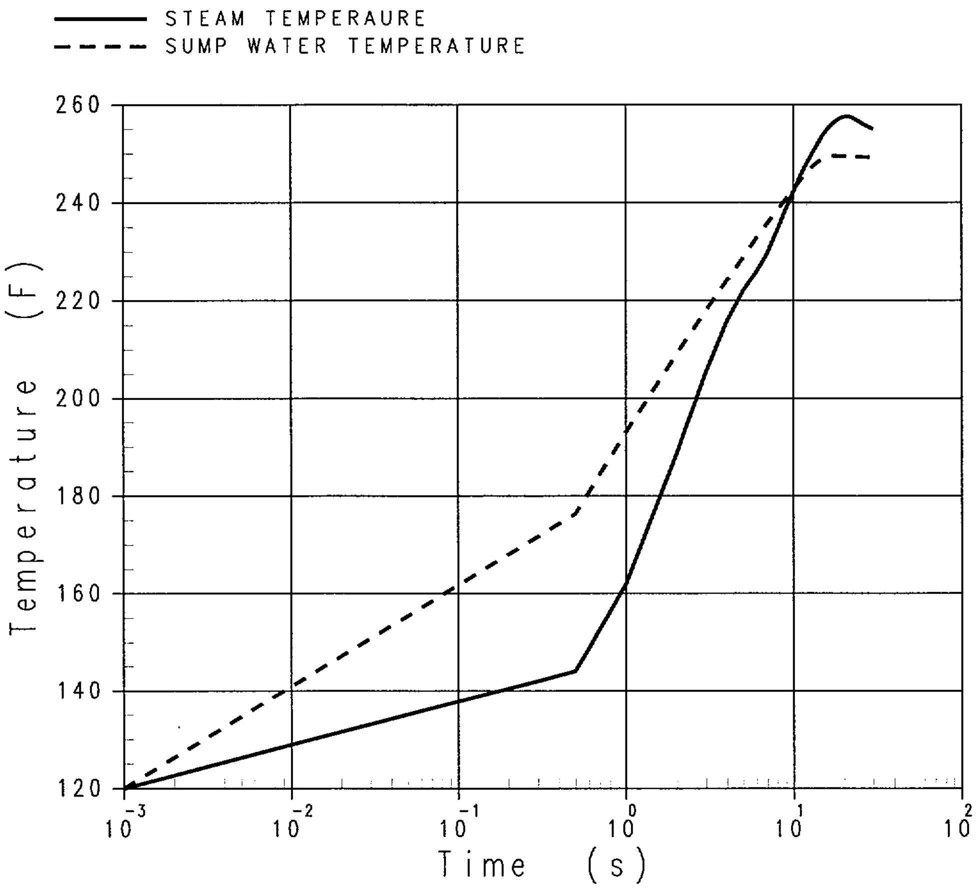

Figure 6.4.3-12

Byron Unit 2 and Braidwood Unit 2 Uprate Project (D5 SG)

Double Ended Hot Leg Break Loss of Offsite Power

Assumptions Steam and Sump Water Temperatures vs. Time

6.5 Main Steamline Break Mass and Energy Releases, Containment Response, and Steam Tunnel Analysis

6.5.1 Main Steamline Break Mass and Energy Releases Inside Containment

6.5.1.1 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the plant operating conditions and the size of the rupture as well as the configuration of the plant steam system and containment design. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the Main Steamline Break (MSLB) Mass and Energy (M&E) releases for use in containment integrity analysis.

6.5.1.2 Input Parameters and Assumptions

To assess the effect of break area on the mass and energy releases from a ruptured steamline, a spectrum of break sizes has been evaluated. At a plant power level of 102% nominal full-load power, three break sizes have been analyzed based on the results of the analyses presented in the Byron/Braidwood Nuclear Plant UFSAR, Section 6.2.1.4. These break types are the following.

1. A full double-ended rupture (DER) downstream of the flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The break area for Units 1 (BWI Steam Generators) is 1.1 ft^2, and the break area for Units 2 (D5 Steam Generators) is 1.4 ft^2.

2. A small break at the steam generator nozzle having an area of 1 ft^2.

3. A small split-rupture that will neither generate a steamline isolation signal from the Engineered Safety Features nor result in water entrainment in the break effluent.

A total of sixteen cases were chosen for power uprate analyses based on the results of the analyses presented in the Byron/Braidwood Nuclear Plant UFSAR, Section 6.2.1.4. Eight cases were run for both Units 1 and Units 2. The important plant conditions and features that were assumed are discussed in the following paragraphs.

Initial Power Level

Several different power levels spanning full- to zero-power conditions have been investigated for Byron/Braidwood Units 1 and 2 as presented in the Byron/Braidwood UFSAR, based on the information in Reference 1. For the uprating analysis, the approach taken was to demonstrate that the containment response at uprated power conditions was bounded by the current containment response. Thus, a power level of 102% uprated power was used for all calculations.

In general, plant initial conditions are assumed to be at their nominal values corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.5.1-1 and 6.5.1-2 identify the values assumed for Reactor Coolant System (RCS) pressure, RCS vessel average temperature, pressurizer water volume, steam generator water level, and feedwater enthalpy at 102% uprated reactor power for both Units 1 and Units 2. Steamline break mass and energy releases assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is more mass and energy available for release into containment.

Single-Failure Assumptions

Each case analyzed considered only one single failure, including a base case with no reactor safeguards failures assumed. Brief descriptions of the single failures that are considered in the current analyses are presented both below and in Reference 1.

a. No Single Failure

No failures were assumed. All automatic reactor safety functions were assumed to occur as designed. This case allows the single failure to be modeled in the containment safeguards system without accounting for multiple single failures.

b. Failure to Completely Isolate All Main Steamlines

The main steamline isolation function is accomplished via the Main Steam Isolation Valves (MSIVs) in each of the four steamlines. The valves close on an isolation signal to terminate steam flow from the associated steam generator. A main steamline rupture upstream of the valve, as postulated for the inside-containment analysis, will create a situation in which the steam generator on the faulted loop cannot be isolated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is prevented by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators.

In addition to continued blowdown from the faulted-loop steam generator after MSIV closure, steam in the unisolable section of the steamline needs to be considered. An MSIV failure can impact the mass and energy releases, since a failed MSIV will result in a larger unisolable steamline volume.

c. Failure of Feedwater Isolation Valve (FIV) in Faulted Loop

The feedwater isolation valve is assumed to close in seven seconds (signal processing time and valve stroke time). If the FIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, the unisolable volume of feedwater piping is increased. The fluid inventory in this additional unisolable feedwater piping is available to flash and be released to the containment as the piping depressurizes.

Main Feedwater System

The rapid depressurization which occurs following a steamline rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. Rapid-closing feedwater isolation valves in the main feedwater lines limit this effect.

Following initiation of the MSLB, main feedwater flow is conservatively modeled by assuming that sufficient feedwater flow is provided to match or exceed the steam flow prior to reactor trip. The initial increase in feedwater flow (until fully isolated) is in response to increases in steam flow following the steamline break. This maximizes the total mass addition prior to feedwater

isolation. The feedwater isolation response time, following the safety injection signal, was assumed to be a total of seven seconds, consisting of two seconds for signal processing plus five seconds for the Feedwater Isolation Valve (FIV) stroke time.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream from the isolation valve may flash to steam if the feedwater temperature is at or exceeds the saturation temperature associated with the pressure of the feedwater piping. This unisolable feedwater line volume is an additional source of high-energy fluid that was assumed to be discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop and minimized for the intact loops. The energy in the unisolable volume is conservatively maximized for this calculation.

Auxiliary Feedwater System

Generally, within the first minute following a steamline break, the Auxiliary Feedwater (AF) System is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment and increase the heat transferred to the secondary fluid. The auxiliary feedwater flow to the faulted and intact steam generators has been assumed to be a constant value, based on maximum AF pump performance, in the steamline break analysis inside containment. A higher AF flowrate to the faulted loop steam generator is assumed, consistent with a depressurizing steam generator. Conversely, a lower AF flowrate is assumed, consistent with the intact loop steam generators remaining at a pressurized condition.

Steam Generator Fluid Mass

A maximum initial steam generator mass in the faulted loop steam generator was used in all analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. For both BWI and D5 steam generators, the initial mass was calculated as the value corresponding to the programmed level +5% narrow-range span and assuming 0% tube plugging, plus a mass uncertainty. This assumption is conservative with respect to the RCS cooldown through the faulted loop steam generator resulting from the steamline break.

Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy, which can be transferred to the steam generator with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available for transfer to the steam generator with the broken steamline. The effects of reverse steam generator heat transfer are included in the results.

Break Flow Model

Piping discharge resistances were not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an f (L / D) = 0 was used].

Steamline Volume Blowdown

The contribution to the mass and energy releases from the secondary plant steam piping was included in the mass and energy release calculations. The flowrate was calculated by the LOFTRAN code (Reference 3) by modeling the limiting flow area of the main steam isolation valve as a break area with the steam header modeled as the source volume. For all steamline break cases analyzed for the power uprating, the unisolable steamline mass is included in the mass exiting the break from the time of steamline isolation until the unisolable mass is completely released to containment.

Main Steamline Isolation

Steamline isolation is assumed in all four loops to terminate the blowdown from the three intact steam generators. A delay time of eight seconds was assumed. The delay time accounts for both signal processing time plus valve stroke time. Full steam flow is conservatively assumed through the valve during the valve stroke time.

Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, and feedwater isolation. [Subsequent analysis of the containment response to the MSLB models the operation of the emergency fan coolers and containment spray.] The protection system actuation signals and associated setpoints that were modeled in the analysis are identified in Table 6.5.1-3. The setpoints used are conservative values with respect to the Byron/Braidwood plant-specific values delineated in the Technical Specifications.

For the 1.1 and 1.4 ft^2 DER MSLB for Units 1 and 2 at 102% power, as well as the smaller breaks at 102% power, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in any loop which initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal.

For the split-rupture steamline breaks, no credit is taken for any mitigation signals received from either the Reactor Protection System or any secondary-side signals produced by the Engineered Safety Features Actuation System. The first protection system signal actuated is Containment Pressure High-1 (2-of-3 channels) which initiates safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal. The second protection system signal actuated is Containment Pressure High-2 (2-of-3 channels) which actuates steamline isolation.

Safety Injection System

Minimum Emergency Core Cooling System (ECCS) flowrates corresponding to the failure of one ECCS train were assumed in this analysis. A minimum ECCS flow is conservative since the reduced boron addition maximizes a return to power resulting from RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full ECCS flow was assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis since the assumed loss of offsite power would reduce the mass and energy

releases. This is due to the loss of forced reactor coolant flow, which results in a consequential reduction in primary-to-secondary heat transfer.

Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick metal masses and heat transfer coefficients.

Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2 model (Reference 2).

Rod Control

The rod control system was conservatively assumed to be in manual operation for all steamline break analyses. Rods in automatic control would step in prior to reactor trip, due to the increased steam flow. This would reduce nuclear power and core heat flux, reducing the primary-to-secondary heat transfer.

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions were used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.1.3 Description of Analysis

The break flows and enthalpies of the steam release through the steamline break inside containment are analyzed with the LOFTRAN computer code. Blowdown mass and energy

releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer.

The Byron/Braidwood NSSS is analyzed using LOFTRAN to determine the transient steam mass and energy releases inside containment following a steamline break event. The mass and energy releases are used as input conditions to the analysis of the containment response.

The following licensing-basis cases of the MSLB inside containment have been analyzed at the uprated power, based on the results of the analyses presented in the Byron/Braidwood Nuclear Plant UFSAR, Section 6.2.1.4.

Case 1: Full double-ended (1.1 ft^2 for Units 1 or 1.4 ft^2 for Units 2) rupture at 102% power; main steamline isolation valve failure assumed

Case 2: Full double-ended (1.1 ft^2 for Units 1 or 1.4 ft^2 for Units 2) rupture at 102% power; feedwater isolation valve failure assumed

Case 3: Small double-ended (1.0 ft^2 for Units 1 or Units 2) rupture at 102% power; no single failure assumed

Case 4: Small double-ended (1.0 ft^2 for Units 1 or Units 2) rupture at 102% power; feedwater isolation valve failure assumed

Case 5: Small double-ended (1.0 ft^2 for Units 1 or Units 2) rupture at 102% power; main steam isolation valve failure assumed

Case 6: Split break (1.0 ft^2 for Units 1 or 0.82 ft^2 for Units 2) at 102% power; no single failure assumed

Case 7: Split break (1.0 ft^2 for Units 1 or 0.82 ft^2 for Units 2) at 102% power; feedwater isolation valve failure assumed

Case 8: Split break (1.0 ft^2 for Units 1 or 0.82 ft^2 for Units 2) at 102% power; main steam isolation valve failure assumed

For the double-ended rupture cases, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the integral flow restrictor area of 1.1 ft^2 for Units 1 (BWI steam generators) and 1.4 ft^2 for Units 2 (D5 steam generators). The reverse-flow cross-sectional flow area of the steam piping associated with the break location is limited by the 2.64 ft^2 flow area of the Main Steamline Isolation Valve (MSIV) located in the steam line between the steam header, which is fed by the three intact steam generators, and the break location. At the time of MSIV closure, the steam flow from the intact-loop steam generators is terminated.

6.5.1.4 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. Additional clarification of the ANS classification of this event is presented in Section 6.2.4 and 6.2.5 of this report, which discuss the core response to a steamline break event. The acceptance criteria associated with the steamline break event resulting in a mass and energy release inside containment is based on an analysis which provides sufficient conservatism to assure that the containment design margin is maintained.

The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model including entrainment, main and auxiliary feedwater flow, steamline and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in Reference 1 and Section 6.5.1.2 of this report.

The mass and energy release data for each of the steamline break cases analyzed are used as input to a containment response calculation to confirm the design parameters of the Byron/Braidwood Units 1 and 2 containment structure and the environmental qualification (EQ) of equipment located inside containment.

6.5.1.5 Results

Using Reference 1 as a basis, including parameter changes associated with the power uprating, the mass and energy release rates for each of the steamline break cases noted in Section 6.5.1.3 were developed for use in containment pressure and temperature response

analysis. The containment pressure and temperature responses were, in turn, used for evaluation of containment integrity and environmental qualification (EQ) of equipment. Tables 6.5.1-4 and 6.5.1-5 provide the sequence of events for Units 1 and Units 2, respectively, for the large double-ended rupture at 102% power with MSIV failure assumed.

6.5.1.6 Conclusions

The mass and energy releases from the sixteen steamline break cases have been analyzed at the uprated power conditions. The assumptions delineated in Section 6.5.1.2 have been included in the steamline break analysis such that the applicable acceptance criteria are met. The steam mass and energy releases discussed in this section have been provided for use in the containment response analysis in support of the Byron/Braidwood power uprating.

6.5.1.7 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary)," April 1984

Table 6.5.1-1
Byron/Braidwood Units 1 and 2
Nominal Plant Parameters for Thermal Uprate*
(MSLB M&E Releases)

Nominal Conditions	Units 1 (BWI SG)	Units 2 (D5 SG)
NSSS Power, MWt	3600.6	3600.6
Core Power, MWt	3586.6	3586.6
Reactor Coolant Pump Heat, MWt	14	14
Reactor Coolant Flow (total), gpm	368,000	368,000
Pressurizer Pressure, psia	2250	2250
Core Bypass, %	8.3	8.3
Reactor Coolant Temperatures, °F		
Core Outlet	625.4	625.4
Vessel Outlet	620.3	620.3
Core Average	592.7	592.7
Vessel Average	588.0*	588.0*
Vessel/Core Inlet	555.7	555.7
Steam Generator		
Steam Temperature, °F	547.5	538.3
Steam Pressure, psia	1024	953
Steam Flow (total), 10^6 lbm/hr	16.07	16.04
Feedwater Temperature, °F	446.6	446.6

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{AVG} window.

2.

Table 6.5.1-2

Byron/Braidwood Units 1 and 2

Initial Condition Assumptions for Thermal Uprate*

MSLB M&E Releases Inside Containment

Parameter	Value	
NSSS Power (% Nominal Uprated)	102	
RCS Average Temperature (°F)	597.1	
RCS Flowrate (gpm)	368,000	
RCS Pressure (psia)	2250	
Pressurizer Water Volume (ft^3)	1066.6	
Feedwater Enthalpy (Btu/lbm)	426.8 (Units 1)	426.7 (Units 2)
SG Water Level (% span)	65 (Units 1)	68.7 (Units 2)

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{AVG} window; the temperature includes the applicable calorimetric uncertainties.

Table 6.5.1-3
Byron/Braidwood Units 1 and 2
Protection System Actuation Signals and
Safety System Setpoints for Thermal Uprate Analysis

MSLB M&E Releases Inside Containment	
<u>Reactor Trip</u> 2/4 Low Pressurizer Pressure – 1857 psia	 Nominal value of 1900 psia with an uncertainty of 43 psi
<u>Safety Injection</u> 2/4 Low Pressurizer Pressure – 1715 psia 2/3 Low Steamline Pressure in any loop – 562 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 Containment Pressure High-1 – 6.8 psia	 Conservatively low value used Conservatively low value used Conservatively high value used
<u>Steamline Isolation</u> 2/3 Low Steamline Pressure in any loop – 562 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 Containment Pressure High-2 – 12.8 psia	 Conservatively low value used in the analysis of the DERs and the small breaks at power Conservatively high value used

4.

Table 6.5.1-4 Byron/Braidwood Units 1 (BWI Steam Generators) 1.1 ft^2 MSLB Hot Full Power With MSIV Failure Assumed Sequence of Events	
Time (sec)	**Event Description**
0.0	Main Steamline Break Occurs
1.8	Low Steamline Pressure Setpoint (562 psia)
3.8	Rod Motion Starts (Low Steamline Pressure actuates SI which initiates Reactor Trip)
8.8	Feedwater Isolation Occurs
9.8	Steamline Isolation Occurs (following receipt of the Low Steamline Pressure signal)
28.9	Safety Injection Initiated (following Low Steamline Pressure signal)
1810.	Mass and Energy Releases Terminate (SG Dryout)

Table 6.5.1-5 Byron/Braidwood Units 2 (D5 Steam Generators) 1.4 ft^2 MSLB Hot Full Power With MSIV Failure Assumed Sequence of Events	
Time (sec)	**Event Description**
0.0	Main Steamline Break Occurs
0.9	Low Steamline Pressure Setpoint (562 psia) Reached
2.9	Rod Motion Starts (Low Steamline Pressure actuates SI which initiates Reactor Trip)
7.9	Feedwater Isolation Occurs
8.9	Steamline Isolation Occurs (following receipt of the Low Steamline Pressure signal)
28.0	Safety Injection Initiated (following Low Steamline Pressure signal)
1808.	Transient Terminate (SG Dryout)

6.5.2 Main Steamline Break Mass and Energy Releases Outside Containment

6.5.2.1 Introduction

Steamline ruptures occurring outside the reactor containment structure may result in significant releases of high-energy fluid to the structures surrounding the steam systems. Superheated steam blowdowns following the steamline break have the potential to raise compartment temperatures outside containment. Early uncovery of the steam generator tube bundle maximizes the enthalpy of the superheated steam released. The impact of the steam releases depends on the plant configuration at the time of the break, plant response to the break, and the size and location of the break. Because of the interrelationship among many of the factors that influence steamline break mass and energy releases, an appropriate determination of a single limiting case with respect to mass and energy releases cannot be made. Therefore, it is necessary to analyze the steamline break event outside containment for a range of conditions.

6.5.2.2 Input Parameters and Assumptions

To determine the effects of plant power level and break area on mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated as part of the methodology development program documented in Reference 4. At plant power levels of 102 percent and 70 percent, various break sizes have been defined, ranging from 0.1 ft^2 to the full rupture of a main steamline.

A full break spectrum at both power levels (102 percent and 70 percent) has been analyzed at the uprated-power conditions for Byron/Braidwood Units 1 (BWI steam generators) and Byron/Braidwood Units 2 (Westinghouse Model D5 steam generators). Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

6.5.2.2.1 Initial Power Level

The initial power assumed for steamline break analyses outside containment affects the mass and energy releases and steam generator tube bundle uncovery in two ways. First, the steam generator mass inventory increases with decreasing power levels; this will tend to delay uncovery of the steam generator tube bundle, although the increased steam pressure at lower power levels will cause faster blowdown at the beginning of the transient. Second, the amount

of stored energy and decay heat, as well as feedwater temperature, are less for lower power levels; this will result in lower primary temperatures and less primary-to-secondary heat transfer during the steamline break event.

Therefore, the following power levels are analyzed.

- Full power - maximum allowable NSSS power plus uncertainty, i.e., 102 percent of rated power; and
- Near full-power - 70 percent of maximum allowable NSSS power.

For this Byron and Braidwood power-uprate analysis, the power levels and steamline break sizes are noted in subsection 6.5.2.3 of this report.

In general, plant initial conditions are assumed to be the nominal values corresponding to the initial power for that case, with appropriate uncertainties included. Table 6.5.2-1 lists nominal 100% power plant conditions. Table 6.5.2-2 lists initial plant condition assumptions for the cases analyzed.

Steamline break mass releases and superheated steam enthalpies assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, the calculated values of the superheated steam enthalpy available for release outside containment are larger than at the low end. The thermal design flowrate has been used for the RCS flow input. This is consistent with the assumptions documented in Reference 1 and with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties and RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the power-uprating program have been applied only to RCS average temperature (9.1°F), steam generator mass (5 percent narrow-range span), and power fraction (2 percent) at full power. Nominal values are adequate for the initial pressurizer pressure and water level. Uncertainty conditions are only applied to those parameters that could increase the enthalpy of superheated steam discharged out of the break.

6.5.2.2.2 Single-Failure Assumptions

The steamline break analyses outside containment assume two separate single failures consistent with the current licensing basis for Byron and Braidwood. The first single failure is one auxiliary feedwater (AF) pump resulting in minimum AF flow to the steam generators. Variations in AF flow can affect steamline break mass and energy releases in a number of ways including break mass flowrate, RCS temperature, tube bundle uncovery time and steam superheating. The minimum AF flow used in the analysis is conservatively based on only one motor-driven AF pump.

The second single failure is the main steamline isolation valve (MSIV) in the loop with the faulted steamline. This permits blowdown of the entire mass inventory of the steam generator in the loop with the faulted steamline, while allowing increased AF flow to all steam generators. This single failure is limited to the steamline with the postulated break. There is no single failure that would permit more than one steam generator to blow down through the pipe break.

6.5.2.2.3 Main Feedwater System

The rapid depressurization that typically occurs following a steamline rupture results in large amounts of water being added to the steam generators through the main feedwater system. However, main feedwater flow has been conservatively modeled by assuming no increase in feedwater flow in response to the increased steam flow following the steamline break. This minimizes total mass addition and the associated cooling effects in the steam generators, which causes the earliest onset of superheated steam released out of the break.

In general, isolation of main feedwater flow is conservatively assumed to be coincident with reactor trip, irrespective of the function that produced the trip signal. This assumption reduces the total mass addition to the steam generators. The main feedwater flow isolation valves are assumed to close instantaneously with no consideration of associated signal processing or valve stroke time.

However, for a subset of the steamline break spectrum, a modified feedwater isolation model has been assumed. To increase the amount of feedwater in the steam generators for the three smallest break sizes for the Units 2 analysis, a conservative value for the low RCS average

temperature setpoint is used with a minimum delay time associated only with signal processing. Receipt of this signal coincident with reactor trip actuates a main feedwater isolation signal. Main feedwater system assumptions used in the analysis are presented in Table 6.5.2-3.

6.5.2.2.4 Auxiliary Feedwater System

Generally, within the first few minutes following a steamline break, the AF system is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available to cover the tube bundle and reduce the amount of superheated steam produced. For this reason, AF flow is minimized while the actuation delay is maximized to accentuate depletion of the initial secondary-side inventory.

The volume of the AF piping up to the isolation valve closest to the steam generator is maximized and purging of the AF piping is assumed. This maximizes the amount of preheated water resident in the AF piping and ensures that this preheated water is injected into the steam generator first. The less dense resident auxiliary feedwater decreases initial mass addition to the faulted-loop steam generator. The large volume also delays the introduction of colder AF into any steam generator, which reduces the cooldown effect on the primary side of the RCS. Auxiliary feedwater system assumptions used in the analysis are presented in Table 6.5.2-3.

6.5.2.2.5 Steam Generator Fluid Mass

A minimum initial fluid mass in all steam generators has been used in each of the analyzed cases. This minimizes the availability of the heat sink afforded by the steam generators and leads to earlier tube bundle uncovery. The initial mass has been calculated as that corresponding to the programmed water level, minus 5 percent narrow-range span (NRS), minus a mass uncertainty. All steam generator fluid masses are calculated assuming 0 percent tube plugging. This assumption is conservative with respect to the RCS cooldown through the steam generators resulting from the steamline break.

6.5.2.2.6 Steam Generator Reverse Heat Transfer

Once steamline isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steamline via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam

generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.5.2.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the mass and energy releases resulting from the steamline ruptures analyzed.

6.5.2.2.8 Steamline Volume Blowdown

There is no contribution to mass and energy releases from the steam in the secondary plant loop piping and header because the initial volume is saturated steam. With the focus of the MSLB analysis outside containment on maximizing superheated steam enthalpy, it is presumed that the saturated steam in the loop piping and header has no adverse effects on the results. The blowdown of steam in this volume serves to delay the time of tube uncovery in the steam generators and is conservatively ignored.

6.5.2.2.9 Main Steamline Isolation

Steamline isolation is assumed to terminate blowdown from the intact-loop steam generators for the MSIV failure cases. For the AF failure cases, steamline isolation is assumed to terminate blowdown from all four steam generators. The main steamline isolation function is accomplished via the MSIV in each of the four steamlines. The MSIV actuation signal is received from a dynamically compensated low steamline pressure signal. A delay time of 8 seconds, accounting for delays associated with signal processing plus MSIV stroke time has been assumed. Unrestricted steam flow through the valve during valve stroke has been assumed.

6.5.2.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident outside containment include reactor trip, safety injection, steamline isolation, and auxiliary feedwater. The protection system actuation signals and associated setpoints that have been modeled in

the analysis are identified in Table 6.5.2-4. The setpoints used are conservative values with respect to the plant-specific values delineated in the Byron and Braidwood Technical Specifications.

At 102 percent power for break sizes 1.2 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AF initiation occurs as a result of the safety injection signal. For intermediate-size breaks, from 1.1 ft^2 to 0.4 ft^2, reactor trip is actuated following the Overpower ΔT signal; for break sizes smaller than 0.4 ft^2, reactor trip is actuated following the Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip, or shortly after reactor trip following the low RCS average temperature signal for the smallest break areas (see Section 6.5.2.2.3). Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure. Auxiliary feedwater flow is initiated following either the safety injection signal or the Low-Low Steam Generator Water Level signal.

At 70 percent power for break sizes 1.4 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AF initiation occurs as a result of the safety injection signal. For break sizes smaller than 1.4 ft^2, reactor trip and motor-driven AF initiation are actuated following a Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip, or shortly after reactor trip following the low RCS average temperature signal for the smallest break areas (see Section 6.5.2.2.3). Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure.

In all cases, the turbine stop valve is assumed to close instantly following the reactor trip signal.

6.5.2.2.11 Safety Injection System

Minimum Emergency Core Cooling System (ECCS) flowrates, corresponding to failure of one ECCS train, have been assumed in this analysis. Minimum ECCS flow is conservative since the reduced boron addition maximizes a return to power resulting from RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full safety injection flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis since the mass and energy releases would be reduced due to loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.5.2.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the reactor coolant temperature drops below that of the reactor coolant piping, reactor vessel, reactor coolant pumps, and steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat does not have a major impact on the calculated mass and energy releases, but the effects are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.5.2.2.13 Core Decay Heat

Core decay heat generation assumed in calculating steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 2).

6.5.2.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses. Rods in automatic control would step in prior to reactor trip due to the increase in steam flow. This would reduce nuclear power and core heat flux. However, sensitivity analyses performed when Reference 4 was written, investigating the effects on steamline break mass and energy releases of manual versus automatic rod control, have shown negligible impact on calculated results.

6.5.2.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize reactivity feedback effects resulting from the steamline break. This results in higher power generation should the reactor return to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.2.3 Description of Analysis

The system transient that provides the break flows and enthalpies of the steam release through the steamline break outside containment has been analyzed with the LOFTRAN (Reference 3) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage, and reverse steam generator heat transfer. The use of the LOFTRAN code for analysis of the MSLB with superheated steam M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 1), which has been reviewed and approved by the NRC for use in analyzing main steamline breaks, and in Reference 4 for MSLBs outside containment.

The Byron and Braidwood NSSS has been analyzed to determine the transient mass releases and associated superheated steam enthalpy values outside containment following a steamline break event. The resulting tables of mass flowrates and steam enthalpies are used as input conditions to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

The following licensing-basis cases of the MSLB outside containment have been analyzed at the noted conditions for the power-uprating program.

For Byron/Braidwood Units 1

- At 102 percent power, break sizes of 4.4, 2.0, 1.4, 1.2, 1.1, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2
- At 70 percent power, break sizes of 4.4, 2.0, 1.4, 1.2, 1.1, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

For Byron/Braidwood Units 2

- At 102 percent power, break sizes of 5.6, 2.0, 1.4, 1.2, 1.1, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

- At 70 percent power, break sizes of 5.6, 2.0, 1.4, 1.2, 1.1, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

Each MSLB outside containment is represented as a nonmechanistic split rupture (crack area). The largest break is postulated as a crack area equivalent to a single-ended pipe rupture. The break flowrate is limited by the total cross-sectional flow area of the four integral flow restrictors; the maximum break size is thus limited to 4.4 ft^2 (BWI steam generators) and 5.6 ft^2 (Westinghouse steam generators) rather than the actual pipe break area. Prior to steamline isolation, the break area is represented by the spectrum noted above. After steamline isolation, the break area is limited by the smaller of the integral steam generator flow restrictor (1.1 ft^2 for BWI steam generators and 1.4 ft^2 for Westinghouse steam generators) or the defined break size.

6.5.2.4 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release outside containment is based on an analysis that provides sufficient conservatism to ensure that the equipment qualification temperature envelope is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, break flow model, steamline and feedwater isolation, and main and auxiliary feedwater flow such that superheated steam resulting from tube bundle uncovery in the steam generators is accounted for and maximized. These assumptions have been included in this steamline break mass and energy release analysis as discussed in subsection 6.5.2.2 of this report. The tables of mass flowrates and steam enthalpy values for each of the steamline break cases analyzed are used as input to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

6.5.2.5 Results

Using the MSLB analysis methodology documented in Reference 4 as a basis, including parameter changes associated with the power uprate, the mass and energy release rates for each steamline break case analyzed have been developed for use in the environmental evaluation of safety-related electrical equipment in the main steam tunnel. Tables 6.5.2-5 through 6.5.2-12 provide the sequence of events for the various steamline break sizes for Byron and Braidwood Units 1 and Units 2, at 102 percent and 70 percent power, with the two separate single failures considered.

6.5.2.6 Conclusions

The mass releases and associated steam enthalpy values from the spectrum of steamline break cases outside containment have been analyzed at the conditions defined by the Byron and Braidwood power-uprate program. The assumptions delineated in subsection 6.5.2.2 have been included in the analysis such that conservative mass and energy releases are calculated. The resulting mass releases and associated steam enthalpy values have been provided for use in the environmental evaluation of safety-related electrical equipment outside containment in support of the Byron and Braidwood power-uprate program.

6.5.2.7 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

3. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

4. WCAP-10961, Rev. 1, (Proprietary), "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," October 1985.

6.

Table 6.5.2-1
Nominal Plant Parameters for Power Uprating*
(MSLB M&E Releases Inside and Outside Containment)

Nominal Conditions	Units 1	Units 2
NSSS Power, MWt	3600.6	3600.6
Core Power, MWt	3586.6	3586.6
Reactor Coolant Pump Heat, MWt	14	14
Reactor Coolant Flow (total), gpm TDF	368,000	368,000
Pressurizer Pressure, psia	2250	2250
Core Bypass, %	8.3	8.3
Reactor Coolant Vessel Average Temperature, °F	588.0*	588.0*
Steam Generator		
Steam Temperature, °F	547.5	538.8
Steam Pressure, psia	1024	953
Steam Flow, 10^6 lbm/hr (Plant Total)	16.07	16.04
Feedwater Temperature, °F	446.6	446.6
Zero-Load Temperature, °F	557	557

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

Table 6.5.2-2 Initial Condition Assumptions for Power Uprating* MSLB M&E Releases Outside Containment		
Initial Conditions	**102% Power**	**70% Power**
RCS Average Temperature (°F)	597.1*	587.8*
RCS Flowrate (gpm TDF)	368,000	368,000
RCS Pressure (psia)	2250	2250
Pressurizer Water Volume (ft^3)	1066.6	891.67
Feedwater Enthalpy (Btu/lbm) Units 1	426.8	381.4
Feedwater Enthalpy (Btu/lbm) Units 2	426.7	381.4
SG Pressure (psia)** Units 1	1106	1116
SG Pressure (psia)** Units 2	1027	1065
SG Water Level (% NRS) Units 1	55	55
SG Water Level (% NRS) Units 2	58.7	58.7

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

8.

Table 6.5.2-3

Main and Auxiliary Feedwater System Assumptions for Power Uprating

MSLB M&E Releases Outside Containment

Main Feedwater System	
Flowrate – both powers	nominal flow to all loops
Unisolable volume from SG nozzle to MFIV (all loops)	none assumed
Auxiliary Feedwater System	
Single-failure assumption, per steam generator	140 gpm
No single failure, per steam generator*	280 gpm
Manual isolation assumption	1200 seconds
Temperature (maximum value)	120°F
Piping volume (faulted loop) [Units 1]	160 ft^3
Piping volume (faulted loop) [Units 2]	60 ft^3
Actuation delay time	60 seconds

* For most steamline break cases analyzed, auxiliary feedwater is isolated 1200 seconds after event initiation. For a subset of the steamline break spectrum (the three smallest break sizes for the Unit 2 analysis), auxiliary feedwater is isolated 1200 seconds after reactor trip (the first indication to the operator that a transient is in progress).

9.

Table 6.5.2-4

Protection System Actuation Signals and Safety System Setpoints for Power Uprating MSLB M&E Releases Outside Containment

Reactor Trip			
2/4 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span*			
2/4 Low Pressurizer Pressure – 1857 psia			
2/4 Power-Range High Neutron Flux – 118% rated thermal power			
2/4 Overtemperature ΔT	K1 = 1.5	K2 = 0.0297	K3 = 0.00181
Dynamic compensation lead - 33 seconds lag - 4 seconds			
2/4 Overpower ΔT	K4 = 1.155	K5 = 0.0	K6 = 0.00245
Dynamic compensation rate lag - 10 seconds			
Safety Injection			
2/4 Low Pressurizer Pressure – 1715 psia			
2/3 Low Steamline Pressure in any loop – 503 psia [Units 1]; 379 psia [Units 2]*			
Dynamic compensation lead - 50 seconds lag - 5 seconds			
Steamline Isolation			
2/3 Low Steamline Pressure in any loop – 503 psia [Units 1]; 379 psia [Units 2]*			
Dynamic compensation lead - 50 seconds lag - 5 seconds			
Feedwater Isolation			
Reactor Trip (conservative assumption)*			
Auxiliary Feedwater Initiation			
2/4 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span*			
Safety Injection			

* For most steamline break cases analyzed, the noted values in this table have been used. For a subset of the steamline break spectrum (the three smallest break sizes for the Unit 2 analysis), modified inputs have been assumed to reduce the temperature in the steam tunnel caused by superheated steam releases. The assumptions for this subset of steamline breaks includes 28.6% NRS, main feedwater isolation on a low RCS average temperature coincident with reactor trip, and a low steam pressure setpoint of 503 psia.

Table 6.5.2-5 B/B Units 1 MSLB Outside Containment M&E Release Time Sequence Summary (AFW single failure) - 102% Power									
Power Level (%)	Break Size (ft2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Time of AFW Actuation (sec)	Steamline Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
102	4.4	LSP[1]	3.1	3.1	LSP	NA	NA	9.1	NA
102	2.0	LSP	4.6	4.6	LSP	NA	NA	10.6	NA
102	1.4	LSP	6.8	6.8	LSP	NA	NA	12.8	NA
102	1.2	LSP	10.8	10.8	LSP	NA	NA	16.8	NA
102	1.1	OPΔT[2]	15.0	15.0	LPP[3]	77.9	110.8	246.8	191.5
102	1.0	OPΔT	15.4	15.4	LPP	82.9	115.8	271.7	208.5
102	0.9	OPΔT	15.9	15.9	LPP	88.2	121.1	301.4	227.5
102	0.8	OPΔT	16.5	16.5	LPP	94.0	126.9	337.9	250.5
102	0.7	OPΔT	17.4	17.4	LPP	103.5	136.4	387.6	283.5
102	0.6	OPΔT	18.7	18.7	LPP	117.2	150.1	450.0	323.5
102	0.5	OPΔT	20.9	20.9	LPP	138.4	171.3	543.4	384.5
102	0.4	OPΔT	26.4	26.4	LPP	172.4	205.3	681.5	471.5
102	0.3	LSGWL[4]	254.5	254.5	LPP	447.1	312.5	806.1	490.5
102	0.2	LSGWL	372.9	372.9	LPP	687.8	430.9	1230.0	683.0
102	0.1	LSGWL	728.0	728.0	LPP	1762.8	786.0	1800. (M)[5]	1466.5

[1] LSP ≡ Low Steam Pressure signal
[2] OPΔT ≡ Overpower ΔT reactor trip signal
[3] LPP ≡ Low Pressurizer Pressure SI signal
[4] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[5] M ≡ Manual actuation

NA refers to not applicable since the noted function does not occur prior to steamline isolation. Once steamline isolation occurs, the break flowrates are terminated for these cases with the AFW single failure

Table 6.5.2-6 B/B Units 1 MSLB Outside Containment M&E Release Time Sequence Summary (MSIV single failure) - 102% Power									
Power Level (%)	**Break Size (ft^2)**	**Reactor Trip Signal**	**Time of Rod Motion (sec)**	**Feedwater Isolation (sec)**	**Safety Injection Signal**	**Time Safety Injection Starts (sec)**	**Time of AFW Actuation (sec)**	**Steamline Isolation (sec)**	**SG Tube Uncovery in Faulted SG (sec)**
102	4.4	LSP[6]	3.1	3.1	LSP	28.2	61.1	9.1	79.5
102	2.0	LSP	4.6	4.6	LSP	29.7	62.6	10.6	83.5
102	1.4	LSP	6.8	6.8	LSP	31.9	64.8	12.8	85.5
102	1.2	LSP	10.8	10.8	LSP	35.9	68.8	16.8	88.5
102	1.1	OPΔT[7]	15.0	15.0	LPP[8]	77.9	110.8	252.9	194.5
102	1.0	OPΔT	15.4	15.4	LPP	82.9	115.8	279.8	211.5
102	0.9	OPΔT	15.9	15.9	LPP	88.2	121.1	312.3	232.5
102	0.8	OPΔT	16.5	16.5	LPP	94.0	126.9	352.2	257.5
102	0.7	OPΔT	17.4	17.4	LPP	103.5	136.4	403.1	292.5
102	0.6	OPΔT	18.7	18.7	LPP	117.2	150.1	479.2	336.5
102	0.5	OPΔT	20.9	20.9	LPP	138.4	171.3	581.7	405.5
102	0.4	OPΔT	26.4	26.4	LPP	172.4	205.3	770.8	462.5
102	0.3	LSGWL[9]	254.5	254.5	LPP	445.5	312.5	951.4	532.5
102	0.2	LSGWL	372.9	372.9	LPP	682.1	430.9	1446.9	687.5
102	0.1	LSGWL	728.0	728.0	LPP	1617.5	786.0	1800. (M)[10]	1042.6

[6] LSP ≡ Low Steam Pressure signal
[7] OPΔT ≡ Overpower ΔT reactor trip signal
[8] LPP ≡ Low Pressurizer Pressure SI signal
[9] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[10] M ≡ Manual actuation

10.

Table 6.5.2-7 B/B Units 1 MSLB Outside Containment M&E Release Time Sequence Summary (AFW single failure) - 70% Power									
Power Level (%)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Time of AFW Actuation (sec)	Steamline Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
70	4.4	LSP[11]	3.0	3.0	LSP	NA	NA	9.0	NA
70	2.0	LSP	4.7	4.7	LSP	NA	NA	10.7	NA
70	1.4	LSP	6.9	6.9	LSP	NA	NA	12.9	NA
70	1.2	LSGWL[12]	75.2	75.2	LPP[13]	133.8	133.2	118.2	NA
70	1.1	LSGWL	80.8	80.8	LPP	142.7	138.8	245.9	183.5
70	1.0	LSGWL	87.5	87.5	LPP	154.2	145.5	271.0	199.5
70	0.9	LSGWL	95.6	95.6	LPP	166.7	153.6	299.0	215.5
70	0.8	LSGWL	105.7	105.7	LPP	183.7	163.7	335.6	237.5
70	0.7	LSGWL	118.6	118.6	LPP	205.4	176.6	382.4	265.5
70	0.6	LSGWL	135.9	135.9	LPP	234.6	193.9	445.1	303.5
70	0.5	LSGWL	159.8	159.8	LPP	275.4	217.8	534.5	356.5
70	0.4	LSGWL	195.8	195.8	LPP	337.5	253.8	671.4	436.5
70	0.3	LSGWL	255.5	255.5	LPP	441.0	313.5	907.2	571.0
70	0.2	LSGWL	374.8	374.8	LPP	664.4	432.8	1366.3	850.0
70	0.1	LSGWL	732.1	732.1	LPP	1517.4	790.1	1800. (M)[14]	NA

NA refers to not applicable since the noted function does not occur prior to steamline isolation. Once steamline isolation occurs, the break flowrates are terminated for these cases with the AFW single failure.

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[11] LSP ≡ Low Steam Pressure signal
[12] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[13] LPP ≡ Low Pressurizer Pressure SI signal
[14] M ≡ Manual actuation

Table 6.5.2-8 B/B Units 1 MSLB Outside Containment M&E Release Time Sequence Summary (MSIV single failure) - 70% Power									
Power Level (%)	**Break Size (ft^2)**	**Reactor Trip Signal**	**Time of Rod Motion (sec)**	**Feedwater Isolation (sec)**	**Safety Injection Signal**	**Time Safety Injection Starts (sec)**	**Time of AFW Actuation (sec)**	**Steamline Isolation (sec)**	**SG Tube Uncovery in Faulted SG (sec)**
70	4.4	LSP[15]	3.0	3.0	LSP	28.1	61.0	9.0	85.5
70	2.0	LSP	4.7	4.7	LSP	29.8	62.7	10.7	90.5
70	1.4	LSP	6.9	6.9	LSP	32.0	64.9	12.9	93.5
70	1.2	LSGWL[16]	75.2	75.2	LPP[17]	133.8	133.2	118.2	147.5
70	1.1	LSGWL	80.8	80.8	LPP	142.7	138.8	250.5	183.5
70	1.0	LSGWL	87.5	87.5	LPP	154.2	145.5	277.9	200.5
70	0.9	LSGWL	95.6	95.6	LPP	166.7	153.6	307.5	217.5
70	0.8	LSGWL	105.7	105.7	LPP	183.7	163.7	347.5	239.5
70	0.7	LSGWL	118.6	118.6	LPP	205.4	176.6	399.0	270.5
70	0.6	LSGWL	135.9	135.9	LPP	234.5	193.9	463.4	309.5
70	0.5	LSGWL	159.8	159.8	LPP	275.0	217.8	574.8	368.5
70	0.4	LSGWL	195.8	195.8	LPP	336.6	253.8	743.3	459.5
70	0.3	LSGWL	255.5	255.5	LPP	438.8	313.5	1086.6	570.0
70	0.2	LSGWL	374.8	374.8	LPP	657.4	432.8	1578.3	689.5
70	0.1	LSGWL	732.0	732.0	LPP	1383.8	790.0	1800. (M)[18]	NA

NA refers to not applicable since the noted function does not occur prior to steamline isolation. Once isolation occurs, the transient is over for this case.

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[15] LSP ≡ Low Steam Pressure signal
[16] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[17] LPP ≡ Low Pressurizer Pressure SI signal
[18] M ≡ Manual actuation

Table 6.5.2-9 B/B Units 2 MSLB Outside Containment M&E Release Time Sequence Summary (AFW single failure) - 102% Power									
Power Level (%)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Time of AFW Actuation (sec)	Steamline Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
102	5.6	LSP[19]	3.1	3.1	LSP	NA	NA	9.1	NA
102	2.0	LSP	4.3	4.3	LSP	NA	NA	10.3	NA
102	1.4	LSP	6.1	6.1	LSP	NA	NA	12.1	NA
102	1.2	LSP	8.3	8.3	LSP	NA	NA	14.3	NA
102	1.1	OPΔT[20]	14.3	14.3	LPP[21]	76.9	109.8	149.5	105.5
102	1.0	OPΔT	14.6	14.6	LPP	80.7	113.6	162.3	112.5
102	0.9	OPΔT	15.0	15.0	LPP	85.5	118.4	180.9	119.5
102	0.8	OPΔT	15.6	15.6	LPP	91.6	124.5	202.6	125.5
102	0.7	OPΔT	16.3	16.3	LPP	99.9	132.8	232.4	133.5
102	0.6	OPΔT	17.5	17.5	LPP	113.4	146.3	269.1	147.5
102	0.5	OPΔT	19.4	19.4	LPP	132.8	165.7	321.0	166.5
102	0.4	OPΔT	24.5	24.5	LPP	164.7	197.1	395.8	198.5
102	0.3	LSGWL[22]	196.9	250.1	LPP	400.7	254.9	470.5	204.5
102	0.2	LSGWL	280.7	375.1	LPP	608.3	338.7	757.9	288.5
102	0.1	LSGWL	531.3	814.2	LPP	1341.2	589.3	1800.(M)[23]	536.6

NA refers to not applicable since the noted function does not occur prior to steamline isolation. Once steamline isolation occurs, the break flowrates are terminated for these cases with the AFW single failure.

