TABLE 15.0-1

•→14

RESULTS SUMMARY OF TRANSIENTS EVENTS APPLICABLE TO BWRs
(For Original Rated Power 2894 Mwt)

14←•

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown	
										No. of Valves First Blowdown	Duration of Blowdown (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of feedwater heater, automatic flow control	112.3	1,046	1,085	1,035	106.07	(2)	a	0	0
15.1.1	15.1-2	Loss of feedwater heater, manual flow control	121.0	1,060	1,099	1,047	113.88	0.12	a	0	0
15.1.2	15.1-3	Feedwater control failure, max demand	176.3	1,191	1,222	1,188	106.45	0.11	a	16	6
15.1.3[3]	15.1-4	Pressure regulator									
		fail - open	104.31	1,127	1,159	1,127	100.27	(2)	a	16	5
15.1.4		Inadvertent opening of safety or relief valve	See Text								
15.1.6		RHR shutdown cooling malfunction decreasing temp	See Text								
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1	15.2-1	Pressure regulation downscale failure	160.8	1,186	1,219	1,182	102.69	0.09	a	16	7

TABLE 15.0-1 (Cont)
(For Original Rated Power 2894 Mwt)

•→14
14←•

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.2.2	15.2-2	Generator load rejection, bypass on	189.3	1,191	1,219	1,185	102.63	0.08	a	16	6
15.2.2	15.2-3	Generator load rejection, bypass off	237.7	1,204	1,232	1,198	104.88	0.11	a	16	7
15.2.3	15.2-4	Turbine trip, bypass on	164.1	1,189	1,217	1,184	100.92	0.07	a	16	6
15.2.3	15.2-5	Turbine trip, bypass off	216.0	1,203	1,231	1,198	103.22	0.09	a	16	7
15.2.4	15.2-6	All MSIV closure	105.15	1,178	1,207	1,174	100.10	(2)	a	16	5
15.2.5	15.2-7	Loss of condenser vacuum	168.7	1,190	1,217	1,184	100.90	(2)	a	16	6
15.2.6	15.2-8	Loss of auxiliary power transformer	104.2	1,171	1,186	1,170	100.05	(2)	a	16	5
15.2.6	15.2-9	Loss of all grid connections	121.08	1,187	1,211	1,182	100.03	(2)	a	16	8
15.2.7	15.2-10	Loss of all feedwater flow	104.2	1,046	1,085	1,035	100.06	(2)	a	0	0
15.2.8		Feedwater piping break	See Table 15.0-3, event 15.6.6								
15.2.9		Failure of RHR shutdown cooling	See Text								
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									

TABLE 15.0-1 (Cont)
(For Original Rated Power 2894 Mwt)

•→14
14←•

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.3.1	15.3-1	Trip of one recirculation pump motor	104.2	1,047	1,085	1,036	100.0	(2)	a	0	0
15.3.1	15.3-2	Trip of both recirculation pump motors	104.2	1,168	1,182	1,165	100.0	(2)	a	16	5
15.3.2	15.3-3	Fast closure of one main recirc valve	104.2	1,049	1,085	1,037	100.0	(2)	a	0	0
15.3.2	15.3-4	Fast closure of two main recirc valves	104.2	1,175	1,188	1,171	100.12	(2)	a	0	0
15.3.3	15.3-5	Seizure of one recirculation pump	104.2	1,167	1,184	1,164	100.14	(2)	c	16	5
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling	See Text						b		
15.4.1.2		RWE - Startup	See Text						b		
15.4.2		RWE - At power	See Text						a		
15.4.3		Control rod mis-operation	See 15.4.1 and 15.4.2								
15.4.4	15.4-3	Abnormal startup of idle recirculation loop	122.8	993	1,007	987	161.14	(2)	a	0	0

TABLE 15.0-1 (Cont)
(For Original Rated Power 2894 Mwt)

•→14
14←•

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.4.5	15.4-4	Fast opening of one main recirc valve	472.4	998	1,009	984	149.56	(2)	a	0	0
15.4.5	15.4-5	Fast opening of both main recirc valves	353.7	982	1,005	978	140.50	(2)	a	0	0
15.4.7		Misplaced bundle accident	See Text						b		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS pump start	104.2	1,046	1,085	1,035	100.12		a	0	0
15.5.3		BWR transients	See appropriate Events in 15.1 and 15.2								

(1) a = moderate
b = infrequent
c = limiting fault

(2) ΔCPR <0.12

(3) The pressure regulator failure-open is no longer a postulated single failure scenario; no Ovation TCPS single failure scenarios are postulated that could result in all valves open. This specific USAR section is maintained for historical purposes.

TABLE 15.0-1A

RESULTS SUMMARY OF TRANSIENTS EVENTS APPLICABLE TO BWRs

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % original NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of feedwater heater, automatic flow control	112.3	1,046	1,085	1,035	106.07	(2)	a	0	0
15.1.2 (NOTES 6,8)	15.1-2	Loss of feedwater heater, manual flow control	121.0	1,060	1,099	1,047	113.88	0.12	a	0	0
15.1.2 (NOTES 1,8)	15.1-3	Feedwater control failure, max demand	176.3	1,191	1,222	1,188	106.45	0.11	a	16	6
15.1.3 (NOTE 10)	15.1-4	Pressure regulator fail - open	105.31	1,127	1,159	1,127	100.27	(2)	a	16	5
15.1.4		Inadvertent opening of safety or relief valve	See Text								
15.1.6		RHR shutdown cooling malfunction decreasing temp	See Text								
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1 (NOTES 2,8)	15.2-1	Pressure regulation downscale failure	160.8	1,186	1,219	1,182	102.69	0.09	a	16	7

TABLE 15.0-1A (Cont)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % original NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.2.2	15.2-2	Generator load rejection, bypass on	189.3	1,191	1,219	1,185	102.63	0.08	a	16	6
15.2.2 (NOTES 3,8)	15.2-3	Generator load rejection, bypass off	237.7	1,204	1,232	1,198	104.88	0.11	a	16	7
15.2.3	15.2-4	Turbine trip, bypass on	164.1	1,189	1,217	1,184	100.92	0.07	a	16	6
15.2.3 (NOTES 4,8)	15.2-5	Turbine trip, bypass off	216.0	1,203	1,231	1,198	103.22	0.09	a	16	7
15.2.4 (NOTES 5,9)	15.2-6	All MSIV closure	118.9	1,229	1,262	1,228	100.1		a	9	
15.2.5	15.2-7	Loss of condenser vacuum	168.7	1,190	1,217	1,184	100.90	(2)	a	16	6
15.2.6	15.2-8	Loss of auxiliary power transformer	104.2	1,171	1,186	1,170	100.05	(2)	a	16	5
15.2.6	15.2-9	Loss of all grid connections	121.08	1,187	1,211	1,182	100.03	(2)	a	16	8
15.2.7	15.2-10	Loss of all feedwater flow	104.2	1,046	1,085	1,035	100.06	(2)	a	0	0
15.2.8		Feedwater piping break	See Table 15.0-3, event 15.6.6								
15.2.9		Failure of RHR shutdown cooling	See Text								
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									

TABLE 15.0-1A (Cont)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % original NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category[1]	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.3.1	15.3-1	Trip of one recirculation pump motor	104.2	1,047	1,085	1,036	100.0	(2)	a	0	0
15.3.1	15.3-2	Trip of both recirculation pump motors	104.2	1,168	1,182	1,165	100.0	(2)	a	16	5
15.3.2	15.3-3	Fast closure of one main recirc valve	104.2	1,049	1,085	1,037	100.0	(2)	a	0	0
15.3.2	15.3-4	Fast closure of two main recirc valves	104.2	1,175	1,188	1,171	100.12	(2)	a	0	0
15.3.3	15.3-5	Seizure of one recirculation pump	104.2	1,167	1,184	1,164	100.14	(2)	c	16	5
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling	See Text						b		
15.4.1.2		RWE - Startup	See Text						b		
15.4.2 (NOTES 7,8)		RWE - At power	See Text						a		
15.4.3		Control rod mis-operation	See 15.4.1 and 15.4.2								
15.4.4	15.4-1	Abnormal startup of idle recirculation loop	122.8	993	1,007	987	161.14	(2)	a	0	0

TABLE 15.0-1A (Cont)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % original NBR	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum Critical Power Ratio	Frequency Category(1)	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.4.5	15.4-2	Fast opening of one main recirc valve	472.4	998	1,009	984	149.56	(2)	a	0	0
15.4.5	15.4-3	Fast opening of both main recirc valves	353.7	982	1,005	978	140.50	(2)	a	0	0
15.4.7		Misplaced bundle accident	See Text						b		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS pump start	104.2	1,046	1,085	1,035	100.12	___	a	0	0
15.5.3		BWR transients	See appropriate Events in 15.1 and 15.2								

(1) a = moderate
b = infrequent
c = limiting fault

(2) ΔCPR <0.12

TABLE 15.0-1A (Cont)

Notes:

(1) The Feedwater Controller Failure case has been re-analyzed at 3039 MWt core power with normal and reduced feedwater temperature. Results of the re-analyses are reported in Table 15.0-1B.

(2) Pressure Regulator Downscale Failure case has been re-analyzed at 3039 MWt core power with normal and reduced feedwater temperature. The results of the reduced feedwater temperature case are reported in Table 15.0-1B. The normal feedwater temperature case is bounded by the reduced feedwater temperature.

(3) The Generator Load Rejection with no Bypass option case has been re-analyzed at 3039 MWt core power with normal and reduced feedwater temperature, at full and partial arc turbine control valve (TCV) options. Results of the re-analyses are reported in Table 15.0-1B. The reduced feedwater temperature case is bounded by the normal feedwater temperature, and the partial arc is bounded by the full arc TCV mode of operation.

(4) The Turbine Trip with no Bypass option case has been re-analyzed at 3039 MWt core power with normal and reduced feedwater temperature, at full and partial arc turbine control valve (TCV) options. Results of the re-analyses are reported in Table 15.0-1B. The reduced feedwater temperature case is bounded by the normal feedwater temperature, and the partial arc is bounded by the full arc TCV mode of operation.

(5) DELETED

(6) The Loss of Feedwater Heater event is re-analyzed at 3100 MWt core power (2% over 3039 MWt). The resulting delta CPR is 0.11. The results of the re-analyses are reported in Table 15.0-1B. This event is described in detail in Section 15.1, and also in Appendix 15B.

(7) The RWE event is re-analyzed at 3039 MWt core power. The resulting delta CPR at full power is 0.16. The event is also described in detail in Section 15.4.