[19] LSP ≡ Low Steam Pressure signal
[20] OPΔT ≡ Overpower ΔT reactor trip signal
[21] LPP ≡ Low Pressurizer Pressure SI signal
[22] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[23] M ≡ Manual actuation

Table 6.5.2-10 B/B Units 2 MSLB Outside Containment M&E Release Time Sequence Summary (MSIV single failure) - 102% Power									
Power Level (%)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Time of AFW Actuation (sec)	Steamline Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
102	5.6	LSP[24]	3.1	3.1	LSP	28.2	61.1	9.1	37.5
102	2.0	LSP	4.3	4.3	LSP	29.4	62.3	10.3	42.5
102	1.4	LSP	6.1	6.1	LSP	31.2	64.1	12.1	43.5
102	1.2	LSP	8.3	8.3	LSP	33.4	66.3	14.3	48.5
102	1.1	OPΔT[25]	14.3	14.3	LPP[26]	76.9	109.8	151.5	105.5
102	1.0	OPΔT	14.6	14.6	LPP	80.7	113.6	165.9	112.5
102	0.9	OPΔT	15.0	15.0	LPP	85.5	118.4	186.6	119.5
102	0.8	OPΔT	15.6	15.6	LPP	91.6	124.5	208.3	125.5
102	0.7	OPΔT	16.3	16.3	LPP	99.9	132.8	238.5	133.5
102	0.6	OPΔT	17.5	17.5	LPP	113.4	146.3	281.3	147.5
102	0.5	OPΔT	19.4	19.4	LPP	132.8	165.7	344.3	166.5
102	0.4	OPΔT	24.5	24.5	LPP	164.7	197.1	439.0	198.5
102	0.3	LSGWL[27]	196.9	250.1	LPP	393.4	254.9	550.2	204.5
102	0.2	LSGWL	280.7	373.9	LPP	565.1	338.7	1049.5	288.5
102	0.1	LSGWL	531.3	753.0	LPP	1131.0	589.3	1800. (M)[28]	536.3

[24] LSP ≡ Low Steam Pressure signal
[25] OPΔT ≡ Overpower ΔT reactor trip signal
[26] LPP ≡ Low Pressurizer Pressure SI signal
[27] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[28] M ≡ Manual actuation

Table 6.5.2-11
B/B Units 2 MSLB Outside Containment M&E Release
Time Sequence Summary (AFW single failure) - 70% Power

Power Level (%)	Break Size (ft2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Time of AFW Actuation (sec)	Steamline Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
70	5.6	LSP[29]	3.0	3.0	LSP	NA	NA	9.0	NA
70	2.0	LSP	4.7	4.7	LSP	NA	NA	10.7	NA
70	1.4	LSP	7.0	7.0	LSP	NA	NA	13.0	NA
70	1.2	LSGWL[30]	102.6	102.6	LPP[31]	161.7	160.6	148.7	127.5
70	1.1	LSGWL	109.9	109.9	LPP	172.9	167.9	163.2	136.5
70	1.0	LSGWL	118.8	118.8	LPP	185.8	176.8	184.6	146.5
70	0.9	LSGWL	129.5	129.5	LPP	202.3	187.5	209.3	160.5
70	0.8	LSGWL	142.9	142.9	LPP	222.2	200.9	233.4	174.5
70	0.7	LSGWL	160.0	160.0	LPP	247.5	218.0	270.2	196.5
70	0.6	LSGWL	182.2	182.2	LPP	282.7	240.8	308.9	220.5
70	0.5	LSGWL	214.7	214.7	LPP	330.0	272.7	368.6	257.5
70	0.4	LSGWL	262.3	262.3	LPP	410.2	320.3	465.3	314.5
70	0.3	LSGWL	251.4	268.9	LPP	440.2	309.4	648.5	332.5
70	0.2	LSGWL	358.1	389.4	LPP	651.8	416.1	1027.0	495.0
70	0.1	LSGWL	676.7	784.5	LPP	1351.1	734.7	1800. (M)[32]	927.2

NA refers to not applicable since the noted function does not occur prior to steamline isolation. Once steamline isolation occurs, the break flowrates are terminated for these cases with the AFW single failure.

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[29] LSP ≡ Low Steam Pressure signal
[30] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[31] LPP ≡ Low Pressurizer Pressure SI signal
[32] M ≡ Manual actuation

Table 6.5.2-12 B/B Units 2 MSLB Outside Containment M&E Release Time Sequence Summary (MSIV single failure) - 70% Power									
Power Level (%)	**Break Size (ft^2)**	**Reactor Trip Signal**	**Time of Rod Motion (sec)**	**Feedwater Isolation (sec)**	**Safety Injection Signal**	**Time Safety Injection Starts (sec)**	**Time of AFW Actuation (sec)**	**Steamline Isolation (sec)**	**SG Tube Uncovery in Faulted SG (sec)**
70	5.6	LSP[33]	3.0	3.0	LSP	28.1	61.0	9.0	56.5
70	2.0	LSP	4.7	4.7	LSP	29.8	62.7	10.7	61.5
70	1.4	LSP	7.0	7.0	LSP	32.1	65.0	13.0	64.5
70	1.2	LSGWL[34]	102.6	102.6	LPP[35]	161.7	160.6	148.7	127.5
70	1.1	LSGWL	109.9	109.9	LPP	172.9	167.9	163.2	136.5
70	1.0	LSGWL	118.8	118.8	LPP	185.8	176.8	184.6	146.5
70	0.9	LSGWL	129.5	129.5	LPP	202.3	187.5	207.0	160.5
70	0.8	LSGWL	142.9	142.9	LPP	222.2	200.9	236.6	174.5
70	0.7	LSGWL	160.0	160.0	LPP	247.7	218.0	277.8	196.5
70	0.6	LSGWL	182.8	182.8	LPP	282.4	240.8	317.5	220.5
70	0.5	LSGWL	214.7	214.7	LPP	330.2	272.7	379.8	257.5
70	0.4	LSGWL	262.3	262.3	LPP	407.9	320.3	502.2	314.5
70	0.3	LSGWL	251.4	268.9	LPP	437.1	309.4	743.7	334.5
70	0.2	LSGWL	358.1	389.4	LPP	627.5	416.1	1415.5	512.5
70	0.1	LSGWL	676.7	779.7	LPP	1208.6	734.7	1800. (M)[36]	831.0

[33] LSP ≡ Low Steam Pressure signal
[34] LSGWL ≡ Low-Low Steam Generator Water Level reactor trip signal
[35] LPP ≡ Low Pressurizer Pressure SI signal
[36] M ≡ Manual actuation

6.5.3 Steam Releases for Radiological Dose Analysis

The vented steam releases have been calculated for the Locked Rotor and Steamline Break events. The following table summarizes the vented steam releases from the operable steam generators as well as auxiliary feedwater flows for the 0-2 hour time period, the 2-8 hour time period and the 8-40 hour time period for each of these events.

Event	Vented Steam Release			Feedwater Flow		
	0-2 hours	**2-8 hours**	**8-40 hours**	**0-2 hours**	**2-8 hours**	**8-40 hours**
Locked Rotor	719,000 lbm	1,109,000 lbm	2,664,000 lbm	867,000 lbm	1,131,000 lbm	2,664,000 lbm
Steamline Break	442,000 lbm	977,000 lbm	2,216,000 lbm	553,000 lbm	993,000 lbm	2,216,000 lbm

For the Steamline Break and Locked Rotor with PORV Failure events, additional steam is released through blowing down the faulted steam generator. This additional total steam mass is 167,000 lbm.

No explicit assumption is considered in these analyses regarding steam generator blowdown system isolation. The implied assumption is that the entire inventory of the steam generators is released to the environment and no loss of inventory through the blowdown line is accounted for. This provides a conservative calculation of the quantity of steam vented during the noted time periods.

The steam releases discussed in this section have been provided as inputs to the radiological dose analyses in support of the Byron and Braidwood core power uprating.

6.5.4 Steamline Break Containment Response Evaluation

6.5.4.1 Introduction

The Byron Units 1 and 2 and Braidwood Units 1 and 2 containment systems are designed such that for all steamline break (SLB) break sizes, up to and including the double-ended severance of a steamline, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical steamline break. The containment response analysis uses the long-term mass and energy release data from Section 6.5.1.

6.5.4.2 Input Parameters and Assumptions

An analysis of containment response to the rupture of a steamline must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis as shown in Table 6.5.4-1. Separate cases were analyzed to maximize containment temperature or to maximize containment pressure, and the initial containment pressure was adjusted to be conservative for each case. The initial relative humidity was set at 20% for all cases, which was shown to be conservative for either peak pressure or peak temperature with plant-specific sensitivity cases.

Also, values for the Refueling Water Storage Tank (RWST) temperature have been specified, along with Containment Spray (CS) pump flowrate and Reactor Containment Fan Cooler (RCFC) heat removal performance. These values are chosen conservatively, as shown in Table 6.5.4-1 and Table 6.5.4-2. The heat sink modeling is specified in Table 6.5.4-3 and Table 6.5.4-4, and is consistent with the values used for the LOCA containment response analysis, as documented in Section 6.4.3.

A series of cases were performed for the SLB containment response. Section 6.5.1 documents the mass and energy (M&E) releases for the uprated full power cases for the BWI and D5 steam generators. The M&E release analyses include cases with no single failure, an MSIV failure or an FIV failure, as discussed in Section 6.5.1.2. For cases in which a single failure is included in the M&E release analysis, no single failure is modeled in the containment response

analysis. For cases with no single failure modeled in the M&E release analysis, the worst single failure in the containment safeguards is assumed. With offsite power available, the limiting single failure in the containment safeguards system is the failure of one RCFC train that results in the loss of 2 RCFC units. Plant-specific sensitivity analyses have shown that failure of one containment spray pump is less limiting than the failure of one RCFC train.

6.5.4.3 Description of Analysis

Calculation of containment pressure and temperature is accomplished by use of the computer code COCO (Reference 1). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for a particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure and temperature transients for previous Byron/Braidwood containment response analyses.

6.5.4.4 Acceptance Criteria

The containment response for a design basis steamline break is an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan (SRP) Section 6.2.1.1.A for long-term containment response, the relevant General Design Criteria (GDC) requirements are listed below.

<u>General Design Criterion 16, Containment Design</u>

To satisfy the requirements of GDC 16, the peak calculated containment pressure must be less than the containment design pressure of 50 psig at Byron and Braidwood.

<u>General Design Criterion 38, Containment Heat Removal</u>

To satisfy the criterion of GDC 38, the calculated pressure at 24 hours must be less than 50% of the peak calculated value.

In addition to the above General Design Criteria, the containment temperature and pressure transients should not exceed the profiles used for equipment qualification.

6.5.4.5 Analysis Results

The peak containment pressure and peak containment temperature are summarized in Tables 6.5.4-5 and 6.5.4-6 for each of the uprated full power cases with offsite power available. Furthermore, the composite pressure transients and composite temperature transients for the uprated cases are compared to the previous composite curves from all limiting cases. Figures 6.5.4-1 and 6.5.4-2 show that the peak containment pressure transient for all of the uprated cases remains bounded at all points in time by the previously-defined composite. A similar comparison is made in Figures 6.5.4-3 and 6.5.4-4 for the containment temperature transient results. While the previously-defined temperature composites bound most of the uprated containment temperature transients, some differences in the timing of the mass and energy release causes some shifts in the timing of the containment temperature transient. The highest peak pressures and temperatures previously calculated are not exceeded by the uprate analysis. Therefore, the peak pressures are 39.3 psig and 38.3 psig for Unit 1 and Unit 2, respectively. The peak temperatures are 333°F and 331°F for Unit 1 and Unit 2, respectively.

6.5.4.6 Conclusions

An evaluation of the steamline break containment pressure and temperature response has been performed as part of the Byron Units 1 and 2 and Braidwood Units 1 and 2 uprate program. The analyses included long-term pressure and temperature profiles for each case, including analyses with mass and energy releases based on the BWI replacement steam generator and the Westinghouse D5 steam generator. The analyzed cases result in a peak containment pressure that is less than the containment design pressure of 50 psig. The long-term pressures are well below 50% of the peak value within 24 hours. Based on these results, the GDC criteria for Byron Units 1 and 2 and Braidwood Units 1 and 2 have been met.

6.5.4.7 References

1. "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974 (Proprietary), WCAP-8326, July 1974 (Non-Proprietary).

12.

Table 6.5.4-1
SLB Containment Response Analysis Initial Containment Conditions and Parameters

RWST water temperature (°F)	120
Initial containment temperature (°F)	120
Initial containment pressure (psia)	
Minimum (Peak Temperature Cases)	14.6
Maximum (Peak Pressure Cases)	15.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.758×10^6
Number of Containment Air Recirculation Fan Coolers	
All containment safeguards	4
Limiting containment safeguards single failure	2
Start of Containment Fan Coolers	
Containment Hi-1 setpoint (psig)	6.8
Delay time from Hi-1 setpoint with offsite power (sec)	27.0
Number of Containment Spray Pumps	
All containment safeguards	2
Limiting containment safeguards single failure	2
Containment Spray Flowrate, total (gpm)	7080
Start of Containment Spray	
Containment Hi-3 setpoint (psig)	24.8
Delay time from Hi-3 setpoint with offsite power (sec)	53.1

13.

Table 6.5.4-2 Reactor Containment Fan Cooler Performance	
Containment Temperature (°F)	**Heat Removal Rate [Btu/sec] Per RCFC**
100	0.00
110	893.54
130	3181.61
160	8057.82
190	14535.02
220	21896.92
250	29430.06
271	34613.85
300	41813.19
350	54225.24

14.

Table 6.5.4-3
Containment Heat Sinks

No.	Description	Material	Thickness (ft)	Surface Area (ft^2)
1	Containment Cylinder Wall	Paint Carbon Steel Concrete	8.30E-04 0.0208 0.75	72,741
2	Containment Dome	Paint Carbon Steel Concrete	8.30E-04 0.0208 0.75	17,550
3	Unlined Concrete – Combined from Containment Floor and the Slab at 425'-0"	Concrete (in contact w/water)	0.75	16,037
4	Lined Concrete – Combined from Containment Floor and Reactor Pool Wall	Stainless Steel Concrete	0.0415 0.75	848
5	Unlined Concrete – Combined from Reactor Cavity, Outside Reactor Wall, and Reactor Pool Wall	Concrete	1.0	4,803
6	Lined Concrete – Secondary Wall	Paint Carbon Steel Concrete	8.30E-04 0.0766 0.75	7,702
7	Lined Concrete – Slab at 425'-0", lining only	Paint Carbon Steel	8.30E-04 0.0625	422.3
8	Unlined Concrete – Combined from Slabs on Steel Beams at El. 412'-0" and 426'-0", Instrument Access Tunnel, and enclosures for SGs, RCPs	Concrete	0.75	69,541
9	Lined Concrete – Slabs on Steel Beams at El. 412'-0" and 426'-0"	Paint Carbon Steel Concrete	8.30E-04 0.004 0.75	3,852
10	Lined Concrete – Enclosures for SGs, RCPs	Paint Carbon Steel Concrete	8.30E-04 0.0710 0.75	1,570

15.

Table 6.5.4-3 (cont.) Containment Heat Sinks				
No.	**Description**	**Material**	**Thickness (ft)**	**Surface Area (ft^2)**
11	Lined Concrete – Refueling Cavity	Stainless Steel Concrete	0.0690 0.75	2,129
12	Miscellaneous Steel Plate, HVAC Hangers, Polar Crane Trolley and Bridge Plates, and NSSS Supports	Paint Carbon Steel	8.30E-04 0.0416	19,791
13	Miscellaneous Steel Plate, Grating, Press. Relief Tank, Polar Crane Bridge Plates, and Return Air Riser	Paint Carbon Steel	8.30E-04 0.0210	94,670
14	Polar Crane Trolley and Bridge Plates and Machinery	Paint Carbon Steel	8.30E-04 0.0760	14,089
15	Combined from Man. Crane Fan, RCFC Fan, and Reactor Cavity Fans	Paint Carbon Steel	8.30E-04 0.0400	21,875
16	Combined from CCFU* Housing, HVAC Hangers, Uninsulated Pipe, Ductwork, and Duct Supports	Paint Carbon Steel	8.30E-04 0.0150	22,528
17	Cable/Conduit Trays	Paint Carbon Steel	8.30E-04 0.0104	27,095
18	Combined from Cable/Conduit Tray Supports, Junction Boxes, and IFME*	Paint Carbon Steel	8.30E-04 8.20E-03	6,385
19	Combined from CFU*, and Miscellaneous Steel Beams and Columns	Paint Carbon Steel	8.30E-04 0.0157	69,856
20	Lined Concrete – Combined from the Instrument Access Tunnel, Reactor Cavity, and Inside Reactor Pool	Stainless Steel Concrete	0.0165 0.75	9,291

*CCFU – Containment Charcoal Filter Unit

*IFME – Incore Flux Mapping Equipment

*CFU – Charcoal Filter Unit

16.

Table 6.5.4-4

Thermophysical Properties of Containment Heat Sinks

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Paint	0.30	28.00
Carbon Steel	27.00	58.80
Stainless Steel	9.00	53.70
Concrete	0.92	22.62

17.

Table 6.5.4-5 Peak Containment Pressures and Temperatures for Byron/Braidwood Unit 1 With Off-Site Power Available			
Break	**Single Failure**	**Peak Pressure (psig) @ Time (sec)**	**Peak Temperature (°F) @ Time (sec)**
Full DER	RCFC	33.6 @ 167	251.5 @ 11
Full DER	FIV	34.4 @ 204	251.5 @ 11
Small DER	RCFC	29.7 @ 167	308.3 @ 67
Small DER	FIV	30.1 @ 231	300.8 @ 36
Small DER	MSIV	30.4 @ 188	327.3 @ 34
Split	RCFC	32.2 @ 217	318.0 @ 69
Split	FIV	32.7 @ 257	308.9 @ 50
Split	MSIV	33.7 @ 252	309.3 @ 50

Table 6.5.4-6 Peak Containment Pressures and Temperatures for Byron/Braidwood Unit 2 With Off-Site Power Available			
Break	**Single Failure**	**Peak Pressure (psig) @ Time (sec)**	**Peak Temperature (°F) @ Time (sec)**
Full DER	RCFC	29.8 @ 124	251.3 @ 11
Full DER	FIV	31.4 @ 150	251.3 @ 11
Small DER	RCFC	28.7 @ 160	303.6 @ 69
Small DER	FIV	27.5 @ 202	296.5 @ 35
Small DER	MSIV	27.8 @ 133	323.3 @ 35
Split	RCFC	28.7 @ 222	307.0 @ 75
Split	FIV	29.5 @ 269	297.7 @ 55
Split	MSIV	30.3 @ 259	297.9 @ 56

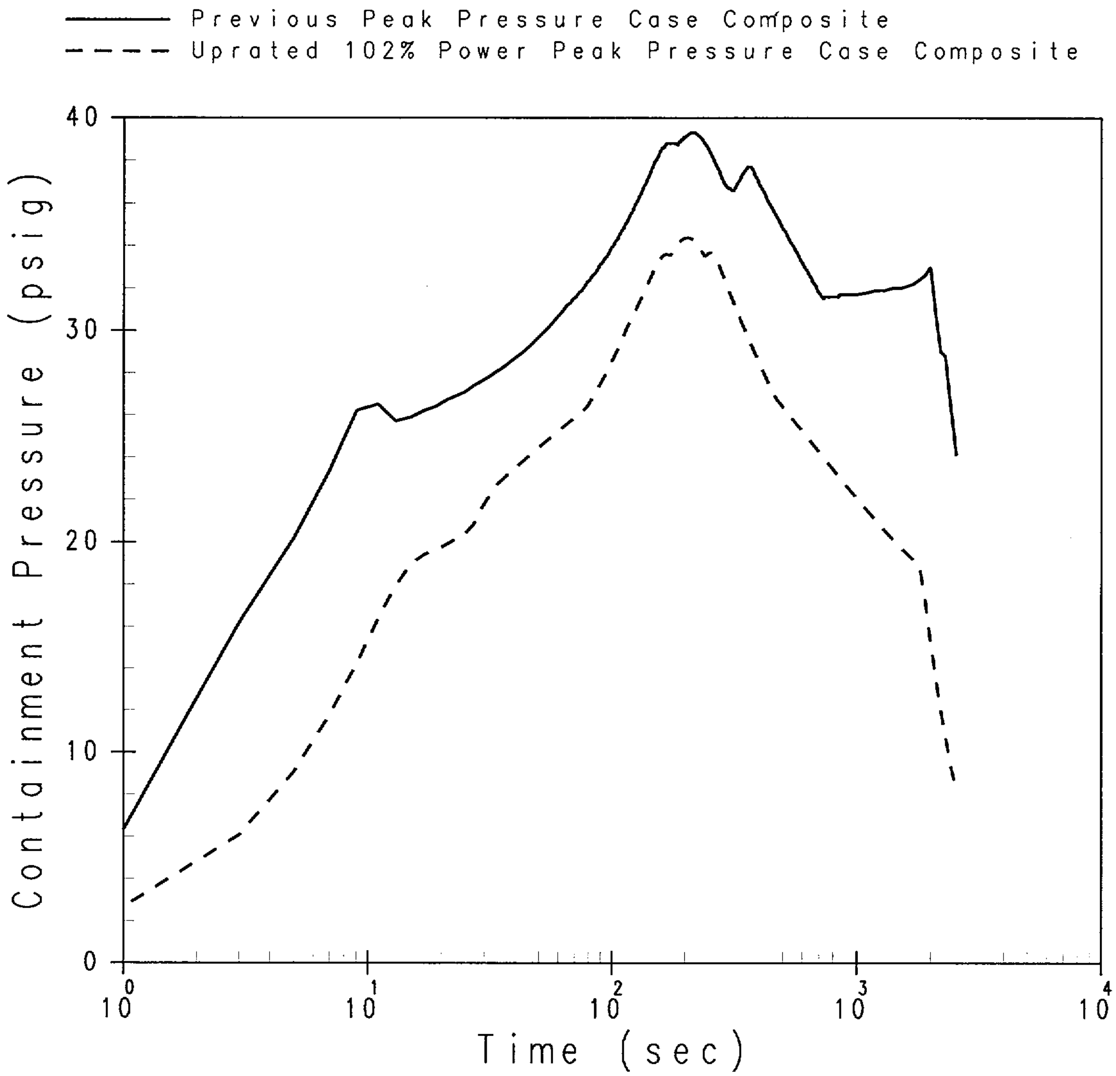

Figure 6.5.4-1

Containment Pressure Composite Curve for Steamline Break

Byron/Braidwood Unit 1 (BWI Steam Generators)

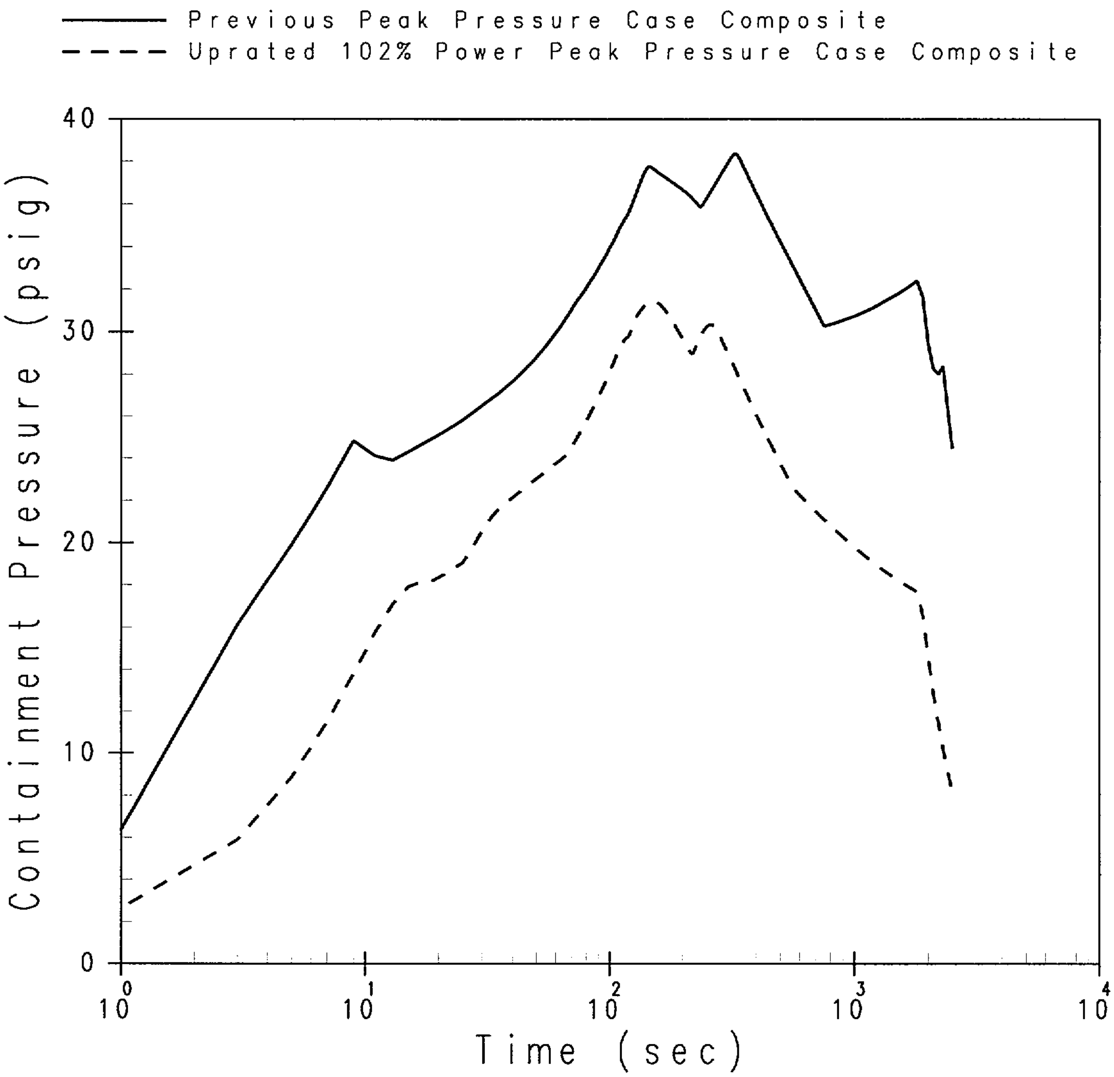

Figure 6.5.4-2

Containment Pressure Composite Curve for Steamline Break

Byron/Braidwood Unit 2 (D5 Steam Generators)

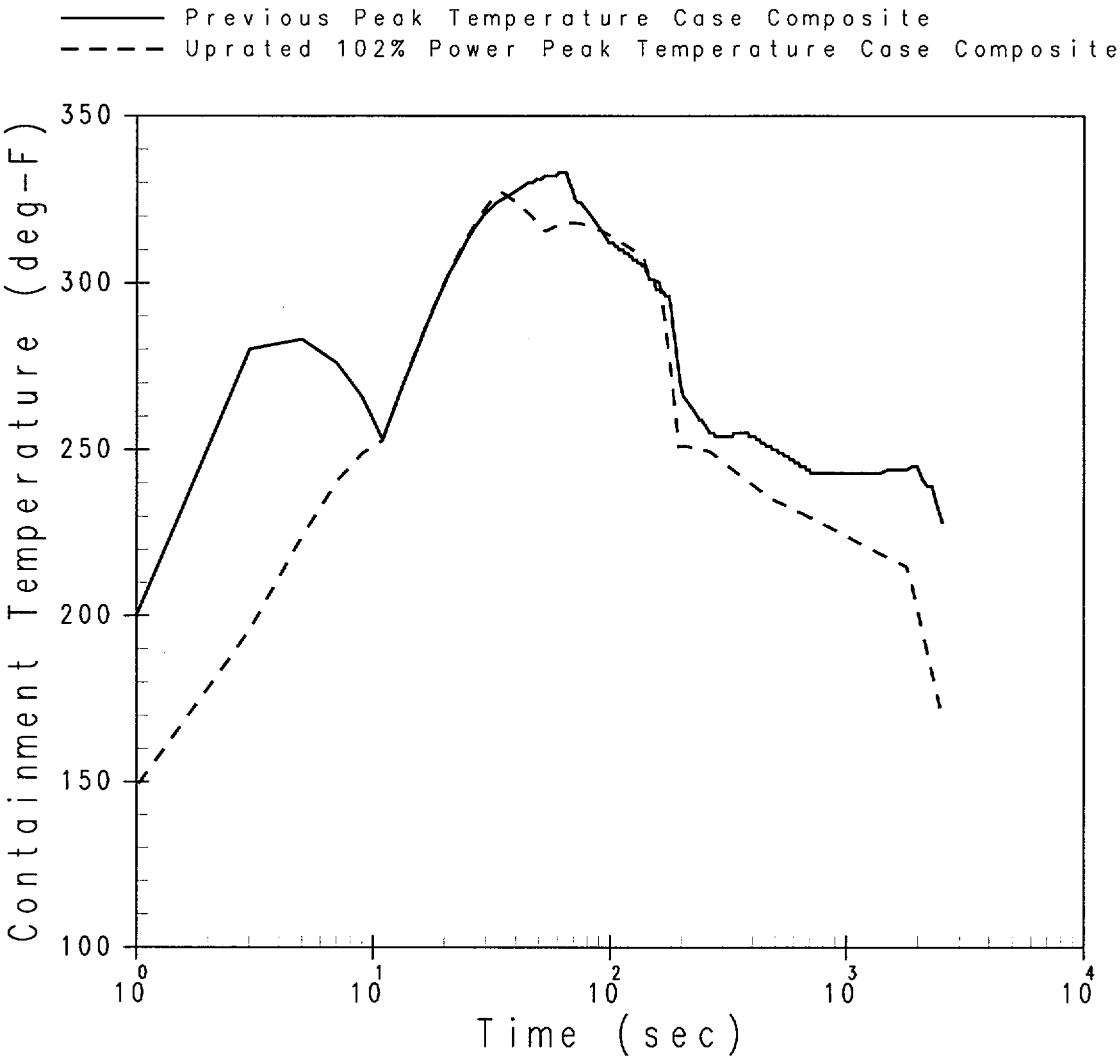

Figure 6.5.4-3

Containment Temperature Composite Curve for Steamline Break

Byron/Braidwood Unit 1 (BWI Steam Generators)

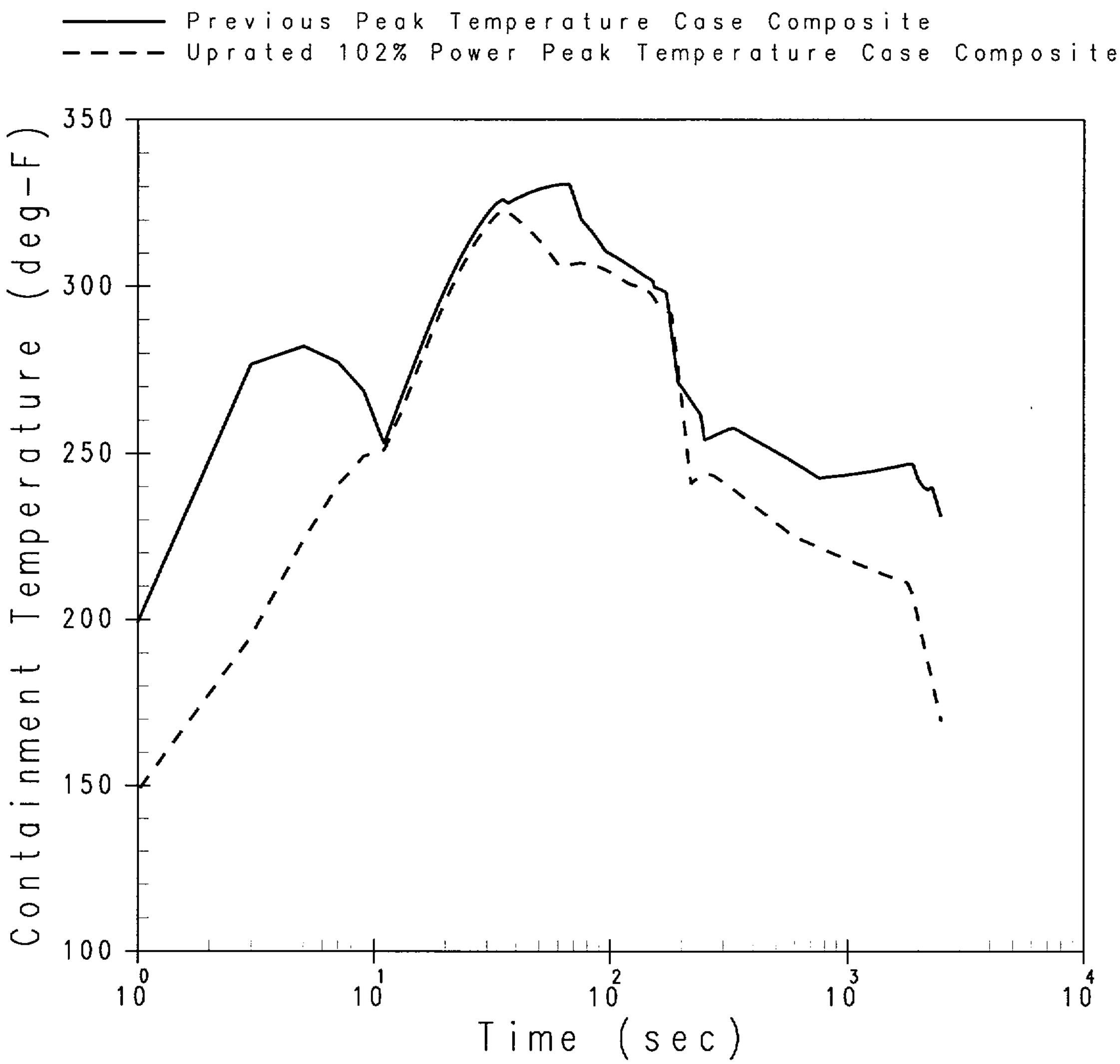

Figure 6.5.4-4

Containment Temperature Composite Curve for Steamline Break

Byron/Braidwood Unit 2 (D5 Steam Generators)

6.5.5 Main Steamline Break Outside Containment Compartment Response

6.5.5.1 Introduction

This section of the report presents the results of a study to determine the effects of superheated steam releases, during postulated main steamline ruptures, on outside containment equipment qualification for Commonwealth Edison Byron Nuclear Generating Station Units 1 and 2 and Braidwood Nuclear Generating Station Units 1 and 2. For this study, the compartment temperature profiles for the steam tunnel main steam valve room were calculated as required by 10 CFR 50.49.

NRC IE Information Notice 84-90, "Main Steam Line Break Effect on Environmental Qualification of Equipment," informed licensees of potential problems related to the release of superheated steam following a postulated main steamline break. Specifically, such superheated blowdowns have the potential to raise the compartment temperatures, and, therefore, the equipment surface and internal temperatures, above those originally used for the environmental qualification of such equipment needed to mitigate the consequences of high energy line breaks.

The report describes the methods and assumptions used in modeling the Byron and Braidwood Units 1 and 2 compartments in the steam tunnel region. The mass and energy releases from the postulated main steamline breaks were presented in Section 6.5.2. The results from these calculated compartment temperature profiles will be presented here.

6.5.5.2 Input Parameters and Assumptions

This study involved calculation of main steamline break mass and energy releases (see Section 6.5.2) and then calculations of the outside containment compartment temperatures resulting from those releases. The Reactor Coolant System (RCS) conditions used for determining the steamline break mass and energy releases were described in Section 6.5.2.2. The input modeling for the actual compartment was provided by ComEd in the form of a RELAP4 model (References 1 and 2).

Based on the information provided in References 1 and 2, the steam tunnel and safety valve room regions for Byron Units 1 & 2 and Braidwood Units 1 & 2 will be assumed to be very similar and, therefore, would only require one COMPACT input model.

The compartment sizes and initial conditions used for input to the COMPACT code (Reference 3) are detailed in Table 6.5.5-1. The seven flow areas in the steam tunnel model are listed in Table 6.5.5-2.

The RELAP4 model had 7 conductors (heat sinks), so the same number will be used for COMPACT. The area of the left surface, area of right surface, and total heat sink volume are provided. The right side is connected to the environment node (i.e., Node 1).

The full area of the inside surface will be used in COMPACT. The heat transfer coefficient stated was 2 Btu/hr-ft^2-°F on the inside (left surface) and 0.1 Btu/hr-ft^2-°F on the outside (right surface). The thickness for each heat sink is provided in Table 1, page 11 of Reference 1, as 2 feet. The heat sink areas used in the analysis are listed in Table 6.5.5-3.

Concrete is the only material type provided. The volumetric heat capacity is given as 22.62 Btu/ft-ft^3 at 100°F. The values used for the specific heat and the density were 0.156 Btu/lbm-°F and 145 lbm/ft^3, respectively. The thermal conductivity of the heat sink was 0.92 Btu/ft-hr-°F. These values were used over the entire range of transient temperatures.

6.5.5.3 Description of Analysis/Evaluation

This analysis of the steam tunnel and safety valve region was performed with the COMPACT code (Reference 3). COMPACT has been found to be acceptable for use in determining the compartment response for outside containment high energy line break transients for equipment qualification (Reference 4).

The mass and energy releases for 60 cases for Byron Unit 1 and Braidwood Unit 1 (BWI steam generators) were provided by the analysis in Section 6.5.2 at the uprated conditions, as were 60 separate cases for Byron Unit 2 and Braidwood Unit 2 (D5 steam generators).

The compartment temperatures for these 120 transients were determined with the COMPACT code and the model described in Section 6.5.5.2 for a transient time of 5000 seconds. Since

the mass and energy releases for all 120 cases terminate prior to 2500 seconds, this duration is sufficient for the compartment temperatures to decrease significantly from the peak values.

6.5.5.4 Acceptance Criteria

The acceptance criteria for the outside containment compartment temperature evaluation is documented in Section 9.3.2.1.

6.5.5.5 Results

The mass and energy releases for 60 cases for Byron Unit 1 and Braidwood Unit 1 (BWI steam generators) were provided by the analysis in Section 6.5.2 at the uprated conditions, as were 60 separate cases for Byron Unit 2 and Braidwood Unit 2 (D5 steam generators).

Tables 6.5.5-4 through 6.5.5-7 show the compartment peak temperature results for Byron/Braidwood Units 1. The peak temperature prior to steamline isolation (SLI) is seen to be 396.0°F from Case 70-C with a MSIV failure (which is a 0.3 ft^2 break case) and the overall peak compartment temperature is 518.4°F from Case 102-L with a MSIV failure (which is a 1.2 ft^2 break case).

Tables 6.5.5-8 through 6.5.5-11 show the compartment peak temperature results for Byron/Braidwood Units 2. Case 70-D with a MSIV failure (which is a 0.4 ft^2 break case) yields the peak temperature prior to steamline isolation of 413.5°F. The overall peak compartment temperature was 502.5°F for Case 102-M with a MSIV failure (which is a 1.4 ft^2 break case).

The COMPACT Temperature Profiles for Node 2 from the Byron and Braidwood Units 1 and 2 Power Uprating for the outside containment steamline break analysis are shown in Figures 6.5.5-1 through 6.5.5-30. Each figure contains the two failure cases at two power levels for the BWI Steam Generator (SG) and the D5 SG. The case names and numbers can be cross referenced to Tables 6.5.5-4 through 6.5.5-11. Node 2 is presented because the break is postulated to occur in that region.

A small sensitivity was performed on the compartment temperature when considering the maximum Reactor Coolant Pump heat. This sensitivity determined that the peaks prior to SLI had a sensitivity of +3.0°F and the overall peak showed a sensitivity of +5.0°F. ***Please note***

that the results above and in Tables 6.5.5-4 through 6.5.5-11 include these temperature penalties but the results provided in Figures 6.5.5-1 through 6.5.5-30 DO NOT include the penalties.

6.5.5.6 Conclusions

The results show that the BWI generator is more limiting for the overall peak compartment temperature outside the containment than the original Westinghouse Model D5 at the uprated conditions by approximately 16°F. However, the results show that the D5 is more limiting for the temperatures prior to steamline isolation (SLI) by 16.5°F.

Based on the information provided in Section 9.3.2.1, the results presented in Table 6.5.5-4 through Table 6.5.5-11 are within the acceptance criteria.

6.5.5.7 References

1. Com Ed Document RSA-B-92-05, Rev. 1, "Byron Long Term Temperature Profiles for Main Steamline Outside Containment," 4-27-93

2. Com Ed Document PSA-B-96-02, Rev. 1, "Steam Tunnel Analysis of Steam Line Break Outside Containment," 6-20-97

3. WCAP-10361, Revision 1, "COMPACT: Compartmentalized Analysis of Containment Transients, A High Speed Computer Code for Multi-Compartment Transients Analysis," June 1987 (Proprietary).

4. Youngblood, B. J. of the NRC, "Transmittal of Draft Copy of the Evaluation on Equipment Qualification Under Superheat Conditions for Sequoyah Units 1 & 2," letter to S.A. White, Tennessee Valley Authority, November 25, 1986.

18.

Table 6.5.5-1 Compartment Sizes and Initial Conditions				
	Volume (ft^3)	**Initial Pressure (psia)**	**Initial Temperature (°F)**	**Relative Humidity (%)**
Node 1 Atmosphere	1.0E + 10	14.7	122.0	30
Node 2 Valve Room 2nd Quadrant	21,805.4	14.7	122.0	30
Node 3 Main Steam Tunnel 2nd Quadrant	13,695.0	14.7	122.0	30
Node 4 Main Steam Tunnel	34,865.0	14.7	122.0	30
Node 5 Main Steam Tunnel 1st Quadrant	35,016.0	14.7	122.0	30
Node 6 Valve Room 1st Quadrant	21,805.0	14.7	122.0	30
Node 7 Main Steam Tunnel 1st Quadrant	17,388.4	14.7	122.0	30
Node 8 Main Steam Tunnel 1st Quadrant	13,529.9	14.7	122.0	30

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Table 6.5.5-2 Steam Tunnel Model Flow Areas				
Flow Area	Upstream Node	Downstream Node	Flow Area (ft^2)	Form Loss Coefficient
1	2	3	146.0	1.5685
2	3	4	199.8	2.1860
3	4	5	199.8	2.7530
4	5	6	146.0	1.5685
5	5	7	373.0	2.2600
6	7	8	270.8	2.2600
7	8	1	270.8	5000.0*

* Flow Area #7 is the only link in the model to the outside atmosphere and the magnitude of loss coefficient indicates that this flow path is closed. No rupture disk or blowout panel is evident.