(8) This event is re-analyzed at 3091 MWt (100% TPO rated core power) per Reference 10, App. E. The results of the re-analyses are reported in Appendix 15B.

(9) This MSIV closure with position scram event is re-analyzed at 3091 MWt consistent with Cycle 19. The results of the re-analyses are reported in Section 15.2.4. The calculated change in Minimum Critical Power Ratio is bounded by the load rejection without bypass and turbine trip without bypass. As such, the MSIV closure event with position scram need not be evaluated each reload. Note that the blowdown through the safety relief valves had not ended by the end of the simulation (~8 seconds after start of the event), but pressure is decreasing rapidly.

(10) The pressure regulator failure-open is no longer a postulated single failure scenario; no Ovation TCPS single failure scenarios are postulated that could result in all valves open. This specific USAR section is maintained for historical purposes.

TABLE 15.0-1B
Summary of Events Analyzed at Power Uprate Conditions

ANALYSIS ID	ANALYSIS NAME	TRANSIENT OUTPUT FILE NAMES (.*)	ODYN PID	SUB EVENTS	POWER (%)	FLOW (%)	STEAM FLOW (%)
100P107F	STANDARD	000E6_E00000_T02_ODYNV09_LRNBP	00264	7SRVOS	100.0	107.0	100.0
100P107F	STANDARD	000E6_E00000_T03_ODYNV09_TTNBP	001A1	7SRVOS	100.0	107.0	100.0
100P107F	STANDARD	000E6_E00000_T04_ODYNV09_FWCF	0019C	7SRVOS	100.0	107.0	100.0
100P107F	STANDARD	000E6_E00000_T05_ODYNV09_PRFDS	00305	7SRVOS	100.0	107.0	100.0
100P107F(1)	STANDARD	00165_E00000_T05_ODYNV09_PRFDS	00A65	7SRVOS	100.0	107.0	88.2

TABLE 15.0-1B (Cont)

TRANSIENT NAME	ODYN PID	ANALYSIS ID	EXPOSURE Mwd/st	PEAK FLUX (N) % ref	PEAK FLUX (Q/A) % init	MAX QFUEL OFUEL PU	MAX NET REACT $	DCPR G1136	DCPR B684W
LRNBP	00264	100P107F	E00000	429.70	116.77	0.59	0.76	0.1931	0.1191
TTNBP	001A1	100P107F	E00000	407.36	114.49	0.53	0.75	0.1822	0.1022
FWCF	0019C	100P107F	E00000	316.77	111.81	0.00	0.69	0.1430	0.0753
PRFDS	00305	100P107F	E00000	145.65	104.81	0.00	0.29	0.0987	0.0464
PRFDS[1]	00A65	100P107X	E00000	146.80	105.71	0.00	0.29	0.1180	0.0505

TABLE 15.0-1B (Cont)

				G1136		B684W	
TRANSIENT NAME	ODYN PID	ANALYSIS ID	EXPOSURE Mwd/st	DCPRB	DCPRA	DCPRB	DCPRA
LRNBP	00264	100P107F	E00000		0.2079		0.1256
TTNBP	001A1	100P107F	E00000		0.1982		0.1086
FWCF	0019C	100P107F	E00000		0.1574		0.0852
PRFDS	00305	100P107F	E00000		0.1120		0.0566
PRFDS[(1)]	00A65	100P107F	E00000		0.1318		0.0597

TABLE 15.0-1B (Cont)

TRANSIENT NAME	ODYN PID	ANALYSIS ID	EXPOSURE Mwd/st	PEAK FLUX Q/A % init	PEAK DOME PRESSURE RATE psi/sec	PEAK PRESSURE DOME psig	PEAK PRESSURE P(V) psig	PEAK PRESSURE P(SL) psig	MIN DELTA P(UCL) psi	MIN DELTA P(SSV) psi	MIN DELTA P(ECL) psi
LRNBP	00264	100P107F	E00000	116.77	301.0	1269.8	1296.4	1265.6	78.6		203.6
TTNBP	001A1	100107F	E00000	114.49	319.6	1268.2	1295.1	1264.1	79.9		204.9
FWCF	0019C	100P107F	E00000	111.81	329.8	1244.2	1267.8	1242.0	107.2		232.2
PRFDS	00305	100P107F	E00000	104.81	114.1	1255.8	1284.0	1253.2	91.0		216.0
PRFDS[1]	00A65	100P107X	E00000	105.71	108.9	1249.3	1277.0	1247.7	98.0		223.0

[1] Second PRFDS case run with Reduced Feedwater Temperature

TABLE 15.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

1. Thermal power level, MWt Warranted value Analysis value	 2,894 3,015
2. Steam flow, lb/hr Warranted value Analysis value	 12.45×10^6 13.07×10^6
3. Core Flow, lb/hr	84.5×10^6
4. Feedwater flow rate[1], lb/sec Warranted value Analysis value	 3,458 3,631
5. Feedwater temperature, °F	425
6. Vessel dome pressure, psig	1,045
7. Vessel core pressure, psig	1,056
8. Turbine bypass capacity, % NBR	10
9. Core coolant inlet enthalpy, Btu/lb	529.9
10. Turbine inlet pressure, psig	960
11. Fuel lattice	P 8X8R
12. Core average gap conductance, Btu/sec-ft^2-°F	0.1892
13. Core leakage flow, %	11
14. Required MCPR operating limit First core Reload core	 1.18 1.19
15. MCPR safety limit First core Reload core	 1.06 1.07
16. Doppler coefficient (-)¢/°F Analysis data[2]	 0.132
17. Void coefficient (-)¢/% rated voids Analysis data for power increase events[2][4] Analysis data for power decrease events[2]	 14.0 4.0
18. Core average rated void fraction, %[2]	42.53
19. Scram reactivity Analysis data[2][4]	 Fig. 15.0-2
20. Control rod drive speed, position versus time	Fig. 15.0-3
21. Jet pump ratio, M	2.47

TABLE 15.0-2 (Cont)

22. SRV capacity, % NBR @ 1,210 psig Manufacturer Quantity installed	109.4 Crosby 16
23. Relief function delay, sec	0.40
24. Relief function response Time constant, sec	 0.10
25. Safety function delay, sec	0.0
26. Safety function response Time constant, sec	 0.2
27. Set points for SRVs Safety function, psig Relief Function, psig	 1175, 1185, 1195 1205, 1215 1125, 1135, 1145 1155
28. Number of valve groupings simulated Safety function, no. Relief function, no.	 5 4
29. SRV reclosure Set point - both modes (% of set point) Maximum safety limit (used in analysis) Maximum operational limit	 98 89
30. High flux trip, % NBR Analysis set point (122 x 1.042)	 127.2
31. High pressure scram set point, psig	1,095
32. Vessel level trips, ft above bottom of separator skirt bottom Level 8 - (L8), ft Level 4 - (L4), ft Level 3 - (L3), ft Level 2 - (L2), ft	 5.88 4.03 1.94 (-)2.86
33. APRM simulated thermal power trip, % NBR Analysis set point (114 x 1.042)	 118.8
34. Time constant, sec	7
35. Nuclear characteristics used in ODYN simulations[4]	End of equilibrium cycle (EOEC)
36. Recirculation pump trip delay, sec	0.14
37. Recirculation pump trip inertia time constant for analysis, sec[3]	Max 5.0 Min 3.0

TABLE 15.0-2 (Cont)

38. Total steam line volume, ft^3	3,275
39. Pressure set point of recirculation pump trip - psig (nominal)	1,135

(1) Includes control rod drive flow

(2) Applies only for events analyzed using model described in Reference 1 to Section 15.1.

(3) The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_O n}{g T_O}$$

where:

t = Inertia time constant (sec)
J_O = Pump motor inertia (lb-ft)
n = Rated pump speed (rps)
g = Gravitational constant (ft/sec)
T_O = Pump shaft torque (lb-ft)

(4) The transient analyses for RBS are based on end of equilibrium cycle (EOEC) nuclear parameters for a pre-control cell core design. These analyses results described in Chapter 15 are bounding for limiting transients relative to the expected performance of the plant at end of cycle 1 conditions for the control cell core.

TABLE 15.0-2A

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS[10]

	Original Power	5% Power Uprate
1. Thermal power level, MWt Warranted value Analysis Value	 2,894 3,015	 - - - - 3039
2. Steam Flow, lb/hr Warranted value Analysis Value	 12.45×10^6 13.07×10^6	 - - - - $13{,}199 \times 10^6$
3. Core Flow, lb/hr	84.5×10^6	68.6×10^6 - 90.4×10^6 [5]
4. Feedwater flow rate[1], lb/sec Warranted value Analysis value	 3,458 3,631	 - - - - 3666.4
5. Feedwater temperature, °F	425	425.7
6. Vessel dome pressure, psig	1,045	1055
7. Vessel core pressure, psig	1,056	1070
8. Turbine bypass capacity, % NBR	10	9.48
9. Core coolant inlet enthalpy, Btu/lb	529.9	531.2
10. Turbine inlet pressure, psig	960	1012[6]
11. Fuel lattice	P8x8R	GE8x8EB GE11
12. Core average gap conductance, Btu/sec-ft^2-°F	0.1892	0.3657[7]
13. Core leakage flow, %	11	13.9[7]
14. Required MCPR operating limit First core Reload core	 1.18 1.19	 - - - - 1.32
15. MCPR safety limit First core Reload core	 1.06 1.07	 - - - - 1.10
16. Doppler coefficient (-)¢/°F Analysis data [2]	0.132	0.139[7]
17. Void coefficient(-)¢/% rated voids Analysis data for power increase events [2][4] Analysis data for power decrease events [2]	 14.0 4.0	 9.94[7] - - - -
18. Core average rated void fraction, %[2]	42.53	40.01[7]
19. Scram reactivity, Analysis data [2] [4]	Fig. 15.0-2	Same
20. Control rod drive speed, position versus time	Fig. 15.0-3	Same
21. Jet pump ratio, M	2.47	2.48

TABLE 15.0-2A (Cont)

	Original Power	5% Power Uprate
22. Installed SRV capacity, % NBR Manufacturer Quantity installed	109.4[(8)] Crosby 16	100.2[(8)] same Same
23. Relief function delay, sec	0.40	Same
24. Relief function response Time constant, sec	0.10	Same
25. Safety function delay, sec	0.0	Same
26. Safety function response Time constant, sec	0.2	Same
27. Set points for SRVs Safety function, psig Relief function, psig	1175, 1185, 1195 1205, 1215 1125, 1135, 1145, 1155	1231, 1241, 1246 1163, 1173, 1183
28. Number of valve groupings simulated Safety function, no. Relief function, no.	5 4	3 3
29. SRV reclosure Set point-both modes (% of set point) Maximum safety limit (used in analysis) Maximum operational limit	98 89	Same Same
30. High flux trip, % NBR Analysis set point (122 x 1.042)	127.2	122.0
31. High pressure scram set point, psig	1,095	1,125
32. Vessel level analysis trips, ft above bottom of separator skirt Level 8 - (L8), ft Level 4 - (L4), ft Level 3 - (L3), ft Level 2 - (L2), ft	5.88 4.03 1.94 (-)2.86	Same Same Same Same
33. APRM simulated thermal power trip, % NBR Analysis set point (114 x 1.042)	118.8	115.0
34. Time constant, sec	7	6.6[(9)]
35. Nuclear characteristics used in ODYN simulations[(4)]	End of equilibrium cycle (EOEC)	Same as Cycle 7/8 core loading
36. Recirculation pump trip delay, sec	0.14	Same
37. Recirculation pump trip inertia time constant for analysis, sec[(3)]	Max 5.0 Min 3.0	Max 6.0[(3)] Same
38. Total steam line volume, ft^3	3243.6	3,275
39. Pressure set point of recirculation pump trip - psig (nominal)	1,135	1157

TABLE 15.0-2A (Cont)

Notes:

(1) Includes control rod drive flow

(2) Applicable only for events analyzed using model described in Reference 1 to Section 15.1.