Table 6.5.5-3 Steam Tunnel Heat Sink Areas						
Heat Sink	Left Node	Right Node	Height (ft)	Width (ft)	Length (ft)	Area (ft^2)
1	2	1	36.33	20.45	29.35	3332.986
2	3	1	19	16.88	43.2	3100.032
3	4	1	19	10.68	171.75	10195.08
4	5	1	20	21.6	81.06	6744.192
5	6	1	36.33	20.45	29.35	3332.986
6	7	1	29	14.9	40.25	3533.95
7	8	1	19	14.74	48.32	3260.634

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Table 6.5.5-4
Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with AFW Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	AFW	102	0.1	1800.0	328.7	330.7	1800.0
B	AFW	102	0.2	1230.0	391.9	395.4	1230.0
C	AFW	102	0.3	806.1	387.1	390.4	807.1
D	AFW	102	0.4	681.5	383.1	387.5	682.5
E	AFW	102	0.5	543.4	383.4	386.4	543.5
F	AFW	102	0.6	450.0	382.8	386.3	450.8
G	AFW	102	0.7	387.6	382.5	385.9	388.9
H	AFW	102	0.8	337.9	381.6	384.7	338.5
I	AFW	102	0.9	301.4	380.3	383.1	301.7
J	AFW	102	1.0	271.7	379.4	383.3	272.7
K	AFW	102	1.1	246.8	379.5	382.6	247.5
L	AFW	102	1.2	16.8	303.3	305.3	10.4
M	AFW	102	1.4	12.8	301.2	303.2	6.9
N	AFW	102	2.0	10.6	301.8	303.8	4.7
O	AFW	102	4.4	9.1	309.3	312.1	9.6

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Table 6.5.5-5 Results for Byron/Braidwood Unit 1 Outside Containment Cases from 102% Power with MSIV Failure							
Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	MSIV	102	0.1	1800.0	325.5	472.5	1974.0
B	MSIV	102	0.2	1446.9	390.2	467.7	1494.0
C	MSIV	102	0.3	951.4	392.2	477.2	999.0
D	MSIV	102	0.4	770.8	391.3	482.7	820.2
E	MSIV	102	0.5	581.7	388.0	488.6	613.5
F	MSIV	102	0.6	479.2	383.8	492.0	508.5
G	MSIV	102	0.7	403.1	380.1	494.2	429.3
H	MSIV	102	0.8	352.2	381.9	496.6	373.9
I	MSIV	102	0.9	312.3	380.9	498.5	332.5
J	MSIV	102	1.0	279.8	380.2	500.2	299.7
K	MSIV	102	1.1	252.9	378.6	501.7	271.5
L	MSIV	102	1.2	16.8	303.3	518.4	1200.2
M	MSIV	102	1.4	12.8	301.2	518.2	1199.9
N	MSIV	102	2.0	10.6	301.8	517.7	1201.5
O	MSIV	102	4.4	9.1	309.3	516.5	1200.0

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Table 6.5.5-6
Results for Byron/Braidwood Unit 1 Outside Containment Cases from 70% Power with AFW Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	AFW	70	0.1	1800.0	308.8	310.8	1799.0
B	AFW	70	0.2	1366.3	388.0	393.0	1366.5
C	AFW	70	0.3	907.2	388.7	393.2	907.5
D	AFW	70	0.4	671.4	387.0	390.5	672.2
E	AFW	70	0.5	534.5	387.0	389.1	535.5
F	AFW	70	0.6	445.1	386.8	389.0	446.5
G	AFW	70	0.7	382.4	385.9	387.9	381.9
H	AFW	70	0.8	335.6	384.7	388.5	336.1
I	AFW	70	0.9	299.0	383.8	387.2	299.5
J	AFW	70	1.0	271.0	382.2	385.8	271.5
K	AFW	70	1.1	245.9	381.7	385.2	246.5
L	AFW	70	1.2	118.2	316.5	318.6	119.1
M	AFW	70	1.4	12.9	300.7	302.7	7.0
N	AFW	70	2.0	10.7	301.1	303.1	4.5
O	AFW	70	4.4	9.0	312.4	315.0	9.6

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Table 6.5.5-7
Results for Byron/Braidwood Unit 1 Outside Containment Cases from 70% Power with MSIV Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	MSIV	70	0.1	1800.0	310.7	459.1	2061.0
B	MSIV	70	0.2	1578.3	392.1	469.9	1623.6
C	MSIV	70	0.3	1086.6	396.0	479.1	1142.0
D	MSIV	70	0.4	743.3	392.2	485.6	782.5
E	MSIV	70	0.5	547.8	372.3	490.9	609.5
F	MSIV	70	0.6	463.4	383.8	493.9	494.5
G	MSIV	70	0.7	399.0	386.5	496.4	423.7
H	MSIV	70	0.8	347.5	385.9	498.5	370.9
I	MSIV	70	0.9	307.5	384.3	500.2	329.5
J	MSIV	70	1.0	277.9	383.5	501.7	298.9
K	MSIV	70	1.1	250.5	381.8	502.8	270.7
L	MSIV	70	1.2	118.2	316.3	501.0	183.5
M	MSIV	70	1.4	12.9	300.7	499.4	626.5
N	MSIV	70	2.0	10.7	301.1	499.1	126.5
O	MSIV	70	4.4	9.0	312.4	498.5	122.1

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Table 6.5.5-8
Results for Byron/Braidwood Unit 2 Outside Containment Cases from 102% Power with AFW Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	AFW	102	0.1	1800.0	343.2	343.2	1660.3
B	AFW	102	0.2	757.9	396.3	396.6	758.1
C	AFW	102	0.3	470.5	385.6	385.6	470.2
D	AFW	102	0.4	395.8	404.2	408.2	396.6
E	AFW	102	0.5	321.0	400.7	404.4	322.0
F	AFW	102	0.6	269.1	398.5	401.9	269.5
G	AFW	102	0.7	232.4	396.5	399.4	232.9
H	AFW	102	0.8	202.6	394.5	398.8	203.9
I	AFW	102	0.9	180.9	392.4	396.5	182.1
J	AFW	102	1.0	162.3	388.4	392.1	162.6
K	AFW	102	1.1	149.5	387.7	389.7	149.5
L	AFW	102	1.2	14.3	308.6	310.6	8.0
M	AFW	102	1.4	12.1	308.1	310.1	6.1
N	AFW	102	2.0	10.3	308.2	310.2	4.0
O	AFW	102	5.6	9.1	318.2	320.6	9.5

25.

Table 6.5.5-9
Results for Byron/Braidwood Unit 2 Outside Containment Cases from 102% Power with MSIV Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	MSIV	102	0.1	1800.0	314.9	446.8	2039.4
B	MSIV	102	0.2	1049.5	395.1	453.7	1133.9
C	MSIV	102	0.3	550.2	389.3	462.7	611.4
D	MSIV	102	0.4	439.0	408.9	472.3	479.5
E	MSIV	102	0.5	344.3	403.8	476.3	376.5
F	MSIV	102	0.6	281.3	399.0	478.8	309.5
G	MSIV	102	0.7	238.5	392.9	480.6	266.5
H	MSIV	102	0.8	208.3	395.7	482.2	232.3
I	MSIV	102	0.9	186.6	393.4	482.5	210.5
J	MSIV	102	1.0	165.9	388.4	484.0	186.8
K	MSIV	102	1.1	151.5	386.6	484.7	170.9
L	MSIV	102	1.2	14.3	308.2	502.2	1200.9
M	MSIV	102	1.4	12.1	308.0	502.5	1200.3
N	MSIV	102	2.0	10.3	308.2	501.6	1200.2
O	MSIV	102	5.6	9.1	318.1	499.8	1197.5

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Table 6.5.5-10
Results for Byron/Braidwood Unit 2 Outside Containment Cases from 70% Power with AFW Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	AFW	70	0.1	1800.0	324.0	324.0	1799.2
B	AFW	70	0.2	1027.0	389.4	390.3	1027.5
C	AFW	70	0.3	648.5	386.6	386.6	647.9
D	AFW	70	0.4	465.3	403.0	406.5	465.5
E	AFW	70	0.5	368.6	395.6	400.2	369.3
F	AFW	70	0.6	308.9	392.2	396.3	309.7
G	AFW	70	0.7	270.2	388.4	393.1	270.5
H	AFW	70	0.8	233.4	380.1	385.1	233.5
I	AFW	70	0.9	209.3	374.0	379.0	209.3
J	AFW	70	1.0	184.6	365.5	370.8	186.5
K	AFW	70	1.1	163.2	353.2	357.3	163.9
L	AFW	70	1.2	148.7	347.4	351.4	150.1
M	AFW	70	1.4	13.0	306.2	308.2	7.3
N	AFW	70	2.0	10.7	306.2	308.2	4.7
O	AFW	70	5.6	9.0	319.5	321.5	8.9

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Table 6.5.5-11
Results for Byron/Braidwood Unit 2 Outside Containment Cases from 70% Power with MSIV Failure

Case	Failure	Power Level (%)	Break Size (ft^2)	Steamline Isolation (sec)	Peak Steam Temp @ or Before SLI (°F)	Peak Steam Temperature (°F)	Time of Peak (sec)
A	MSIV	70	0.1	1800.0	313.0	446.7	2163.6
B	MSIV	70	0.2	1415.5	397.1	455.1	1509.3
C	MSIV	70	0.3	743.7	388.5	463.5	804.0
D	MSIV	70	0.4	502.2	413.5	474.6	539.3
E	MSIV	70	0.5	379.8	396.6	478.6	412.9
F	MSIV	70	0.6	317.5	395.3	481.7	345.3
G	MSIV	70	0.7	277.8	393.3	483.7	304.5
H	MSIV	70	0.8	236.6	383.6	484.5	261.9
I	MSIV	70	0.9	207.0	369.7	485.4	232.8
J	MSIV	70	1.0	184.6	367.4	486.2	210.3
K	MSIV	70	1.1	163.2	352.5	486.7	188.5
L	MSIV	70	1.2	148.7	345.9	487.2	172.9
M	MSIV	70	1.4	13.0	306.2	481.4	514.3
N	MSIV	70	2.0	10.7	306.2	481.1	502.1
O	MSIV	70	5.6	9.0	319.5	481.2	86.3

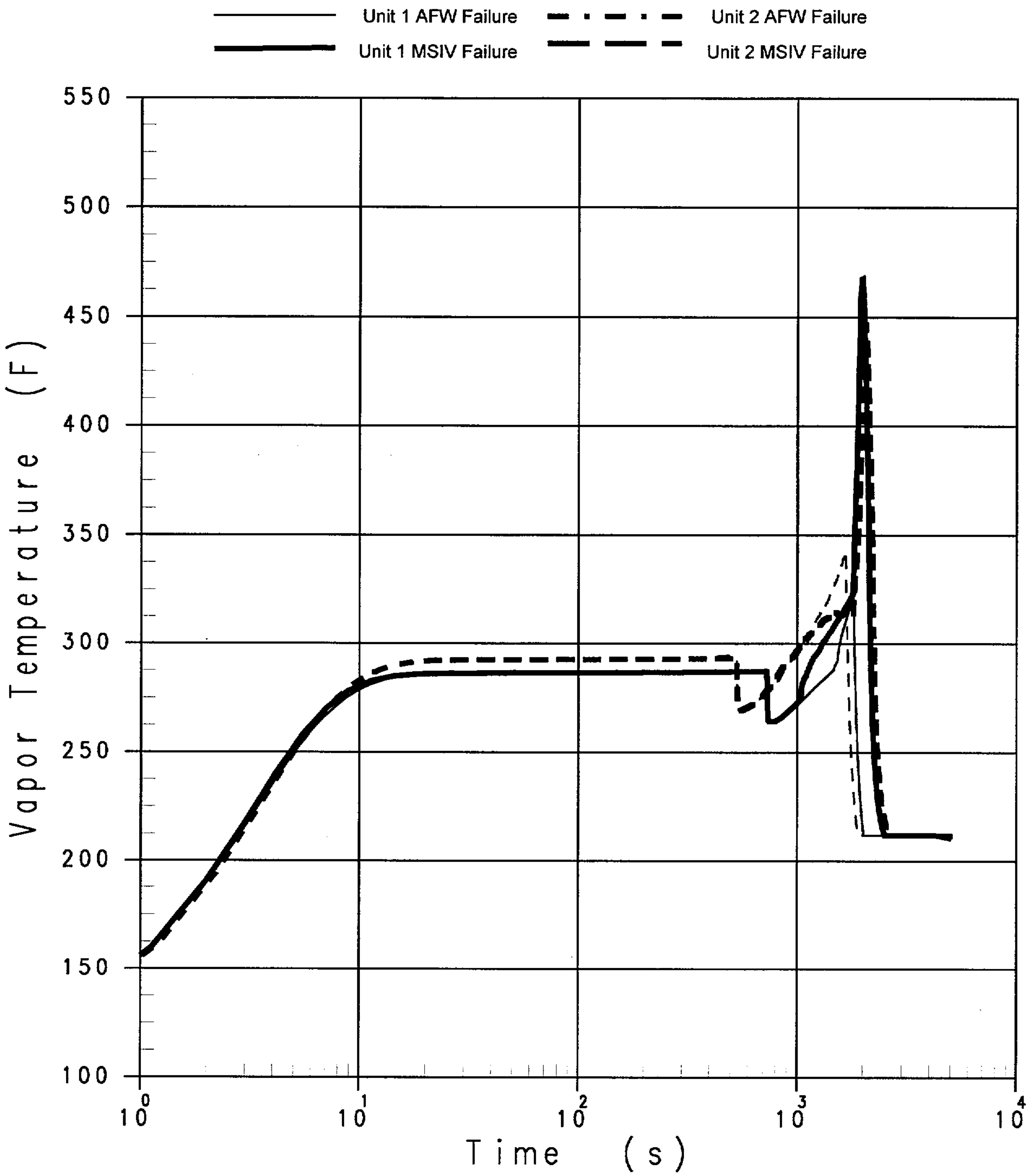


Figure 6.5.5-1

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-A Vapor Temperature for Steam Tunnel Node 2

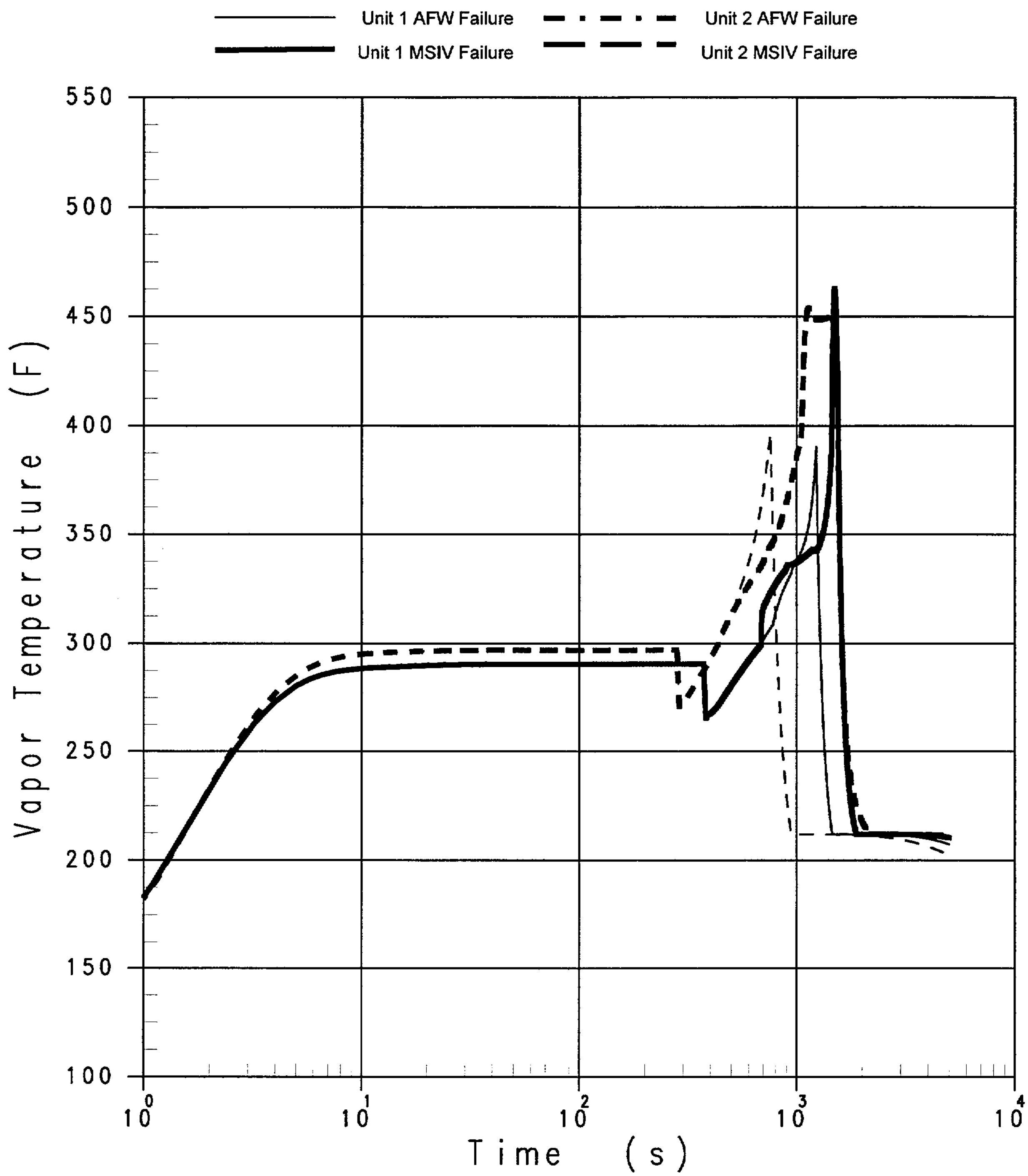


Figure 6.5.5-2

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-B

Vapor Temperature for Steam Tunnel Node 2

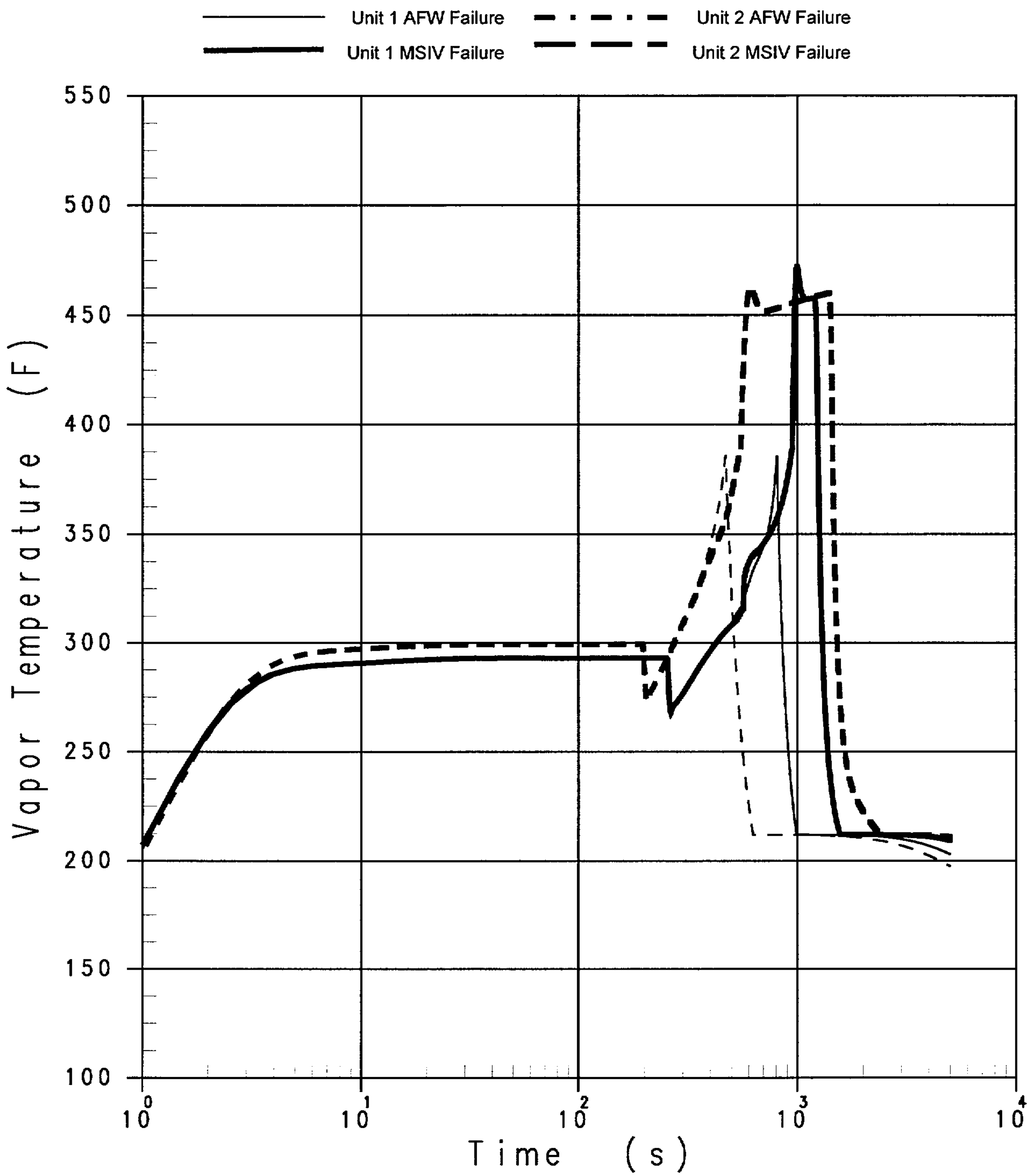


Figure 6.5.5-3

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-C

Vapor Temperature for Steam Tunnel Node 2

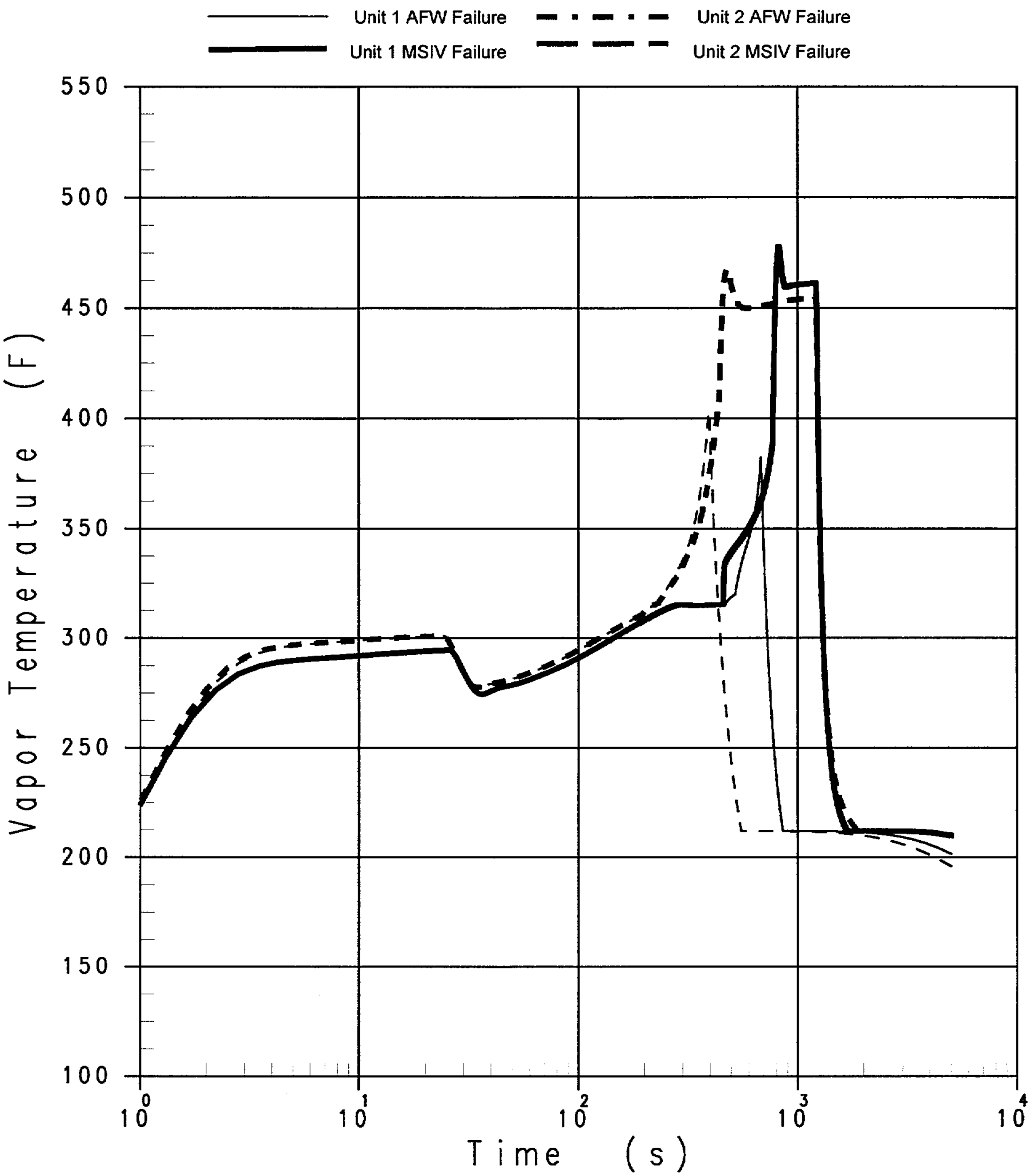


Figure 6.5.5-4

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-D

Vapor Temperature for Steam Tunnel Node 2

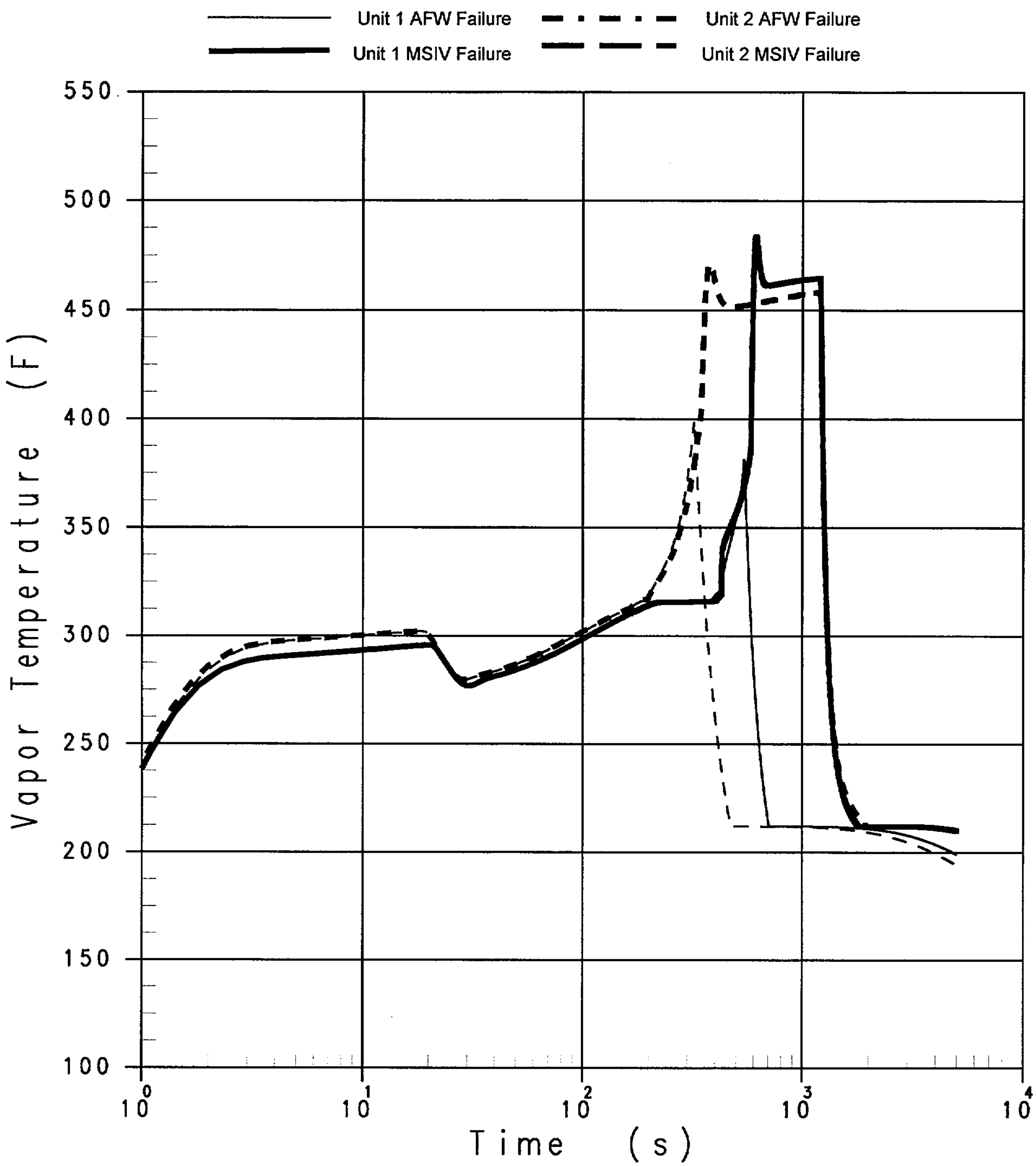


Figure 6.5.5-5

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-E

Vapor Temperature for Steam Tunnel Node 2

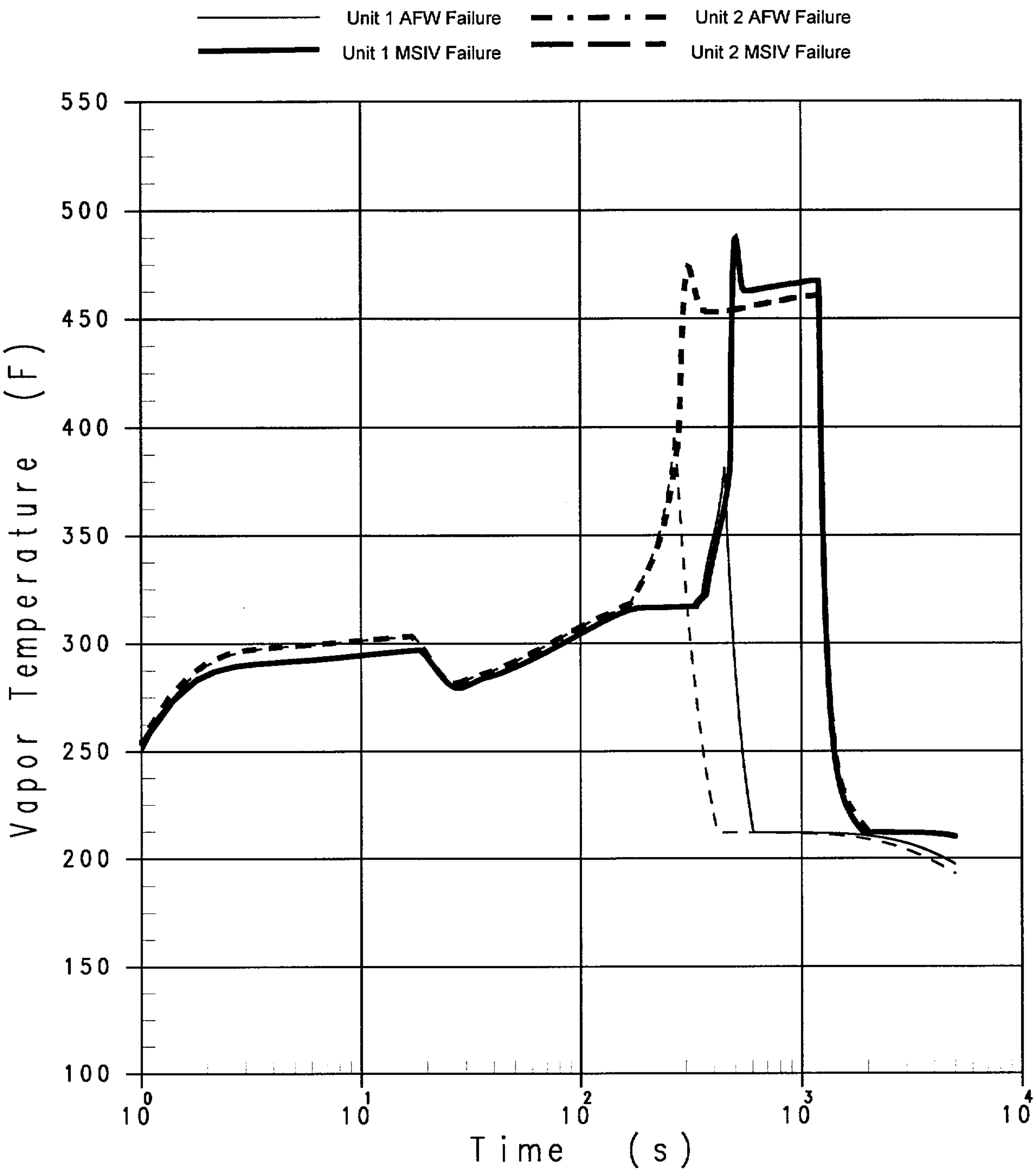


Figure 6.5.5-6

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-F

Vapor Temperature for Steam Tunnel Node 2

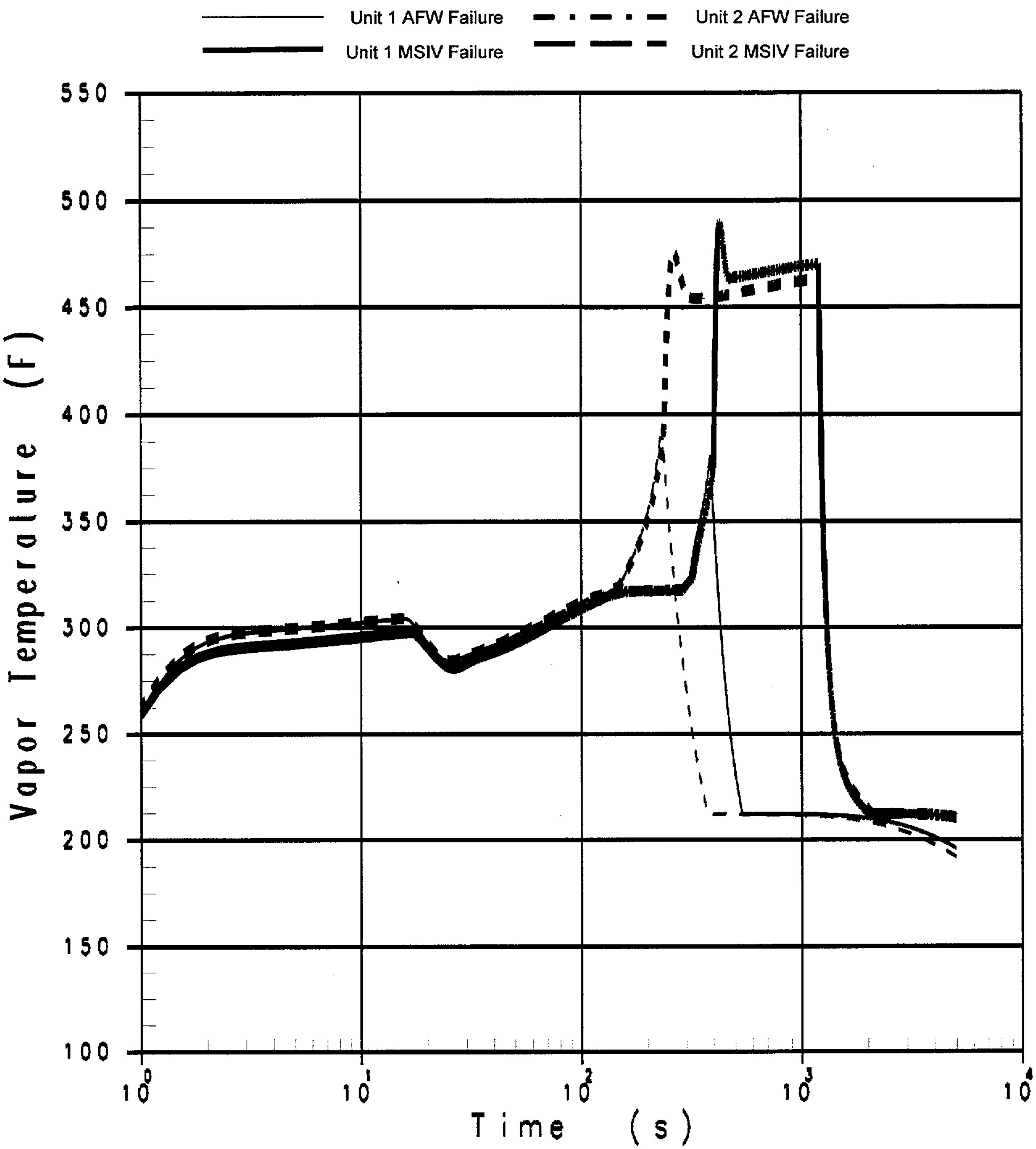


Figure 6.5.5-7

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-G

Vapor Temperature for Steam Tunnel Node 2

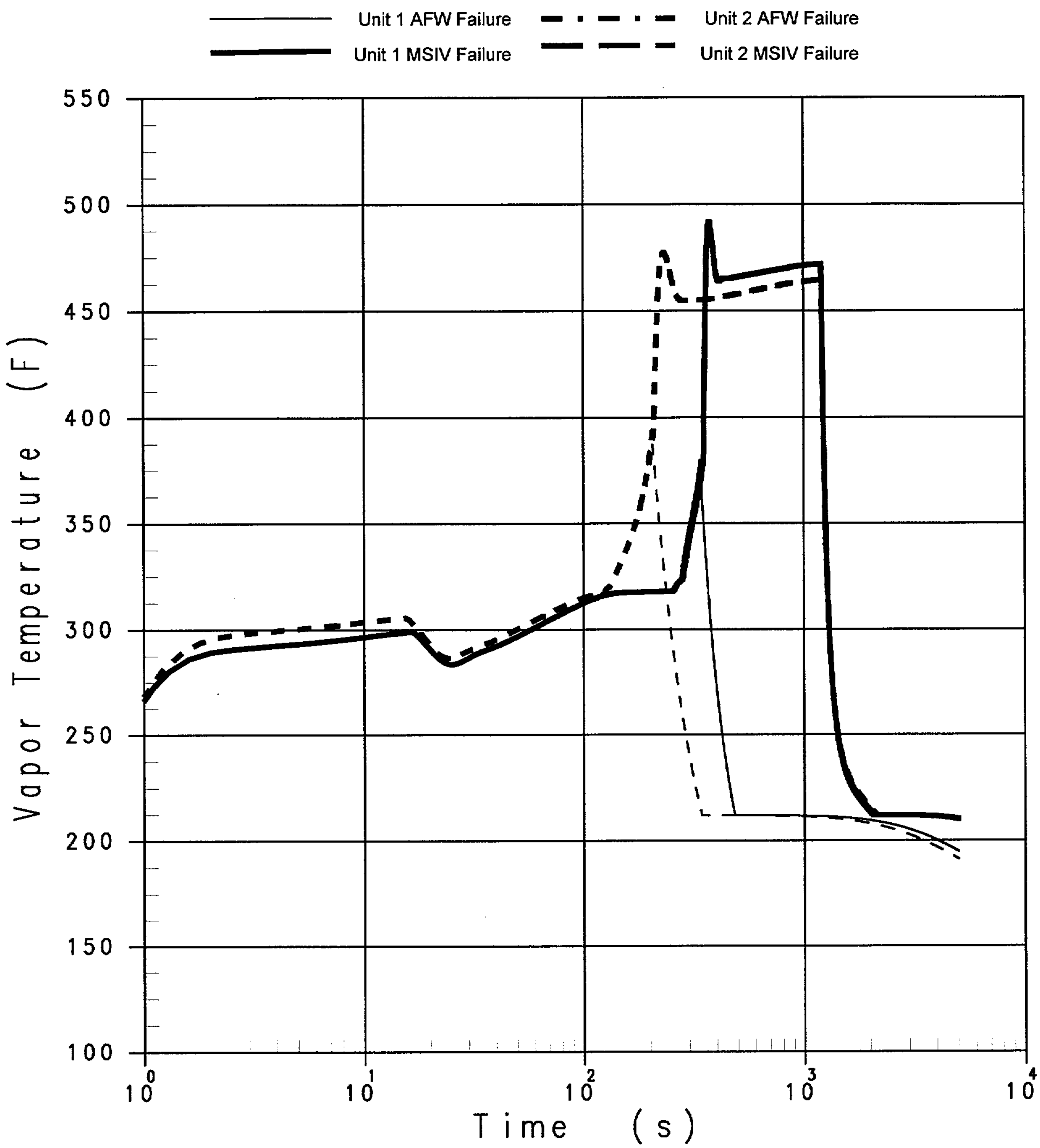


Figure 6.5.5-8

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-H

Vapor Temperature for Steam Tunnel Node 2

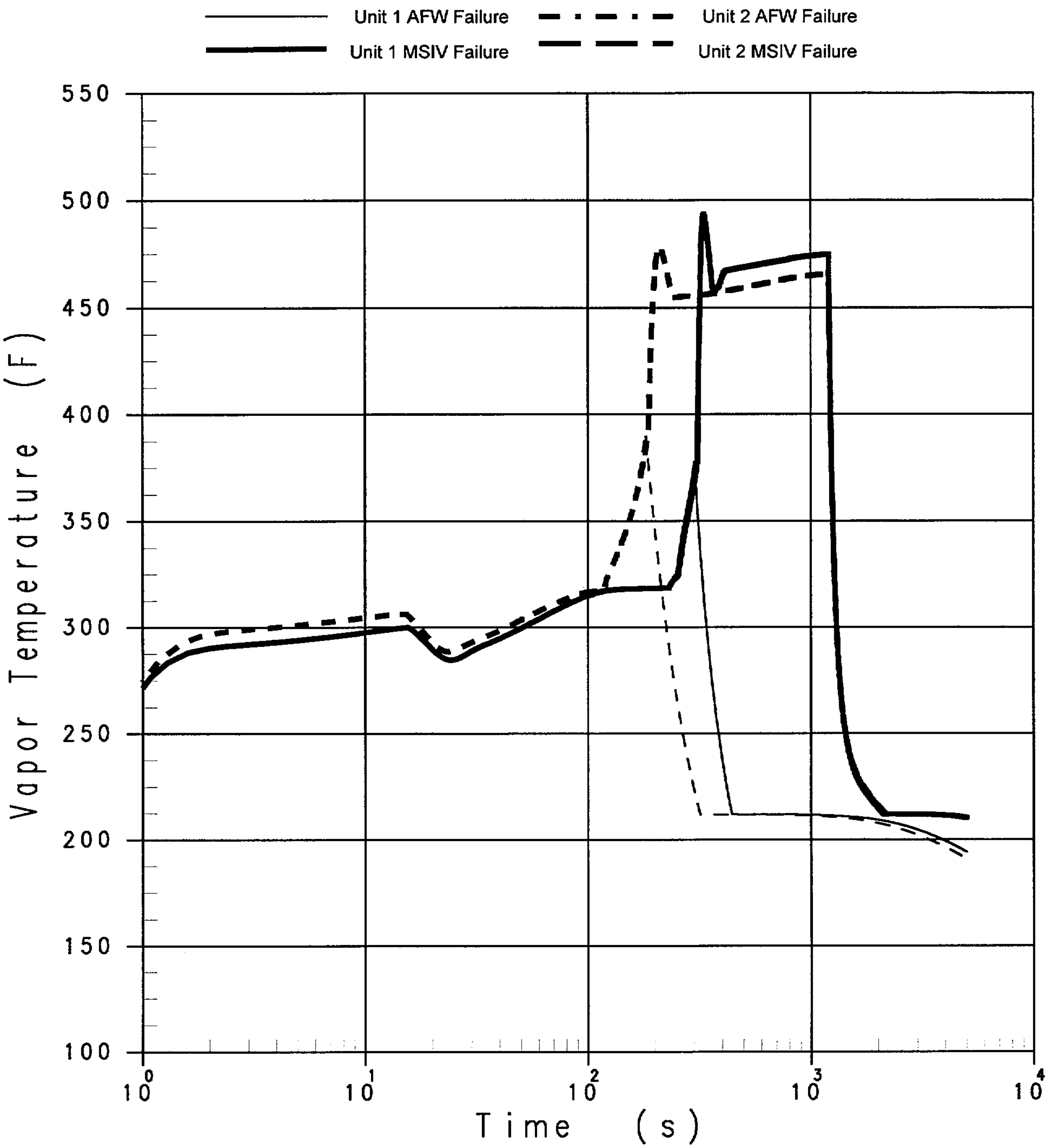


Figure 6.5.5-9

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-I

Vapor Temperature for Steam Tunnel Node 2

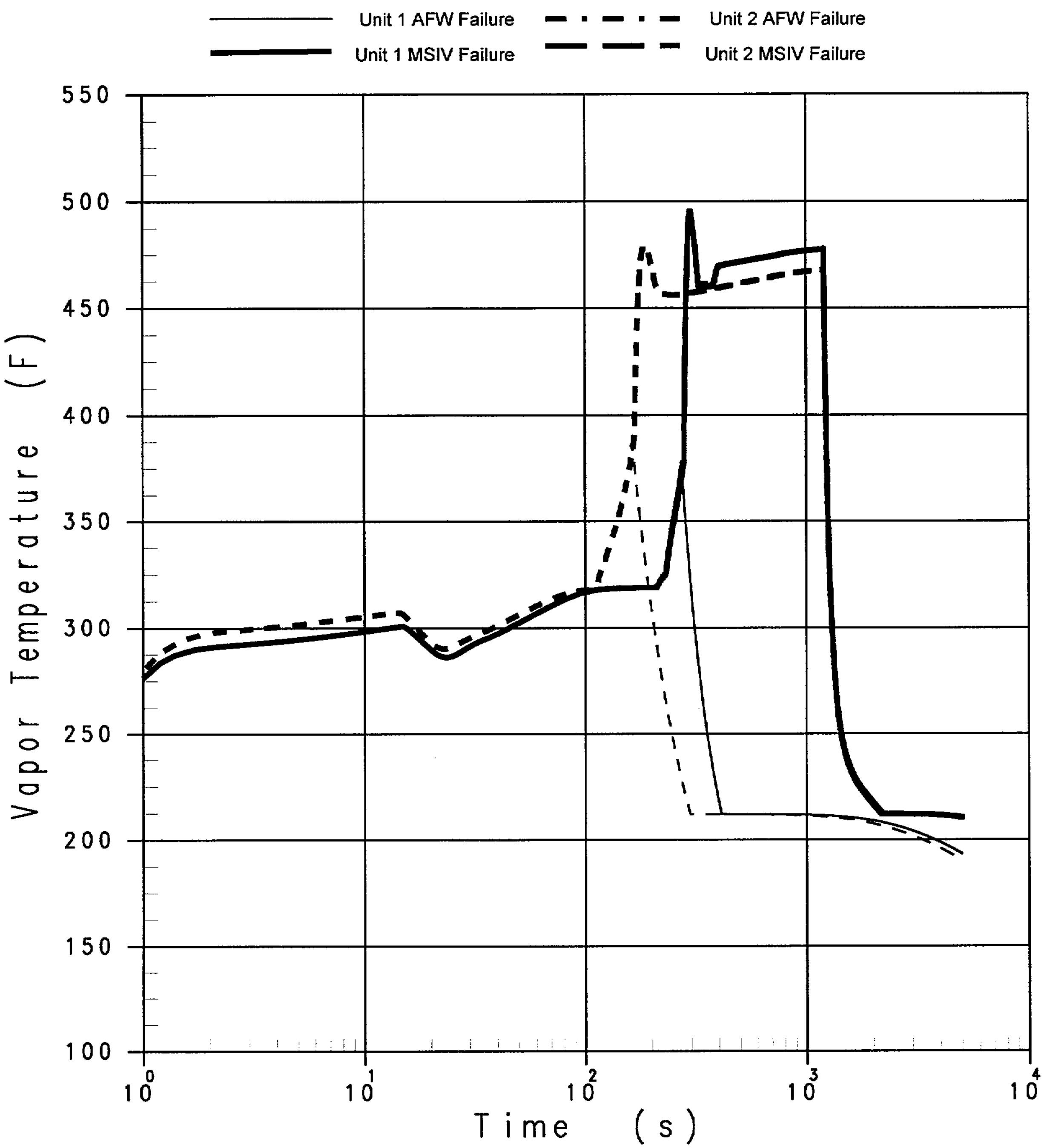


Figure 6.5.5-10

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-J

Vapor Temperature for Steam Tunnel Node 2

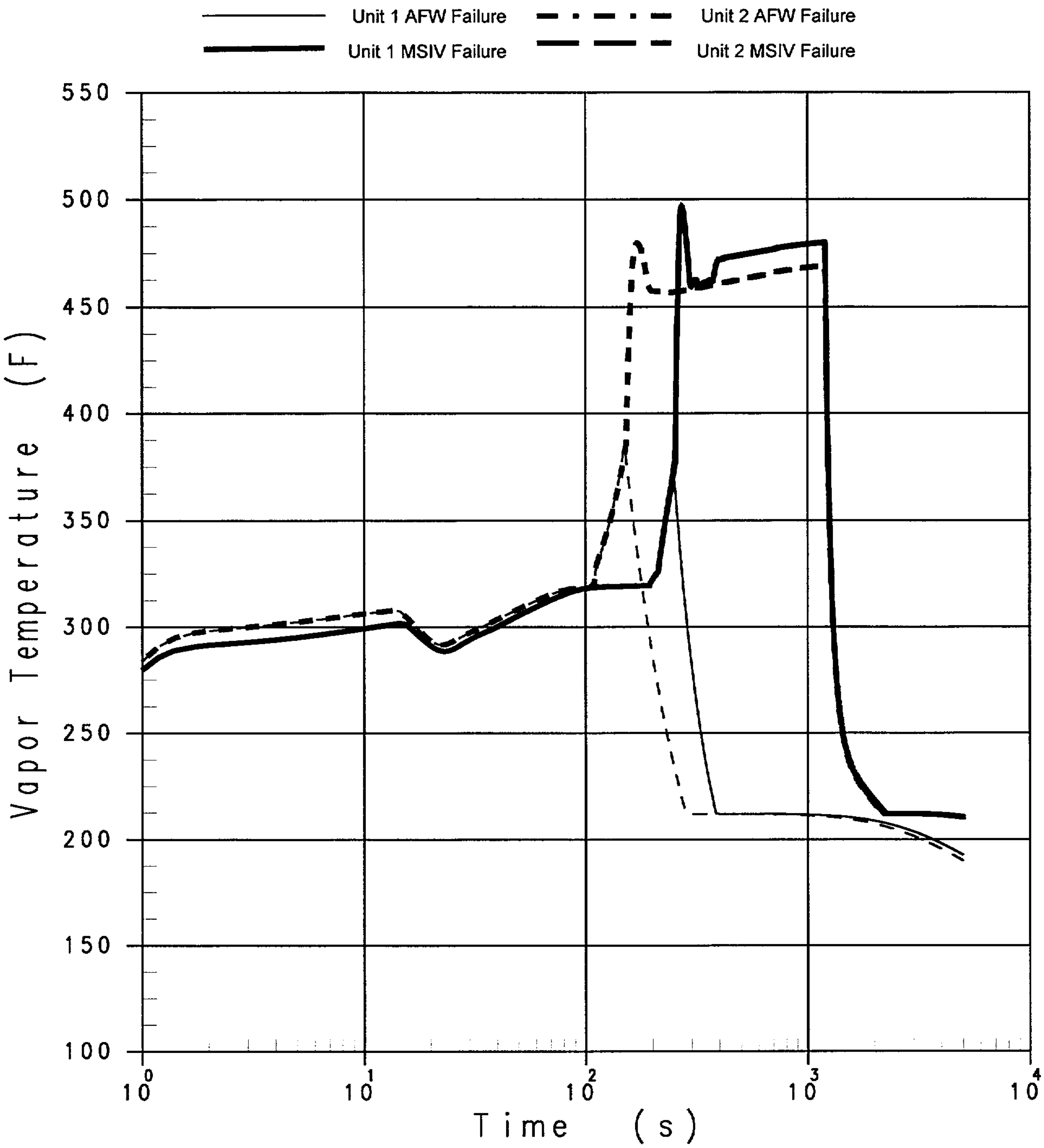


Figure 6.5.5-11

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-K

Vapor Temperature for Steam Tunnel Node 2

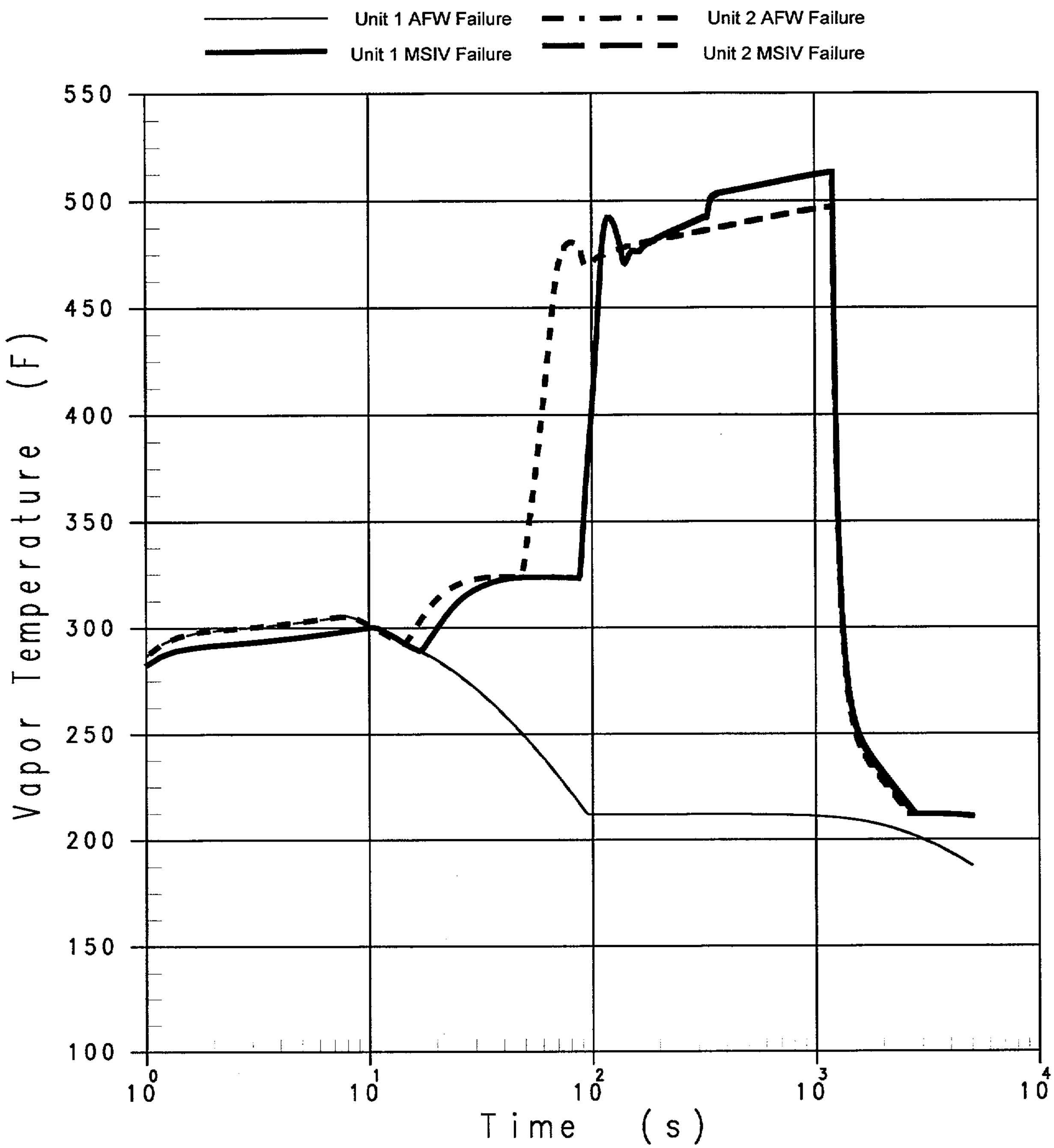


Figure 6.5.5-12

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-L

Vapor Temperature for Steam Tunnel Node 2

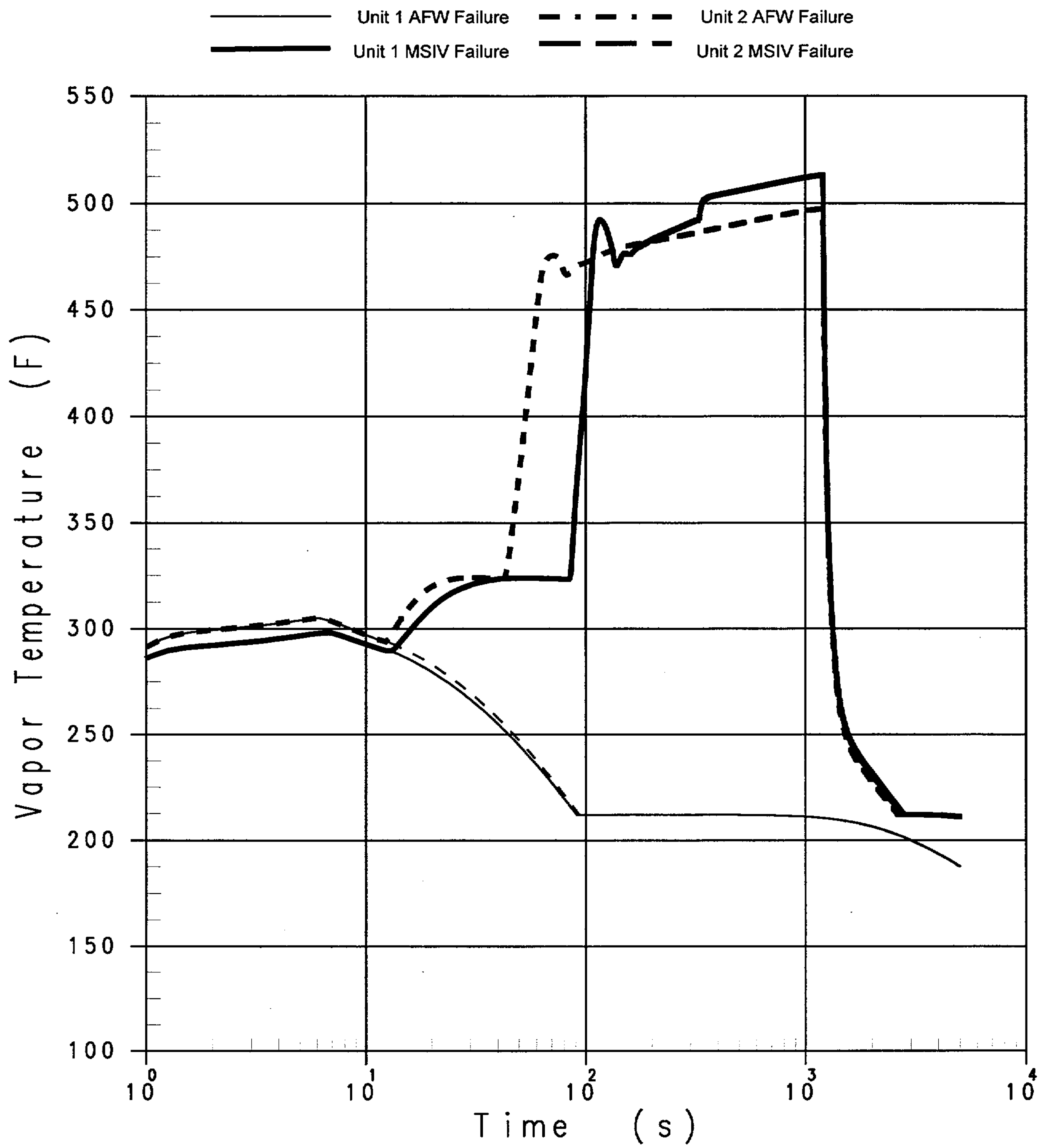


Figure 6.5.5-13

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-M

Vapor Temperature for Steam Tunnel Node 2

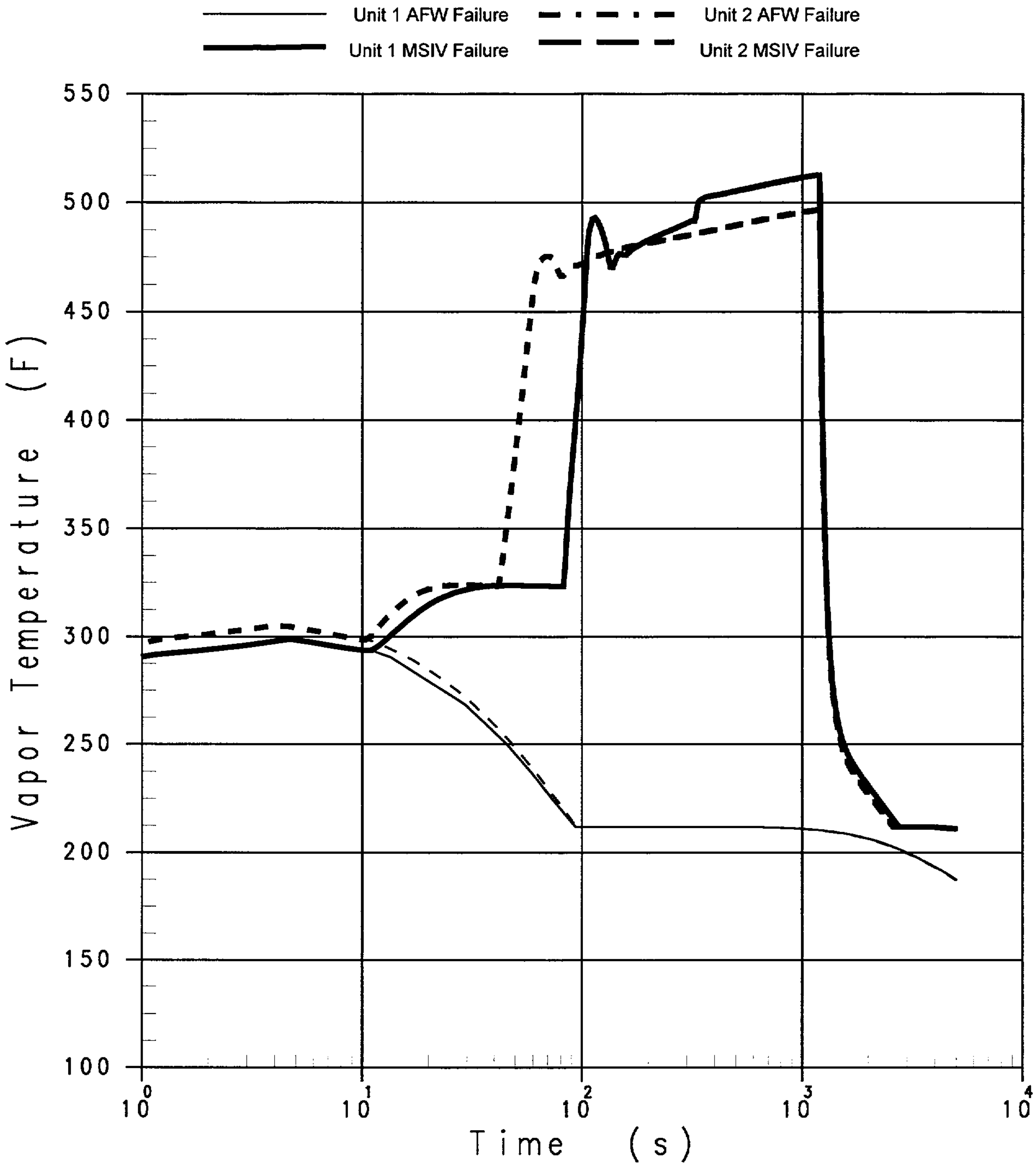


Figure 6.5.5-14

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-N

Vapor Temperature for Steam Tunnel Node 2

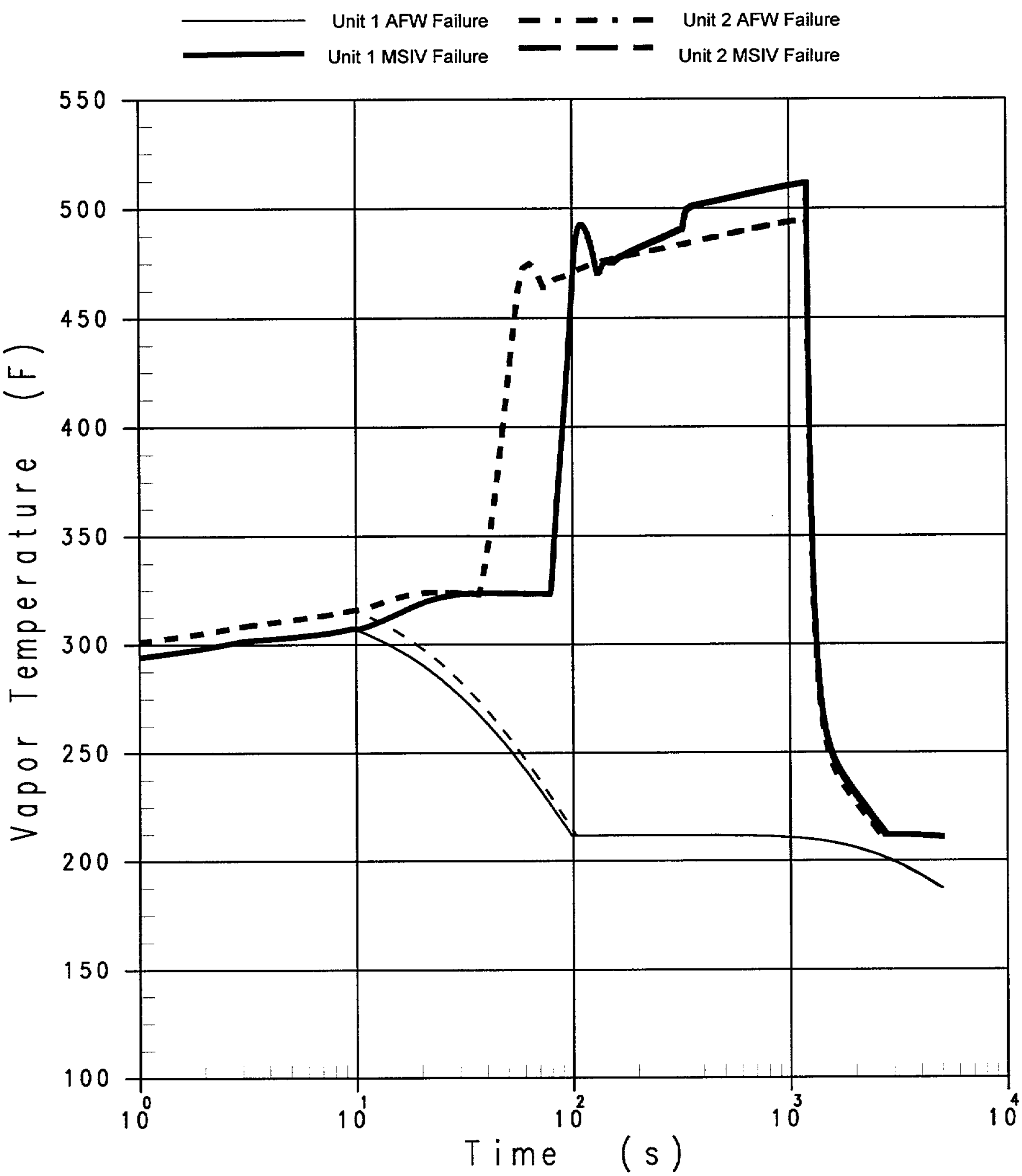


Figure 6.5.5-15

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 102-O

Vapor Temperature for Steam Tunnel Node 2

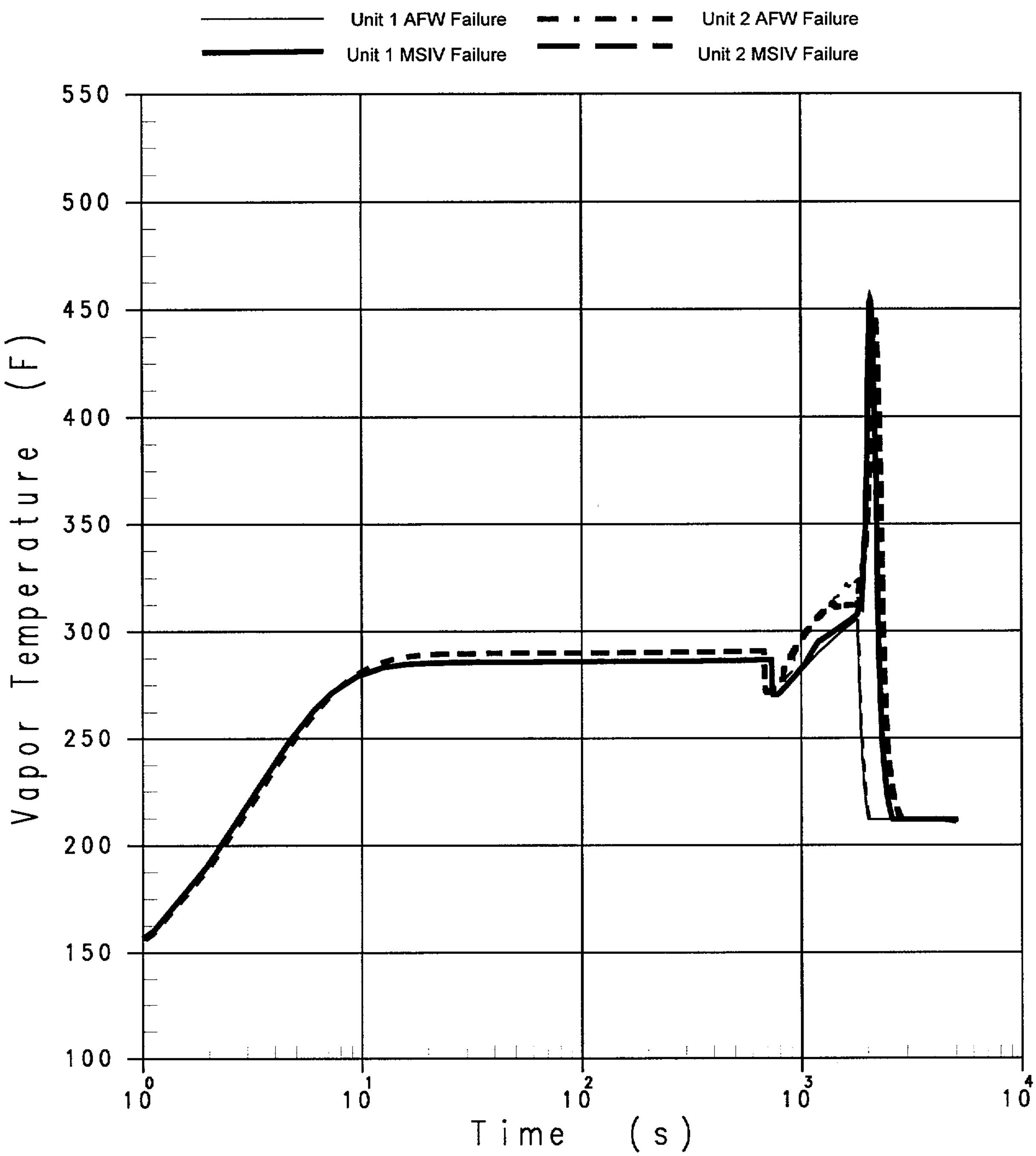


Figure 6.5.5-16

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-A

Vapor Temperature for Steam Tunnel Node 2

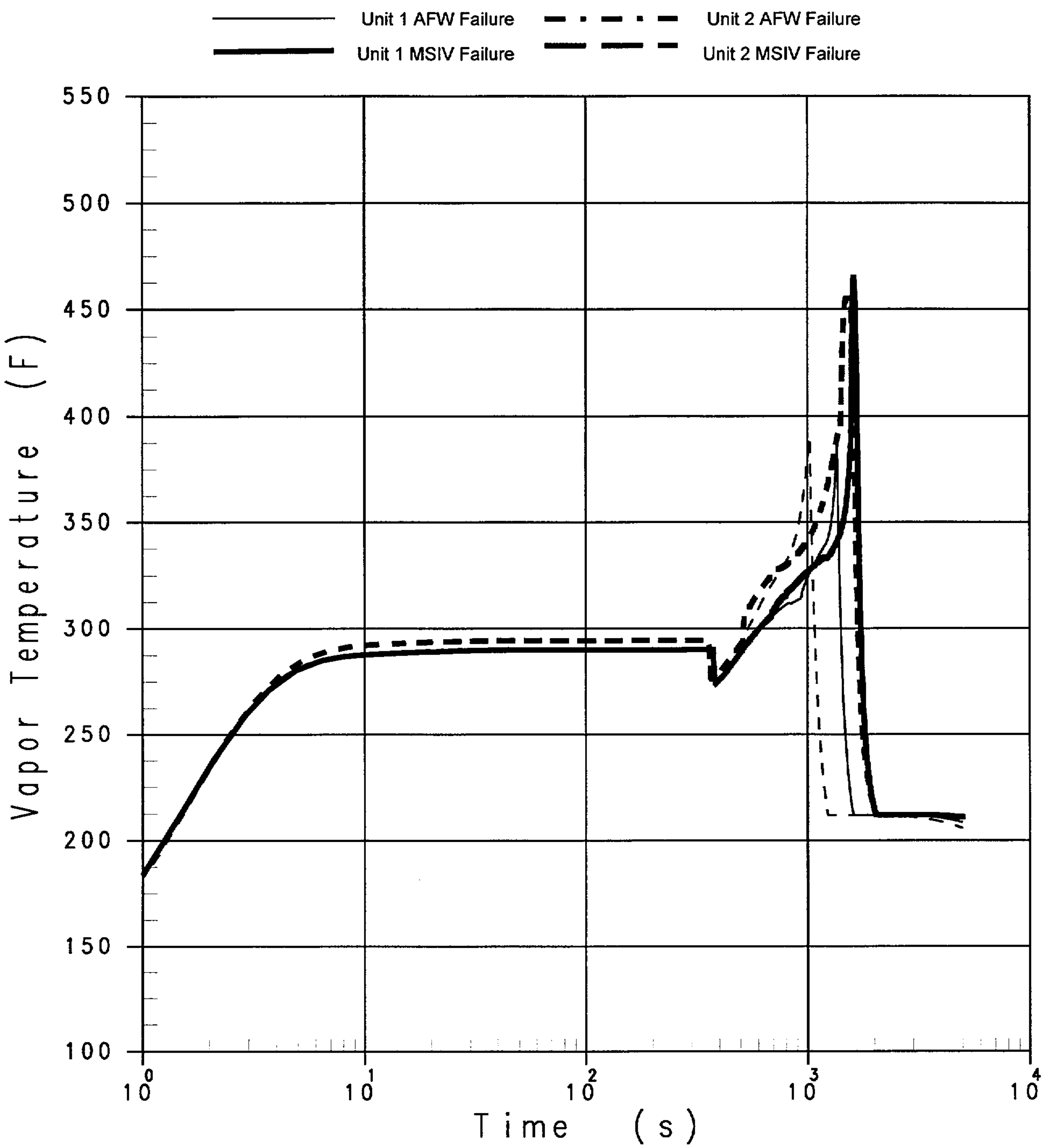


Figure 6.5.5-17

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-B

Vapor Temperature for Steam Tunnel Node 2

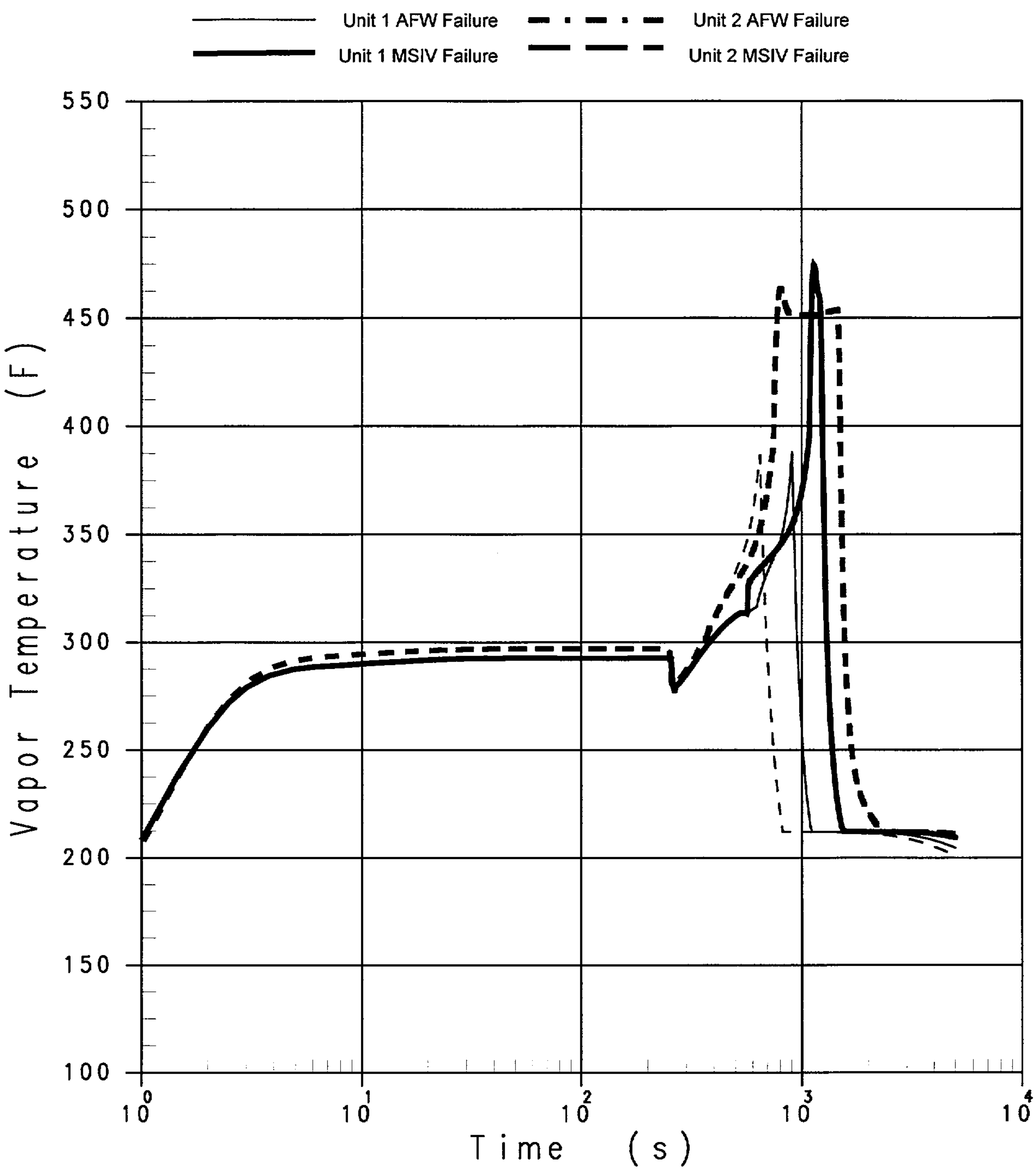


Figure 6.5.5-18

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-C

Vapor Temperature for Steam Tunnel Node 2

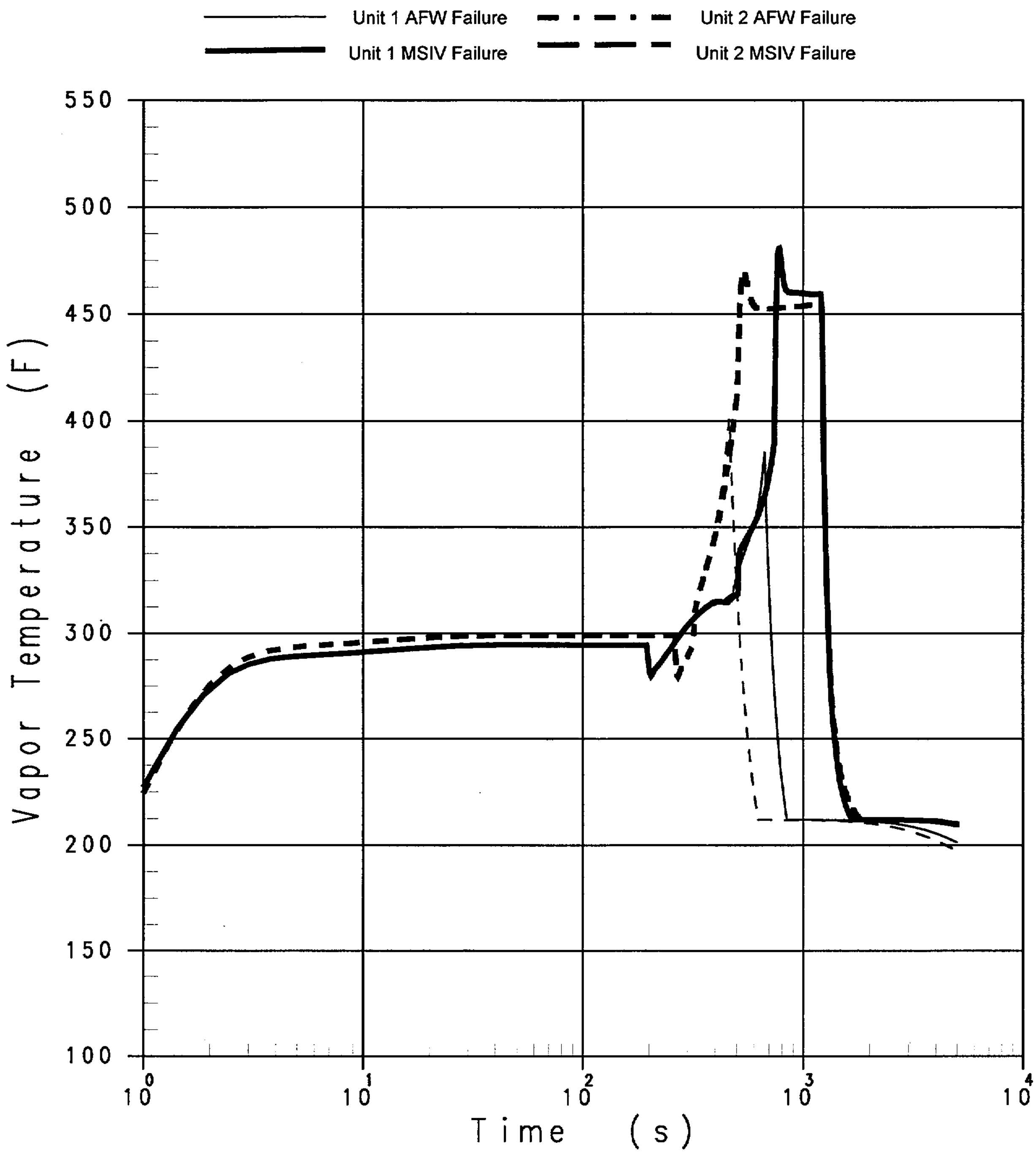


Figure 6.5.5-19

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-D

Vapor Temperature for Steam Tunnel Node 2

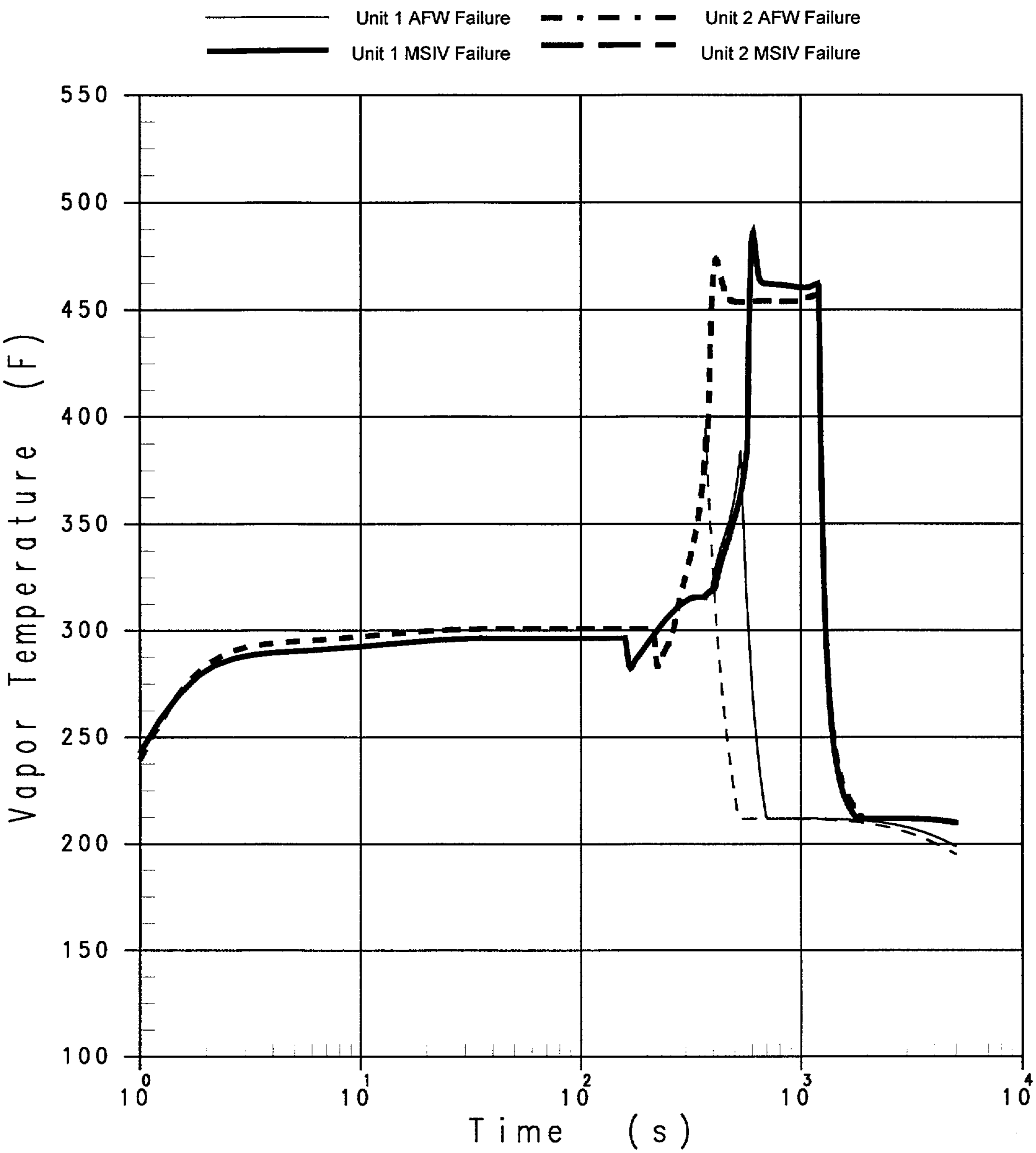


Figure 6.5.5-20

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-E

Vapor Temperature for Steam Tunnel Node 2

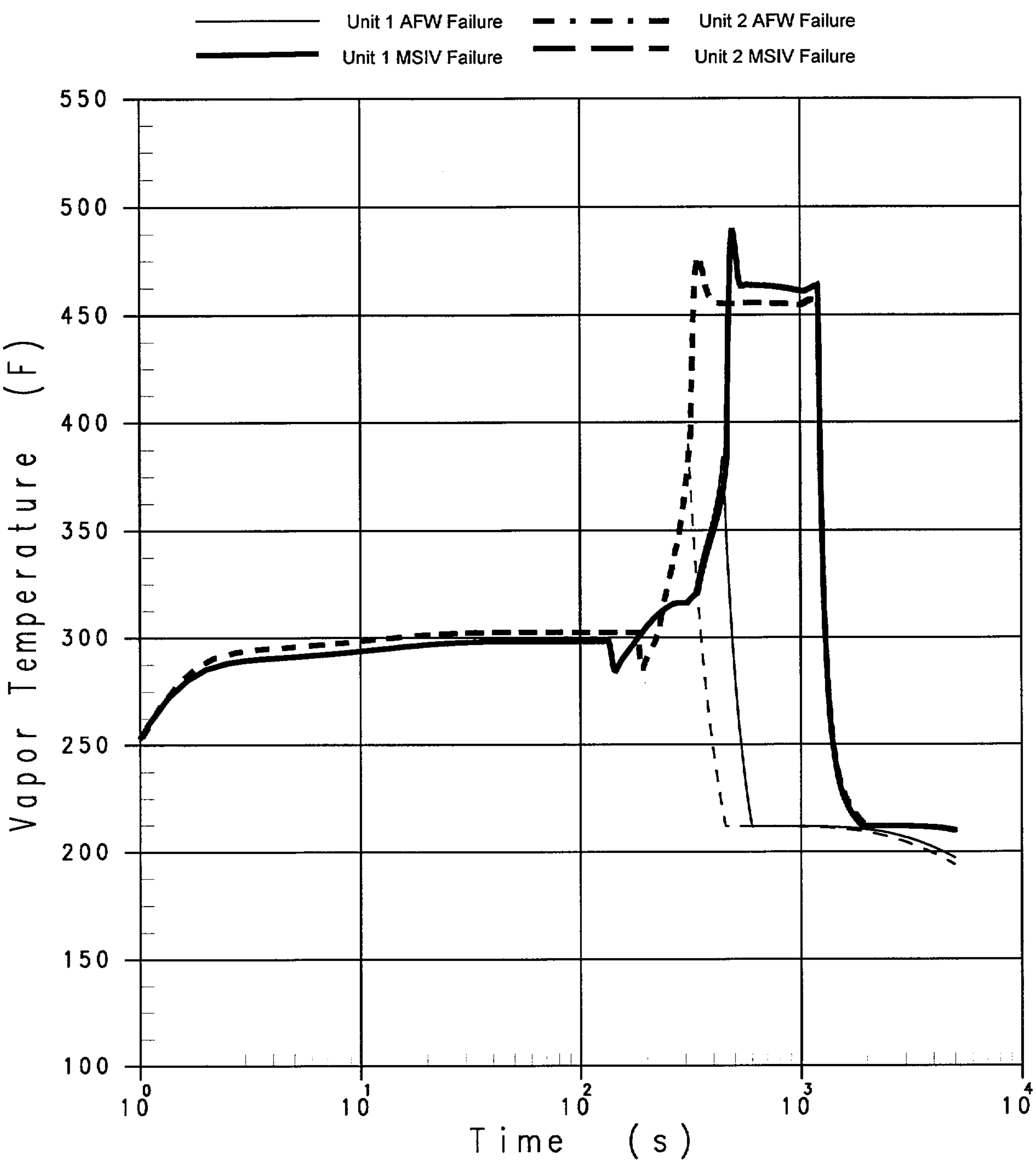


Figure 6.5.5-21

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-F

Vapor Temperature for Steam Tunnel Node 2

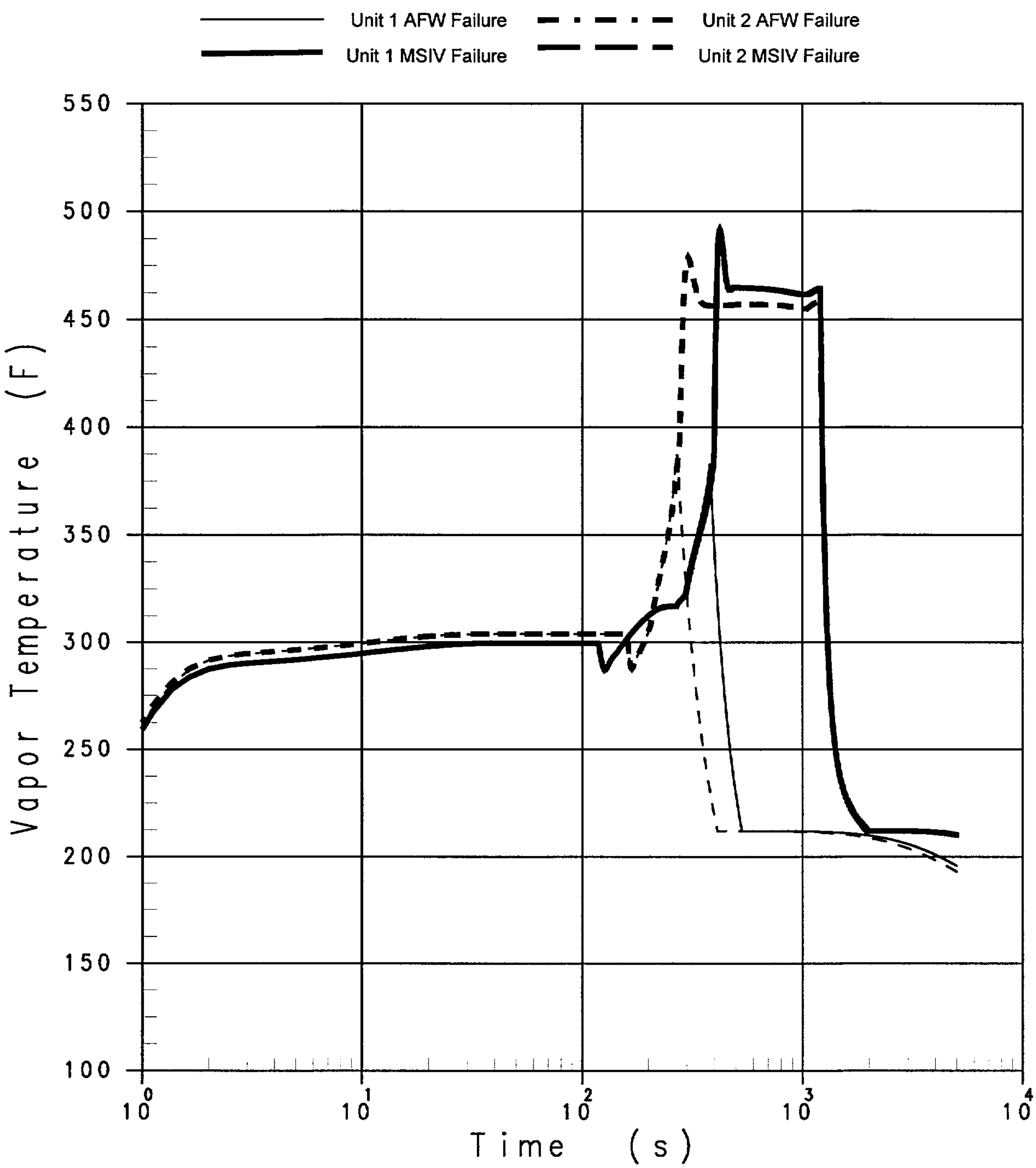


Figure 6.5.5-22

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-G

Vapor Temperature for Steam Tunnel Node 2

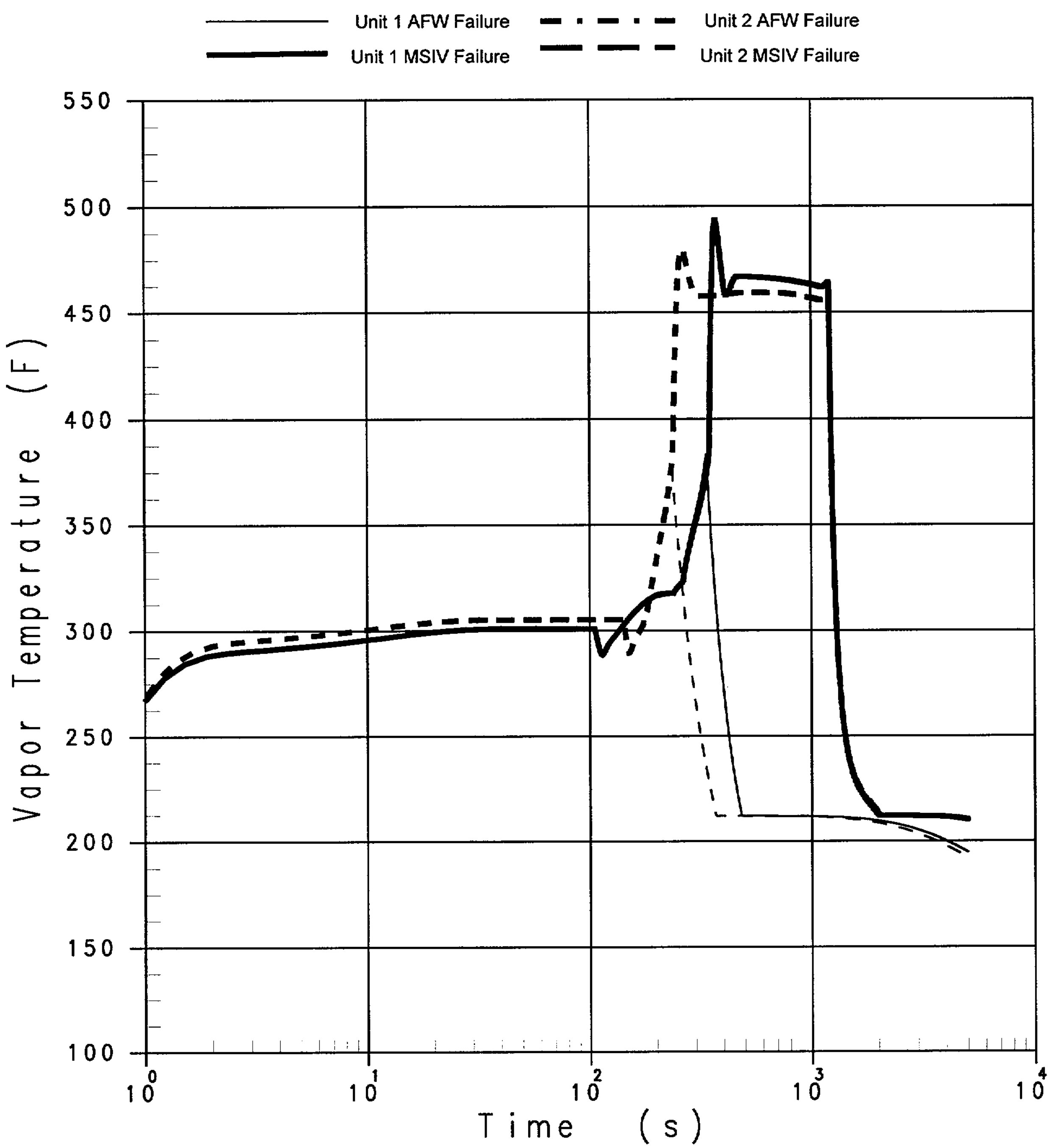


Figure 6.5.5-23

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-H

Vapor Temperature for Steam Tunnel Node 2

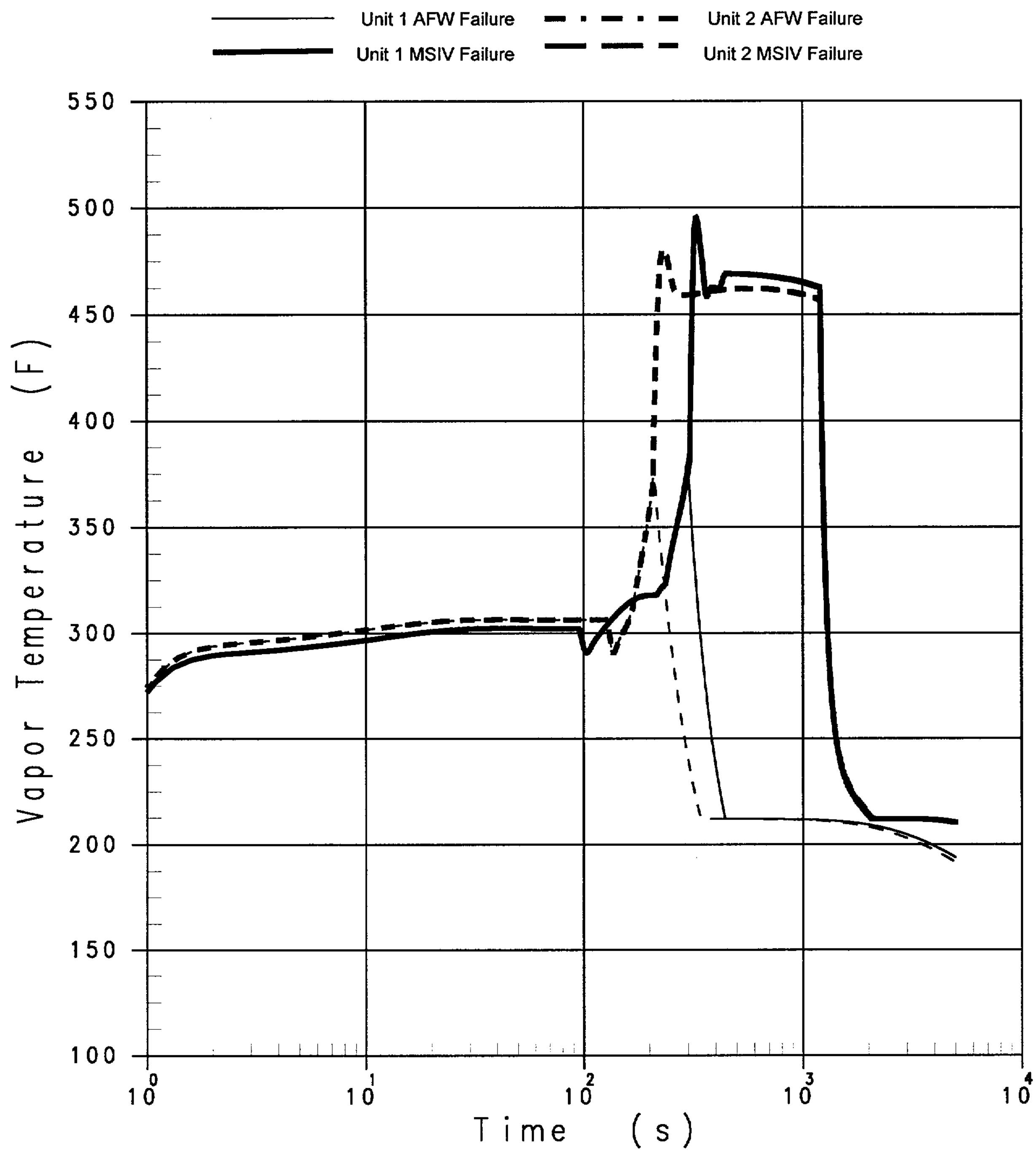


Figure 6.5.5-24

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-I

Vapor Temperature for Steam Tunnel Node 2

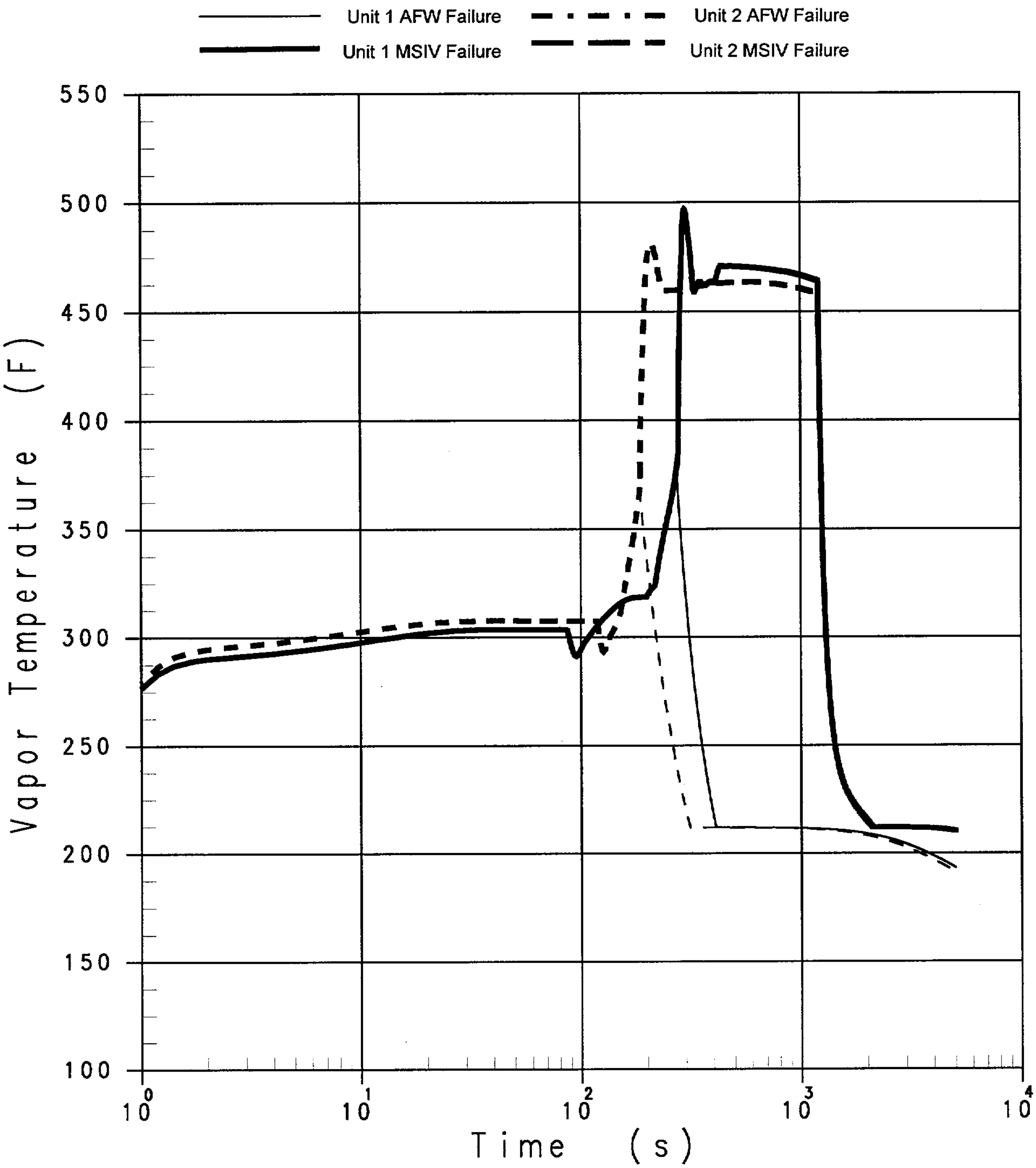


Figure 6.5.5-25

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-J

Vapor Temperature for Steam Tunnel Node 2

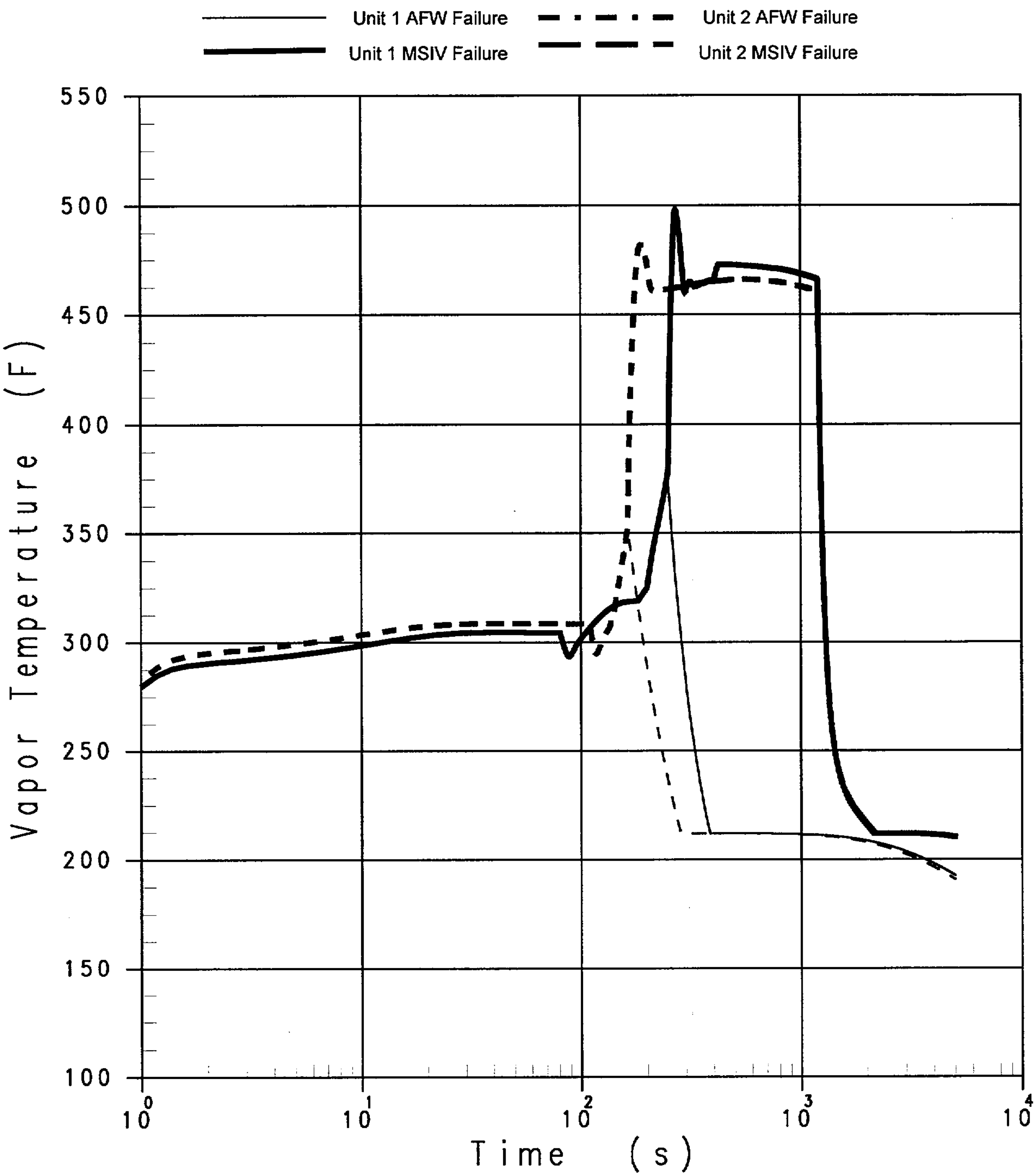


Figure 6.5.5-26

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-K

Vapor Temperature for Steam Tunnel Node 2

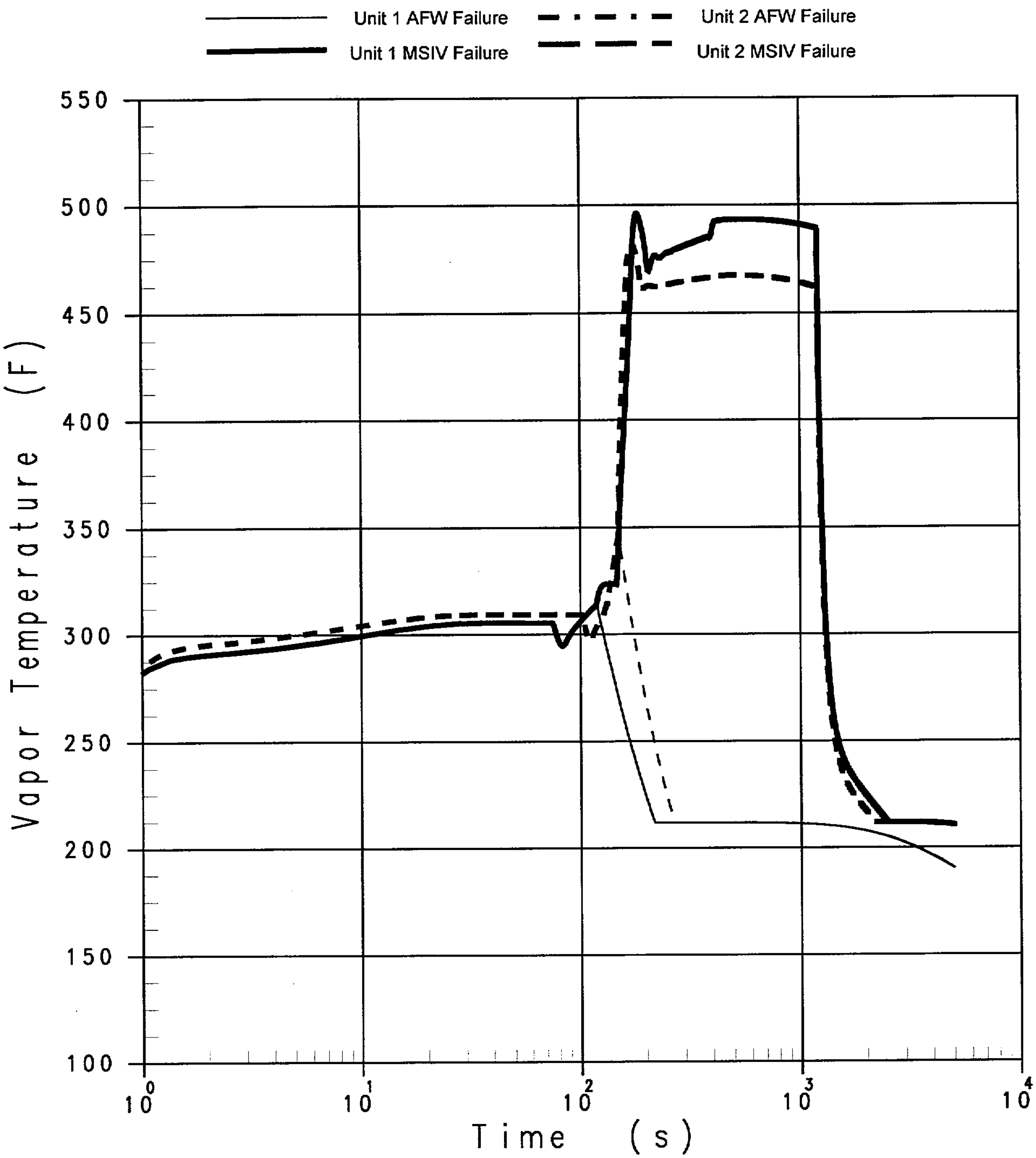


Figure 6.5.5-27

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-L

Vapor Temperature for Steam Tunnel Node 2

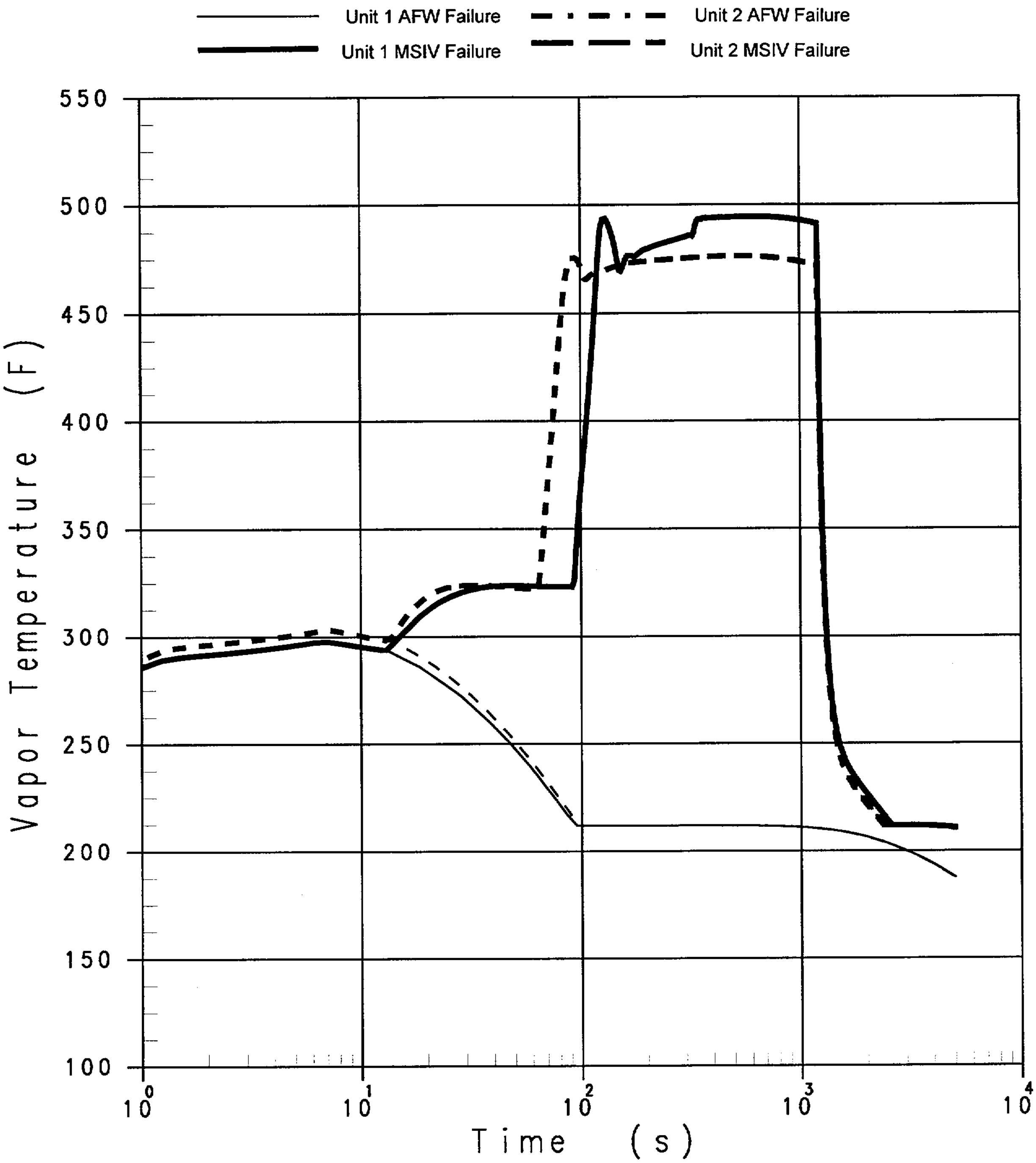


Figure 6.5.5-28

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-M

Vapor Temperature for Steam Tunnel Node 2

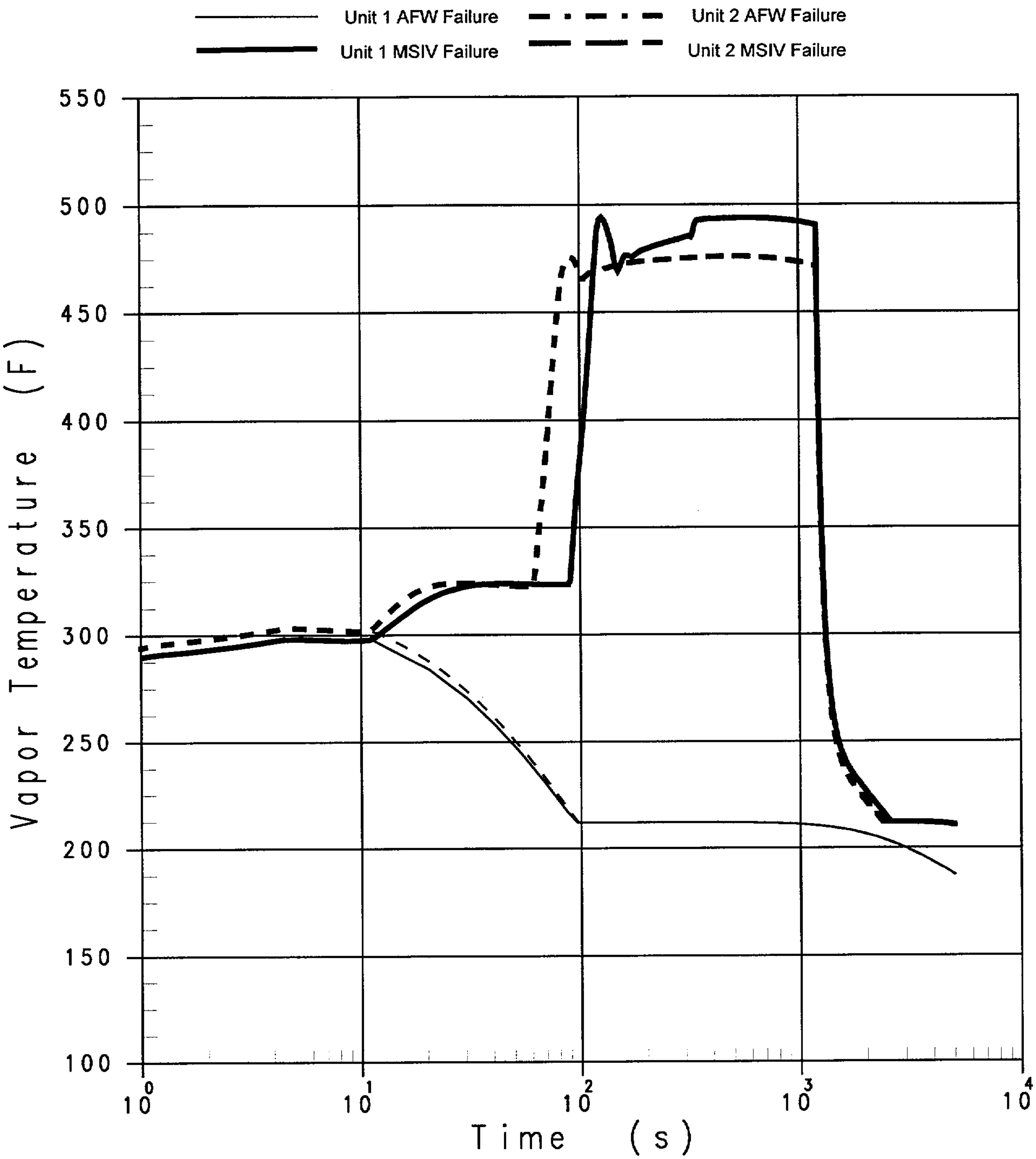


Figure 6.5.5-29

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-N

Vapor Temperature for Steam Tunnel Node 2

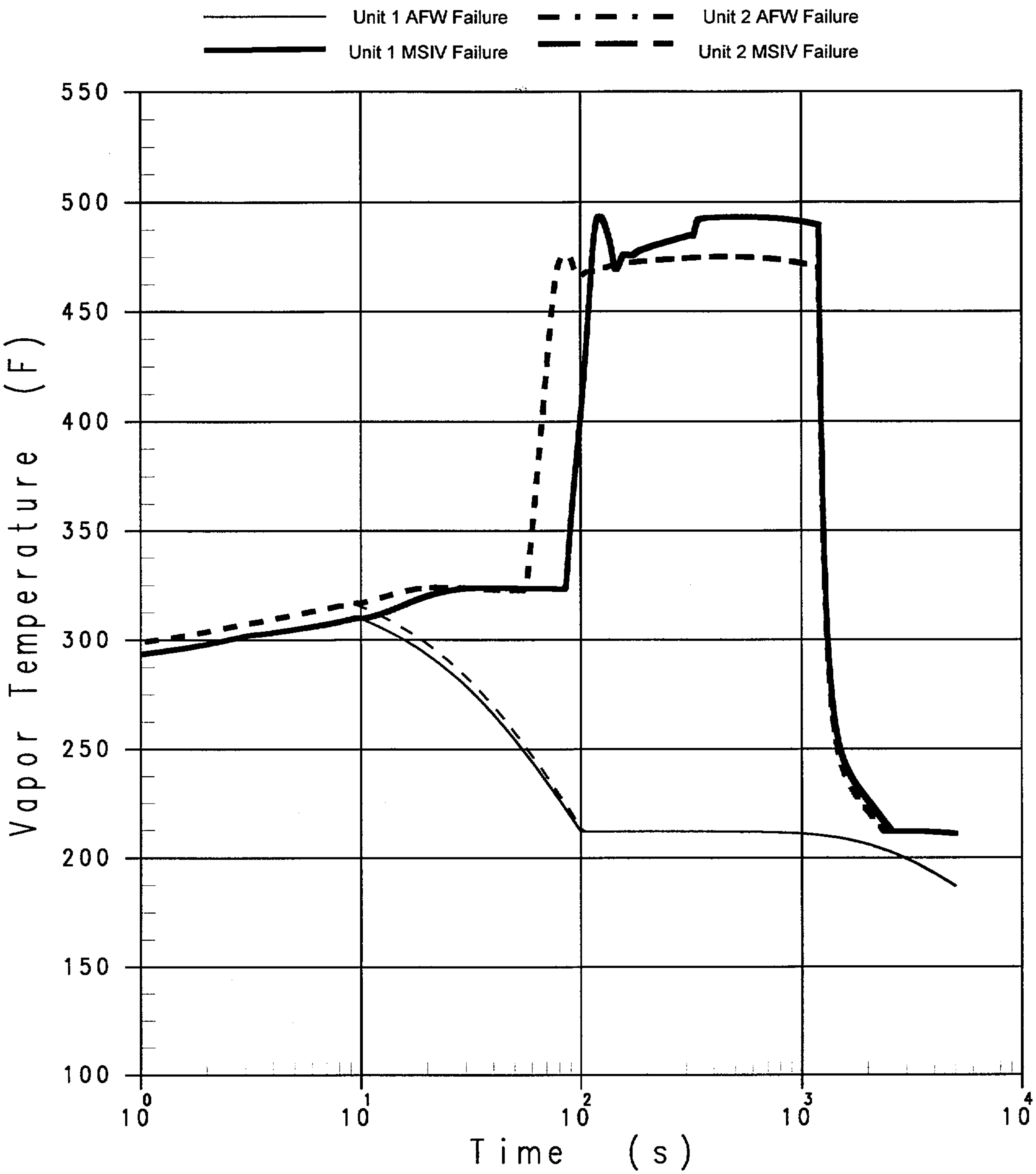


Figure 6.5.5-30

Byron/Braidwood Power Uprate Program Compartment Temperatures for Case 70-O Vapor Temperature for Steam Tunnel Node 2

6.6 LOCA Hydraulic Forces Evaluation

6.6.1 Introduction

A Loss of Coolant Accident (LOCA) Hydraulic Forces evaluation was performed in support of the Power Uprate for Byron/Braidwood Units 1 and 2. The LOCA Hydraulic Forces evaluation provides input for determining the structural integrity of the Reactor Pressure Vessel, Reactor Coolant System (RCS) Loop Piping, and Steam Generators during a LOCA transient. This section provides a summary of the LOCA Hydraulic Forces (LHF) evaluation.

6.6.2 Input Parameters and Assumptions

The following table provides parameters important for the LHF evaluation.

Parameter	Value	Uncertainty
Coldest Reactor Vessel Inlet Temperature	542.0 °F	9.1 °F
Coldest Reactor Vessel Outlet Temperature	608.0 °F	9.1 °F
Reactor Coolant System Pressure	2250 psia	43 psi

These parameters are based upon Thermal Design Flow Rate Per Loop of 92,000 gpm and an uprated reactor power of 3586.6 MWt. For LOCA Hydraulic Forces, higher pressure and lower temperature at the break produce greater forcing functions. For the current Westinghouse analyses, the maximum temperature uncertainty was subtracted from the minimum full-power RCS temperatures at thermal design flow, and the maximum pressurizer pressure uncertainty was added to the nominal full-power pressure, for conservatism.

Loads induced by initial RCS mass flow are typically small compared to maximum forces induced by the acoustical wave generated by RCS ruptures. Small changes to these initial loads are insignificant compared to LOCA transient hydraulic forces. For most of the analyses, cold leg breaks produce the largest and most limiting LOCA forces. As lower temperatures at the break are limiting, flow rates consistent with the minimum T_{COLD} are postulated.

6.6.3 Description of Evaluation

The current LOCA hydraulic forces calculations for all four Byron and Braidwood units were evaluated to determine their applicability at conditions consistent with the uprated reactor power of 3586.6 MWt. The evaluation addressed the current LOCA hydraulic forces analyses applicable to the vessel, loop piping, and steam generators. Westinghouse performed the current applicable analyses of vessel forces, and Byron/Braidwood Units 2 loop and steam generator LOCA forces. Framatome Technologies performed the current applicable analyses of Byron/Braidwood Units 1 loop and steam generator LOCA forces.

The operating conditions assumed, break sizes assumed and break locations considered in the current analyses were evaluated against the uprated power conditions and the break sizes and locations required under current leak-before-break (LBB) analyses approved by the NRC (References 1 and 2) for all four Byron and Braidwood units. As allowed under 10 CFR 50, Appendix A, General Design Criteria 4 (GDC-4), the main coolant loop piping (Reference 1), the safety injection (SI) line piping connections to the accumulators (Reference 2), and the reactor coolant (RC) bypass lines (Reference 2), have been excluded from consideration for dynamic effects associated with postulated pipe rupture.

Where the current LOCA forces analyses were not performed for operating conditions which bound the uprated conditions, the postulated ruptures which produced the largest forcing functions may be eliminated from consideration for dynamic effects in accordance with GDC-4 (References 1 and 2). In these cases, credit for the effect of reduction in required postulated break area (allowed under LBB) was taken to offset the adverse impact of uprated power operating conditions. That is, using established sensitivities to LOCA forces, the reduction in LOCA forces associated with the change in break area from the analyses-of-record to the current required break areas was shown to exceed the increase in LOCA forces caused by the reduction in temperatures and increases in pressures possible at the uprated power conditions.

For LOCA forces analyses already performed to conditions which bound those at uprated power, no reduction in postulated break area was necessary to demonstrate their continued applicability.

Because no new analyses were performed, no new methods or computer codes were applied, and no existing codes or methods were judged inappropriate. The current LOCA forcing functions from the analyses-of-record have been shown to remain applicable for conditions consistent with the power uprating, either directly, or by using credit for conservatism in the assumed break areas to compensate for differences between the analysis conditions and the uprating conditions. Consequently, no margins of safety in these analyses have been reduced.