(3) The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_o n}{gT_o}$$

where:

t = Inertia time constant (sec)
J_o = Pump motor inertia (lb-ft)
n = Rated pump speed (rps)
g = Gravitational constant (ft/sec)
T_o = Pump shaft torque (lb-ft)

The 6 second inertia characteristic has been conservatively assumed in T-G trip RPT analysis for several reload cycles (per OPL-3).

(4) The transient analyses for RBS are based on end of equilibrium cycle (EOEC) nuclear parameters for a pre-control cell core design. These analyses results described in Chapter 15 are bounding for limiting transients relative to the expected performance of the plant at end of cycle 1 conditions for the control cell core.

(5) Transients were performed at the core flow range of 81% to 107% of rated or 68.6 to 90.4 Million lb-hr (rated core flow is 84.5 Mlb/hr).

(6) Turbine inlet pressure is measured at Turbine Stop Valve (TSV) inlet conditions.

(7) Values taken from PANACEA/ODYN/CRNC at increased core flow (ICF) condition (100%P/107%F).

(8) The Safety and Relief valve setpoints were obtained from OPL-3 at uprate conditions, where the capacities are based at a reference pressure of 1080 psig. The capacities originally developed for this USAR table were based on a reference pressure of 1210 psig. The pre-uprate OPL-3 information exchange has been shifted to the 1080 psig reference pressure for several cycles.

(9) The 6.6 value for the SIP time constant has been in use for several reload cycles (per OPL-3).

(10) The input parameters and initial conditions for the cycle-specific analyses are given in Attachment B to Appendix 15B.

TABLE 15.0-3

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Rods	
		GE-Calculated Value	NRC Worst-Case Assumption
15.3.3	Seizure of one recirculation pump	None	
15.3.4	Recirculation pump shaft break	None	
•→8			
15.4.9	Rod drop accident	<770	770**
8←•			
15.6.2	Instrument line break	None	None
15.6.4	Steam system pipe break outside containment	None	None
15.6.5	LOCA within RCPB	None	100%
15.6.6	Feedwater line break-outside containment	None	None
15.7.1.1	Main condenser gas treatment system failure	N/A	N/A
15.7.3	Liquid radwaste tank failure	N/A	N/A
15.7.4	Fuel handling accident	<125	125
15.7.5	Cask drop accident	None	None
15.8	ATWS	*	

___

* Special event still under negotiation
•→8
** See Appendix 15B2.3 for reload core conditions
8←•

TABLE 15.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATER,
AUTO FLOW CONTROL (FIGURE 15.1-1)

Time (sec)	Event
0	Initiate a 100°F temperature reduction in the feedwater system
5	Initial effect of unheated feedwater starts to raise core power level but the automatic flow control system automatically reduces core flow to maintain initial steam flow
40	Reactor variables settle into new steady state

TABLE 15.1-2

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATER, MANUAL FLOW CONTROL (FIGURE 15.1-2)

Time (sec)	Event
0	Initiate a 100°F temperature reduction into the feedwater system
5	Initial effect of unheated feedwater starts to raise core power level and steam flow
10	Turbine control valves start to open to regulate pressure
61.9	Initiation of reactor scram on high simulated thermal power
73.0	Narrow range (NR) sensed water level reaches Level 3 (L3) set point
73.2	Trip of recirculation pump power source to low frequency MG speed; RPT initiates due to Level 3 Trip (not included in simulation)
>80(est)	Wide range (WR) sensed water level reaches Level 2 (L2) set point
>80	Recirculation pumps trip off due to Level 2 RPT
>110(est)	HPCS/RCIC flow enters vessel (not simulated)
>120(est)	Reactor variables settle into limit cycle.

TABLE 15.1-3

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND (FIGURE 15.1-3)

•→14

Time (sec)	Event
0	Initiate simulated failure of 108 percent upper limit on feedwater flow at a system design pressure of 1,065 psig
32.3	L8 vessel level set point initiates reactor scram and trips main turbine and feedwater pumps
32.4	RPT actuated by stop valve position switches
32.4	Main turbine bypass valves opened due to turbine trip
33.8	SRVs open due to high pressure
36	Water level dropped to low water level setpoint (Level 2)
>66 (est)	RCIC and HPCS flow into vessel (not simulated)

14←•

TABLE 15.1-4

SEQUENCE OF EVENTS FOR PRESSURE REGULATOR FAILURE - OPEN TO 130% (FIGURE 15.1-4)
(NOTE 1)

Time (sec)	Event
0	Simulate steam flow demand to 130 percent
0.5	Main turbine bypass fully opens
8	Turbine control valves wide open
19	Low turbine inlet pressure trip initiates main steam isolation
19.5	MSIV closure initiates reactor scram
22.0	Vessel water level reaches L3 set point. Recirculation pumps trip to low frequency M/G sets.
25.5(est)	SRVs open
26(est)	Vessel water level reaches L2 set point. Recirculation pumps trip due to Level 2 RPT signal.
30.5	SRVs close
41.18	Group 1 SRVs open again to relieve decay heat
46.18	Group 1 SRVs close again
>50 (est)	HPCS and RCIC flow enters vessel (not simulated)

NOTE 1: The pressure regulator failure-open is no longer a postulated single failure scenario; no Ovation TCPS single failure scenarios are postulated that could result in all valves open. This specific USAR section is maintained for historical purposes.

TABLE 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

Time-sec	Event
0	Initiate opening of one SRV
0.5 (est)	Relief flow reaches full flow
15 (est)	System establishes new steady-state operation

TABLE 15.1-6

SEQUENCE OF EVENTS FOR INADVERTENT RHR
SHUTDOWN COOLING OPERATION

Approximate Elapsed Time	Event
0	Reactor at states B or D (Appendix 15A) when RHR shutdown cooling inadvertently activated
0-10 min	Slow rise in reactor power
+10 min	Operator may take action to limit power rise; Flux scram occurs if no action is taken

TABLE 15.2-1

SEQUENCE OF EVENTS FOR PRESSURE REGULATION
DOWNSCALE FAILURE (FIGURE 15.2-1)

Time (sec)	Event
0	Simulate zero steam flow demand to main turbine and bypass valves
0	Turbine control valves start to close
•→14 0.86	Neutron flux reaches high flux scram set point and initiates a reactor scram
2.07	Reactor pressure reaches high pressure setpoint and initiates recirculation pump trip
2.64	SRVs open

•→13
*Shown for information only. Starting with Cycle 10 this event is no longer required to be analyzed.
13←• 14←•

TABLE 15.2-2

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITH BYPASS (FIGURE 15.2-2)

Time (sec)	Event
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load
0	Turbine-generator load rejection sensing devices trip to initiate TCV fast closure and main turbine bypass system operation
0	Turbine control valve (TCV) fast closure initiates scram trip and RPT
0.07	TCVs closed
0.10	Turbine bypass valves start to open
1.30	SRVs open due to high pressure
7.66	SRVs close

TABLE 15.2-3

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITH FAILURE OF BYPASS (FIGURE 15.2-3)

Time (sec)	Event
(-)0.015 (approx.)	T-G detection of loss of electrical load
0	T-G load rejection sensing devices trip to initiate TCV fast closure
0	Turbine bypass valves fail to operate
0	TCV fast closure initiates scram trip and RPT
•→14	
0.08	TCVs closed
1.53	SRVs open due to high pressure
14←•	

TABLE 15.2-4

SEQUENCE OF EVENTS FOR TURBINE TRIP WITH BYPASS (FIGURE 15.2-4)

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves
0	Turbine trip initiates bypass operation
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip and RPT
0.10	Turbine stop valves close
0.10	Turbine bypass valves start to open to regulate pressure
1.34	SRVs open due to high pressure
7.60	SRVs close

TABLE 15.2-5

SEQUENCE OF EVENTS FOR TURBINE TRIP WITH FAILURE OF BYPASS (FIGURE 15.2-5)

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves
0	Turbine bypass valves fail to operate
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip and RPT
0.10	Turbine stop valves close
•→14	
1.58	SRVs open due to high pressure
14←•	

TABLE 15.2-6

SEQUENCE OF EVENTS FOR CLOSURE OF ALL MSIVs (FIGURE 15.2-6)

Time (sec)	Event
0	Initiate closure of all MSIVs
0.45	MSIVs reach 85% open
0.45	MSIV position trip scram initiated
3.6	SRVs open due to high pressure
>8 (est)	SRVs close

TABLE 15.2-7

RADIOLOGICAL CONSEQUENCES OF MSIV CLOSURE

●→14

Isotope	Maximum Concentration Released (Ci/hr)	Restricted Area Boundary Concentration (μ Ci/cc)	ECL[1]	Percent of ECL
I-131	1.2-4[2]	1.1-13	2.0-10	5.6-2
I-132	1.0-3	9.4-13	2.0-8	4.7-3
I-133	1.5-3	1.4-12	1.0-9	1.4-1
I-134	1.1-3	1.0-12	6.0-8	1.7-3
I-135	1.3-3	1.2-12	6.0-9	2.0-2
Br-83	4.1-6	3.7-15	9.0-8	4.1-6
Br-84	1.3-6	1.2-15	8.0-8	1.5-6
Br-85	7.1-8	6.5-17	1.0-9	6.5-6
Kr-83m	2.8-1	2.5-10	5.0-5	5.1-4
Kr-85m	5.8-1	5.3-10	1.0-7	5.3-1
Kr-85	1.4-2	1.3-11	7.0-7	1.9-3
Kr-87	1.2+0	1.1-9	2.0-8	5.7+0
Kr-88	1.7+0	1.6-9	9.0-9	1.8+1
Xe-131m	2.9-3	2.7-12	2.0-6	1.4-4
Xe-133m	4.9-2	4.5-11	6.0-7	7.5-3
Xe-133	1.4+0	1.3-9	5.0-7	2.5-1
Xe-135m	8.0+0	7.3-9	4.0-8	1.8+1
Xe-135	4.3+0	3.9-9	7.0-8	5.6+0
Xe-138	1.6+0	1.5-9	2.0-8	7.3+0

Total percent of ECL = 55.4

---

(1) Effluent Concentration Limits (airborne) in unrestricted areas from 10CFR20, Appendix B, Table 2 Column 1.

(2) 1.2-4 = 1.2×10^{-4}

14←●

TABLE 15.2-8

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

Cause		Estimated Vacuum Decay Rate
1.	Failure or isolation of steam jet air ejectors	<1 in Hg/min
2.	Loss of sealing steam to shaft gland seals	Approximately 1 to 2 in Hg/min
3.	Opening of vacuum breaker valves	Approximately 2 to 12 in Hg/min
4.	Loss of one or more circulating water pumps	Approximately 4 to 24 in Hg/min

TABLE 15.2-9

SEQUENCE OF EVENTS FOR LOSS OF CONDENSER VACUUM (FIGURE 15.2-7)

Time (sec)	Event
-3.0 (est)	Initiate simulated loss of condenser vacuum at 2 in of Hg/sec
0.0 (est)	Low condenser vacuum main turbine trip actuated
0.08	Main turbine trip initiates RPT and scram
1.34	SRVs open due to high pressure
5.0	Low condenser vacuum initiates MSIV closure
5.6	Low condenser vacuum initiates bypass valve closure
7.60	SRVs close
8.13	Group 1 SRVs open again to relieve decay heat
14.03	Group 1 SRVs close again
17.42	Group 1 SRVs open again to relieve decay heat
23.45	Group 1 SRVs close again

TABLE 15.2-10

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (in of Hg)	Protective Action Initiated
27 to 28	Normal vacuum range
20 to 23	Main turbine trip (stop valve closures)
7 to 10	Main steam isolation valve (MSIV) closure and bypass valve closure

TABLE 15.2-11

SEQUENCE OF EVENTS FOR LOSS OF NORMAL AND PREFERRED STATION SERVICE TRANSFORMERS (FIGURE 15.2-8)

Time (sec)	Event
0	Loss of normal and preferred station service transformers occurs.
0	Recirculation system pump motors are tripped.
0	Feedwater and condensate pumps are tripped.
2.00	Reactor scram and closure of MSIV occur due to loss of power to the solenoids.
5.10	SRVs open due to high pressure
10.12	SRVs close
12.10	Group 1 SRVs cycle open and close on pressure
21	Vessel water level reaches Level 2 set point
>45 (est)	HPCS and RCIC flow enters vessel (not simulated)

TABLE 15.2-12

SEQUENCE OF EVENTS FOR LOSS OF ALL GRID CONNECTIONS (FIGURE 15.2-9)

Time (sec)	Event
(-)0.015 (approx.)	Loss of grid causes T-G to detect a loss of electrical load.
0	Turbine control valve fast closure is initiated.
0	T-G power-load unbalance (PLU) trip initiates main turbine bypass system operation.
0	Recirculation system pump motors are tripped.
0	TCV fast closure initiates a reactor scram trip.
0.07	TCVs closed.
0.11	Turbine bypass valves open.
1.33	SRVs open due to high pressure.
2.00	Closure of MSIV due to loss of power.
9.03	SRVs close.
19 (est)	Vessel water level reaches Level 2 set point.
>45(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-13

SEQUENCE OF EVENTS FOR LOSS OF ALL FEEDWATER FLOW (FIGURE 15.2-10)

Time (sec)	Event
0	Trip of all feedwater pumps initiated.
2.33	Vessel water level reaches level 4 and initiates recirculation flow runback.
4.88	Feedwater flow decays to zero.
8.71	Vessel water level (L3) trip initiates scram trip and recirculation pumps trip to low frequency M/G set.
24(est)	Vessel water level reaches Level 2.
24(est)	Recirculation pumps trip due to Level 2 RPT signal.
>50(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-14

•→14 •→13

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING
Configuration for Activity C1 (a)

Time (sec)	Event
13←•	
0	Reactor is operating at 100.3 percent rated when loss of offsite power occurs initiating power plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
600	Suppression pool cooling initiated to prevent overheating from SRV actuation.
1465	Controlled depressurization initiated (100°F/hr) using selected safety/relief valves.
8900	Blowdown to approximately 100 psig completed.
8900	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
9200	ADS valves are opened to complete blowdown to suppression pool, and RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.
11,600	Cold Shutdown achieved (200 degrees F RPV Temperature)
16,146	Maximum Suppression Pool Temperature (183.1 degrees F)

14←•

TABLE 15.2-14a

•→14

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING
Configuration for Activity C1 (b)

Time (sec)	Event
0	Reactor is operating at 100.3 percent rated power when loss of offsite power occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
600	Suppression pool cooling initiated to prevent overheating from SRV actuation.
1465	Controlled depressurization initiated (100°F/hr) using selected safety/relief valves.
8900	Blowdown to approximately 100 psig completed.
8900	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
9200	ADS valves are opened to complete blowdown to suppression pool, and RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.
13,368	Maximum Suppression Pool Temperature (177.9 degrees F)
40,586	Cold Shutdown achieved (200 degrees F RPV Temperature)

14←•

TABLE 15.2-15

INPUT PARAMETERS FOR EVALUATION OF FAILURE
OF RHR SHUTDOWN COOLING

Initial Conditions

•→14

Rated power (%)	100.3
Suppression pool water volume (ft^3)	1.228E5
RHR Hx constant (Btu/sec/°F)	390
Vessel pressure (psia)	1072
Vessel temperature (°F)	553
Primary coolant inventory (lbm)	4.598E5
Pool temperature (°F)	100
Service water temperature (°F)	95
Vessel heat capacity (Btu/lbm/°F)	0.123
HPCS flow rate (lbm/sec)	676.8
Maximum at 0 psid, vessel to drywell pressure difference	
LPCI flow rate per loop (lbm/sec)	686.1
Maximum at 0 psid, vessel-to-drywell pressure difference	
LPCI Pump Heat (HP)	700
HPCS Pump Heat (HP)	2500

14←•

TABLE 15.3-1

SEQUENCE OF EVENTS FOR TRIP OF ONE RECIRCULATION PUMP
(FIGURE 15.3-1)

Time (sec)	Event
0	Trip of one recirculation pump initiated
5	Jet pump diffuser flow reverses in the tripped loop
~39.0	Core flow and power level stabilize at new equilibrium conditions.

TABLE 15.3-2

SEQUENCE OF EVENTS FOR TRIP OF TWO RECIRCULATION PUMPS
(FIGURE 15.3-2)

Time (sec)	Event
0	Trip of both recirculation pumps initiated
4.2	Vessel water level (L8) trip initiates scram, turbine trip and feedwater pump trip
4.3	Turbine trip initiates bypass operation
5.8	SRVs open due to high pressure
11.2	SRVs close
17.2	Vessel water level (L2) set point reached
47.2(est)	HPCS and RCIC flow enters vessel (not simulated)

TABLE 15.3-3

SEQUENCE OF EVENTS FOR FAST CLOSURE OF ONE MAIN RECIRCULATION VALVE (FIGURE 15.3-3)

Time (sec)	Event
0	Initiate fast closure of one main recirculation valve
2	Jet pump diffuser flow reverses in the affected loop
40(est)	Core flow and power approach new equilibrium conditions

TABLE 15.3-4

SEQUENCE OF EVENTS FOR FAST CLOSURE OF TWO MAIN RECIRCULATION VALVES (FIGURE 15.3-4)

Time (sec)	Event
0	Initiate fast closure of both main recirculation valves
5.15	Vessel level (L8) trip initiates scram and turbine trip
5.15	Feedwater pumps tripped off
5.30	Turbine trip initiates bypass operation
6.46	SRVs open due to high pressure
12.31	SRVs close
17.50	Vessel water level reaches Level 2 set point
47.50(est)	HPCS and RCIC flow enters vessel (not simulated)

TABLE 15.3-5

SEQUENCE OF EVENTS FOR RECIRCULATION PUMP SEIZURE
(FIGURE 15.3-5)

Time (sec)	Event
0	Single pump seizure was initiated.
0.8	Jet pump diffuser flow reverses in seized loop.
3.11	Vessel level (L8) trip initiates reactor scram.
3.11	Vessel level (L8) trip initiates turbine and feedwater pump trips.
3.30	Turbine trip initiates bypass operation.
3.35	Turbine trip initiates recirculation pumps trip.
4.85	SRVs open due to high pressure.
10.2	SRVs close.
12.7	Vessel water level reaches Level 2 set point.
42.7(est)	HPCS/RCIC flow enters the vessel (not simulated).

TABLE 15.4-l

SEQUENCE OF EVENTS FOR
ROD WITHDRAWAL ERROR IN POWER RANGE

Elapsed Time	Event
0	Core is operating on thermal limits with a typical control rod pattern.
0	Operator selects and withdraws a single rod or gang of rods continuously.
~1 sec	The local power in the vicinity of the withdrawn rod (or gang) increases. Total core power output increases.
~4* sec	RWL blocks further withdrawal.
~25 sec	Core stabilizes at slightly higher core power level.

*Based on a l.0-foot RWL increment.