All changes to the units which have impacted the existing analyses have been considered in evaluating the uprated power conditions for Westinghouse analyses. The applicability of the LOCA forces from the existing Westinghouse analyses have not been affected by changes in knowledge, regulation, guidance or plant configuration. The applicability of the Framatome Technologies LOCA forces analysis is also assumed to remain unaffected by changes in knowledge, regulation, guidance or plant configuration, since the Framatome analyses are more recent, and the older Westinghouse analyses were not affected by any such issues.

6.6.4 Acceptance Criteria

LOCA Hydraulic Forces are provided as input to structural qualification analyses, and as such have no independent regulatory acceptance criteria. The structural analyses performed using these forcing functions are done to demonstrate compliance with 10 CFR 50, Appendix A, General Design Criteria 4.

6.6.5 Results

The evaluations concluded that the current vessel, loop piping, and steam generator LOCA hydraulic forces remain bounding for Byron/Braidwood Units 1 and 2 at conditions consistent with the power uprating. Vessel forces for all four units and loop forces for both Units 2 were found to be bounding due to conservatism in the break areas postulated which compensated for effects of decreased temperature and increased pressure. Loop forces for both Units 1 and steam generator forces for all four units were found to have been done at conditions which bound the uprated power conditions. The results of the evaluation were transmitted to the appropriate structural qualifications groups. The acceptance of the LOCA hydraulic forces is incorporated with the acceptance of the structural qualifications for the uprate program.

6.6.6 Conclusions

The conclusions of the evaluations performed were that all of the currently applicable LOCA forces analyses for vessel, loop piping, and steam generators remain applicable for the structural qualification analyses at uprated power conditions.

6.6.7 References

1. USNRC letter "SAFETY EVALUATION (SE) REGARDING LEAK-BEFORE-BREAK ANALYSIS – BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2," October 25, 1996, from Ramin R. Assa (NRC) to Irene Johnson (ComEd).

2. USNRC letter "SAFETY EVALUATION OF LEAK-BEFORE-BREAK METHODOLOGY APPLICABLE TO ACCUMULATOR PIPING AND REACTOR COOLANT BYPASS PIPING," April 19, 1991, from Anthony H. Hsia (NRC) to Thomas J. Kovach (ComEd).

6.7 Radiological Consequences Evaluations (Doses)

6.7.1 Introduction

The radiological consequences for the following design basis accidents were reanalyzed to support the power uprating effort:

- Main Steamline Break
- Locked Reactor Coolant Pump (RCP) Rotor
- Locked RCP Rotor with Power-Operated Relief Valve (PORV) Failure
- Rod Ejection
- Small Line Break Outside Containment
- Steam Generator Tube Rupture
- Large-Break Loss-of-Coolant Accident (LOCA)
- Small-Break Loss-of-Coolant Accident
- Waste Gas Decay Tank Rupture
- Liquid Waste Tank Failure
- Fuel Handling Accident

All of these accidents except for the small-break LOCA are currently addressed in the Byron and Braidwood UFSAR.

For each accident, the thyroid and whole body doses are determined at the exclusion area boundary (EAB) for the 0- to 2-hour period and at the low population zone boundary (LPZ) for the duration of the accident. Also, the thyroid, whole-body, and beta-skin doses are determined for the control room personnel (CR) for the large-break and small- break LOCA events.

6.7.1.1 Input Assumptions

The assumptions and inputs described in this section are common to various analyses discussed in the following sections. Each accident and the specific input assumptions are described in detail in Sections 6.7.2 through 6.7.12.

The dose conversion factors (DCFs) used in determining the thyroid dose are from International Commission on Radiological Protection (ICRP) Publication 30 (Reference 1). The average

disintegration energies used in determining the whole-body and beta-skin doses from airborne iodine isotopes are from ICRP Publication 38 (Reference 2). The DCFs used in determining the whole-body and beta-skin doses from the noble gas activity in the air are from Regulatory Guide 1.109 (Reference 3). The model presented in the UFSAR used a semi-infinite cloud model for whole body and beta skin doses for noble gases. This model uses average gamma and beta ray energies per disintegration and does not account for the shielding provided by the "dead" skin layer. The whole body and skin dose model has been changed from the semi-infinite cloud model to that of Regulatory Guide 1.109. The decay constants for iodine and noble gas nuclides are also obtained from Reference 2. The nuclide data are all listed in Table 6.7.1-1.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 6.7.1-2.

The offsite dose acceptance limits are based on 10 CFR Part 100 guidance of 300 rem thyroid and 25 rem whole body. Depending on the event, the acceptane limit is 100 percent of 10 CFR 100 or a fraction of these guidelines. Some events are designated as having a dose limit that is "well within" 10 CFR 100 where this is defined as 25 percent of those limits (75 rem thyroid and 6 rem whole body) and other events are specified with a dose limit that is a "small fraction" of 10 CFR 100 where this is defined as 10 percent of those limits (30 rem thyroid and 2.5 rem whole body).

Parameters modeled in the control room personnel dose calculations are provided in Table 6.7.1-3. These parameters include normal operational flowrates, emergency operation flowrates, control room volume, filter efficiencies, control room operator breathing rates, and dose limits.

The core fission product activity is provided in Table 6.7.1-4 for iodine and noble gas nuclides. The nominal reactor coolant activity, based on 1% fuel defects for noble gases and 1.0 μCi/gm Dose Equivalent I-131 (DE I-131) for iodine, is provided in Table 6.7.1-5.

6.7.1.2 Iodine Spiking Models

A number of accident analyses take iodine spiking into consideration (e.g., Main Steamline Break, Steam Generator Tube Rupture, Small Line Break Outside Containment, etc.).

For the pre-existing iodine spike, it is assumed that a reactor transient has occurred prior to the accident and has raised the primary coolant iodine concentration to 60 μCi/gm of DE I-131 (this is the Technical Specification limit for transient elevated iodine activity in the primary coolant). For the accident-initiated iodine spike, it is assumed that the reactor trip associated with the accident creates an iodine spike which increases the iodine release rate from the fuel to the reactor coolant to a value 500 times greater than the maximum equilibrium release rate (where the equilibrium release rate is that rate corresponding to maintaining a primary coolant concentration of 1.0 μCi/gm of DE I-131, which is the maximum concentration allowed by the Technical Specifications for continuous operation).

The primary coolant iodine concentrations associated with a pre-existing iodine spike are provided in Table 6.7.1-6, as are the iodine appearance rates associated with an accident-initiated iodine spike.

6.7.1.3 Computer Code

The TITAN5 code used in the dose calculations is a Westinghouse Electric Company code in long-standing use for determining radiological consequences of postulated accidents. This code has been used previously for Byron and Braidwood licensing activities. The NRC, in its confirmatory analyses performed on numerous applications, has obtained results consistent with those obtained using the TITAN5 code.

6.7.1.4 References

1. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979.

2. International Commission on Radiological Protection, "Radionuclide Transformations, Energy and Intensity of Emissions," ICRP Publication 38, 1983.

3. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance, with 10 CFR Part 50, Appendix I, Revision 1, October 1977.

Table 6.7.1-1 Nuclide Parameters				
Nuclide	Decay Constant (hr^{-1})	Thyroid Dose Conversion Factor (rem/Ci)	Gamma Disintegration Energy (MeV/dis)	Beta Disintegration Energy (MeV/dis)
I-131	0.00359	1.07E6	0.381	0.192
I-132	0.301	6.29E3	2.28	0.496
I-133	0.0333	1.81E5	0.607	0.41
I-134	0.791	1.07E3	2.62	0.623
I-135	0.105	3.14E4	1.58	0.367
Nuclide	Decay Constant (hr^{-1})	Thyroid Dose Conversion Factor (rem/Ci)	Whole Body Dose Conversation Factor (rem-M^3/Ci-Sec)	Beta-Skin Dose Conversion Factor (rem-M^3/Ci-Sec)
Kr-85m	0.155	N/A	0.0371	0.0463
Kr-85	7.38E-6	N/A	0.00051	0.0425
Kr-87	0.545	N/A	0.188	0.3085
Kr-88	0.244	N/A	0.466	0.0751
Xe-131m	0.00243	N/A	0.0029	0.015
Xe-133m	0.0132	N/A	0.00796	0.0315
Xe-133	0.00551	N/A	0.00932	0.0097
Xe-135m	2.72	N/A	0.0989	0.0225
Xe-135	0.0763	N/A	0.0574	0.059
Xe-138	2.93	N/A	0.28	0.131

2.