TABLE 15.4-2

SEQUENCE OF EVENTS FOR ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP (FIGURE 15.4-1)

Time (sec)	Event
0	Start pump motor
1.26	Jet pump diffuser flows on started pump side become positive
2.73	Pump motor at full speed and drive flow at about 25% of rated
21.5 (est)	Last of cold water leaves recirculation drive loop
22.0	Peak value of core inlet subcooling
50.0 (est)	Reactor variables settle into new steady state

TABLE 15.4-3

SEQUENCE OF EVENTS FOR FAST OPENING OF ONE RECIRCULATION VALVE (FIGURE 15.4-2)

Time (sec)	Event
0	Simulate failure of single loop control
0.97	Reactor APRM high flux scram trip initiated
3.5 (est)	TCVs start to close upon falling turbine pressure
7.9 (est)	TCVs closed; turbine pressure below pressure regulator set points
>1OO (est)	Reactor variables settle into new steady state

TABLE 15.4-4

SEQUENCE OF EVENTS FOR FAST OPENING OF TWO RECIRCULATION VALVES (FIGURE 15.4-3)

Time (sec)	Event
0	Initiate failure of master controller
1.0	Reactor APRM high flux scram trip initiated
4.0 (est)	TCVs start to close upon falling turbine pressure
8.0 (est)	TCVs closed; turbine pressure below pressure regulator set points
>lOO (est)	Reactor variables settle into new steady state

TABLE 15.4-5

SEQUENCE OF EVENTS FOR MISPLACED BUNDLE ACCIDENT

1. During core loading operation, a bundle is placed in the wrong location.
2. Subsequently, the bundle intended for this location is placed in the location of the previous bundle.
3. During core verification procedure, the two errors are not observed.
4. Plant is brought to full power operation without detecting misplaced bundle.
5. Plant continues to operate throughout the cycle.

TABLE 15.4-6

INPUT PARAMETERS AND INITIAL CONDITIONS FOR
FUEL BUNDLE LOADING ERROR

●→8

	Input Parameters	Initial Conditions *
1.	Power, % rated	100
2.	Flow, % rated	100
3.	MCPR operating limit (est)	1.18
4.	MLHGR operating limit, kw/ft	13.4
5.	Core exposure	End of cycle

8←●

NOTE: Core conditions are assumed to be normal for a hot, operating core at EOC.

●→8
* See Appendix 15B for reload core conditions
8←●

TABLE 15.4-7

•→8

RESULTS OF WORST FUEL BUNDLE LOADING
ERROR ANALYSIS (INITIAL CORE)

8←•

1.	MCPR limit	1.18
2.	MCPR with misplaced bundle	1.08
3.	Δ CPR for event	0.10
4.	LHGR limit	13.4
5.	LHGR with misplaced bundle	14.7
6.	Δ LHGR for event	1.3

•→8

NOTE: See Appendix 15B for reload core conditions

8←•

TABLE 15.4-8

SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time (sec)	Event
	Reactor is operated at 50 percent rod density pattern
	Maximum worth control rod blade becomes decoupled from the CRD
	Operator selects and withdraws the CRD of the decoupled rod either individually or along with other control rods assigned to the RCIS group
	Decoupled control rod sticks in the fully inserted or an intermediate bank position
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations
<1	Reactor goes on a positive period and the initial power increase is terminated by the Doppler coefficient
<1	APRM 120 percent power signal scrams reactor
<5	Scram terminates accident

TABLE 15.4-9

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR ROD WORTH COMPLIANCE CALCULATION

Input Parameters	Initial Conditions
1. Reactor power, % Rated	0.0
2. Reactor flow, % Rated	0.0
3. Core average exposure, MWd/t	0.0
4. Control rod fraction	Approx. 0.50
5. Average fuel temperature, °C	286
6. Average moderator temperature °C	286
7. Xenon state	None

TABLE 15.4-10

INCREMENT WORTH OF THE MOST REACTIVE ROD USING A BANK POSITION WITHDRAWAL SEQUENCE[1]

Core Condition (MWD/T)	Control Rod Group	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase In k_{eff}
0.0	7	04	(24,49)	00-08	0.0012
0.0	7	08	(24,49)	00-12	0.0032
0.0	7	12	(24/49)	00-48	0.0079
0.0	7	48	(24,49)	00-48	0.0005

[1] The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:

a. BOC 1, 0.0 GWD/St average exposure
b. Hot startup
c. No xenon
d. Rod groups 1-6 withdrawn
e. Sequence A

TABLE 15.4-11

•→14 •→8 8←• 14←•

CONTROL ROD DROP ACCIDENT RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
A. Data and assumptions used to estimate radioactive source from postulated accident	
1. Power Level	3100 MWt
2. Number of damaged rods	
100% Power Event	850 GE 8x8
Low Power Event (gap release)	50 GE 8x8
3. Total Rods in core	
GE 8x8	38,688 (62 rods per assembly)
GE 9x9	46,176 (74 rods per assembly)
GE 10x10	57,408 (92 rods per assembly)
4. Number of assemblies damaged	
Design Basis - Maximum Fuel Damage (based on 8x8)	850/62 = 13.7
Limited CRDA (Based on 8x8)	50/62 = 0.8
Note: For the CRDA scenario GE8 fuel is bounding. This is confirmed each reload cycle.	
5. Core Activity available for release	Table 15.4-11A
6. Radial Peaking Factor	2.00
7. Assumed % fuel melt	
Design Basis - Maximum Fuel Damage	100%
Limited CRDA	0%
8. Gap Activity Release Fractions	Per RG 1.183, Table 3 and Appendix C 10% noble gases, 10% iodines, 12% alkali metals
9. Fuel Melt Release Fractions	Per RG 1.183, Appendix C 100% noble gases, 50% iodines
10. Fuel Release Duration	
Design Basis- Maximum Fuel Damage	Instantaneous
Limited CRDA	10 sec. burst
B. Data and assumptions used to estimate activity released to the environment.	
1. Condenser Leak Rate	
Design Basis CRDA	Per RG 1.183, Appendix C 1% per day for 24 hours,
Limited CRDA	4000 cfm for 20 minutes, 1% per day for next 24 hours
2. Condenser Iodine Release Fractions	Per RG 1.183, Appendix C 97% Elemental, 3% Organic

TABLE 15.4-11 (Cont)

•→14 •→8 8←• 14←•

Description of Input/Assumption	Design Basis Input and/or Assumption
3. Condenser Radioactive Decay During Holdup	Credited
4. Condenser Volume	106,460 ft^3
C. Dispersion Data	
1. EAB X/Q Data	
0-2 hrs	7.51E-04 sec/m^3
2. LPZ X/Q Data	
0-8 hrs	7.79E-05 sec/m^3
8-24 hrs	5.23E-05 sec/m^3
1-4 days	2.21E-05 sec/m^3
4-30 days	6.40E-06 sec/m^3
3. Control Room X/Q Data	
Main Air Intake	
0-2 hrs	3.02E-3 sec/m^3
2-8 hrs	2.47E-3 sec/m^3
8-24 hrs	1.05E-3 sec/m^3
1-4 days	9.01E-4 sec/m^3
4-30 days	6.74E-4 sec/m^3
D. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm
3. Outside Air Ventilation Rate	1700 cfm
4. Intake Iodine Filter Efficiency	
Design Basis CRDA	
Aerosol	0%
Elemental and Organic	0%
Limited CRDA	
Aerosol	99%
Elemental and Organic	98%
5. Time for Control Room Ventilation Isolation per Operator Action	
Design Basis CRDA	Not credited
Limited CRDA	20 minutes
6. Emergency Mode Recirculation Rate (Post-isolation Mode)	2000 cfm
7. Control Room Breathing Rates and Occupancy Factors	Per RG 1.183

•→14 •→2 2←• 14←•

TABLE 15.4-11A

RBS CRDA CORE ACTIVITY

Isotope	EOC Core Inventory (Ci/MWt)
Kr-85	3.66E+02
Kr-85m	7.02E+03
Kr-87	1.35E+04
Kr-88	1.89E+04
Rb-86	6.31E+01
I-131	2.70E+04
I-132	3.92E+04
I-133	5.52E+04
I-134	6.06E+04
I-135	5.17E+04
Xe-133	5.26E+04
Xe-135	1.99E+04
Cs-134	6.11E+03
Cs-136	2.00E+03
Cs-137	3.95E+03

TABLE 15.4-12

CRDA ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	100% Power Event
Kr-85	1.61E+01
Kr-85m	3.08E+02
Kr-87	5.93E+02
Kr-88	8.30E+02
Rb-86	3.33E-05
I-131	5.93E+00
I-132	8.61E+00
I-133	1.21E+01
I-134	1.33E+01
I-135	1.14E+01
Xe-133	2.31E+03
Xe-135	8.74E+02
Cs-134	3.22E-03
Cs-136	1.05E-03
Cs-137	2.08E-03

•→14
14←•

NOTE: 1.26E+02 = 1.26×10^{2}

TABLE 15.4-13

•→14
14←•

CONTROL ROD DROP ACCIDENT RADIOLOGICAL CONSEQUENCES

Receptor	Regulatory Limit (TEDE)	Design Basis Event Dose (TEDE)	Limited CRDA Dose (TEDE)
EAB	6.3	1.0	4.91
LPZ	6.3	0.4	0.51
Control Room	5	4.4	1.30

•→2
2←•

RBS USAR

TABLE 15.5-1

SEQUENCE OF EVENTS FOR INADVERTENT STARTUP OF HPCS
(FIGURE 15.5-1)

Time (sec)	Event
0	Simulate HPCS cold water injection
3	Full flow established for HPCS
7	Depressurization effect stabilized

TABLE 15.6-1

SEQUENCE OF EVENTS FOR STEAM LINE BREAK OUTSIDE CONTAINMENT*

Time (sec)	Event
0	Guillotine break of one main steam line outside primary containment
0.5 (Approx.)	High steam line flow signal initiates closure of MSIVs
<1.0	Reactor begins scram
≤5.5	MSIVs fully closed
●→10	
~60	SRVs open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1,000 psi
~190	RCIC and HPCS would initiate on low water level, L2, (RCIC considered unavailable, HPCS assumed single failure and therefore not available)
~800	ADS receives signal to initiate on low water level, L1; ADS bypass timer starts
~1300	All ADS timers timed out. ADS valves are actuated initiating rapid depressurization of vessel
~1400	Reactor water level above core begins to drop slowly due to loss of steam through the SRVs; reactor pressure still at approximately 1,000 psi
~1420	LPCS system initiates injection
~1450	LPCI system initiates injection
~1600	Reactor vessel water level recovers back to initial level: no fuel rod heatup and no fuel rod failure.
10←●	

*See also Section 6.3.3.7.7.