Table 6.7.1-2
Offsite Breathing Rates
and Atmospheric Dispersion Factors

Time	Offsite Breathing Rates (m^3/sec)	
0 - 8 hours	3.47E-4	
8 - 24 hours	1.75E-4	
>24 hours	2.32E-4	
	Offsite Atmospheric Dispersion Factors (sec/m^3)	
	Byron	**Braidwood**
Exclusion Area Boundary (0 - 2 hr)	5.7E-4	7.7 E-4
Low Population Zone		
0 – 8 hours	1.7E-5	7.1E-5
8 – 24 hours	2.4E-6	1.4E-5
24 – 96 hours	1.1E-6	7.1E-6
>96 hours	7.6E-7	4.1E-6

3.

Table 6.7.1-3

Control Room Parameters

Breathing Rate - Duration of the Event	3.47E-4 m^3/sec
HVAC Volume CR Volume	230,837 ft^3 70,275 ft^3
Occupancy Factors 0 - 24 hours 1 - 4 days 4 - 30 days	 1.0 0.6 0.4
Normal Ventilation Flow Rates Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate (Not modeled - no impact on analyses)	 0.0 SCFM 0.0 SCFM 6000 SCFM 43,500 SCFM
Emergency Ventilation System Flow Rates Filtered Makeup Air Flow Rate Min Filtered Recirculation Flow Rate Max Unfiltered Inleakage	 6000 SCFM ±10% 39,150 CFM 100 SCFM
Filter Efficiencies for Intake Flow Elemental Organic Particulate	 99% 99% 99%
Filter Efficiencies for Recirculation Flow Elemental Organic Particulate	 90% 90% 80%
Delay to Switchover of HVAC from Normal Operation to Emergency Operation after Receipt of an Isolation Signal	15 seconds
Thyroid Dose Acceptance Criteria Whole Body Dose Acceptance Criteria Beta Skin Dose Acceptance Criteria	30 rem 5 rem 30 rem 75 rem (with credit for protective clothing)

4.

Table 6.7.1-4
Core Total Fission Product Activities
Based on 3658.3 MWt (102% of 3586.6 MWt)

Isotope	Activity (Ci)
I-131	9.74E7
I-132	1.40E8
I-133	1.97E8
I-134	2.17E8
I-135	1.85E8
Kr-85m	2.50E7
Kr-85	1.02E6
Kr-87	4.79E7
Kr-88	6.74E7
Xe-131m	1.09E6
Xe-133m	6.17E6
Xe-133	1.97E8
Xe-135m	3.88E7
Xe-135	4.00E7
Xe-138	1.62E8

Table 6.7.1-5 RCS Coolant Concentrations Based on 1.0 μCi/gm DE I-131 for Iodines and 1% Fuel Defects for Noble Gases	
Nuclide	**Activity (μCi/gm)**
I-131	0.742
I-132	0.979
I-133	1.350
I-134	0.243
I-135	0.842
Kr-85m	1.80
Kr-85	7.11
Kr-87	1.15
Kr-88	3.35
Xe-131m	3.31
Xe-133m	3.65
Xe-133	251
Xe-135m	0.488
Xe-135	7.72
Xe-138	0.663

6.

Table 6.7.1-6
Iodine Spiking Data

Isotope	Primary Coolant Concentration for Pre-existing Spike (μCi/gm)	Iodine Appearance Rate into Primary Coolant for Accident-Initiated Spike (Ci/min)
I-131	44.5	208
I-132	58.7	877
I-133	81.0	462
I-134	14.6	463
I-135	50.5	413

6.7.2 Steamline Break Radiological Consequences

The complete severance of a main steamline outside containment is assumed to occur. The affected steam generator will rapidly depressurize and release iodine activity initially contained in the secondary coolant and primary coolant activity (iodines and noble gases), transferred via steam generator tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact steam generators and the activity transferred to the secondary coolant due to tube leakage is released to atmosphere through either the Power Operated Relief Valves (PORVs) or the main steam safety valves (MSSVs). The steamline break outside containment will bound any break inside containment since the outside-containment break provides a means for direct release to the environment. This section describes the assumptions and analyses performed to determine the offsite doses resulting from the release of activity associated with this event.

6.7.2.1 Input Parameters and Assumptions

The analysis of the main steamline break (MSLB) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). The activity available for release to the environment includes the iodine assumed to be initially present in the secondary coolant and the activity in the primary coolant (both iodine and noble gases) that could leak into the secondary coolant due to steam generator tube leakage.

The iodine activity concentration of the secondary coolant at the time the MSLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of dose equivalent (DE) I-131.

The MSLB is analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity and the other in which an iodine spike is assumed to be initiated by the accident. Based on having 10 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted in 6.0 hours and the accident-initiated spike is terminated at that time.

The noble gas activity concentration in the Reactor Coolant System (RCS) at the time the accident occurs is based on operation with a fuel defect level of 1.0%.

The amount of primary to secondary tube leakage is assumed to be 0.5 gpm in the faulted steam generator and 0.218 gpm in each of the intact steam generators. The tube leakage in both the faulted and intact steam generators is assumed to persist for 40 hours following initiation of the event.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The steam generator connected to the broken steamline is assumed to boil dry within two minutes following the MSLB. The entire liquid inventory of this steam generator is assumed to be steamed off and all of the iodine initially in this steam generator is released to the environment. Also, iodine carried over to the faulted steam generator by tube leakage is assumed to be released directly to the environment, with no credit taken for iodine retention in the steam generator. An iodine partition factor in the intact steam generators of 0.01 (curies I/gm steam)/(curies I/gm water) is used (Reference 1).

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

At 40 hours after onset of the accident, the residual heat removal system is assumed to remove all decay heat and there are no further steam releases to atmosphere from the secondary system.

No fuel failure (DNB or melt) is calculated to occur for the steam line break event (Sections 6.2.4 and 6.2.5).

The major assumptions and parameters used in this analysis are itemized in Table 6.7.2-1.

6.7.2.2 Acceptance Criteria

The offsite dose limits for an MSLB with a pre-existing iodine spike are provided in the SRP 15.1.5 Appendix A (Reference 1) as being the limits defined in 10CFR100 (i.e., 25 rem whole body and 300 rem thyroid). For an MSLB with an accident-initiated iodine spike, the acceptance criterion provided in Reference 1 is a “small fraction” of the 10CFR100 guideline values where “small fraction” is defined as 10 percent (i.e., 2.5 rem whole body and 30 rem thyroid).

6.7.2.3 Results and Conclusions for Byron Station

The offsite doses due to the MSLB with a pre-existing iodine spike are:

	Thyroid	Whole Body
Exclusion Area Boundary	4.8 rem	0.02 rem
Low Population Zone	0.5 rem	0.002 rem

The offsite doses due to the MSLB with an accident-initiated iodine spike are:

	Thyroid	Whole Body
Exclusion Area Boundary	6.3 rem	0.05 rem
Low Population Zone	2.7 rem	0.02 rem

The acceptance criteria are met.

6.7.2.4 Results and Conclusions for Braidwood Station

The offsite doses due to the MSLB with a pre-existing iodine spike are:

	Thyroid	Whole Body
Exclusion Area Boundary	6.5 rem	0.03 rem
Low Population Zone	2.2 rem	0.006 rem

The offsite doses due to the MSLB with an accident-initiated iodine spike are:

	Thyroid	Whole Body
Exclusion Area Boundary	8.5 rem	0.06 rem
Low Population Zone	12.2 rem	0.05 rem

The acceptance criteria are met.

6.7.2.5 References

1. NUREG-0800, Standard Review Plan, Section 15.1.5, Appendix, A, "Radiological Consequences of Main Steam Line Failures Outside of a PWR," Revision 2, July 1981.

Table 6.7.2-1 **Assumptions Used for Steamline Break Dose Analysis**	
Nuclide Parameters	See Table 6.7.1-1
Primary Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level (See Table 6.7.1-5)
Primary Coolant Iodine Activity Prior to Accident	
Pre-Existing Spike	60 μCi/gm of DE I-131 (See Table 6.7.1-6)
Accident-Initiated Spike	1.0 μCi/gm of DE I-131 (see Table 6.7.1-5)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate (See Table 6.7.1-6)
Duration of Accident-Initiated Spike	6.0 hours
Secondary Coolant Iodine Activity Prior to Accident	0.1 μCi/gm of DE I-131 (1/10 of Table 6.7.1-5 values)
Faulted SG Tube Leak Rate During Accident	0.5 gpm
Intact SGs Tube Leak Rate During Accident	0.654 gpm total for 3 SGs
SG Iodine Partition Factor	
Intact SG	0.01
Faulted SG	1.0
Duration of Activity Release from Secondary System	40 hours
Offsite Power	Lost
Steam Release from Intact SGs to Environment	
0-2 hours	442,000 lbm
2-8 hours	977,000 lbm
8-40 hours	2,216,000 lbm
Steam Release from Faulted SG to Environment (During First Two Minutes)	167,000 lbm
Offsite Breathing Rates	See Table 6.7.1-2
Offsite Atmospheric Dispersion Factors	See Table 6.7.1-2

6.7.3 Locked Rotor Accident

An instantaneous seizure of a reactor coolant pump (RCP) rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop. Fuel cladding damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products transfer from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming from the steam generators following the accident.

Following the guidelines in the Standard Review Plan (Reference 1), if the minimum DNBR falls below the limit, fuel failure must be assumed. In this case the gap activity of those rods would be released to the reactor coolant system and then be available for release via tube leakage and subsequent steaming.

6.7.3.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 6.7.3-1.

Source Term

The analysis of the locked rotor radiological consequences assumes a pre-existing iodine spike in the reactor coolant system. For the pre-existing iodine spike, it is assumed that a reactor transient has occurred prior to the event that has raised the RCS iodine concentration to 60 μCi/gm of dose equivalent I-131.

The noble gas activity concentration in the primary coolant when the accident occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant when the locked rotor occurs is assumed to be 0.10 μCi/gm of dose equivalent I-131.

As a result of the locked rotor event, only 0.1% of the fuel rods in the core undergo DNB (Section 6.2.12). In determining the offsite doses following the locked rotor event, it is conservatively assumed that 5% of the fuel rods in the core suffer sufficient damage that all of

their gap activity is released to the primary coolant. The percentage of the total core activity assumed to be in the fuel-cladding gap is 10%.

Iodine Chemical Form

The iodine is conservatively assumed to all be in the elemental form.

Release Pathway

Activity is released to the environment by way of primary to secondary leakage and steaming from the secondary side to the environment. The total primary to secondary steam generator tube leak rate used in the analysis is 1.0 gpm.

The Residual Heat Removal System is conservatively assumed to remove all decay heat at 40 hours into the accident and there are no further releases to the environment after that time.

Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 2).

6.7.3.2 Acceptance Criteria

The offsite dose limits for the Locked Rotor event are provided in the SRP (Reference 1) as being a "small fraction" of the 10CFR100 guideline values of 25 rem whole body and 300 rem thyroid, where "small fraction" is not defined but is assumed to be consistent with the definition of "small fraction" in other sections of the SRP as being 10 percent (i.e., 2.5 rem whole body and 30 rem thyroid).

6.7.3.3 Results and Conclusions for the Byron Station

The offsite doses due to the Locked Rotor are:

	Thyroid	Whole Body
Exclusion Area Boundary	4.1 rem	0.4 rem
Low Population Zone	1.4 rem	0.03 rem

The acceptance criteria are met.

6.7.3.4 Results and Conclusions for Braidwood Station

The offsite doses due to the Locked Rotor are:

	Thyroid	Whole Body
Exclusion Area Boundary	5.5 rem	0.5 rem
Low Population Zone	6.7 rem	0.1 rem

The acceptance criteria are met.

6.7.3.5 References

1. NUREG-0800, Standard Review Plan 15.3-3 – 15.3-4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," Rev. 2, July 1981.

2. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Revision 2, July 1981.

Table 6.7.3-1

Locked Rotor Accident Input Parameters and Assumptions

Source Term	
Core activity	See Table 6.7.1-4
Fission product gap fractions	
Iodines	10%
Noble gases	10%
Nuclide parameters	See Table 6.7.1-1
Fraction of fuel rods in core failing	5%
Iodine chemical form	All as elemental
Primary coolant activity before fuel failure	
Iodines	60 μCi/gm dose equivalent I-131 (see Table 6.7.1-6)
Noble Gases	Based on operation with 1.0% fuel defects (see Table 6.7.1-5)
Secondary coolant iodine activity at beginning of event	0.1 μCi/gm dose equivalent I-131 (10% of Table 6.7.1-5 values)
Release Path	
Primary coolant mass	2.063E8 gm
Secondary coolant mass	1.356E8 gm
Primary to Secondary leak rate (total)	1.0 gal/min
Steaming rate from the secondary side	
0 - 2 hr	2.72E6 gm/min
2 - 8 hr	1.40E6 gm/min
8 - 40 hr	6.30E5 gm/min
Steaming partition coefficient for iodine	0.01
Termination of releases	40 hours
Atmospheric Dispersion Factors (sec/m^3)	See Table 6.7.1-2
Breathing Rates	See Table 6.7.1-2

6.7.4 Locked Rotor with Power-Operated Relief Valve Failure

An instantaneous seizure of a reactor coolant pump (RCP) rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop. Fuel cladding damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products transfer from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident, and some of this activity is released to atmosphere as a result of steaming from the steam generators following the accident.

For the dose analysis, a power-operated relief valve (PORV) is assumed to fail open resulting in an uncontrolled blowdown of steam from one of the steam generators. The block valve for the failed open relief valve is assumed to be closed 20 minutes into the accident.

Following the guidelines in the Standard Review Plan (Reference 1), if the minimum DNBR falls below the limit, fuel failure must be assumed. In this case the gap activity of those rods would be released to the reactor coolant system and then be available for release via tube leakage and subsequent steaming.

6.7.4.1 Input Parameters and Assumptions

Major assumptions and parameters used in the analysis are itemized in Table 6.7.4-1.

Source Term

The analysis of the locked rotor radiological consequences assumes a pre-existing iodine spike in the reactor coolant system. For the pre-existing iodine spike, it is assumed that a reactor transient has occurred prior to the event that has raised the RCS iodine concentration to 60 μCi/gm of dose equivalent I-131.

The noble gas activity concentration in the primary coolant when the accident occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant when the locked rotor occurs is assumed to be 0.10 μCi/gm of dose equivalent I-131.

As a result of the locked rotor event, only 0.1% of the fuel rods in the core undergo DNB (Section 6.2.12). In determining the offsite doses following the locked rotor with a failed-open PORV, it is conservatively assumed that 2% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the primary coolant. (This is a reduction from the 5% assumed for the locked rotor without PORV failure; see Section 6.7.3.) The percentage of the total core activity assumed to be in the fuel-cladding gap is 10%.

Iodine Chemical Form

All iodine is conservatively assumed to be in elemental form.

Release Pathway

Activity is released to the environment by way of primary to secondary leakage and steaming from the secondary side to the environment. The primary to secondary steam generator tube leak rate used in the analysis is 0.5 gpm for the faulted steam generator and 0.218 gpm for each of the intact steam generators.

A PORV is assumed to fail open resulting in an uncontrolled blowdown of steam from one of the steam generators. The block valve for the failed-open PORV is assumed to be closed 20 minutes into the accident. This timing is consistent with that used in the UFSAR analysis for steam generator tube rupture.

The Residual Heat Removal System is conservatively assumed to remove all decay heat at 40 hours into the accident and there are no further releases to the environment after that time.

Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the intact steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 2). For the faulted steam generator (having the failed-open PORV), it

is conservatively assumed that there is a partition factor of 1.0 such that all of the iodine activity initially in the steam generator and all of the iodine activity in the primary coolant leaking into the steam generator is released to the environment.

6.7.4.2 Acceptance Criteria

The offsite dose limits for the Locked Rotor event are provided in the SRP (Reference 1) as being a "small fraction" of the 10CFR100 guideline values of 25 rem whole body and 300 rem thyroid, where "small fraction" is not defined but is assumed to be consistent with the definition of "small fraction" in other sections of the SRP as being 10 percent (i.e., 2.5 rem whole body and 30 rem thyroid). The dose at the Exclusion Area Boundary is considered for the first two hours of the event and the dose at the Low Population Zone outer boundary is determined for the duration of accident releases.

6.7.4.3 Results and Conclusions for the Byron Station

The offsite doses due to the Locked Rotor with failed-open PORV are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	14.3 rem	0.3 rem
Low Population Zone	0.9 rem	0.02 rem

The acceptance criteria are met.

6.7.4.4 Results and Conclusions for the Braidwood Station

The offsite doses due to the Locked Rotor with failed-open PORV are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	19.2 rem	0.3 rem
Low Population Zone	4.0 rem	0.06 rem

The acceptance criteria are met.

6.7.4.5 References

1. NUREG-0800, Standard Review Plan 15.3-3 – 15.3-4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," Rev. 2, July 1981.

2. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Revision 2, July 1981.

9.

Table 6.7.4-1

Locked Rotor Accident with Failed-Open PORV Input Parameters and Assumptions

Source Term	
Core activity	See Table 6.7.1-4
Fission product gap fractions	
Iodines	10%
Noble gases	10%
Nuclide parameters	See Table 6.7.1-1
Fraction of fuel rods in core failing	2%
Iodine chemical form	All as elemental
Primary coolant activity before fuel failure	
Iodines	60 μCi/gm dose equivalent I-131 (see Table 6.7.1-6)
Noble Gases	Based on operation with 1.0% fuel defects (see Table 6.7.1-5)
Secondary coolant iodine activity at beginning of event	0.1 μCi/gm dose equivalent I-131 (10% of Table 6.7.1-5 values)
Release Path	
Primary coolant mass	2.063E8 gm
Secondary coolant mass	
Intact SGs	1.017E8 gm
SG with failed-open PORV	7.575E7 gm
Primary to Secondary leak rate (per intact steam generator)	0.218 gal/min
Primary to Secondary leak rate (faulted steam generator)	0.5 gal/min
Steaming rate through failed-open PORV	
0 – 20 minutes	3.788E6 gm/min
> 20 minutes	0.0
Steaming rate from the secondary side (for intact SGs)	
0 - 2 hr	2.72E6 gm/min
2 - 8 hr	1.40E6 gm/min
8 - 40 hr	6.30E5 gm/min
Partition coefficient for iodine	
For steaming through failed-open PORV	1.0
For steaming through intact steam generators	0.01
Termination of releases	40 hours
Atmospheric Dispersion Factors	See Table 6.7.1-2
Breathing Rates	See Table 6.7.1-2

6.7.5 Rod Ejection Accident

It is assumed that a mechanical failure of a control rod drive mechanism pressure housing has occurred, resulting in ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melting (pellet centerline) are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive primary coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser, the atmospheric relief valves, or the main steam safety valves. Also, iodine activity is contained in the secondary coolant prior to the accident and some of this activity would be released to the atmosphere as a result of steaming from the steam generators following the accident. Finally, radioactive primary coolant is discharged to the containment via spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

6.7.5.1 Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 6.7.5-1.

<u>Source Term</u>

Per Reference 1, less than 10% of the fuel rods in the core would undergo DNB as a result of the rod ejection accident. In determining the offsite doses following this accident, it is conservatively assumed that 15% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. As specified in Regulatory Guide 1.77 (Reference 2), 10% of the core activity is assumed to be in the fuel/clad gap.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod ejection accident. This amounts to 0.375% of the core and the melting takes place in the centerline of the affected rods. The 0.375% of the fuel assumes that 15% of the rods in the core enter DNB. Of the rods that enter DNB, 50% are assumed to experience some melting of the fuel (7.5% of the core). Of the rods experiencing melting, 50% of the axial length of the rod is assumed to experience melting (3.75% of the core). It is further assumed that only 10% of

the radial portion of the rod experiences melting (0.375% of the total core). Only 50% of the iodines in the melted fuel are released (Reference 2).

A pre-existing iodine spike in the reactor coolant is assumed to have increased the primary coolant iodine concentration to 60 μCi/gm of dose equivalent I-131 prior to the rod ejection accident. The noble gas activity concentrations in the RCS when the accident occurs is based on operation with a fuel defect level of one percent. The iodine activity concentration of the secondary coolant when the rod ejection accident occurs is assumed to be equivalent to 0.10 μCi/gm of dose equivalent I-131.

Half of the iodines released to the containment are assumed to plate out on the containment surfaces.

Iodine Chemical Form

For the containment leakage release path, the iodine is assumed to be 91% elemental, 4% organic, and 5% particulate consistent with guidance for the LOCA (Reference 3). For the primary-to-secondary leakage and steaming release path, it is conservatively assumed that all iodine is in elemental form.

Release Pathways

Conservatively, all the iodine and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to the primary-to-secondary steam generator tube leakage. Primary-to-secondary tube leakage and steaming from the steam generators continues until the reactor coolant system pressure drops below the secondary pressure. A bounding time of 4000 seconds was selected for this analysis, although the analysis shows that this would occur well before then.

The primary-to-secondary steam generator tube leak used in the analysis is 1.0 gpm (total). Although the primary-to-secondary pressure differential drops throughout the event, a constant leakage rate is assumed.

When determining the offsite doses due to containment leakage, all of the iodine and noble gas activity is assumed to be in the containment. A containment leak rate of 0.1 wt% per day is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half this value, or 0.05 wt% per day (Reference 3).

Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 4).

For the containment leakage pathway, half of the iodine released is assumed to plate out on the containment surfaces. No credit is taken for any additional removal by either continued deposition onto containment surfaces or by containment spray operation which would remove airborne iodine.

6.7.5.2 Acceptance Criteria

The offsite dose limits for the rod ejection accident are "well within" (i.e. 25%) the dose guidelines of 10 CFR 100 (Reference 5). These limits are 6 rem whole body and 75 rem thyroid at the Exclusion Area Boundary (for the first 2 hours) and at the Low Population Zone outer boundary (for the duration of accident releases) per SRP 15.4.8 Appendix A (Reference 6).

6.7.5.3 Results and Conclusions for Byron Station

The offsite doses due to the Rod Ejection Accident are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	42.2 rem	1.0 rem
Low Population Zone	4.4 rem	0.05 rem

The acceptance criteria are met.

6.7.5.4 Results and Conclusions for Braidwood Station

The offsite doses due to the Rod Ejection Accident are:

	Thyroid	Whole Body
Exclusion Area Boundary	57.0 rem	1.4 rem
Low Population Zone	19.8 rem	0.2 rem

The acceptance criteria are met.

6.7.5.5 References

1. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

2. U. S. AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

3. U. S. AEC Regulatory Guide 1.4, "Assumptions Used for the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", Revision 2, June 1974.

4. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Revision 2, July 1981.

5. 10 CFR 100.11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance.

6. NUREG-0800, Standard Review Plan 15.4.8 Appendix A, "Radiological Consequences of a Control Rod Ejection Accident (PWR)," Revision 1, December 1981.

10.

Table 6.7.5-1
Assumptions Used for Rod Ejection Accident

Source Term	
Core activity	See Table 6.7.1-4
Nuclide parameters	See Table 6.7.1-1
Fission product gap fractions	10%
Fraction of fuel rods in core failing	15%
Fraction of fuel melting	0.375%
Activity release from melted fuel	
Iodines	50%
Noble gases	100%
Primary coolant activity before fuel failure	
Iodines	60 μCi/gm DE I-131 (see Table 6.7.1-6)
Noble Gas	1% fuel defects (see Table 6.7.1-5)
Secondary coolant iodine activity	0.1 μCi/gm DE I-131 (10% of values in Table 6.7.1-5)
Containment Leakage Release Path	
Fraction of iodine assumed to plate out	50%
Iodine chemical form	91% elemental, 4% organic, 5% particulate
Removal coefficients	None assumed
Leak rate	
0 - 24 hours	0.1 wt% per day
> 24 hours	0.05 wt% per day
Duration of releases	30 days
Steam Generator Steaming Release Path	
Primary coolant mass	2.063E8 gm
Secondary coolant mass	1.356E8 gm
Primary-to-Secondary leak rate (total)	1.0 gpm
Steaming rate from the secondary side	
0 – 200 seconds	3000 lb/sec
200 – 4000 seconds	500 lb/sec
> 4000 seconds	0 lb/sec
Iodine Chemical Form	100% elemental
Steaming partition coefficient	0.01
Atmospheric Dispersion Factors (sec/m^3)	See Table 6.7.1-2
Breathing Rates	See Table 6.7.1-2

6.7.6 Small Line Break Outside Containment

Rupture of the letdown line outside containment is assumed to occur. The noble gases and a portion of the iodine activity contained in the spilled coolant is released to atmosphere. This section describes the assumptions and analyses performed to determine the offsite doses resulting from the release of activity associated with this event.

6.7.6.1 Input Parameters and Assumptions

The analysis of the radiological consequences uses the analytical methods and assumptions outlined in Standard Review Plan (SRP) Section 15.6.2 (Reference 1). The activity available for release to the environment is the activity in the primary coolant (both iodine and noble gases).

The letdown line break is analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity, and the other in which an iodine spike is assumed to be initiated by the accident.

The noble gas activity concentration in the RCS when the accident occurs is based on a fuel defect level of 1.0%.

Letdown flow rate is assumed to be 140 gpm.

An iodine partition factor of 0.1 (curies/gm steam)/(curies/gm water) is assumed, consistent with SRP Section 15.6.5, Appendix B (Reference 2).

All noble gas activity in the spilled liquid is assumed to be immediately released to the outside atmosphere.

At 15 minutes into the accident, it is assumed that the operator acts to isolate the letdown line, thus terminating releases.

The major assumptions and parameters used in this analysis are itemized in Table 6.7.6-1.

6.7.6.2 Acceptance Criteria

The offsite dose limits for a small line break outside containment are provided in SRP 15.6.2 (Reference 1) as being a "small fraction" of the 10 CFR 100 guideline values, where "small fraction" is defined as 10 percent (i.e., 2.5 rem whole body and 30 rem thyroid).

6.7.6.3 Results and Conclusions for Byron Station

The offsite doses due to the small line break outside containment with a pre-existing iodine spike are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	1.0 rem	0.03 rem
Low Population Zone	0.03 rem	0.0008 rem

The offsite doses due to the small line break outside containment with an accident-initiated iodine spike are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	0.3 rem	0.03 rem
Low Population Zone	0.007 rem	0.0007 rem

The acceptance criteria are met.

6.7.6.4 Results and Conclusions for Braidwood Station

The offsite doses due to the small line break outside containment with a pre-existing iodine spike are:

	Thyroid	**Whole Body**
Exclusion Area Boundary	1.4 rem	0.04 rem
Low Population Zone	0.2 rem	0.004 rem

The offsite doses due to the small line break outside containment with an accident-initiated iodine spike are:

	Thyroid	Whole Body
Exclusion Area Boundary	0.3 rem	0.04 rem
Low Population Zone	0.03 rem	0.003 rem

The acceptance criteria are met.

6.7.6.5 References

1. NUREG-0800, Standard Review Plan 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981.

2. NUREG-0800, Standard Review Plan 15.6.5, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment," Revision 1, July 1981.

Table 6.7.6-1

Assumptions Used for Small Line Break Outside Containment Dose Analysis

Nuclide Parameters	See Table 6.7.1-1
Primary Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level (See Table 6.7.1-5)
Primary Coolant Iodine Activity Prior to Accident	
Pre-Existing Spike	60 μCi/gm of DE I-131 (See Table 6.7.1-6)
Accident-Initiated Spike	1.0 μCi/gm of DE I-131 (see Table 6.7.1-5)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate (See Table 6.7.1-6)
Letdown flow rate	140 gpm
Duration of Activity Release	905 seconds
Letdown water temperature (nominal)	290°F
Partitioning of iodine for spilled water	0.1
Auxiliary Building Filter Efficiency	90%
Breathing Rates	See Table 6.7.1-2
Atmospheric Dispersion Factors	See Table 6.7.1-2

6.7.7 Steam Generator Tube Rupture Transient Offsite Dose Calculations

A description of the inputs and initial conditions, analysis method, and the SGTR event is given in Section 6.3.

6.7.7.1 Input Parameters and Assumptions

The analysis of the radionuclide release during a postulated steam generator tube rupture accident is based on the guidance given in Section 15.6.3 of Reference 1. As indicated in the SRP, two distinct accident scenarios are considered: 1) the preaccident iodine spike case, where it is assumed that an iodine spike has occurred sometime prior to the steam generator tube rupture, and 2) the concurrent iodine spike case, where it is assumed that the tube rupture causes an iodine spike.

The primary and secondary coolant iodine activity concentrations used for these two cases are discussed below.

Preaccident Iodine Spike Case:

1. For the preaccident iodine spike case, the initial primary coolant iodine activity concentration is based on the maximum technical specification limit of 60 µCi/gm DE I-131 as per section III.6.(a) of the SRP (see Table 6.7.1-6).

2. Prior to the accident, the iodine concentration in the secondary coolant is based on the design basis assumption of 0.1 µCi/gm DE I-131 (see Table 6.7.1-6).

3. The initial primary coolant noble gas activity concentration is based on a 1% failed fuel defect level (see Table 6.7.1-57).

Concurrent Iodine Spike Case:

1. For the concurrent iodine spike case, it is assumed that the iodine release rate from the fuel rods increases to a value 500 times greater than the assumed normal operation release rate (see Table 6.7.1-6). The duration of the spike is conservatively assumed to

be 8 hours. The initial primary coolant iodine activity concentration is based on the maximum technical specification limit of 1 μCi/gm DE I-131 (see Table 6.7.1-5).

2. Prior to the accident, the iodine concentration in the secondary coolant is based on the design basis assumption of 0.1 μCi/gm DE I-131.

3. The initial primary coolant noble gas activity concentration is based on a 1% failed fuel defect level (see Table 6.7.1-57).

The offsite dose calculations are performed for two time periods: dose at exclusion area boundary (EAB) which is released between 0 to 2 hours and dose at low population zone (LPZ) which is released over the course of the accident. Therefore, steam releases are generally calculated for two time periods: from event initiation to 2 hours after the event, and from 2 hours after the event to the time when the residual heat removal (RHR) system is in service. After RHR is put into service, steaming is no longer required for plant cooldown and offsite release is terminated. For the SGTR event, steam releases are broken into three portions: event initiation to break flow termination, break flow termination to 2 hours after event initiation, and 2 hours after event initiation to RHR in service. Prior to termination of break flow, releases through the ruptured and intact steam generators are calculated by RETRAN-02. The steam releases from the other two time periods are determined from mass and energy balances. A listing of offsite dose case input is provided in Table 6.7.7-1.

6.7.7.2 Acceptance Criteria

The doses remain within the SGTR dose guidelines in 10 CFR 100 and Standard Review Plan 15.6.3.

6.7.7.3 Results and Conclusions

The offsite dose results for thyroid and whole body are given in Tables 6.7.7-2 and 6.7.7-3 respectively. The analysis includes the dose due to steam release from break flow termination to 8 hours after event initiation. While RHR is put into service 40 hours after event initiation, the steam release and associated dose from hour 8 to hour 40 are considered negligible and were not calculated per approved methodology (Reference 2). The radiological consequences are well within the 10 CFR 100 guidelines. Figures 6.7-1 and 6.7-2 provide the ruptured tube flow

as a function of time for Units 1 and 2 respectively. Figures 6.7-3 and 6.7-4 provide the average flashing fraction as a function of time for Units 1 and 2 respectively.

6.7.7.4 References

1. "Radiological Consequences of Steam Generator Tube Failure (PWR)," NRC Standard Review Plan (SRP), NUREG-0800, Rev. 2, July 1981.

2. "Revised Steam Generator Tube Rupture Analysis – Byron Unit 2, and Braidwood Unit 2 (TAC Nos. M97316 and M97318)," NRC Letter, Stewart N. Bailey to Oliver D. Kingsley, dated May 25, 1999.

Table 6.7.7-1 – Input Parameters for SGTR Offsite Dose Cases

1). Steam Released to the Environment as a Function of Time:

	Unit 1	Unit 2
Faulted SG		
0-2 Hours	9.75E4 lbm	9.42E4 lbm
2-8 Hours	2.69E4 lbm	2.66E4 lbm
Intact SGs		
0-2 Hours	5.53E5 lbm	5.44E5 lbm
2-8 Hours	1.20E6 lbm	1.17E6 lbm

2). Atmospheric Dispersion Factors (sec/m^3)

	Byron	Braidwood
EAB (0-2 hours)	5.7E-4	7.7E-4
LPZ (0-8 hours)	1.7E-5	7.1E-5

3). Primary to Secondary Leakrate = 1 gpm total for intact SG's

4). Breathing Rate (0 – 8 hours) = 3.47E-4 (m^3/sec)

5). Primary Mass — Unit 1 = 563,843 lbm

Unit 2 = 501,746 lbm

6). Secondary Mass — Unit 1 = 110,669 lbm

Unit 2 = 70,303 lbm

7). Partition Factors — 0.01 Iodine

1.0 Noble Gases

Table 6.7.7-2 - Offsite Dose Results (Thyroid)

Case	Unit 1 (rem)	Unit 2 (rem)	Acceptance Criteria (rem) SRP 15.6.3
Byron Preaccident Iodine Spike - EAB	10.4250	16.1987	300.0
Byron Preaccident Iodine Spike - LPZ	0.3280	0.5008	300.0
Byron Concurrent Iodine Spike - EAB	7.3148	10.0321	30.0
Byron Concurrent Iodine Spike - LPZ	0.2679	0.3640	30.0
Braidwood Preaccident Iodine Spike - EAB	14.0828	21.8823	300.0
Braidwood Preaccident Iodine Spike - LPZ	1.3697	2.0913	300.0
Braidwood Concurrent Iodine Spike - EAB	9.8810	13.5528	30.0
Braidwood Concurrent Iodine Spike - LPZ	1.1189	1.5203	30.0

Table 6.7.7-3 - Offsite Dose Results (Whole Body)

Case	Unit 1 (rem)	Unit 2 (rem)	Acceptance Criteria (rem) SRP 15.6.3
Byron Preaccident Iodine Spike - EAB	0.1619	0.2122	25.0
Byron Preaccident Iodine Spike - LPZ	0.0052	0.0067	25.0
Byron Concurrent Iodine Spike – EAB	0.2024	0.2586	2.5
Byron Concurrent Iodine Spike – LPZ	0.0082	0.0104	2.5
Braidwood Preaccident Iodine Spike - EAB	0.2188	0.2866	25.0
Braidwood Preaccident Iodine Spike - LPZ	0.0208	0.0270	25.0
Braidwood Concurrent Iodine Spike - EAB	0.2735	0.3494	2.5
Braidwood Concurrent Iodine Spike - LPZ	0.0265	0.0338	2.5

Figure 6.7-1 – Ruptured Tube Flow – Unit 1

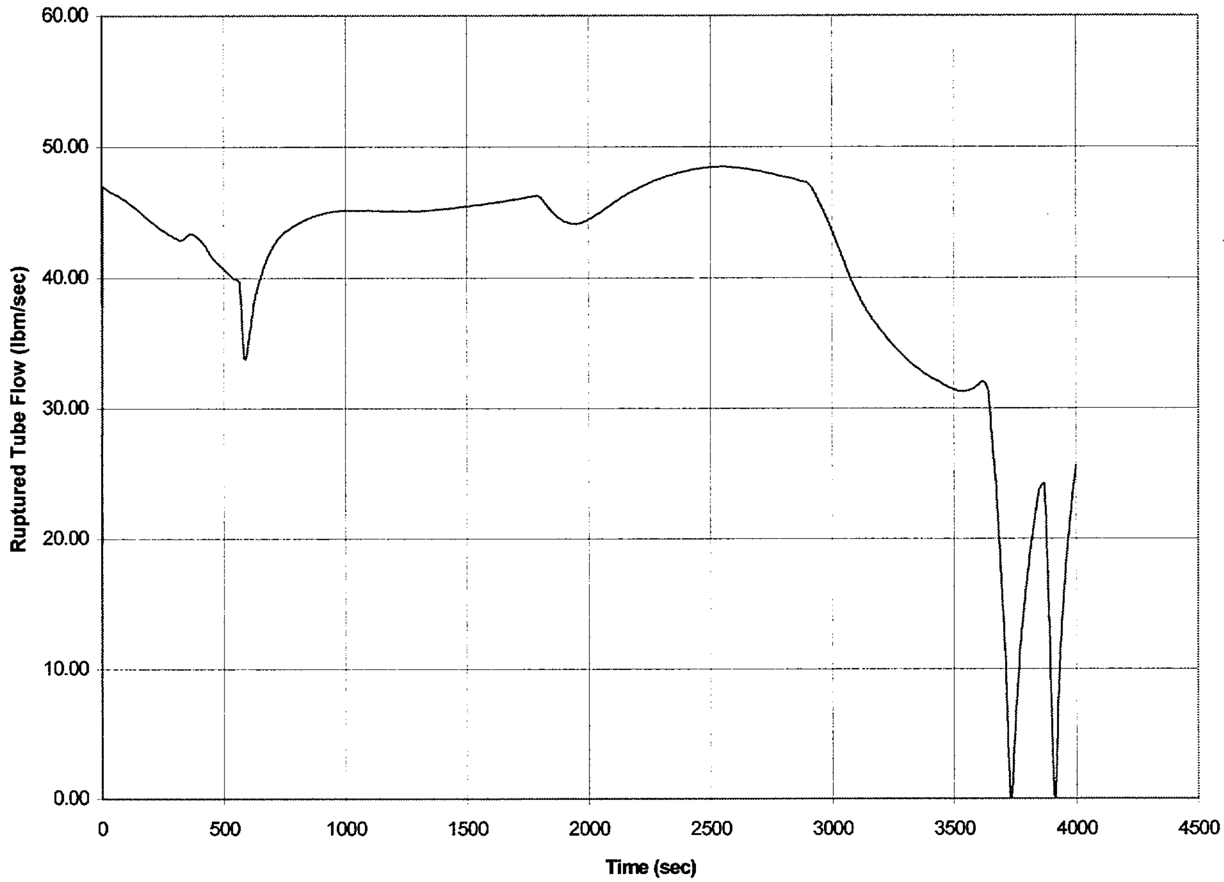


Figure 6.7-2 – Ruptured Tube Flow – Unit 2

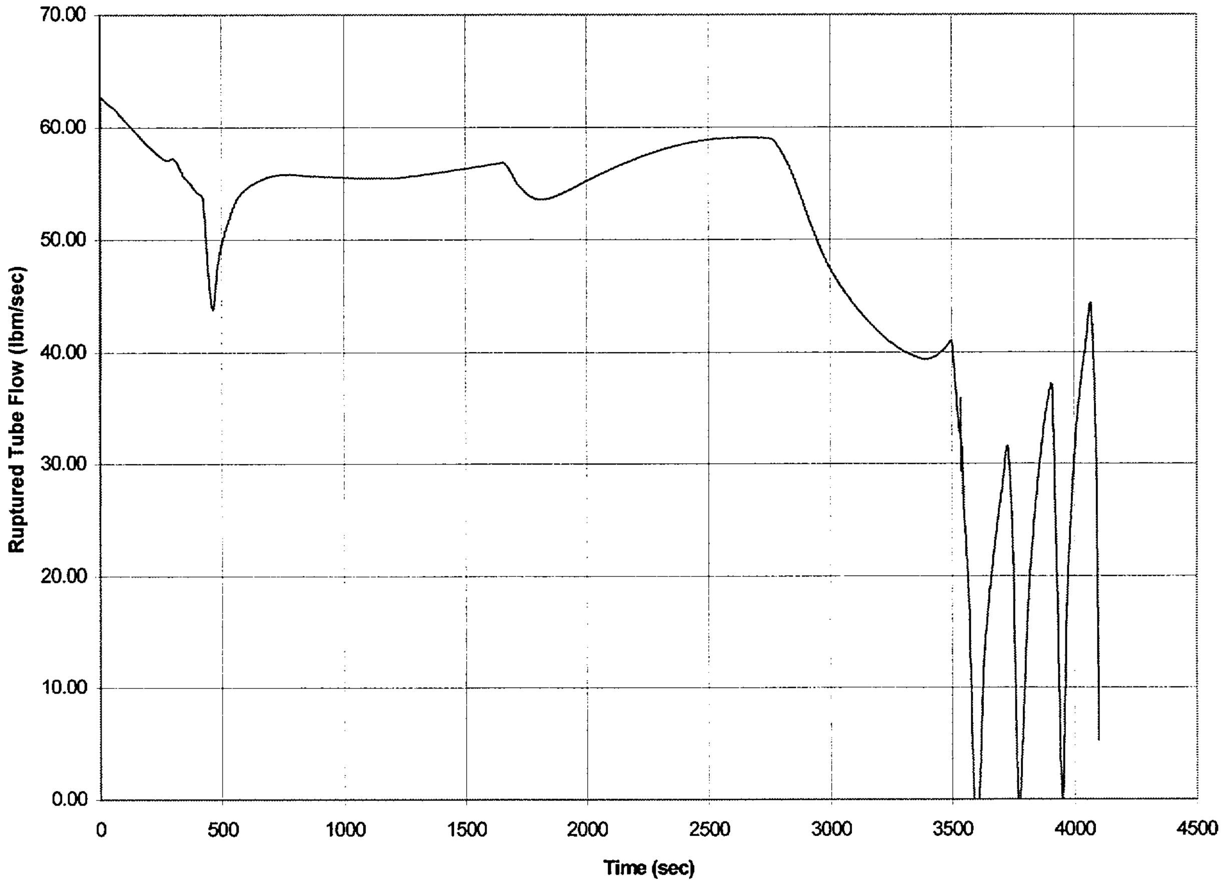


Figure 6.7-3 – Average Flashing Fraction – Unit 1

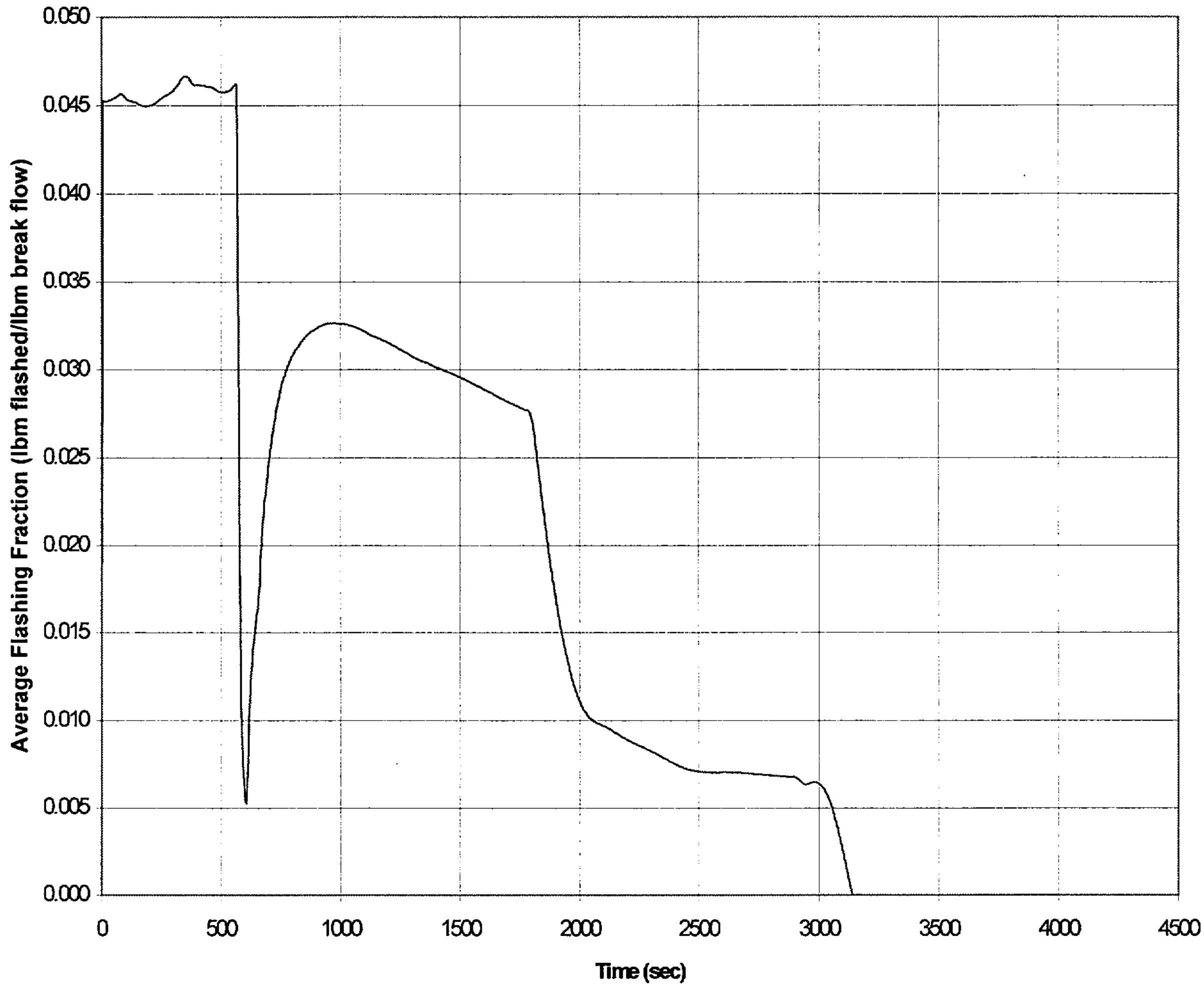
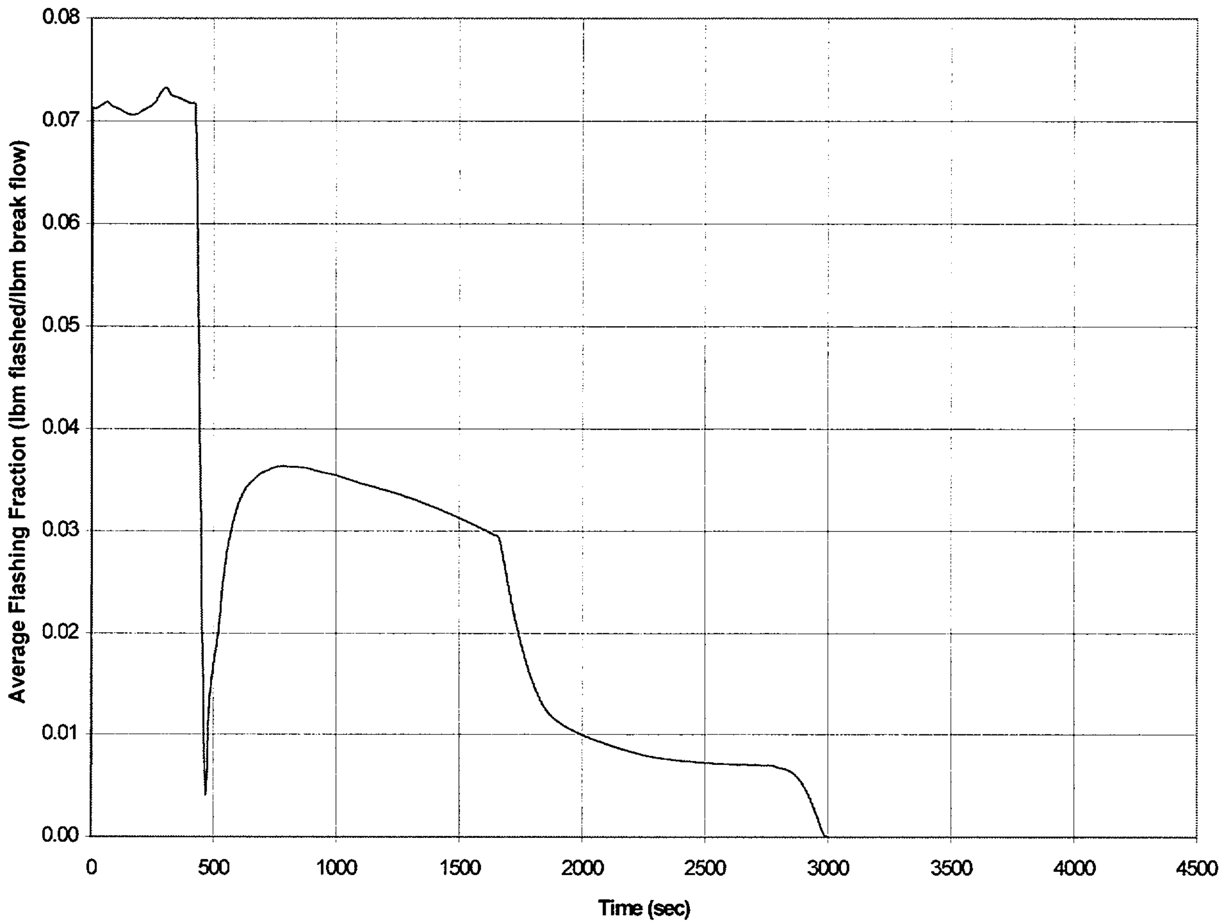


Figure 6.7-4 – Average Flashing Fraction – Unit 2



6.7.8 Large-Break Loss of Coolant Accident

An abrupt failure of a reactor coolant pipe is assumed to occur, and it is also assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and then to the environment by means of containment leakage or leakage from the emergency core cooling system as it recirculates sump solution outside containment.

6.7.8.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.7.8-1. Activity from the damaged core is released into the containment. The analysis considers the release of activity from the containment via containment leakage and leakage from external recirculation of the sump solution. The following sections address topics of significant interest.

While the current UFSAR addresses additional activity release due to the assumed venting of the containment to reduce hydrogen concentration, this release pathway has been eliminated from consideration. With redundant, safety-grade hydrogen recombiners available on the plant site, failure of one recombiner would not result in lost recombination capability.

The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths considered.

6.7.8.1.1 Source Term

The reactor coolant activity is assumed to be insignificant compared with the release from the core, and is not included in the analysis.

Instantaneous melting of the core and release of activity to the containment is assumed in the analysis. Release of only iodines and noble gases are assumed for calculation of the whole body and thyroid doses. Of the total core iodines provided in Table 6.7.1-4, 50% is assumed to be retained in the melted fuel. Therefore, only 50% of the iodines are released from the fuel to the containment atmosphere. For the noble gases, 100% of the total core activity

(Table 6.7.1-4) is assumed to be released to the containment atmosphere and available for release to the environment via containment leakage.

For the containment leakage analysis, 50% of the iodine released from the fuel is assumed to plate-out on the containment surfaces. Therefore only 25% of the total core iodines are assumed to be in the containment atmosphere (Reference 1) until removed by sprays, radioactive decay or leakage from the containment. For the ECCS leakage analysis, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

The iodine exists in the form of 91% elemental, 4% organic and 5% particulate in the containment atmosphere (Reference 1). The iodine species in the sump solution for the ECCS leakage and passive failure cases is conservatively assumed to be 100% elemental. All noble gas activity is assumed to be in the containment atmosphere.

6.7.8.1.2 Containment Modeling

The containment is modeled as two discrete volumes representing the sprayed and unsprayed regions. The volumes are conservatively assumed to be mixed by the flow of two of the four deck fans.

The containment leakage analyses do not model the time required for isolation of the containment, since there would be very little activity in the containment that early in the event.

6.7.8.1.3 Removal of Activity from the Containment Atmosphere

Credit for removal of elemental and particulate iodine from the containment atmosphere is taken only for containment sprays and radioactive decay. The noble gases and organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate following the LOCA. Spray injection is initiated at 90 seconds. When the RWST drains to a predetermined setpoint level, the operators switch to recirculation of sump liquid to provide a source for the sprays. The switchover is assumed to take 10 minutes. During this 10 minutes, the analysis does not credit

any spray removal in the containment. The analysis conservatively assumes that the switchover is initiated at 1254 seconds from the start of the spray injection.

Containment Spray Removal of Elemental Iodine

The Standard Review Plan (Reference 2) identifies a methodology to determine spray removal of elemental iodine. The removal rate constant is determined by:

$$\lambda_s = 6K_gTF/VD$$

where

λ_s = Removal rate constant due to spray removal, hr^{-1}

K_g = Gas phase mass transfer coefficient, ft/min

T = Time of fall of the spray drops, min

F = Volume flow rate of sprays, ft^3/hr

V = Containment sprayed volume, ft^3

D = Mass-mean diameter of the spray drops, ft

The upper limit specified for this model is 20 hr^{-1}.

Parameters are listed below and were chosen to bound the current plant configuration:

K_g = 9.84 ft/min

T = 11.5 sec

F = 2950 gpm

V = 2.35E6 ft^3

D = 0.105 cm

These parameters and appropriate conversion factors were used to calculate the elemental spray removal coefficients. The calculated value is well above the identified upper limit of 20 hr^{-1}.

Removal of elemental iodine from the containment atmosphere is assumed to be terminated when the spray injection phase is terminated or the airborne inventory drops to one percent of

the total elemental iodine released to the containment (this is a DF of 100), although Reference 2 specifies a DF of 200. Since plate-out on the containment surfaces of 50% was assumed in the analysis, this provides an overall DF of 200 for the elemental iodine release from the fuel. From the analysis, the DF of 100 is not reached by the end of the spray injection phase; the achieved DF is just short of 75.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 2.

The first order spray removal rate constant for particulates is written as follows:

$$\lambda_p = 3hFE/2VD$$

where

h = Drop Fall Height
F = Spray Flow Rate
V = Volume Sprayed
E = Single Drop Collection Efficiency
D = Average Spray Drop Diameter

Parameters are listed below and were chosen to bound the current plant configuration:

H = 141 ft
F = 2950 gpm
V = 2.35E6 ft^3

The E/D term depends upon the particle size distribution and spray drop size. From Reference 2, it is conservative to use 10 m^{-1} for E/D until the point when the inventory in the atmosphere is reduced to 2% of its original (DF of 50), and the value is then reduced to 1.0 m^{-1}.

These parameters and appropriate conversion factors were used to calculate the particulate spray removal coefficients. The conservative value used in the analysis is listed in Table 6.7.8-1. When the airborne inventory drops to two percent of the total particulate iodine

released to the containment (a DF of 50), the removal coefficient is reduced by a factor of 10. In the analysis, this occurs just after one hour.

The analysis assumes that recirculation spray continues for minimum of 2 hours from the start of the event.

6.7.8.1.4 ECCS Leakage

After the injection phase is over, the containment sump solution is recirculated outside containment and may leak, resulting in activity release. The analysis considers a leak of 7820 cc/hr, which is twice the UFSAR limit of 3910 cc/hr.

The concentration of activity in the recirculating sump solution is determined based on the calculated sump volume over the course of the transient. Although recirculation is not initiated until the RWST has drained to the pre-determined setpoint level, the analysis conservatively considers this leakage from the start of the event. All iodine activity in the recirculated water is assumed to be elemental in form.

The Byron and Braidwood plants have an ESF ventilation system providing filtration of air discharged from portions of the auxiliary building which contain the ECCS recirculation path. Therefore, a passive failure in the ECCS recirculation system is not considered.

6.7.8.2 Acceptance Criteria

The offsite dose limits for the large-break LOCA are that they meet the dose guidelines of 10 CFR 100 (Reference 3). These limits are 25 rem whole body and 300 rem thyroid at the Exclusion Area Boundary (for the first 2 hours) and at the Low Population Zone outer boundary (for the duration of the accident - taken as 30 days). The control room dose acceptance criteria are 5 rem whole body and 30 rem thyroid for the 30 day duration of the event (Reference 4). The acceptance criterion for the control room beta-skin dose is specified by Reference 4 to be 30 rem (or 75 rem if credit is taken for protective clothing). As shown in Sections 6.7.8.3 and 6.7.8.4, Byron and Braidwood Stations meet the 30 rem acceptance criteria.

6.7.8.3 Byron Station Results and Conclusions

The offsite and control room doses due to the large-break LOCA from the containment leakage release pathway are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	64 rem	2.6 rem	N/A
Low Population Zone	6.6 rem	0.2 rem	N/A
Control Room	18 rem	1.8 rem	29.7 rem

The offsite and control room doses due to the large-break LOCA from the ECCS recirculation leakage release pathway are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	1.6 rem	7.5E-3 rem	N/A
Low Population Zone	0.31 rem	5.9E-4 rem	N/A
Control Room	0.34 rem	1.4E-5 rem	1.3E-4 rem

The acceptance criteria are met for both the exclusion area boundary and low population zone doses. For the control room, the thyroid, whole body, and beta-skin doses meet the acceptance criteria.

6.7.8.4 Braidwood Station Results and Conclusions

The offsite and control room doses due to the large-break LOCA from the containment leakage release pathway are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	86 rem	3.4 rem	N/A
Low Population Zone	31 rem	0.7 rem	N/A
Control Room	18 rem	1.8 rem	29.5 rem

The offsite and control room doses due to the large-break LOCA from the ECCS recirculation leakage release pathway are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	2.1 rem	1.1E-2 rem	N/A
Low Population Zone	1.6 rem	2.7E-3 rem	N/A
Control Room	0.3 rem	1.3E-5 rem	1.3E-4 rem

The acceptance criteria are met for both the exclusion area boundary and low population zone doses. For the control room, the thyroid, whole body, and beta-skin doses meet the acceptance criteria.

6.7.8.5 References

1. U. S. Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Rev. 2, June 1974.

2. NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.

3. 10CFR100.11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance.

4. NUREG–0800, Standard Review Plan 6-4, “Control Room Habitability System,” Rev. 2, July 1981

Table 6.7.8-1

Assumptions Used for Large-Break LOCA Analysis

Source Term	
Core Activity	See Table 6.7.1-4
Nuclide Parameters	See Table 6.7.1-1
Activity Release from the Fuel	
Iodines	50%
Noble Gases	100%
Iodine Plate-out on Containment Surfaces	50%
Iodine chemical form in containment	
Elemental	91%
Organic	4%
Particulate (cesium iodide)	5%
Iodine chemical form in sump & recirculating liquid	100% elemental
Containment	
Containment Volume	
Sprayed	2.35E6 ft^3
Unsprayed	5.0E5 ft^3
Deck Fan Operation	
Number of fans operating	2
Flow rate	65,000 cfm per fan
Leak Rate	
0 – 24 hours	0.1 wt% per day
> 24 hours	0.05 wt% per day
Duration of releases	30 days
Spray Operation	
Time to initiate sprays	0.025 hours
Termination of spray injection phase	0.373 hours
Beginning of recirculation spray phase	0.54 hours
Terminate recirculation spray	2 hours
Spray Flow Rate (both injection and recirculation)	2950 gpm
Spray Fall Height	141 feet

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Table 6.7.8-1 (cont.)

Assumptions Used for Large-Break LOCA Analysis

Spray Removal Coefficients		
Elemental iodine (injection spray phase)	20.0 hr^{-1}	
Particulates (until DF = 50)	6.0 hr^{-1}	
Particulates (after DF = 50)	0.6 hr^{-1}	
Elemental iodine DF limit	100	
Sump Volume		
At beginning of ECCS recirculation (0.193 hr)	38,979 ft^3	
At beginning of spray recirculation (0.373 hr)	58,506 ft^3	
ECCS Leak Rate (0 – 30 days)	7820 cc/hr	
Passive Failure Leak Rate (24 – 24.5 hours)	N/A	
Iodine partioning from ECCS leakage	10%	
Auxiliary Building HVAC Exhaust Filter Efficiency (for elemental iodine)	90%	
Breathing Rates (offsite)	See Table 6.7.1-2	
Control Room Parameters	See Table 6.7.1-3	
Atmospheric Dispersion Factors		
Offsite	See Table 6.7.1-2	
Control room – containment leakage		
(containment ground level release)	**Byron**	**Braidwood**
0 – 2 hr	6.1E-3 sec/m^3	6.2E-3 sec/m^3
2 – 8 hr	5.3E-3 sec/m^3	5.37E-3 sec/m^3
8 – 24 hr	2.68E-3 sec/m^3	2.79E-3 sec/m^3
24 – 96 hr	2.0E-3 sec/m^3	1.82E-3 sec/m^3
96 – 720 hr	1.53E-3 sec/m^3	1.32E-3 sec/m^3
Control room – ECCS recirculation leakage		
	Byron	**Braidwood**
0 – 2 hr	2.28E-3 sec/m^3	2.48E-3 sec/m^3
2 – 8 hr	1.91E-3 sec/m^3	1.87E-3 sec/m^3
8 – 24 hr	8.88E-4 sec/m^3	8.11E-4 sec/m^3
24 – 96 hr	5.97E-4 sec/m^3	5.04E-4 sec/m^3
96 – 720 hr	4.77E-4 sec/m^3	3.91E-4 sec/m^3

6.7.9 Small-Break Loss of Coolant Accident

The following discussion on SBLOCA is not part of B/B Station licensing basis and is provided for information only.

An abrupt failure of the primary coolant system is assumed to occur. It is also assumed that the break is small enough that the containment spray system is not actuated by high containment pressure, but that the core experiences some cladding damage such that the fission product gap activity of damaged fuel rods is released. The analysis conservatively assumes that the gap activity of all rods in the core is released. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. There is also a release path through the steam generators (primary-to-secondary leakage) until the primary system depressurizes to below the secondary system pressure.

6.7.9.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.7.9-1. The following sections address topics of significant interest that benefit from extended discussion.

<u>Source Term</u>

The fission product gap fraction (fraction of core activity in the pellet-to-cladding gap) is assumed to be 10%. In addition, although it does not significantly increase the source term, the contribution from the initial primary coolant inventory prior to the accident is included.

For the containment leakage pathway, it is assumed that all of the noble gases and iodines are instantaneously released into the containment atmosphere (no retention of iodine in the primary coolant is assumed).

For the primary-to-secondary leakage pathway, the activity released from the fuel is conservatively assumed to remain in the primary coolant (transfers to the containment are ignored) and available to leak into the secondary coolant. The initial activity in the secondary coolant is trivial compared with the activity assumed to leak into the secondary side and is not included in the analysis.

Iodine Chemical Form

The chemical form of the iodine initially in the containment atmosphere is 95.5% elemental, 2% organic and 2.5% particulate (based on Reference 1 but eliminating the assumption of 50% plate-out of elemental iodine). For the primary-to-secondary leakage pathway, the iodine is conservatively assumed to be 100% elemental.

Release Pathways

Conservatively, all the iodine and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment) when determining doses due to primary-to-secondary steam generator tube leakage. The primary-to-secondary tube leakage and steaming from the steam generators continues until the reactor coolant system pressure drops below the secondary pressure at 4000 seconds.

The primary-to-secondary steam generator tube leak rate used in the analysis is 1.0 gpm (total). Although the primary-to-secondary pressure differential drops throughout the event, the flow rate is maintained constant.

When determining the doses due to containment leakage, all of the iodine and noble gas activity is assumed to be in the containment. A containment leak rate of 0.1 wt% per day is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half, or 0.05 wt% per day (Reference 1).

Removal Coefficients

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

An iodine partition factor in the steam generators of 0.01 curies/gm steam per curies/gm water is used (Reference 2).

Deposition removal of elemental iodine from the containment atmosphere is determined using the model described in the Standard Review Plan (SRP) Section 6.5.2 (Reference 3).

The first order deposition removal rate constant for elemental iodine is written as follows:

$$\lambda_d = kA/V$$

where

k = Mass transfer coefficient = 4.9 m/hr (Reference 3)

A = Area available for deposition = ft^2

V = Containment volume = ft^3

The resulting deposition removal coefficient is greater than 2 hr^{-1}. A value of 2.0 hr^{-1} is conservatively used in the calculation. Consistent with the SRP Section 6.5.2 (Reference 3), removal of elemental iodine is terminated when a DF of 200 is reached.

It is assumed that the containment spray system is not actuated (operation of the containment spray system would rapidly remove airborne particulates and elemental iodine).

6.7.9.2 Acceptance Criteria

The offsite dose limits for the small-break LOCA are that they meet the dose guidelines of 10 CFR 100 (Reference 4). These limits are 25 rem whole body and 300 rem thyroid at the Exclusion Area Boundary (for the first 2 hours) and at the Low Population Zone outer boundary (for the duration of the accident - taken as 30 days). The control room dose acceptance criteria are 5 rem whole body, 30 rem thyroid, and 30 rem beta-skin (Reference 5).

6.7.9.3 Byron Station Results and Conclusions

The doses due to the small-break LOCA from the containment leakage and steam generator steaming release pathways are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	193 rem	5.2 rem	N/A
Low Population Zone	7.6 rem	0.2 rem	N/A
Control Room	12.4 rem	1.0 rem	13.3 rem

The acceptance criteria are met.

6.7.9.4 Braidwood Station Results and Conclusions

The doses due to the small-break LOCA from the containment leakage and steam generator steaming release pathways are:

	Thyroid	Whole Body	Beta-Skin
Exclusion Area Boundary	260 rem	7.0 rem	N/A
Low Population Zone	32.5 rem	0.7 rem	N/A
Control Room	12.3 rem	1.0 rem	13.6 rem

The acceptance criteria are met.

6.7.9.5 References

1. U. S. Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Rev. 2, June 1974.

2. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)," Revision 2, July 1981.

3. NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.

4. 10CFR100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".

5. NUREG-0800, Standard Review Plan 6.4, "Control Room Habitability System," Revision 2, July 1981.

Table 6.7.9-1

Assumptions Used for Small-Break LOCA Analysis

Source Term		
Core Activity	See Table 6.7.1-4	
Nuclide Parameters	See Table 6.7.1-1	
Fission product gap fractions		
Iodines	10%	
Noble Gases	10%	
Fraction of fuel rods in core failing	100%	
Iodine chemical form in containment		
Elemental	95.5%	
Organic	2%	
Particulate (cesium iodide)	2.5%	
Iodine Chemical Form (steam generator steaming path)		
Iodine chemical form	100% elemental	
Containment Leakage Release Path		
Deposition removal coefficient (elemental iodine)	2.0 hr^{-1}	
DF limit for elemental iodine removal	200	
Leak rate		
0 – 24 hours	0.1 wt% per day	
> 24 hours	0.05 wt% per day	
Duration of releases	30 days	
Steam Generator Steaming Release Path		
Primary coolant mass	2.063E8 gm	
Secondary coolant mass (total of 4 steam generators)	1.356E8 gm	
Primary to Secondary leak rate	1 gpm	
Steaming rate from the secondary side		
0 – 200 seconds	3000 lb/sec	
200 – 4000 seconds	500 lb/sec	
> 4000 seconds	none	
Steaming partition coefficient	0.01	
Breathing Rates	See Table 6.7.1-2	
Control Room Parameters	See Table 6.7.1-3	
Atmospheric Dispersion Factors		
Offsite	See Table 6.7.1-2	
Control room – containment leakage (containment ground level release)	**Byron**	**Braidwood**
0 – 2 hr	6.1E-3 sec/m^3	6.2E-3 sec/m^3
2 – 8 hr	5.3E-3 sec/m^3	5.37E-3 sec/m^3
8 – 24 hr	2.68E-3 sec/m^3	2.79E-3 sec/m^3
24 – 96 hr	2.0E-3 sec/m^3	1.82E-3 sec/m^3
96 – 720 hr	1.53E-3 sec/m^3	1.32E-3 sec/m^3
Control room – plant stack release	**Byron**	**Braidwood**
0 – 2 hr	3.98E-3 sec/m^3	4.08E-3 sec/m^3
2 – 8 hr	3.48E-3 sec/m^3	3.43E-3 sec/m^3
8 – 24 hr	1.64E-3 sec/m^3	1.69E-3 sec/m^3
24 – 96 hr	1.04E-3 sec/m^3	9.78E-4 sec/m^3
96 – 720 hr	8.96E-4 sec/m^3	6.56E-4 sec/m^3

6.7.10 Gas Decay Tank Rupture Radiological Consequences

For the gas decay tank rupture analysis, there is assumed to be a failure that results in the release of the contents of one gas decay tank.

6.7.10.1 Input Parameters and Assumptions

The major assumptions and parameters used to determine the doses due to gas decay tank failure are given in Table 6.7.10-1.

The inventory of gases in the tank is conservatively determined by assuming that a single unit is operated for a full cycle without purging gases from the volume control tank, thus maximizing the coolant noble gas inventory. The unit is then assumed to be degassed for shutdown with all gaseous activity transferred to a single gas decay tank (no initial activity is considered in the gas decay tank). Consideration is made of radioactive decay during the process of transferring the noble gases to the decay tank. However, once the maximum inventory of a particular nuclide is achieved in the tank, no further decay of that nuclide is taken into account. This approach conservatively maximizes the inventory for each nuclide. The gas decay tank inventory resulting from this unit shutdown degassing process is provided in Table 6.7.10-1.

A failure in the gaseous waste processing system is assumed to result in release of the tank inventory with a conservatively short release duration of one minute.

6.7.10.2 Acceptance Criteria

Per the current licensing basis in the Byron & Braidwood UFSAR, the offsite doses must be "well within" the guidelines of 10 CFR 100. This term is defined as being 25% of the 10 CFR 100 limits, or 75 rem thyroid and 6 rem whole body at the exclusion area boundary and at the low population zone (Reference 1).

6.7.10.3 Byron Station Results and Conclusions

The offsite doses due to the gas decay tank rupture are:

	Whole Body
Exclusion Area Boundary	0.54 rem
Low Population Zone	0.02 rem

The acceptance criteria are met.

6.7.10.4 Braidwood Station Results and Conclusions

The offsite doses due to the gas decay tank rupture are:

	Whole Body
Exclusion Area Boundary	0.73 rem
Low Population Zone	0.07 rem

The acceptance criteria are met.

6.7.10.5 References

1. NUREG-0800, Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, December 1981.

2. Safety Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure" March 1972.

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Table 6.7.10-1

Assumptions Used for Gas Decay Tank Rupture Dose Analysis

Gas Decay Tank Inventory (Ci)	
Kr-85M	1.75E2
Kr-85	5.24E3
Kr-87	4.41E1
Kr-88	2.22E2
XE-131m	1.07E3
Xe-133m	9.24E2
Xe133	7.51E4
X-135m	7.71E1
Xe-135	1.06E3
Xe-138	5.98E0
Nuclide Parameters	Table 6.7.1-1
Duration of release	One minute
Atmospheric Dispersion Factors	Table 6.7.1-2

6.7.11 Liquid Waste Tank Rupture

Two separate tank failure scenarios are considered: one involving a boron recycle holdup tank and the other a spent resin storage tank. The iodine and noble gas activity released from the failed tank passes to the outside atmosphere through the auxiliary building ventilation system with no credit for removal of any of the activity.

6.7.11.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 6.7.11-1.

<u>Source Term in the Boron Recycle Holdup Tank</u>

The inventory of gases in the tank is based on transferring all primary coolant activity to the tank with the letdown line operating at maximum purification flow. No transfer of activity to the gas decay tanks is taken into account.

The inventory of iodine in the tank is based on the assumptions that the primary coolant iodine activity is at the Technical Specification limit for equilibrium operation (1.0 μCi/gram dose equivalent I-131) and that the mixed bed demineralizer in the letdown line is operating with a decontamination factor of 10 (i.e., 90 percent of the iodine is removed by the demineralizer). The tank is assumed to be filled to 80% of capacity with water consistent with SRP guidance (Reference 1).

<u>Source Term in the Spent Resin Storage Tank</u>

The tank is assumed to be filled to 80% of capacity with a resin and water slurry consistent with Reference 1 guidance. The resin is conservatively assumed to be all from the letdown line mixed bed demineralizer. It is also assumed that the primary coolant passing through the demineralizer is at the Technical Specification iodine concentration limit for equilibrium operation (1.0 μCi/gm dose equivalent I-131). The demineralizer is assumed to remove all iodine from the process flow.

It is conservatively assumed that spent resin from both units is being directed to one tank and that resin is transferred to the spent resin storage tank once an equilibrium level of iodine is

attained on the demineralizer bed. This results in a conservatively short time to fill the tank with spent resin. Decay is taken into account only after the resin is transferred to the spent resin storage tank. One percent of the iodine stored on the resin is assumed to enter the water in the spent resin storage tank.

No noble gases are considered in the source term, since the tank is vented and no significant amount of activity would accumulate in the tank gas space.

Iodine Form

For both tank failure models, all iodine is conservatively assumed to be entirely in elemental form.

Release Model

For each tank failure analysis, consistent with the guidance in the Standard Review Plan (SRP) Section 15.6.5, Appendix B (Reference 2), 10% of the iodine in the water is assumed to become airborne. For the case of the recycle holdup tank, it is assumed that all of the noble gas activity is released.

All of the activity released from the failed tank is assumed to be released to the environment within five minutes. No credit is taken for any reduction in iodine releases by filters in the auxiliary building ventilation flow path.

6.7.11.2 Acceptance Criteria

Per the current licensing basis in the Byron & Braidwood UFSAR, the offsite doses must be "well within" the guidelines of 10 CFR 100. This term is defined as being 25% of the 10 CFR 100 limits, or 75 rem thyroid and 6 rem whole body at the exclusion area boundary and at the low population zone (Reference 3).

6.7.11.3 Byron Station Results and Conclusions

The offsite doses due to the recycle holdup tank rupture are:

	Thyroid	Whole Body
Exclusion Area Boundary	0.85 rem	0.44 rem
Low Population Zone	0.026 rem	0.014 rem

The offsite doses due to the spent resin storage tank rupture are:

	Thyroid	Whole Body
Exclusion Area Boundary	0.45 rem	1.6E-4 rem
Low Population Zone	0.014 rem	4.7E-6 rem

The acceptance criteria are met.

6.7.11.4 Braidwood Station Results and Conclusions

The offsite doses due to the recycle holdup tank rupture are:

	Thyroid	Whole Body
Exclusion Area Boundary	1.2 rem	0.6 rem
Low Population Zone	0.11 rem	0.055 rem

The offsite doses due to the spent resin storage tank rupture are:

	Thyroid	Whole Body
Exclusion Area Boundary	0.61 rem	2.1E-4 rem
Low Population Zone	0.056 rem	2.0E-5 rem

The acceptance criteria are met.

6.7.11.5 References

1. NUREG-0800, Standard Review Plan 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," Revision 2, July 1981.

2. NUREG-0800, Standard Review Plan 15.6.5, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment," Revision 1, July 1981.

3. NUREG-0800, Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, December 1981.

4. Safety Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," March 1972.

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Table 6.7.11-1 Assumptions Used for Liquid Waste Tank Failure Dose Analysis	
Nuclide Parameters	See Table 6.7.1-1
Primary Coolant Noble Gas Activity	1.0% Fuel Defect Level (See Table 6.7.1-5)
Primary Coolant Initial Iodine Activity	1.0 μCi/gm of DE I-131 (see Table 6.7.1-5)
Breathing Rates	See Table 6.7.1-2
Atmospheric Dispersion Factors	See Table 6.7.1-2
Boron Recycle Holdup Tank Model	
Tank Volume	125,000 gallons
Fraction of Tank Filled with Water	80%
Letdown flow rate	132 gpm
Partitioning of iodine for spilled water	10%
Duration of Activity Release	5 minutes
Spent Resin Storage Tank Model	
Tank Volume	5000 gallons
Fraction of Tank Filled with Resin/Water Slurry	80%
Letdown flow rate	132 gpm
Demineralizer resin volume	39 cubic feet
Letdown Line Mixed Bed Demineralizer DF	infinite
Partitioning of iodine for spilled water	10%
Duration of Activity Release	5 minutes

6.7.12 Fuel Handling Accident

A fuel assembly is assumed to be dropped and damaged during refueling. While a fuel handling accident (FHA) could occur either inside or outside the containment, the Byron & Braidwood licensing basis is that doses are calculated only for the accident occurring outside the containment. This is based on arguments that show that, in the event of an FHA inside the containment, the containment would be automatically isolated before any activity would be released. A large fraction of the iodine activity released from the damaged fuel is retained in the spent fuel pool. The iodine and noble gas activity released from the water pool passes to the outside atmosphere through the auxiliary building ventilation system which removes a fraction of the iodine.

6.7.12.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 6.7.12-1. All of the activity released from the damaged fuel to the auxiliary building air space is assumed to be released to the environment over a two-hour period.

<u>Source Term</u>

The FHA analysis is based on the assumption that the accident occurs 48 hours after shutdown. The average assembly fission product inventories for this decay time is provided in Table 6.7.12-2. The 48 hour decay time bounds the current Technical Requirements Manual limit that does not allow fuel movement until 100 hours after shutdown.

The calculation of the radiological consequences following an FHA uses the gap fractions consistent with Regulatory Guide 1.25 (Reference 1) and NUREG/CR-5009 (Reference 2), i.e., 12% for I-131, 30% for Kr-85 and 10% for all other iodines and noble gases.

It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that all damaged fuel rods have been operated at 1.70 times the core average power.

Iodine Chemical Form

The analysis assumes that the iodine is 99.75% elemental and 0.25% organic consistent with the existing licensing basis analysis and Regulatory Guide 1.25 (Reference 1).

Pool Scrubbing Removal of Activity

As the gases released from the damaged fuel rods rise through the spent fuel pool, the water absorbs elemental iodine. Per Reference 1, a decontamination factor (DF) of 133 is used for elemental iodine. Reference 1 states that use of this DF is based on having a minimum of 23 feet of water and on a fuel pressure of ≤1200 psig. Per the Technical Specifications, there is a minimum of 23 feet of water above the spent fuel racks. The fuel rod pressure may exceed 1200 psig but would not exceed 1500 psig. Evaluation of this increase in fuel rod pressure has determined that it is not sufficient to change the level of conservatism associated with the specification of a DF of 133. Use of a DF of 133 thus remains valid. The DF for organic iodine and noble gases is 1.0.

Filtration of Release Paths

The activity released to the auxiliary building atmosphere is assumed to be released to the environment over a 2 hour period. The releases pass through charcoal filters which are assumed to remove 90% of the elemental iodine and 30% of the organic.

6.7.12.2 Acceptance Criteria

From the Standard Review Plan 15.7.4 (Reference 3), the offsite dose limits for the FHA are to be "well within" the dose guidelines of 10 CFR 100, where "well within" is defined as 6 rem whole body and 75 rem thyroid at the Exclusion Area Boundary (for the first 2 hours) and at the Low Population Zone outer boundary (for the duration of the accident).

6.7.12.3 Byron Station Results and Conclusions

The offsite doses due to the fuel handling accident are:

	Thyroid	Whole Body
Exclusion Area Boundary	55 rem	1.4 rem
Low Population Zone	1.7 rem	0.03 rem

The acceptance criteria are met.

6.7.12.4 Braidwood Station Results and Conclusions

The offsite doses due to the fuel handling accident are:

	Thyroid	Whole Body
Exclusion Area Boundary	74 rem	1.8 rem
Low Population Zone	6.8 rem	0.17 rem

The acceptance criteria are met.

6.7.12.5 References

1. U. S. AEC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 23, 1972.

2. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light water Power Reactors," February 1988.

3. NUREG-0800, Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, December 1981.

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Table 6.7.12-1 **Assumptions Used for Fuel Handling Accident Analysis**	
Average assembly fission product activity	See Table 6.7.12-2
Radial peaking factor	1.70
Fuel rod gap fraction	
I-131	12%
Other iodines	10%
Kr-85	30%
Other noble gases	10%
Fuel damaged	one assembly
Time after shutdown	48 hours
Iodine species	
Elemental	99.75%
Organic	0.25%
Water depth	23 feet
Pool scrubbing factor	
Elemental iodine	133
Organic iodine	1
Noble gases	1
Filter efficiency	
Elemental iodine	90%
Organic iodine	30%
Filter bypass	1.0%
Atmospheric Dispersion Factors	See Table 6.7.1-2
Breathing Rates	See Table 6.7.1-2

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Table 6.7.12-2 Average Fuel Assembly Fission Product Inventory	
	Isotopic Inventory, curies
Iodine	
I-131	4.330E5
I-132	4.820E5
I-133	2.112E5
I-134	0.000E0
I-135	6.244E3
Noble Gases	
Kr-85m	7.819E1
Kr-85	5.306E3
Kr-87	0.000E0
Kr-88	2.843E0
Xe-131m	5.601E3
Xe-133m	2.347E4
Xe-133	9.031E5
Xe-135m	1.000E3
Xe-135	5.456E4
Xe-138	0.000E0

7.0 NUCLEAR FUEL

This chapter discusses the analyses performed in support of the uprate project in the nuclear fuel and fuel-related areas. Specifically, it addresses fuel thermal-hydraulic design, fuel core design, fuel rod performance, heat generation rates, neutron fluence, and source terms. The results and conclusions of each analysis can be found within the applicable subsection.

The Reactor Coolant System (RCS) at Byron Units 1 and 2 and Braidwood Units 1 and 2 are similar. The analyses performed accounted for known differences relating to the installed steam generators at Units 1 (BWI replacements) and Units 2 (original D5) and are applicable to all four units.

7.1 Core Thermal-Hydraulic Design

7.1.1 Introduction and Background

This section describes the core thermal-hydraulic analyses and evaluations performed in support of Byron/Braidwood Units 1 and 2 operation at an uprated power level of 3586.6 MWt (core) over a range of RCS temperatures.

7.1.2 Input Parameters and Assumptions

Table 7.1-1 summarizes the thermal-hydraulic design parameters used in the DNBR analyses. The core inlet temperature used in the DNBR analyses is based on the upper bound of the RCS temperature range for the power uprate conditions. Use of the upper bound temperature is conservative for the DNBR analyses. The DNBR analyses also assume that the uprated core designs are composed of VANTAGE 5/VANTAGE+ fuel.

7.1.3 Description of Analyses and Evaluations

7.1.3.1 Calculation Methods

The thermal-hydraulic design criteria and methods for the power uprate remain the same as those presented in the Byron/Braidwood UFSAR (Reference 1).

As discussed in Reference 1, the design method employed to meet the DNB design basis for the VANTAGE 5/VANTAGE+ fuel is the Revised Thermal Design Procedure (RTDP) (Reference 2). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values were determined such that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, or transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2).

Uncertainties in plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and RCS flow) are shown in Table 7.1-2. The current plant operating parameter uncertainties, and the uncertainties used in the uprate RTDP analyses, are also presented in Table 7.1-2 for comparison. Only the random portion of each plant operating parameter uncertainty is included in the statistical combination for RTDP. Any adverse instrumentation bias is treated either as a direct DNBR penalty or a direct analysis input.

The RTDP design limit DNBR values specified in Reference 1 for Byron/Braidwood Units 1 and 2 were revised for the power uprate to 1.24/1.25 (thimble cell/typical cell).

In addition to the above considerations for uncertainties, additional DNBR margin was maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. Sufficient DNBR margin was maintained in the safety analysis DNBR limits to offset the rod bow DNBR penalty. The net remaining DNBR margin, after consideration of this penalty, is available for operating and design flexibility. Table 7.1-3 lists the DNBR limits and the DNBR margin summary for the Byron/Braidwood RTDP analyses that support the power uprate.

As noted in Reference 1, the Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

7.1.3.2 DNB Performance

The current DNBR analyses of record for the Byron/Braidwood Units are primarily those that were performed to support the core peaking factor increase when using VANTAGE 5/VANTAGE+ fuel. All DNBR analyses performed for the VANTAGE 5/VANTAGE+ uprate included an uprated core power level of 3586.6 MWt and are bounding for operation at the current core power level of 3411 MWt. A comparison of the current UFSAR thermal-hydraulic parameters and the power uprate parameters is shown in Table 7.1-1.

To support the operation of Byron/Braidwood Units 1 and 2 at power uprate conditions, DNBR reanalysis was required to define new core limits, axial offset limits, and Condition II accident acceptability. The analyses to support the power uprate and related items incorporated coincident with the power uprate are addressed below. The related items include RCS flow values based on current operating values, design flexibility to remove thimble plugs in the core, and continued applicability of uncertainty and instrumentation bias parameters from the plant.

7.1.3.2.1 Loss of Flow

The DNB analysis of the Loss of Flow accident was performed for power uprate conditions assuming that the reactor protection system functions and assuming there are regulated failures, as required by procedure. Several cases, including Partial Loss of Flow, Frequency Decay, and Complete Loss of Flow, were checked to ensure the limiting scenario was identified. The effect of updated fuel temperatures was included in the analysis of this event (Section 7.1.3.4). The Frequency Decay case resulted in the lowest minimum DNBR. The minimum DNBRs calculated for each of the three cases were greater than the new safety analysis DNBRs listed in Table 7.1-3, thereby demonstrating compliance to the DNB design criterion for this event.

7.1.3.2.2 Locked Rotor

The analysis of the Locked Rotor accident was performed for power uprate conditions assuming that the reactor protection system functions and assuming there are regulated failures, as required by procedure. The Locked Rotor accident is classified as a Condition IV event, but was evaluated consistent with the more limiting Condition II DNB criterion. To

estimate the radiation release possible as a consequence of the accident, DNB calculations were performed to quantify the inventory of rods that would experience DNB and be presumed conservatively to fail. Acceptability of the system to mitigate these consequences was demonstrated by the containment radiation release analysis, assuming the number of failed rods, to be within site dose limits.

For the Byron/Braidwood units, and based on the safety analysis limit DNBR, the analysis indicates that a maximum of 0.1% of the rods would be in DNB due to the Locked Rotor accident. The dose calculations for the Locked Rotor Accident are contained in Sections 6.7.3 and 6.7.4. The 0.1% rods in DNB is well below the 2% limit used in the dose calculations.

7.1.3.2.3 Feedwater Malfunction

The DNB analysis of the Feedwater Malfunction event was performed for power uprate conditions assuming that the reactor protection system functions and assuming there are regulated failures, as required by procedure.

The Feedwater Malfunction analyses considered a variety of scenarios including single- and multi-loop excessive flow cases and multi-loop temperature reduction cases for both steam generator designs. Also, manual and automatic rod control were considered, and it was determined that manual rod control cases were more limiting. Several of these cases were analyzed for compliance to DNB criterion to ensure the limiting scenario was identified. A case for single-loop, hot full power, excessive flow in manual rod control was identified as the limiting case. The minimum DNBR's calculated for each of the eight cases were well above the new safety analysis DNBRs listed in Table 7.1-3, thereby demonstrating compliance to the DNB design criterion for this event.

7.1.3.2.4 Dropped Rod

Dropped Rod Limit Lines were calculated to address the acceptability of the plant's response to this accident scenario. The limit lines were calculated based on the reference power shape (nominally the WCAP-9500 shape as is currently applied). The loci of points which would result in the safety limit DNBR being reached were defined for a wide span of core conditions (inlet temperature, power, and pressure).

There is no explicit DNBR calculation for the Dropped Rod event. Calculation of the effects of the accident on the core is checked cycle by cycle. The effects on core conditions, including power distribution, are demonstrated to remain within the bounds represented by the Dropped Rod Limit lines. By so doing, continued compliance to the DNB criterion is inferred.

7.1.3.2.5 Steamline Break

The DNB analysis of the Steamline Break event was performed for power uprate conditions assuming that the reactor protection system functions and assuming there are regulated failures, as required by procedure. Cases were analyzed both for Hot Zero Power and Hot Full Power preconditions. For each of these bases, an appropriate methodology was applied.

For the Hot Full Power cases, the RTDP methodology was used. For acceptability, calculated DNBRs must be above the design limit DNBR defined by a convolution of uncertainties on core condition parameters. For the Hot Zero Power cases, the RDTP methodology is not appropriate, so the mechanistic Standard Thermal Design Procedure was applied. For the STDP application, the DNBR limit is the approved DNBR limit of 1.45, which has been acknowledged by the USNRC as sufficiently high to assure DNB criterion acceptance, given the documented limitation of the model.

The minimum DNBRs calculated for both these accident cases were well above the safety analysis DNBRs listed in Table 7.1-3 for their respective bases.

7.1.3.2.6 Rod Withdrawal from Subcritical

The DNB analysis of the rod withdrawal from subcritical accident was performed for power uprate conditions assuming that the reactor protection system functions and assuming there are regulated failures, as required by procedure.

By nature of the accident, a bottom-skewed power shape was conservatively applied. A power excursion, due to the removed rod bank, would develop more prominently in the lower part of the core. For this calculation, a conservative generic power shape was applied. To preserve applicability of the Critical Heat Flux Correlation, two calculations are required for this accident. For fuel assembly spans below the first mixing vane grid, the W-3 correlation is applied. For fuel assembly spans above the mixing grid, the WRB-2 correlation is applied, consistent with

other DNBR confirmation calculations. Also, because of the zero power precondition of this event, the methodology that convolutes uncertainty terms to set limits is not appropriate, so the mechanistic Standard Thermal Design Procedure (STDP) was applied. For the STDP application, the DNBR limit applied is the correlation limit DNBR, given that uncertainties are mechanistically applied on the calculation input. For the W-3 correlation, this value is 1.30. For the WRB-2 correlation, this value is 1.17.

Calculations have been completed for each span and the results indicate that the predicted DNBR remains above the respective correlation limit DNBR, thereby demonstrating compliance to the DNB design criterion for this event.

7.1.3.3 Hydraulic Evaluation

The impact of power uprate conditions on the fuel hydraulic analyses was evaluated. The increased coolant density associated with the low end of the RCS temperature range does not have a significant impact on the hydraulic analyses. The fraction of flow that bypasses the core through the thimble guide tubes is unchanged from the value in Reference 3. Fuel assembly lift forces were evaluated for the power uprate parameters and remain bounded by existing analyses for the VANTAGE 5/VANTAGE+ fuel. The use of ZIRLO cladding does not affect the VANTAGE 5/VANTAGE+ hydraulic resistance. The power uprate conditions do not affect this conclusion.