TABLE 15.6-2

MAIN STEAM LINE BREAK RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Maximum Pre-accident Spike Iodine Concentration	4 µCi/gm DE I-131
3. Maximum Equilibrium Iodine Concentration	0.2 µCi/gm DE I-131
4. Noble Gas Source Term	Based on 310,000 µCi/sec at 30 minutes, corrected to time equal zero.
5. Alkali Metals	Reactor coolant activity design concentration ratioed to account for 102% power.
II. Data and assumptions used to estimate activity released to the environment.	
1. Mass Release	Steam, 11,620 lbm Liquid, 68,942 lbm
Note: This data corresponds to that calculated for initial licensing of the plant. Analyses demonstrate these values bound the hot standby conditions for Power Uprated conditions.	
2. Iodine Carryover Fraction	4%
3. Break Isolation Time	5.5 seconds
4. Building Release Rate	Instantaneous ground level release with no credit for plateout, holdup, or dilution.
5. Iodine Species Release Fractions to Environment	Per RG 1.183, Appendix D, 95% Aerosol, 4.85 % Elemental, 0.15% Organic
6. Activity Released to Environment	Table 15.6-3

TABLE 15.6-2 (Cont)

Description of Input/Assumption	Design Basis Input and/or Assumption
III. Dispersion Data	
1. EAB X/Q Data	
0-2 hrs	6.33E-04 sec/m^3
2. LPZ X/Q Data	
0-8 hrs	7.57E-05 sec/m^3
8-24 hrs	5.08E-05 sec/m^3
1-4 days	2.13E-05 sec/m^3
4-30 days	6.24E-06 sec/m^3
3. Control Room X/Q Data *	
0-2 hrs	1.42E-03 sec/m^3
2-8 hrs	1.08E-03 sec/m^3
8-24 hrs	4.57E-04 sec/m^3
1-4 days	3.50E-04 sec/m^3
4-30 days	2.58E-04 sec/m^3
* Since no credit is taken for the ESF filter trains, the X/Q values assumed in this analysis were based on the Main Air Intake.	
IV. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm
3. Outside Air Ventilation Rate	1700 cfm
4. Emergency Mode Filtered Intake/Unfiltered Inleakage Rate (1700 cfm ventilation rate + 300 cfm inleakage rate)	2000 cfm

•→11 •→3 3←• 11←•

TABLE 15.6-3

MAIN STEAM LINE BREAK ACTIVITY RELEASED TO ENVIRONMENT

Isotope	0.2μCi/gm DE* I-131 Case (Ci)	4μCi/gm DE I-131 Case (Ci)
I-131	1.70E+00	3.40E+01
I-132	2.51E+01	5.02E+02
I-133	2.27E+01	4.53E+02
I-134	3.97E+01	7.94E+02
I-135	2.19E+01	4.37E+02
Cs-134	5.42E-03	5.42E-03
Cs-136	3.51E-03	3.51E-03
Cs-137	1.40E-02	1.40E-02
Kr-85m	5.22E-02	5.22E-02
Kr-85	1.63E-04	1.63E-04
Kr-87	1.79E-01	1.79E-01
Kr-88	1.79E-01	1.79E-01
Xe-133	6.58E-02	6.58E-02
Xe-135	1.95E-01	1.95E-01

•→14 •→11
14←•
11←•

* Dose Equivalent

NOTE: 1.9E+01 = 1.9×10^{1}

TABLE 15.6-4

MAIN STEAM LINE BREAK ACCIDENT RADIOLOGICAL CONSEQUENCES

Case	EAB Dose (REM TEDE)	LPZ Dose (REM TEDE)	Regulatory Limit (REM TEDE)
4µCi/gm DE* I-131	1.4	0.2	25
0.2µCi/gm DE I-131	<0.1	<0.1	2.5

Case	Control Room Dose (REM TEDE)	Regulatory Limit (REM TEDE)
4µCi/gm DE I-131	2.2	5
0.2µCi/gm DE I-131	0.2	5

•→14 •→11 •→3

3←• 11←• 14←•

* Dose Equivalent

TABLE 15.6-5

LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL CONSEQUENCES ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/ or Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accident	
•→14	
A. Power level	3,100 MWt
B. Core Activity available for release	Table 15.6-6
C. Gap Activity Release Fractions	Per Table 1 of RG 1.183
D. Release fission product species and chemical form	Per RG 1.183, Section 3.5
•→13 •→10	
II. Release Rates	
A. Primary Containment Leakage Rate	
0-24 hours	0.325 volume % per day
1-30 days	0.179 volume % per day
B. Secondary Containment Bypass Leakage Rate	
0-24 hours	580,000 cc/hr @ Pa (0.341 cfm)
1-30 days	319,000 cc/hr @ Pa (0.188 cfm)
C. Main Steam Line Leakage	
0-25 minutes	150 scfh = 2.52 cfm*
25 minutes - 30 days	0 scfh
D. Engineered Safety Features Leakage	1 gpm

* 150 scfh was converted based on a maximum DW temperature of 330° F. The pressure assumed was 7.6 psig (power uprate reports show that the drywell pressure is 22.8 psia @ 121 seconds decreases steadily to ~19 psia at 10 minutes).

Description of Input/Assumption	Design Basis Input and/ or Assumptions
III. Dispersion Data	
A. EAB X/Q Data	See Table 15.6-5A
B. LPZ X/Q Data	See Table 15.6-5A
C. Control Room X/Q Data	See Table 15.6-5A
IV. Control Room Parameters	
A. Unfiltered In-Leakage Rate	300 cfm
B. Outside Air Ventilation Rate	
Actual	2000 cfm
Assumed	2000-300 = 1700 cfm
C. Filter Initiation Time	20 minutes
D. CR ESF Iodine Filter Efficiency	
Elemental/Organic (Charcoal)	98%
Particulate (HEPA)	99%
E. CR Breathing Rates and Occupancy Factors	Per RG 1.183

Table 15.6-5 (Cont)

V. Standby Gas Treatment Parameters	
A. Positive Pressure Period	30 minutes
B. Flow Rates	
Annulus	2,500 cfm
Auxiliary Building	10,000 cfm
C. SGTS Iodine Filter Efficiency	
Elemental/Organic (Charcoal)	90%
Particulate (HEPA)	99%
VI. Building Volumes	
A. Drywell	2.36E+05 ft^3
B. Containment	1.19E+06 ft^3
C. Annulus***	3.57E+05 ft^3
D. Auxiliary Building*	1.16E+06 ft^3
E. Control Room	1.88E+05 ft^3
F. Suppression Pool**	1.25E+05 ft^3

* Only 50% of the auxiliary building volume was credited in the actual analysis (values listed are actual volumes)

** The Actual analysis conservatively assumed a volume of 120,000 ft^3.

*** The annulus volume in this analysis was 1.0 ft^3 to minimize mixing of the annulus atmosphere reflecting disabling of the annulus mixing system.

VII. Containment Mixing Data	
A. Blowdown Data	
(Drywell ⇒ Containment)	
0-10 minutes	4.74E+05 cfm
10 minutes +	0 cfm
B. Hydrogen Mixing Data	
(Drywell ⇔ Containment)	
0-25 minutes	0 cfm
25 minutes - 1.9 hours	600 cfm
1.9 hours - 30 days (Infinite mixing)	1.0E+08 cfm
C. Steaming Data	
(Drywell ⇒ Containment)	
0-25 minutes	0 cfm
25 minutes - 1.9 hours	3000 cfm
1.9 hours - 30 days (Infinite mixing)	(Included in Hydrogen Mixing)
VIII. Misc. Data	
A. Dose Conversion Factors	Based on Federal Guidance Report 11 & 12
B. Off-Site Breathing Rates	Based on RG 1.183, Section 4.1.3
A. Drywell Plateout Coefficients	
Elemental	1.01hr^{-1}
Particulate	RADTRAD default - Powers (10) Model
D. ESF Leakage - Halogen Flashing Fraction	0.10

●→11 10←● 11←● 13←● 14←●

1.16E+06 = 1.16x10^6

Table 15.6-5A

X/Q VALUES USED IN LOCA ANALYSIS

Release Point	EAB*	LPZ	MCR
SGTS/Containment			
0-2 hours	6.05E-4	7.49E-5	2.55E-4
2-8 hours	6.05E-4	7.49E-5	1.92E-4
8-24 hours	6.05E-4	5.02E-5	8.09E-5
1-4 days	6.05E-4	2.10E-5	6.22E-5
4-30 days	6.05E-4	6.13E-6	5.09E-5
Turbine Building			
0-2 hours	7.51E-4	7.79E-5	4.66E-4
2-8 hours	7.51E-4	7.79E-5	3.83E-4
8-24 hours	7.51E-4	5.23E-5	1.67E-4
1-4 days	7.51E-4	2.21E-5	1.27E-4
4-30 days	7.51E-4	6.40E-6	9.33E-5

* The 0-2 hour values conservatively apply for the duration of the accident to ensure the “maximum” 2 hour dose is calculated as required per RG 1.183.

NOTE: 2.55E-4 = 2.55×10^{-4}

TABLE 15.6-6

●→14 ●→13 ●→10

BWR CORE INVENTORIES

AST Group	Isotope	Core Inventory (CI/MWT)
5	Co-58	2.61E+02
5	Co-60	4.50E+02
1	Kr-85	3.66E+02
1	Kr-85m	7.02E+03
1	Kr-87	1.35E+04
1	Kr-88	1.89E+04
3	Rb-86	6.31E+01
4	Sr-89	2.54E+04
4	Sr-90	2.91E+03
4	Sr-91	3.20E+04
4	Sr-92	3.47E+04
6	Y-90	3.08E+03
6	Y-91	3.28E+04
6	Y-92	3.49E+04
6	Y-93	4.04E+04
6	Zr-95	4.78E+04
6	Zr-97	4.98E+04
6	Nb-95	4.80E+04
5	Mo-99	5.13E+04
5	Tc-99m	4.51E+04
5	Ru-103	4.19E+04
5	Ru-105	2.90E+04
5	Ru-106	1.61E+04
5	Rh-105	2.67E+04
4	Sb-127	2.92E+03
4	Sb-129	8.77E+03
4	Te-127	2.95E+03
4	Te-127m	3.92E+02
4	Te-129	8.62E+03

10←● 13←● 14←●

Table 15.6-6 (Cont)

AST Group	Isotope	Core Inventory (CI/MWT)
4	Te-129m	1.28E+03
4	Te-131m	3.92E+03
4	Te-132	3.84E+04
2	I-131	2.70E+04
2	I-132	3.92E+04
2	I-133	5.52E+04
2	I-134	6.06E+04
2	I-135	5.17E+04
1	Xe-133	5.26E+04
1	Xe-135	1.99E+04
3	Cs-134	6.11E+03
3	Cs-136	2.00E+03
3	Cs-137	3.95E+03
4	Ba-139	4.92E+04
4	Ba-140	4.74E+04
6	La-140	4.88E+04
6	La-141	4.49E+04
6	La-142	4.33E+04
7	Ce-141	4.49E+04
7	Ce-143	4.15E+04
7	Ce-144	3.69E+04
6	Pr-143	4.06E+04
6	Nd-147	1.80E+04
7	Np-239	5.42E+05
7	Pu-238	1.10E+02
7	Pu-239	1.28E+01
7	Pu-240	1.66E+01
7	Pu-241	5.26E+03
6	Am-241	6.55E+00
6	Cm-242	1.51E+03
6	Cm-244	7.62E+01

NOTE: 6.81E+03 = 6.81×10^3

TABLE 15.6-7

●→11 11←● ●→13 ●→10

LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL CONSEQUENCES

Release Descriptions	EAB	LPZ	MCR
Containment/Secondary Containment	4.0	2.2	0.46
Secondary Containment Bypass/MSIV	13.4	6.2	3.12
ESF Liquid Leakage	0.4	0.6	0.16
Total	17.8	9.0	3.7
Regulatory Limit	25.0	25.0	5.00

●→14 10←● 13←● 14←●

●→13 ●→11 11←● 13←●

TABLE 15.6-8

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time (sec)	Event
0	One feedwater line breaks
0+	Feedwater line check valves isolate the reactor from the break
5 (approx)	Reactor scram on low water level
<30	HPCS and RCIC start on low-low water level, L2, and are expected to maintain the water level above the low-low-low level, L1, trip, and eventually restore it to the normal elevation
1 to 2 hr	Normal reactor cooldown procedure established

TABLE 15.7-1

SEQUENCE OF EVENTS FOR MAIN CONDENSER
OFF GAS TREATMENT SYSTEM FAILURE

Approximate Elapsed Time	Event
0 sec	Event begins - system fails.
0 sec	Noble gases are released.
< 1 min	Area radiation alarms alert plant personnel.
< 1 min	Operator actions begin with:
	1. Initiation of appropriate system isolations
	2. Manual scram actuation
	3. Assurance of reactor shutdown cooling.
1 hr	Steam jet air ejector is shut down.

TABLE 15.7-2

GASEOUS RADWASTE SYSTEM FAILURE - PARAMETERS TABULATED
FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source for postulated accidents	
A. Power level	NA
B. Burnup	NA
C. Fuel damage	None
D. Release of activity	Table 15.7-3
E. Iodine fractions	NA
1. Organic	NA
2. Elemental	NA
3. Particulate	NA
F. Reactor coolant activity before the accident	NA
II. Data and assumptions used to estimate activity released	
A. Containment leak rate (%/day)	NA
B. Secondary containment leak rate(%/day)	NA
C. Valve movement times	NA
D. Absorption and filtration efficiencies	NA
1. Organic iodine	NA
2. Elemented iodine	NA
3. Particulate iodine	NA
4. Particulate fission products	NA
E. Recirculation system parameters	NA
1. Flow rate	NA
2. Mixing efficiency	NA
3. Filter efficiency	NA
F. Containment volumes	NA
G. All other pertinent data and assumptions	None

TABLE 15.7-2 (Cont)

	Design Basis Assumptions
III. Dispersion data	
•→3	
A. EAB distance (m)	894
B. X/Qs for EAB (sec/m-)	8.58-4
3←•	
IV. Dose data	
A. Method of dose calculation	NA
B. Dose conversion assumptions	Reg. Guides 1.98 and 1.109
C. Peak activity concentrations in containment	NA
D. Doses	Table 15.7-4

NOTE: 8.58-4 = 8.58×10^{-4}

TABLE 15.7-3

EQUIPMENT FAILURE RELEASE ASSUMPTIONS RELEASE FRACTIONS ASSUMED FOR DESIGN BASIS ANALYSIS

Equipment Piece	Noble Gases	Particulate Daughters	Radioiodine
Holdup pipe	1.00	1.00	NA
Prefilter	NA	0.01	NA
Charcoal adsorbers	1.00	0.01	NA
Steam jet air ejector (1-hr release)	1.00	1.00	NA

TABLE 15.7-4

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE (DESIGN BASIS ANALYSIS) OFFSITE RADIOLOGICAL EFFECTS

	Dose at Exclusion Area Boundary	
Noble Gases	β Skin (rem)	γ Whole Body (rem)
Charcoal bed	6.0-1	6.8-1
Holdup pipe	1.2-2	2.3-2
Steam jet air ejector (1-hr release)	3.8-1	5.1-1
Particulates		
•→14		
Prefilter	-	5.6-2
Charcoal bed	-	1.9-4
Holdup pipe	-	3.2-4
Steam jet air ejector (1-hr release)	-	2.1-3
14←•		
TOTAL	9.9-1	1.3+0

NOTE: 6.0-1 = 6.0×10^{-1}

TABLE 15.7-5

ACCIDENT ANALYSIS ASSUMPTIONS
RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE
(RELEASE TO ATMOSPHERE)

1. Design bases activities of Section 11.1, and flows and fractions of primary coolant activities of Section 11.2 are used to develop source terms.

2. Noble gases are not considered due to constant venting of radwaste system tanks.

3. Halogen partition factor 0.001

4. Exclusion area boundary

	χ/Q (sec/m^3)	Breathing Rate (m^3/sec)
0-2 hr	8.58-4	3.47-4

NOTE: 8.58-4 = 8.58×10^{-4}

TABLE 15.7-6

LIQUID RADWASTE SYSTEM TANKS HALOGEN INVENTORIES

	Br-83 (Ci)	Br-84 (Ci)	Br-85 (Ci)	I-131 (Ci)	I-132 (Ci)	I-133 (Ci)	I-134 (Ci)	I-135 (Ci)
Floor drain collector tanks 2A, B, and C (total)	8.4-4	1.8-4	1.1-5	5.1-3	9.9-3	4.2-2	4.2-3	1.7-2
●→14								
Waste collector tanks 1A, B, C, and D (total)	3.6-1	7.2-2	3.6-3	3.6+0	3.3+0	2.0+1	1.9+0	7.6+0
Recovery sample tanks 4A, B, C, and D (total)	1.9-4	1.5-5	6.8-8	2.7-3	1.8-3	1.4-2	6.2-4	4.8-3
14←●								
Phase separator and backwash tanks 6A, B, and 7	2.1+1	4.6+0	2.7-1	1.6+3	6.2+2	1.6+3	1.1+2	4.4+2
Regenerant waste tanks 3A and B	3.2+1	7.0+0	4.0-1	4.0+2	2.8+2	2.2+3	2.2+2	7.0+2
Waste and regenerant evaporators EV-1 and 2	1.5+1	7.4-1	4.0-3	5.4+3	3.4+2	1.1+4	3.8+1	9.6+2
●→14								
Total in all tanks	6.8+1	1.2+1	6.8-1	7.4+3	1.2+3	1.5+4	3.7+2	2.1+3
14←●								

NOTE: 8.4-4 = 8.4×10^{-4}

TABLE 15.7-7

OFFSITE DOSES RESULTING FROM LIQUID RADWASTE SYSTEM TANKS RUPTURE

Exclusion area boundary	Thyroid (rem)	Whole Body Gamma (rem)	Beta (rem)
0-2 hr	5.1+0	4.0-3	1.8-3

NOTE: 4.0-3 = $4.0x10^{-3}$

TABLE 15.7-8

ACCIDENT ANALYSIS DATA
LIQUID RADWASTE TANK RUPTURE RELEASE TO GROUNDWATER

Regenerant Waste Evaporator

Feed rate	25.7 gpm
Concentration factor	42.8
Total operating volume	4,200 gal
Feed stream (regenerant waste tank)	See Table 15.7-9

Nearest Municipal Surface Water Supply - Bayou Lafourche, Louisiana

Travel time	8.72 yr
Dilution factor	1.72+10

NOTE: $1.72 +10 = 1.72x10^{10}$

TABLE 15.7-9

REGENERANT WASTE TANK INVENTORY

Isotope	μCi/cc	Isotope	μCi/cc
Na-24	5.4-2	Ag-110m	1.4-3
P-32	3.9-3	Te-129m	8.1-3
Cr-51	1.2-1	Te-131m	5.6-3
Mn-54	1.4-3	Te-132	2.4-1
Mn-56	5.3-2	Ba-139	1.1-1
Fe-55	2.1-2	Ba-140	2.0-1
Fe-59	1.8-3	Ba-141	3.0-2
Co-58	1.2-1	Ba-142	1.7-2
Co-60	1.1-2	La-142	3.2-2
Ni-63	2.1-5	Ce-141	3.7-3
Ni-65	3.1-4	Ce-143	1.8-3
Cu-64	1.4-1	Ce-144	8.6-4
Zn-65	4.2-3	Pr-143	4.8-3
Zn-69m	1.0-2	Nd-147	3.5-4
Sr-89	7.3-2	W-187	2.7-2
Sr-90	5.6-3	Np-239	3.8+0
Sr-91	3.1-1	Br-83	3.7-1
Sr-92	1.6-1	Br-84	8.3-2
Y-91	8.7-3	Br-85	4.7-3
Y-92	2.0-1	I-131	4.9+0
Y-93	9.1-2	I-132	3.5+0
Zr-95	1.0-3	I-133	3.1+1
Zr-97	2.4-4	I-134	2.6+0
Nb-95	9.8-4	I-135	8.5+0
Nb-98	6.6-3	Rb-89	2.4-3
Mo-99	3.8-1	Cs-134	3.8-3
Tc-99m	2.7-1	Cs-136	2.3-3
Tc-101	3.2-2	Cs-137	1.0-2
Tc-104	4.6-2	Cs-138	5.5-2
Ru-103	2.7-3		
Ru-105	1.8-2		
Ru-106	3.8-4		

NOTE: 5.4-2 = 5.4×10^{-2}

TABLE 15.7-10

RADWASTE EQUIPMENT FAILURE
ACCIDENT RADIOACTIVITY CONCENTRATIONS
AT BAYOU LaFOURCHE WATER SUPPLY

Isotope	Final Activity (μCi/cc)	Fraction of Maximum Permissible Concentrations*
I-129	1.5-22	2.4-15
Sr-90	1.1-11	3.8-05
Y-90	1.1-11	5.7-07
Ru-106	2.4-15	2.4-10
Cs-134	5.0-13	5.6-08
Cs-137	2.1-11	1.0-06
Ce-144	9.5-16	9.5-11
Pm-147	1.0-15	5.1-12
Mn-54	3.2-15	3.2-11
Fe-55	5.6-12	7.0-09
Co-60	8.7-12	1.7-07
Zn-65	1.3-15	1.3-11
Ag-110m	5.9-16	2.0-11
Ni-63	4.9-14	1.6-09
TOTAL		4.0-05

*Maximum permissible concentrations are from 10CFR20, Appendix B, Table II. column 2.