7.1.3.4 Fuel Temperatures

The fuel temperatures for the power uprate safety analysis for VANTAGE 5/VANTAGE+ fuel were based on ZIRLO cladding and low Rod Internal Pressure (RIP) design. The ZIRLO fuel temperatures are higher than the previous Zircaloy-4 fuel temperatures used in the Reference 4 analyses, but the bases taken there are bounding for Zircaloy-4 fuel under uprated conditions. The ZIRLO fuel temperatures were calculated with approved fuel performance models (Reference 5) for conditions which bound the power uprate parameters. The ZIRLO analysis also addressed use of IFBA (integral fuel burnable absorber). The IFBA product used in the analysis was 1.0 X IFBA with 100 psig backfill. Fuel temperatures for 1.0 X IFBA with 100 psig backfill pressure bound 1.5 X and 2.0 X IFBA with either 100 psig or 200 psig backfill. The ZIRLO IFBA and non-IFBA fuel temperatures were used as initial conditions for LOCA and

non-LOCA transients. Also, based on the ZIRLO fuel temperature analysis, the linear power limit to preclude fuel centerline melting of 22.4 kW/ft continues to be supported. The calculated fuel temperature took credit for design changes made to the rods for the purpose of gaining margin in Rod Internal Pressure (RIP).

7.1.4 Acceptance Criteria

The acceptance criteria are contained in each section under 7.1.3.2 (DNB Performance).

7.1.5 Results/Conclusions

Core thermal-hydraulic analyses and evaluations were performed in support of Byron and Braidwood Units 1 and 2 operation at an uprated power level of 3586.6 MWt (core) over a range of RCS temperatures. The results showed that the core thermal-hydraulic design criteria listed in Reference 1 are satisfied.

7.1.6 References

1. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

2. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

3. 99CB-G-0149, "Thimble Plug Elimination for Byron Units 1 and 2 and Braidwood Units 1 and 2," Akers, 17 September 1999.

4. 96CB-G-0126, "Safety Evaluation Report Byron/Braidwood Units 1 and 2 Increased Peaking Factors," Weber, 1 August 1996.

5. WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," R. A. Weiner, et al., August 1988.

6. WCAP-9500-A, "Reference Core Report - 17 x 17 Optimized Fuel Assembly," S. L. Davidson and J. A. Iorii, May 1982.

7. Letter from E. P. Rahe (Westinghouse) to Miller (NRC), dated March 19, 1982, NS-EPR-2573, WCAP-9500 and WCAPs-9401/9402, "NRC SER Mixed Core Compatibility Items."

8. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Supplemental Acceptance No. 2 for Referencing Topical Report WCAP-9500," January 1983.

9. Schueren, P. and McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837-P-A, January 1990.

10. Letter from S. R. Tritch (Westinghouse) to R. C. Jones (NRC), "VANTAGE 5 DNB Transition Core Effects," ET-NRC-91-3618, September 1991.

11. WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," S. L. Davidson and W. R. Kramer (Ed.), September 1985.

Table 7.1-1 (Sheet 1 of 2)

Thermal-Hydraulic Design Parameters For B/B Units 1 And 2

Thermal and Hydraulic Design Parameters	Current UFSAR Parameters (Reference 1)	Power Uprate Parameters
Reactor Core Heat Output, MWt	3411	3586.6
Reactor Core Heat Output, 10^6 Btu/hr	11,639	12,238.2
Heat Generated in Fuel, %	97.4	97.4
Pressurizer Pressure, Nominal, psia	2250	2250
F_{H}, Nuclear Enthalpy Rise Hot Channel Factor	1.70	1.70
Part Power Multiplier for F_{H}	[1+0.3(1-P)]	[1+0.3(1-P)]
Minimum DNBR at Nominal Conditions (Using RTDP)		
Typical Flow Channel	2.39[a]	2.25
Thimble (Cold Wall) Flow Channel	2.26[a]	2.16
Design Limit DNBR		
Typical Flow Channel	1.25	1.25
Thimble (Cold Wall) Flow Channel	1.25	1.24
DNB Correlation[b]	WRB-2	WRB-2
Vessel Minimum Measured Flow Rate, MMF, (including Bypass)		
10^6 lbm/hr	136.2	141.8
gpm	366,000	380,900
Vessel Thermal Design Flow Rate, TDF, (including Bypass)		
10^6 lbm/hr	133.6	137.2
gpm	358,800	368,000

(a) The minimum nominal DNBRs from the VANTAGE 5 RTSR (Reference 1) are conservatively listed in the current UFSAR.

(b) See Chapter 4.4 of Reference 1 for the use of the W-3 DNB correlation.

Table 7.1-1 (Sheet 2 of 2) Thermal-Hydraulic Design Parameters for B/B Units 1 and 2		
HFP Nominal Coolant Conditions	**Current UFSAR Parameters (Reference 1)**	**Power Uprate Parameters**
Core Flow Rate (excluding Bypass, based on TDF)		
10^6 lbm/hr	125.2	126.6
gpm	336,196	337,456
Fuel Assembly Flow Area for Heat Transfer, ft^2	54.14	54.14
Core Inlet Mass Velocity (Based on TDF), 10^6 lbm/hr	2.31	2.53
Nominal Vessel/Core Inlet Temperature, °F	556.9	556.7
Vessel Average Temperature, °F	588.4	588.0
Core Average Temperature, °F	592.2	591.7
Vessel Outlet Temperature, °F[(a)]	619.9	619.3
Average Temperature Rise in Vessel, °F	63.0	62.6
Average Temperature Rise in Core, °F	66.8	66.3
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	57,505	57,505
Average Heat Flux, Btu/hr-ft^2	197,180	207,327
Average Linear Power, kw/ft	5.45	5.73
Peak Linear Power for Normal Operation, kw/ft	14.2	14.9
Temperature Limit for Prevention of Centerline Melt, EF	4,700	4,700

(a) Not an explicit UFSAR value; value obtained from sum of the vessel inlet temperature and the average temperature rise in the vessel.

Table 7.1-2 B/B Plant Uncertainties Used in DNBR Analyses		
Parameters	**Current UFSAR Value (Reference 1)**	**Power Uprate Value**
Pressurizer Pressure (control)	± 43.0 psi	± 43.0 psi
Temperature (rod control)[(a)]	± 7.6°F	± 7.6°F
Power Measurement	±2.0% RTP	±2.0% RTP
RCS Flow Measurement	± 3.5% flow	± 3.5% flow

(a) A 1.5°F reduction on temperature measurement has been taken on calculated core limits, according to the bias reported by the plant(s).

Table 7.1-3 DNBR Margin Summary for B/B Units 1 and 2 Uprated Power RTDP Analyses	
Design Limit DNBR	
Typical Cell	1.25
Thimble Cell	1.24
Safety Analysis Limit DNBR	
Typical Cell	1.33
Thimble Cell	1.33
DNBR Margin (Between Design and Safety Analysis Limit DNBR)	
Typical Cell	6.0%
Thimble Cell	6.7%
DNBR Penalties	
Rod Bow[a]	1.3%
Transition Core	0.0%
Net DNBR Margin (minimum)	4.7%
Safety Analysis Limit DNBR for SLB Accident (typical and thimble cell)	1.45

(a) 0% rod bow penalty for 10 inch spans, 1.3% penalty for 20 inch spans.

7.2 Fuel Core Design

7.2.1 Introduction and Background

The nuclear design portion of the Byron/Braidwood Uprate Program core analysis has two objectives. First, the effect of the uprate on the key safety parameters must be determined. These safety parameters are used as input to the FSAR Chapter 15 accident analyses. Second, the plant Technical Specifications/COLR parameters that apply to nuclear design must be reviewed to determine if they remain appropriate or must be altered.

7.2.2 Input Parameters and Assumptions

The nuclear design analyses demonstrates the acceptability of operation at a core power level of 3586.6 MWt consistent with parameters in Section 2.0.

7.2.3 Description of Analyses and Evaluations

To satisfy these objectives, conceptual models were developed that followed the uprate transition to an equilibrium cycle. Different fuel management strategies were assumed in developing the models. The uprate assumed a power level increase of 5% during the three transition cycles and in the equilibrium cycle. Key safety parameters were then evaluated such that the expected ranges of variation in the parameters were determined. The key safety parameters referred to here are those described in the standard reload design methodology (Reference 1). Some of these parameters, such as shutdown margin, are sensitive to the fuel management and loading pattern characteristics.

The observed variation in these loading pattern (LP) dependent parameters during the transition to an equilibrium cycle with uprate conditions are typical of the normal cycle-to-cycle variations for non-transition fuel reloads. Many of the key safety parameters fall into this LP-dependent category.

The PMTC fuel management strategy reduces burnable absorber requirements, improves fuel economy, and increases nuclear design flexibility. The reduction in burnable absorbers is limited, however, by the reactivity holddown required for long fuel cycles to satisfy post-LOCA

subcriticality concern and the Unfavorable Exposure Time associated with a positive MTC at HZP design.

7.2.3.1 Methodology

The methods and core models used in the Byron/Braidwood Uprate analyses are described in References 1 and 2. These licensed methods and models have been used for Byron/ Braidwood and other previous Westinghouse reload designs fuel with and without uprating. No changes to the nuclear design philosophy, methods, or models, are necessary due to the uprating.

The reload design philosophy used by Westinghouse includes evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the FSAR safety evaluation for each reload cycle. This philosophy is described in References 1 and 2. These key safety parameters will be evaluated for each Byron/Braidwood reload cycle. If one or more of the key parameters fall outside the bounds assumed in the safety analyses, the affected transients will be reevaluated and the results documented in the Reload Safety Evaluation report (RSE) for that cycle. The main objective of the uprating core analyses is to determine, prior to the cycle specific reload design, if the previously used bounds for the key safety parameters remain applicable. The results of these analyses are described below.

7.2.3.2 Physics Characteristics and Key Safety Parameters

Conceptual core loading patterns were constructed to be representative of future Byron/ Braidwood cores. Table 7.2-1 compares the safety parameter ranges considered for the Byron/Braidwood current designs and for the uprate.

The comparison in Table 7.2-1 shows that the uprated core does not have any marked deviations from the core design at 3411 MWt. Of note is an increase in the positive density coefficient to a value of 0.54 delta-k/gm/cc (Steamline Break and Feedwater malfunction at .33 delta-k/gm/cc) due to the increased enrichments in the feed fuel necessary to produce the additional power. This parameter is confirmed each cycle to be less than the Safety Analysis value.

Shutdown margin and maximum boron concentrations are two parameters that are loading pattern dependent and the core design must be developed such that these constraints are met. The shutdown margin requirement of 1300 pcm is primarily a function of the power defect from full power to HZP at the time of trip and the fuel type which is placed under control rod locations. The power defect is set by the enrichments required to achieve the design cycle length and operating temperature. The core design can govern the amount of shutdown margin by increasing the amount of fresh fuel in control rod locations. As the uprate conditions significantly increase the power defect, the required amount of shutdown margin is a loading pattern constraint that must be demonstrated to consider the loading pattern acceptable. Maximum boron concentration is also a function of the feed enrichment needed to achieve the cycle lifetime but also of the fuel management strategy used for the loading pattern. As the maximum boron concentrations are initial or final conditions, they are also a design constraint that must be considered at the time of loading pattern development.

7.2.3.3 Power Distributions and Peaking Factors

Loading patterns were developed and modeled based on the projected energy requirements for the power uprating. These models are not intended to represent limiting loading patterns but were developed with the intent to show that enough margin exits between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores.

7.2.3.4 Radial Power Distribution Impacts

Assembly average power at BOL, MOL and EOL were calculated using the uprated core models for different fuel management techniques. The impact on the radial power distribution due to the uprated conditions is small when compared to loading patterns of similar fuel management style at nominal power conditions. The impacts of the above radial power distribution differences on rod worths, and on off-nominal condition peaking factors are small and are well within normal cycle-to-cycle variation in these parameters.

7.2.3.5 Axial Power Distribution and FQ(z) Impacts

The axial power distribution impact of the at power uprate conditions show only a small axial sensitivity to the uprate.

As part of the reload design process, a cycle specific FAC analysis based on CAOC or RAOC operation check is performed which implicitly includes the axial impacts of the uprating. Load follow simulations are performed through the power range to generate axial power shapes that are typical of Condition I operation. The results of the FAC analysis for this report shows that the total peaking factor (Fq) is acceptable. Therefore, it is expected that all reload cores at uprated conditions will also be acceptable.

7.2.4 Conclusions

In summary, implementation of the power uprate will not cause changes to the current nuclear design bases given in the FSAR. The impact of the reduced temperatures on peaking factors, rod worths, reactivity coefficients, shutdown margin and kinetics parameters is well within normal cycle-to-cycle variation of these values or is controlled by the core design, and will be addressed on a cycle specific basis consistent with Reload Safety Evaluation Methodology. The ranges of key safety parameters as reported in Table 7.2-1 remain valid and bounding for the uprate condition.

Modifications to the existing Technical Specifications are required as a result of the nuclear design related aspects of the power uprating.

7.2.5 References

1. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

2. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

Table 7.2-1
Byron/Braidwood Uprating Program Key Safety Parameters

Safety Parameters	Byron/Braidwood Current Values	Byron/Braidwood Power Uprate
Reactor Core Power (MWt)	3411	3586.6
Vessel T_{avg} HFP (°F)	569.1 to 588.4	575.0 to 588.0
RCS Pressure (psia)	2250	2250
Core Average Linear Heat Rate (Kw/ft)	5.45	5.73
Most Positive MTC (pcm/°F)	+7	+7
Most Positive MDC (delta-k/g/cc)	0.43	0.54
Doppler Temperature Coefficient (pcm/°F)	-0.91 to -2.9	-0.91 to -2.9
Doppler Only Power Coefficients (pcm/% Power) Least Negative	-9.55 to -6.05	-9.55 to -6.05
Doppler Only Power Coefficients (pcm/% Power) Most Negative	-19.4 to -12.6	-19.4 to -12.6
Beta-Effective	.0044 - .0075	.0044 - .0075
Shutdown Margin (pcm)	1300	1300
Nuclear Design FD_H	1.574/1.70	1.574/1.70

7.3 Fuel Rod Design and Performance

7.3.1 Introduction

Fuel rod design analyses were performed to assess the potential impacts that the uprated operating conditions of the Byron and Braidwood Units 1 and 2 would have on meeting fuel rod design criteria.

7.3.2 Description of Analyses, Acceptance Criteria, and Results

The fuel rod design analyses modeled ZIRLO™ clad fuel rods irradiated for up to four cycles at power uprated conditions. Representative rod power histories and axial power shapes, generated by the NRC-approved advanced nodal code (ANC) were analyzed. The NRC-approved Westinghouse PAD 3.4 fuel performance models (References 1 and 2) were used in the analyses. PAD is the principle design tool for evaluating fuel rod performance. PAD calculates the inter-related effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

The following sections summarize the impact of the core power uprating on the fuel rod design criteria most impacted by the uprated core power. The fuel rod design criteria considered include rod internal pressure, clad corrosion, clad stress, and clad strain criteria. Other fuel rod design criteria are not significantly impacted by a core power uprating.

7.3.2.1 Rod Internal Pressure

<u>Design Basis</u>

The fuel system will not be damaged due to excessive fuel rod internal pressure.

<u>Acceptance Limit</u>

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady state operation or cause extensive departure from nucleate boiling (DNB) propagation to occur.

Design Evaluation

The analyses performed showed that the rod internal pressure criteria is most impacted by the uprated core power level. The higher power levels result in higher fuel operating temperatures with a potential for increased fission gas release. Analysis of the representative rod power histories indicated that the higher duty fuel rods have this potential of increased fission gas release resulting in higher rod internal pressures. The rod internal pressure criterion can be met under uprated core conditions by adjusting cycle-specific core design.

7.3.2.2 Clad Corrosion

Design Basis

The fuel system will not be damaged due to excessive fuel clad oxidation. The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, due to hydrogen embrittlement caused by formation of zirconium hydride platelets.

Acceptance Limit

The calculated fuel clad temperature (metal-oxide interface temperature) will be less than 780°F for ZIRLO clad fuel during steady state operation. For Condition II events, the calculated fuel clad temperature will not exceed 850°F for ZIRLO clad fuel. The hydrogen pickup level in the fuel clad will be less than or equal to 600 ppm at the end of fuel operation.

Design Evaluation

The power uprating conditions result in increased operating temperatures for the fuel clad due to the increased fuel rod average power rating. Since the corrosion process is a strong function of fuel clad temperature, the power uprating will impact these criteria. Analysis of the representative rod power histories indicated that the corrosion design criteria will be satisfied for the higher duty fuel rods at the uprated core conditions.

7.3.2.3 Clad Stress and Strain

Design Basis

The fuel system will not be damaged due to excessive fuel clad stress and strain.

Acceptance Limit

The volume average effective stress calculated with the Von Mises equation, considering interference due to uniform cylindrical fuel pellet-clad contact, caused by fuel pellet thermal expansion, fuel pellet swelling, uniform fuel clad creep, and pressure differences, is less than the 0.2% offset yield stress with due consideration to temperature and irradiation effects under Condition II events. The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain increase, due to uniform cylindrical fuel pellet thermal expansion during a transient, is less than 1% from the pre-transient value.

Design Evaluation

The Westinghouse PAD 3.4 fuel performance models (References 1 and 2) are used to evaluate fuel clad stress and strain limits. The local power duty during Condition II events is a key factor in evaluating the margin to fuel clad stress and strain limits. The fuel duty at the power uprated conditions is more limiting, resulting in an increase in the cladding stress and strain levels. However, the fuel analyses results show that the core power uprating will not impact the fuel's capability to meet the clad stress and strain limits.

The cycle-specific fuel performance analyses will consider any improved fuel performance models and methods licensed and approved by the NRC available at the time of the specific cycle design. These cycle-specific evaluations support the Reload Safety Evaluation (RSE) performed for each cycle of operation.

7.3.3 Conclusions

The fuel rod design criteria most impacted by a change in core power rating have been evaluated. The evaluations indicate that all fuel rod design criteria can be met at the uprated core conditions by adjusting cycle-specific core design.

Cycle-specific core designs and fuel performance analyses are performed for each reload cycle. These cycle-specific analyses are performed to ensure that all fuel rod design criteria will be satisfied for the specific operating conditions of that cycle.

Although the uprated analyses described in this section were performed for ZIRLO clad fuel, the cycle-specific fuel performance analyses consider each specific fuel region (whether ZIRLO clad fuel design or older fuel designs with different fuel features) in the core during that cycle. These analyses ensure that all fuel rod design criteria are met for each fuel region.

7.3.4 References

1. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

2. Davidson, S. L., Nuher, D. L., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A, April 1995.

7.4 Reactor Internals Heat Generation Rates

7.4.1 Introduction and Background

The presence of radiation-induced heat generation in reactor internals components, in conjunction with the various reactor coolant fluid temperatures, results in thermal gradients within and between the components. These thermal gradients cause thermal stress and thermal growth which must be considered in the design and analysis of the various components. The primary design considerations are (1) to ensure that thermal growth is consistent with the functional requirements of the components, and (2) to ensure that the applicable ASME Code requirements are satisfied. In order to satisfy these requirements, the reactor internals components must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components subjected to significant radiation-induced heat generation are the upper and lower core plates, lower core support, core baffle plates, former plates, core barrel, neutron pad, baffle-former bolts, and barrel-former bolts. However, due to relatively low heat generation rates (generally less than 50 Btu/hr-lbm), the upper core plate, lower core support, and neutron pad experience little, if any, temperature rise relative to the surrounding reactor coolant.

This section provides a description of the methodology used to determine the radiation-induced heat generation rates for the lower core plate and the remaining radial reactor internals components that are significantly impacted.

7.4.2 Lower Core Plate Heat Generation Rates

7.4.2.1 Input Parameters and Assumptions

Radiation-induced heat generation rates were determined for both long- and short-term conditions. Long-term heat generation rates are intended to represent time-average behavior that can be used for fatigue analysis, whereas the short-term results are intended to provide conservative values for use in the calculation of maximum temperatures and thermal stresses. For the long-term heat generation rate evaluation, radial and axial core power distributions applicable to the Byron and Braidwood uprated power level were employed. For the short-term

heat generation rate evaluation, the Byron and Braidwood uprated radial power distribution was coupled with a conservative design basis bottom-peaked axial power distribution taken from Reference 1. Both the long- and short-term heating rate evaluations were completed at the uprated power level.

7.4.2.2 Description of Analysis

The lower core plate heat generation analyses were carried out with the DORT (version DOORS3.1) two-dimensional discrete ordinates transport code. The calculations were performed in r,z geometry using the equivalent volume cylindrical core concept. The varying amounts of structure located axially below the core were approximated as a number of homogenized geometric regions, each with the appropriate volume fraction of stainless steel, water, and other structural materials. In the axial direction, this r,z model included the lower half of the reactor core and extended axially downward to one foot below the lower core plate. Radially, the model extended from the center of the reactor core to the inner radius of the reactor pressure vessel.

7.4.2.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to the reactor internals evaluation.

7.4.2.4 Results

The results of the radiation-induced heat generation rate calculations were provided as inputs for the reactor internals evaluations described in Section 5.2. The volume-averaged heat generation rates for the lower core plate are summarized as follows:

	Heating Rate (Btu/hr-lbm)	
Location	**Current Analysis**	**Reference 1**
Long-Term		
Lower Core Plate A	343	249
Lower Core Plate B	45	52
Short-Term		
Lower Core Plate A	1406	822
Lower Core Plate B	177	201

In the above tabulation, region A refers to the cylindrical portion of the lower core plate located below the active fuel, and region B refers to the annular portion of the plate located radially outboard of the active fuel.

7.4.3 Core Barrel, Baffle Plates, and Neutron Pad Heat Generation Rates

7.4.3.1 Input Parameters and Assumptions

Design basis heat generation rates applicable to the Byron and Braidwood radial internals are contained in Appendix E of Reference 1. The core power distributions upon which those calculations were based were derived from statistical studies of 23 independent fuel cycles from 10 four-loop reactors. These power distributions represented an upper tolerance limit for beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95% probability with a 95% confidence level. Most of the evaluated fuel cycles were based on an out-in fuel loading strategy (fresh fuel on the periphery) which, when combined with the statistical processing of the data, resulted in a design basis core power distribution that tended to be biased high on the periphery. This high bias on the periphery was desired by the reactor internals analysts to ensure conservative, but not unrealistic, design calculations for the critical baffle-barrel region of the reactor internals.

The evaluation of radiation-induced heat generation rates for the Byron and Braidwood radial internals was based on a comparison of the results calculated for the fuel cycle design provided for the uprated conditions with results based on the use of the design basis power distribution.

7.4.3.2 Description of Analysis

An assessment was made of the effect of the design core power distribution for the uprating program on the heat generation rates in the core baffle plates and core barrel. The approach taken was to use scaling factors which consider the fact that heat generation rates in the radial internals regions are dominated by radiation leakage from the periphery of the core; and, that, to a close approximation, the heat generation rate in a given region is proportional to the power produced in the adjacent fuel region. The scaling functions defining the impact of individual fuel assemblies on the heat generation rates in the regions of interest were determined from a series of two-dimensional discrete ordinates calculations for the four-loop reactor geometry of

the Byron and Braidwood design, combined with the core power distribution defined for the uprating, to produce heat generation rate distributions for the individual fuel assemblies. The individual assembly results were then combined by superposition to create the resultant composite distribution.

7.4.3.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to the reactor internals evaluations.

7.4.3.4 Results

The heat generation rates for the radial components were provided as input for the reactor internals analysis described in Section 5.2. The volume-averaged heat generation rates for the radial internals components are summarized as follows:

	Region Average Long Term Heating Rate [Btu/hr-lbm]	
Location	**Current Analysis**	**Reference 1**
Baffle Plate 1	426	945
Baffle Plate 2	511	1070
Baffle Plate 3	464	996
Baffle Plate 4	313	802
Core Barrel	74.5	186

7.4.4 References

1. A. H. Fero, "Reactor Internals Heat Generation Rates and Neutron Fluences," WCAP-9620, Revision 1, December 1983.

7.5 Neutron Fluence

7.5.1 Introduction

In the assessment of the state of embrittlement of Light Water Reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of the materials comprising the beltline region of the vessel is required. This exposure evaluation must, in general, include assessments not only at locations of maximum exposure at the inner diameter of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

In order to satisfy the requirements of 10CFR50, Appendix G, for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system, fast neutron exposure levels must be defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials comprising the beltline region. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10CFR50, Appendix G. In the determination of values of Reference Temperature – Pressurized Thermal Shock (RT_{PTS}) for comparison with the applicable pressurized thermal shock screening criterion as defined in 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," maximum neutron exposure levels experienced by each of the beltline materials are required. These maximum levels occur at the vessel inner radius.

The methodology used, to determine the fast neutron (Energy > 1.0 MeV) exposure of the Byron Units 1 and 2 and Braidwood Units 1 and 2 pressure vessels, was developed from the guidance provided in ASTM Standard E853, "Analysis and Interpretation of Light Water Reactor Surveillance Results," and Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The analytical methodology has received NRC approval as documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves," January 1996.

7.5.2 Description of Analysis/Evaluations and Input Assumptions

The fast neutron exposure calculations for the Byron and Braidwood reactor geometries were completed using a combination of both forward and adjoint-discrete ordinates transport techniques. In this approach, a single-reference forward calculation based on plant specific geometry and a representative core source distribution was employed to define the relative energy distribution of neutrons and gamma rays for use in relating the results of location-specific adjoint evaluations to other key positions within the reactor vessel wall. In conjunction with this reference forward computation, a series of adjoint calculations were used to establish the means to compute absolute exposure rates using fuel cycle-specific core power distributions.

All of the transport calculations were completed using an S_8 order of angular quadrature and the P_3 legendre expansion of the cross-sections from the BUGLE-96 ENDF/B-VI based cross-section library. The importance functions generated from the individual adjoint calculations established the basis for absolute exposure projections and comparison with measurement. When combined with fuel cycle-specific neutron source distributions, these importance functions yielded absolute calculations of fast neutron exposure at the locations of interest for each fuel cycle. They also established the means for a direct comparison with measurements from in-vessel surveillance capsules withdrawn as an integral part of the reactor vessel surveillance program established for each of the respective reactors. This cycle-dependent data from the adjoint analyses was then integrated to provide fluence projections over the operating lifetime of the respective pressure vessels. All calculations were completed for an uprated power level of 3586.6 MWt.

7.5.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to reactor vessel analyses (see 5.1.2 and 5.1.3).

7.5.4 Results

The results of the fast neutron exposure evaluations for Byron Unit 1 and Unit 2, and Braidwood Unit 1 and Unit 2 are provided in Tables 7.5.3-1 through 7.5.3-8, respectively. These

projections account for the power uprating and are based on the assumption that the power uprate was initiated following Cycle 8 for Byron Unit 1 and Cycle 7 for Byron Unit 2, Braidwood Unit 1, and Braidwood Unit 2. These cycles correspond to the last in-vessel surveillance capsule withdrawal for each unit. The results of analyses performed in support of these capsule withdrawals, as well as evaluations of neutron dosimetry from all capsules withdrawn to date, are provided in References 1 through 4.

In Tables 7.5.3-1 through 7.5.3-8, the calculated fast neutron fluence (Energy > 1.0 MeV) is given for all materials comprising the beltline region of the pressure vessel, based on uprated conditions. For comparison purposes, the Best Estimate fluence projections, derived from least squares evaluations of surveillance capsule dosimetry and the results of the transport calculations, are also listed in Tables 7.5.3-1 through 7.5.3-8. These tables show that the calculated neutron exposure exceeds the Best Estimate values from 7% to 14%.

7.5.5 References

1. T. J. Laubham, et al., "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15123, Rev. 1, January 1999.

2. T. J. Laubham, et al., "Analysis of Capsule X from Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-15176, Rev. 0, March 1999.

3. E. Terek, et al., "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15316, Rev. 1, December 1999.

4. T. J. Laubham, et al., "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-15369, Rev. 0, March 2000.

Table 7.5.3-1

Azimuthal Variations of the Neutron Exposure Projections on the Reactor Vessel Clad/Base Metal Interface at Core Midplane

Byron 1

Best Estimate

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
9.24 EFPY	2.92E+18	4.37E+18	4.98E+18	5.01E+18
12 EFPY	3.77E+18	5.62E+18	6.46E+18	6.51E+18
16 EFPY	4.99E+18	7.44E+18	8.60E+18	8.69E+18
32 EFPY	9.90E+18	1.47E+19	1.72E+19	1.74E+19
54 EFPY	1.66E+19	2.47E+19	2.90E+19	2.94E+19

Byron 1

Calculated

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
9.24 EFPY	3.38E+18	5.06E+18	5.76E+18	5.79E+18
12 EFPY	4.36E+18	6.51E+18	7.47E+18	7.54E+18
16 EFPY	5.78E+18	8.61E+18	9.96E+18	1.01E+19
32 EFPY	1.15E+19	1.70E+19	1.99E+19	2.02E+19
54 EFPY	1.93E+19	2.85E+19	3.35E+19	3.41E+19

Note:

a. Maximum neutron exposure projection reported for 30° vessel location representing the octant containing the 12.5° neutron pad span.

Table 7.5.3-2

Neutron Fluence Projections on the Reactor Vessel Clad/Base Metal Interface for Selected Circumferential Weld Locations Along the 45° Azimuth

Byron 1

Best Estimate

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]				
Weld Location	**9.24 EFPY**	**12 EFPY**	**16 EFPY**	**32 EFPY**	**54 EFPY**
WR-20	7.23E+15	9.39E+15	1.25E+16	2.50E+16	4.21E+16
WR-19	1.02E+16	1.32E+16	1.76E+16	3.52E+16	5.94E+16
WR-34	1.51E+18	1.96E+18	2.61E+18	5.22E+18	8.80E+18
WR-18	4.86E+18	6.30E+18	8.40E+18	1.68E+19	2.83E+19
WR-29	6.91E+14	8.97E+14	1.19E+15	2.39E+15	4.03E+15

Byron 1

Calculated

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]				
Weld Location	**9.24 EFPY**	**12 EFPY**	**16 EFPY**	**32 EFPY**	**54 EFPY**
WR-20	8.37E+15	1.09E+16	1.45E+16	2.89E+16	4.88E+16
WR-19	1.18E+16	1.53E+16	2.04E+16	4.07E+16	6.87E+16
WR-34	1.75E+18	2.27E+18	3.02E+18	6.04E+18	1.02E+19
WR-18	5.62E+18	7.30E+18	9.72E+18	1.94E+19	3.28E+19
WR-29	8.00E+14	1.04E+15	1.38E+15	2.76E+15	4.66E+15

Notes:

WR-20: Outlet Nozzle Shell Forging Circumferential Weld

WR-19: Inlet Nozzle Shell Forging Circumferential Weld

WR-34: Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld

WR-18: Intermediate Shell Forging to Lower Shell Forging Circumferential Weld

WR-29: Lower Shell Forging Circumferential Weld

Table 7.5.3-3

Azimuthal Variations of the Neutron Exposure Projections on the Reactor Vessel Clad/Base Metal Interface at Core Midplane

Byron 2

Best Estimate

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
8.57 EFPY	2.89E+18	4.29E+18	4.90E+18	4.94E+18
12 EFPY	4.03E+18	6.01E+18	6.92E+18	6.98E+18
16 EFPY	5.37E+18	8.01E+18	9.27E+18	9.35E+18
32 EFPY	1.07E+19	1.60E+19	1.87E+19	1.88E+19
48 EFPY	1.60E+19	2.41E+19	2.81E+19	2.83E+19
54 EFPY	1.80E+19	2.71E+19	3.16E+19	3.19E+19

Byron 2

Calculated

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
8.57 EFPY	3.17E+18	4.70E+18	5.37E+18	5.42E+18
12 EFPY	4.42E+18	6.58E+18	7.58E+18	7.64E+18
16 EFPY	5.88E+18	8.78E+18	1.02E+19	1.02E+19
32 EFPY	1.17E+19	1.76E+19	2.05E+19	2.06E+19
48 EFPY	1.76E+19	2.64E+19	3.08E+19	3.10E+19
54 EFPY	1.98E+19	2.97E+19	3.46E+19	3.49E+19

Note:

a. Maximum neutron exposure projection reported for 30° vessel location representing the octant containing the 12.5° neutron pad span.

Table 7.5.3-4

Neutron Fluence Projections on the Reactor Vessel Clad/Base Metal Interface for Selected Circumferential Weld Locations Along the 45° Azimuth

Byron 2

Best Estimate

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 8.57 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	5.09E+15	7.16E+15	9.59E+15	1.93E+16	2.90E+16	3.26E+16
WR-19	7.17E+15	1.01E+16	1.35E+16	2.71E+16	4.08E+16	4.59E+16
WR-34	1.26E+18	1.77E+18	2.37E+18	4.76E+18	7.16E+18	8.06E+18
WR-18	4.89E+18	6.88E+18	9.21E+18	1.85E+19	2.78E+19	3.13E+19
WR-29	4.98E+14	7.01E+14	9.39E+14	1.89E+15	2.84E+15	3.19E+15

Byron 2

Calculated

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 8.57 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	5.58E+15	7.85E+15	1.05E+16	2.11E+16	3.17E+16	3.57E+16
WR-19	7.85E+15	1.11E+16	1.48E+16	2.97E+16	4.47E+16	5.03E+16
WR-34	1.38E+18	1.94E+18	2.60E+18	5.22E+18	7.84E+18	8.83E+18
WR-18	5.36E+18	7.54E+18	1.01E+19	2.03E+19	3.05E+19	3.43E+19
WR-29	5.46E+14	7.69E+14	1.03E+15	2.07E+15	3.11E+15	3.50E+15

Notes:
WR-20: Outlet Nozzle Shell Forging Circumferential Weld
WR-19: Inlet Nozzle Shell Forging Circumferential Weld
WR-34: Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld
WR-18: Intermediate Shell Forging to Lower Shell Forging Circumferential Weld
WR-29: Lower Shell Forging Circumferential Weld

Table 7.5.3-5
Azimuthal Variations of the Neutron Exposure Projections on the Reactor Vessel Clad/Base Metal Interface at Core Midplane

Braidwood 1
Best Estimate

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
7.61 EFPY	2.55E+18	3.81E+18	4.48E+18	4.62E+18
12 EFPY	4.02E+18	5.95E+18	7.04E+18	7.20E+18
16 EFPY	5.35E+18	7.90E+18	9.38E+18	9.56E+18
32 EFPY	1.07E+19	1.57E+19	1.87E+19	1.90E+19
48 EFPY	1.60E+19	2.35E+19	2.81E+19	2.84E+19
54 EFPY	1.80E+19	2.65E+19	3.16E+19	3.19E+19

Braidwood 1
Calculated

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
7.61 EFPY	2.75E+18	4.11E+18	4.83E+18	4.98E+18
12 EFPY	4.33E+18	6.42E+18	7.60E+18	7.77E+18
16 EFPY	5.77E+18	8.53E+18	1.01E+19	1.03E+19
32 EFPY	1.15E+19	1.70E+19	2.02E+19	2.05E+19
48 EFPY	1.73E+19	2.54E+19	3.03E+19	3.06E+19
54 EFPY	1.95E+19	2.85E+19	3.41E+19	3.44E+19

Note:

a. Maximum neutron exposure projection reported for 30° vessel location representing the octant containing the 12.5° neutron pad span.

Table 7.5.3-6

Neutron Fluence Projections on the Reactor Vessel Clad/Base Metal Interface for Selected Circumferential Weld Locations Along the 45° Azimuth

Braidwood 1

Best Estimate

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 7.61 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	6.42E+15	9.98E+15	1.32E+16	2.62E+16	3.92E+16	4.41E+16
WR-19	9.04E+15	1.41E+16	1.86E+16	3.69E+16	5.52E+16	6.21E+16
WR-34	1.38E+18	2.15E+18	2.84E+18	5.63E+18	8.42E+18	9.47E+18
WR-18	4.52E+18	7.03E+18	9.31E+18	1.85E+19	2.76E+19	3.10E+19
WR-29	6.22E+14	9.67E+14	1.28E+15	2.54E+15	3.80E+15	4.27E+15

Braidwood 1

Calculated

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 7.61 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	6.92E+15	1.08E+16	1.43E+16	2.83E+16	4.23E+16	4.75E+16
WR-19	9.75E+15	1.52E+16	2.01E+16	3.98E+16	5.96E+16	6.70E+16
WR-34	1.49E+18	2.31E+18	3.07E+18	6.08E+18	9.09E+18	1.02E+19
WR-18	4.88E+18	7.58E+18	1.00E+19	1.99E+19	2.98E+19	3.35E+19
WR-29	6.71E+14	1.04E+15	1.38E+15	2.74E+15	4.09E+15	4.60E+15

Notes:

WR-20: Outlet Nozzle Shell Forging Circumferential Weld
WR-19: Inlet Nozzle Shell Forging Circumferential Weld
WR-34: Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld
WR-18: Intermediate Shell Forging to Lower Shell Forging Circumferential Weld
WR-29: Lower Shell Forging Circumferential Weld

Table 7.5.3-7
Azimuthal Variations of the Neutron Exposure Projections on the Reactor Vessel Clad/Base Metal Interface at Core Midplane

Braidwood 2

Best Estimate

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
8.53 EFPY	2.72E+18	4.08E+18	4.80E+18	4.86E+18
12 EFPY	3.79E+18	5.72E+18	6.71E+18	6.69E+18
16 EFPY	5.02E+18	7.61E+18	8.91E+18	8.79E+18
32 EFPY	9.96E+18	1.51E+19	1.77E+19	1.72E+19
48 EFPY	1.49E+19	2.27E+19	2.65E+19	2.56E+19
54 EFPY	1.67E+19	2.55E+19	2.98E+19	2.87E+19

Braidwood 2

Calculated

	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	0°	**15°**	**30°[a]**	**45°**
8.53 EFPY	3.01E+18	4.53E+18	5.32E+18	5.39E+18
12 EFPY	4.20E+18	6.34E+18	7.43E+18	7.41E+18
16 EFPY	5.56E+18	8.43E+18	9.87E+18	9.74E+18
32 EFPY	1.10E+19	1.68E+19	1.96E+19	1.90E+19
48 EFPY	1.65E+19	2.51E+19	2.94E+19	2.83E+19
54 EFPY	1.86E+19	2.83E+19	3.30E+19	3.18E+19

Note:

a. Maximum neutron exposure projection reported for 30° vessel location representing the octant containing the 12.5° neutron pad span.

Table 7.5.3-8

Neutron Fluence Projections on the Reactor Vessel Clad/Base Metal Interface for Selected Circumferential Weld Locations Along the 45° Azimuth

Braidwood 2

Best Estimate

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 8.53 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	6.34E+15	8.74E+15	1.16E+16	2.30E+16	3.44E+16	3.87E+16
WR-19	8.94E+15	1.23E+16	1.63E+16	3.24E+16	4.85E+16	5.45E+16
WR-34	1.41E+18	1.95E+18	2.58E+18	5.12E+18	7.66E+18	8.61E+18
WR-18	4.72E+18	6.49E+18	8.61E+18	1.71E+19	2.56E+19	2.87E+19
WR-29	6.24E+14	8.60E+14	1.14E+15	2.26E+15	3.38E+15	3.80E+15

Braidwood 2

Calculated

Weld Location	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 8.53 EFPY	12 EFPY	16 EFPY	32 EFPY	48 EFPY	54 EFPY
WR-20	7.03E+15	9.68E+15	1.28E+16	2.55E+16	3.81E+16	4.28E+16
WR-19	9.90E+15	1.36E+16	1.81E+16	3.59E+16	5.37E+16	6.04E+16
WR-34	1.57E+18	2.16E+18	2.86E+18	5.67E+18	8.49E+18	9.54E+18
WR-18	5.22E+18	7.20E+18	9.54E+18	1.89E+19	2.83E+19	3.18E+19
WR-29	6.91E+14	9.52E+14	1.26E+15	2.51E+15	3.75E+15	4.21E+15

Notes:

WR-20: Outlet Nozzle Shell Forging Circumferential Weld

WR-19: Inlet Nozzle Shell Forging Circumferential Weld

WR-34: Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld

WR-18: Intermediate Shell Forging to Lower Shell Forging Circumferential Weld

WR-29: Lower Shell Forging Circumferential Weld

7.6 Radiation Source Terms

7.6.1 Introduction and Background

This section describes the input parameters and methodology used in the calculation of radiation source terms applicable to the uprating program for the Byron and Braidwood plants. Radiation source terms for several different accident and normal operating conditions were determined for the power uprate conditions. These source terms were used as input to dose and balance-of-plant analyses. The reanalyzed areas included the following:

Total core inventory.
Reactor coolant system fission products.
Gas decay tank activities.

Each of these source term calculations is discussed in subsequent subsections.

7.6.2 Total Core Inventory

7.6.2.1 Input Parameters and Assumptions

The assumptions and input parameters used in the determination of the total core inventory are summarized in Tables 7.6-1 and 7.6-2.

7.6.2.2 Description of Analysis

Fuel burnup and fission product production were modeled via the ORIGEN2 code (References 1 through 5). ORIGEN2 is a versatile point-depletion and radioactive decay code for use in simulating nuclear fuel cycles and calculating the nuclide concentration and characteristics of materials contained therein. The code considers the transmutation of all isotopes in the material. For the relatively high fluxes in the core region of the reactor, burn-in and burn-out of isotopes can have an important effect. This is particularly true for fuel cycle designs with high burnup regions. These important effects are modeled in the ORIGEN2 calculations.

For the Byron and Braidwood uprating program, a representative equilibrium fuel cycle operating at the uprated power conditions was modeled in the ORIGEN2 calculations. The definition of this equilibrium cycle is provided in Table 7.6-2.

The ORIGEN2 analysis for the uprating program modeled a single fuel assembly from each region of the core. Burnup calculations, reflecting each of the appropriate power histories, were performed, and the total inventory for each region, at the end of the equilibrium cycle, was then determined by multiplying the individual assembly isotopic inventory by the number of assemblies in the respective regions. Finally, the results for each region of the core were summed to produce the total core inventory. This methodology was previously submitted to the NRC and reviewed in Reference 6.

7.6.2.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to various radiological evaluations.

7.6.2.4 Results

The total core inventory of actinide and fission product activities was provided as an input to various radiological evaluations in Section 6.7.

7.6.3 Reactor Coolant System Fission Product Activities

7.6.3.1 Input Parameters and Assumptions

The parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the expected coolant cleanup flow rate, are presented in Tables 7.6.2 and 7.6-3. In the reactor coolant system activity calculations, small cladding defects (equivalent to 1% of the fuel rods) were assumed to be present in all regions of the equilibrium fuel cycle. Therefore, fission product escape rate coefficients, based on an average fuel temperature, were used in the analysis.

7.6.3.2 Description of Analysis

The fission product inventory in the reactor coolant, during operation of the equilibrium fuel cycle with defects in 1% of the fuel rods, was computed. In the calculations, there was no credit taken for reduction of fission product concentrations due to pressurizer operation. Likewise, no credit was taken for fission product removal due to purge of the volume control tank. Further, in determining the reactor coolant system inventory for individual isotopes, the maximum activity occuring at any time during the fuel cycle was taken in each case. Therefore, the total set of fission product concentrations does not represent any particular time during the fuel cycle, but rather, a composite of the maximum activity concentration exhibited by each isotope. This overall approach represents a conservative treatment of the reactor coolant system.

7.6.3.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to various radiological evaluations.

7.6.3.4 Results of Analyses

The reactor coolant system fission product activities were provided as input to various radiological evaluations in Section 6.7. A summary of the results of this evaluation is given in Table 7.6-4.

7.6.4 Gas Decay Tank Activities

7.6.4.1 Input Parameters and Assumptions

Radiological inventories for the gas decay tanks (GDT) were determined in a manner similar to that used to determine reactor coolant system activities. For conservatism, the entire calculated inventory, expressed as a volumetric activity, was assumed to be placed in a single gas decay tank. Additional input parameters used in the gas decay tank source calculation are provided in Tables 7.6-1 through 7.6-3.

7.6.4.2 Description of Analyses

Gas decay tank activities were calculated for thirteen noble gas nuclides. For the calculation, a continuous volume control tank purge rate of 0.7 standard cubic feet per minute (scfm) was assumed. For the gas decay tank, this assumption represents a conservative treatment, resulting in a gas decay tank inventory that remains applicable for high pressure or periodic purge modes of operation. The isotopic inventories resulting from this calculation represent the design activities for the uprated conditions with 1% defective fuel and the volume control tank purge system operating.

7.6.4.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to various radiological evaluations.

7.6.4.4 Results of Analyses

The gas decay tank activities were provided as an input to the gas decay tank rupture radiological consequences analysis in Section 6.7.10. A summary of the results of this evaluation is given in Table 7.6-5.

7.6.5 References

1. CCC-371/ORIGEN2 Version 2.1.

2. RSIC Computer Code Collection: ORIGEN2.1 – Isotopic Generation and Depletion Code-Matrix Exponential Method.

3. A. G. Croff, "A User's Manual for the ORIGEN2 Computer Code," ORNL/TM-7175, July 1980.

4. A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," Nuclear Technology, Volume 62, September 1983.

5. T. M. Lloyd, "Conversion/Configuration/Validation of the ORIGEN2 Code, Version 2.1," Westinghouse Calculation Note RSAC-M-813, October 11, 1993.

6. Safety evaluation by the office of Nuclear Reactor Regulation related to Amendment No. 137 to Facility Operating License No. NPF-2 and Amendment No. 129 to Facility Operating License No. NPF-8 Southern Nuclear Operating Company Inc., et al,. Joseph M. Farley Nuclear Plant, Units 1 and 2 Docket Nos. 50-348 and 50-364.