NOTE 1.5-22 = $1.5x10^{-22}$

TABLE 15.7-11

FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to Estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Number of damaged rods (GE 9x9)	150
3. Total number of rods in core	46,176 (Limiting GE 9x9 Fuel)
4. Core Activity available for release	Table 15.7-11A
5. Radial peaking factor	2.00
6. Gap Activity Release Fractions	RG 1.183
7. Release fission product species and chemical form	RG 1.183, Appendix B
8. Decay time	24 hours
II. Data and assumptions used to Estimate Activity released to the environment.	
1. Building Release Rate	2 hour linear release rate
2. Halogen Decontamination Factor	200
III. Dispersion Data	
1. EAB X/Q Data	
0-2 hrs	8.58E-04 sec/m^3
2. LPZ X/Q Data	
0-8 hrs	1.13E-04 sec/m^3
8-24 hrs	7.89E-05 sec/m^3
1-4 days	3.65E-05 sec/m^3
4-30 days	1.21E-05 sec/m^3
3. Control Room X/Q Data	
0-20 mm	1.62E-03 sec/m^3
20 min-8 hrs	4.05E-04 sec/m^3
8-24 hrs	3.00E-04 sec/m^3
1-4 days	1.01E-04 sec/m^3
4-30 days	1.62E-05 sec/m^3
IV. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm
3. Outside Air Ventilation Rate	1700 cfm
4. CR ESF Iodine Filter Efficiency	0% (Not credited)
5. Control Room Breathing Rates and Occupancy Factors	RG 1.183

TABLE 15.7-11A

FUEL HANDLING ACCIDENT CORE ACTIVITY AT REACTOR SHUTDOWN
(i.e., Decay Time = 0 hours)

Isotope	EOC Core Inventory (Ci/MWt)
I-131	2.70E+04
I-132	3.92E+04
I-133	5.52E+04
I-134	6.06E+04
I-135	5.17E+04
Kr-85	3.66E+02
Kr-85m	7.02E+03
Kr-87	1.35E+04
Kr-88	1.89E+04
Xe-133	5.26E+04
Xe-135	1.99E+04

TABLE 15.7-12

FUEL HANDLING ACCIDENT ACTIVITY RELEASED TO ENVIRONMENT

Isotope	Gap Activity [Ci] (t=0 hrs)	Gap Activity [Ci] (t=24 hrs)	Released to Environment [Ci]
Kr-83m	#N/A	#N/A	#N/A
Kr-85	7.37E+02	7.37E+02	7.37E+02
Kr-85m	7.07E+03	1.72E+02	1.72E+02
Kr-87	1.36E+04	2.83E-02	2.83E-02
Kr-88	1.90E+04	5.44E+01	5.44E+01
Kr-89	#N/A	#N/A	#N/A
I-131	4.35E+04	3.99E+04	2.00E+02
I-132	3.95E+04	2.85E+01	1.43E-01
I-133	5.56E+04	2.50E+04	1.25E+02
I-134	6.10E+04	3.50E-04	1.75E-06
I-135	5.21E+04	4.20E+03	2.10E+01
I-136	#N/A	#N/A	#N/A
Xe-131m	#N/A	#N/A	#N/A
Xe-133m	#N/A	#N/A	#N/A
Xe-133	5.30E+04	4.64E+04	4.64E+04
Xe-135m	#N/A	#N/A	#N/A
Xe-135	2.00E+04	3.21E+03	3.21E+03
Xe-137	#N/A	#N/A	#N/A
Xe-138	#N/A	#N/A	#N/A

•→13 •→10 10←• 13←•

NOTE: 1.90E+02 = 1.90×10^{2}

TABLE 15.7-13

FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

Receptor	Regulatory Limit (REM TEDE)	FHA Dose (REM TEDE)
EAB	6.3	2.6
LPZ	6.3	0.4
Control Room	5	1.7

●→13 ●→10 ●→8 ●→2 2←● 13←●
8←● 10←●

TABLE 15.7-14

MAXIMUM CONDENSATE STORAGE TANK INVENTORY

Isotope	μCi/cc	Isotope	μCi/cc
Br-83	3.2-13	Cs-134	7.2-6
I-129	6.3-17	Cs-136	3.8-6
I-131	3.7-4	Cs-137	1.8-5
I-132	1.9-4	Ba-137m	1.7-5
I-133	5.3-4	Ba-139	2.3-19
I-135	2.6-6	Ba-140	1.8-4
Sr-89	7.2-5	Ba-141	5.8-77
Sr-90	5.9-5	La-140	1.4-4
Sr-91	8.6-6	La-141	7.2-10
Sr-92	1.3-11	La-142	1.3-18
Y-90	3.3-5	CE-141	5.9-6
Y-91m	5.7-6	Ce-143	7.8-7
Y-91	1.7-5	Ce-144	8.5-7
Y-92	4.4-9	Pr-143	4.4-6
Y-93	3.0-6	Pr-144	8.5-7
Zr-95	9.9-7	Nd-147	3.3-7
Zr-97	3.7-8	Pm-147	8.3-10
Nb-95m	9.0-9	Na-24	2.0-5
Nb-95	1.0-6	P-32	1.1-5
Nb-97m	3.6-8	Cr-51	3.5-4
Nb-97	4.0-8	Mn-54	4.6-6
Mo-99	2.7-4	Mn-56	6.2-12
Tc-99m	2.6-4	Fe-55	6.5-5
Ru-103	2.6-6	Fe-59	5.1-6
Ru-105	1.9-9	Co-58	3.4-4
Rh-103m	2.6-6	Ni-63	6.5-8
Rh-105m	1.9-9	Ni-65	2.9-14
Rh-105	6.1-10	Cu-64	3.3-5
Rh-106	4.0-7	Zn-65	1.3-5
Te-129m	7.8-6	Zn-69m	2.8-6
Te-129	7.9-6	Ag-110m	4.0-6
Te-131m	2.2-6	Ag-110	8.0-8
Te-131	4.5-7	W-187	2.4-5
Te-132	1.8-4	Np-239	2.4-3

NOTE: 3.2-13 = 3.2×10^{-13}

TABLE 15.7-15

CONDENSATE STORAGE TANK RUPTURE ACCIDENT RADIOACTIVITY CONCENTRATIONS AT BAYOU LAFOURCHE WATER SUPPLY

Isotope	Final Activity (μCi/cc)	Fraction of Maximum Permissible Concentration*
Br-83	8.0-17	2.7-11
I-129	1.6-20	2.6-13
I-131	9.3-8	3.1-1
I-132	4.6-8	5.8-3
I-133	1.3-7	1.3-1
I-135	6.5-10	1.6-4
Sr-89	1.8-8	6.0-3
Sr-90	1.5-8	5.0-2
Sr-91	2.2-9	3.1-5
Sr-92	3.3-15	4.7-11
Y-90	8.3-9	4.1-4
Y-91m	1.4-9	4.7-7
Y-91	4.3-9	1.4-4
Y-92	1.1-12	1.9-8
Y-93	7.4-10	2.5-5
Zr-95	2.5-10	4.1-6
Zr-97	9.4-12	4.7-7
Nb-95m	2.3-12	7.5-7
Nb-95	2.6-10	2.6-6
Nb-97	1.0-11	1.1-8
Mo-99	6.8-8	3.4-4
Tc-99m	6.5-8	1.1-5
Ru-103	6.4-10	8.0-6
Ru-105	4.7-13	4.7-9
Ru-106	1.0-10	1.0-5
Rh-103m	6.4-10	6.4-8
Rh-105	1.5-13	1.5-9
Te-129m	2.0-9	6.6-5
Te-129	2.0-9	2.5-6
Te-131m	5.6-10	9.4-6
Te-132	4.5-8	1.5-3
Cs-134	1.8-9	2.0-4
Cs-136	9.6-10	1.1-5
Cs-137	4.6-9	2.3-4
Ba-140	4.6-8	1.5-3
La-140	3.6-8	1.8-3
La-141	1.8-13	6.0-8
Ce-141	1.5-9	1.6-5
Ce-143	2.0-10	4.9-6
Ce-144	2.1-10	2.1-5

TABLE 15.7-15 (Cont)

Isotope	Final Activity (μCi/cc)	Fraction of Maximum Permissible Concentration*
Pr-143	1.1-9	2.2-5
Nd-147	8.4-11	1.4-6
Pm-147	2.1-13	1.0-9
Na-24	4.9-9	2.4-5
P-32	2.8-9	1.4-4
Cr-51	8.7-8	4.4-5
Mn-54	1.1-9	1.1-5
Mn-56	1.6-15	1.6-11
Fe-55	1.6-8	2.0-5
Fe-59	1.3-9	2.2-5
Co-58	8.5-8	8.5-4
Co-60	8.8-9	1.8-4
Ni-63	1.6-11	5.4-7
Ni-65	7.2-18	7.2-14
Cu-64	8.3-9	2.8-5
Zn-65	3.2-9	3.2-5
Zn-69m	7.0-10	9.9-6
Ag-110m	1.0-9	3.4-5
W-187	5.9-9	8.4-5
Np-239	6.0-7	6.0-3
Total		5.2-1

*Maximum permissible concentrations are from 10CFR20, Appendix B, Table II, Column 2

NOTE: 8.0-17 = 8.0×10^{-17}

TABLE 15.7-16

CONDENSATE STORAGE TANK RUPTURE ACCIDENT RADIOACTIVITY CONCENTRATIONS AT WELL 56 (GROUNDWATER)

Isotope	Final Activity (μCi/cc)	Fraction of Maximum Permissible Concentration*
I-129	1.3-19	2.1-12
Sr-90	4.9-8	1.6-1
Y-90	4.9-8	2.4-3
Ru-106	5.0-13	5.0-8
Cs-134	2.8-10	3.1-5
Cs-137	1.5-8	7.6-4
Ce-144	1.5-13	1.5-8
Pm-147	3.7-13	1.8-9
Mn-54	1.8-12	1.8-8
Fe-55	5.4-9	6.8-6
Co-60	1.0-8	2.0-4
Zn-65	5.2-13	5.2-9
Ag-110m	2.4-13	7.9-9
Ni-63	6.2-11	2.1-6
Total		1.6-1

*Maximum permissible concentrations are from 10CFR20, Appendix B, Table II, Column 2

NOTE: 1.3-19 = 1.3×10^{-19}

THIS TABLE HAS BEEN DELETED

TABLE 15.7-17

FUEL HANDLING ACCIDENT - (INSIDE CONTAINMENT) PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS DURING TYPE C LEAK RATE TESTING

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT

THIS TABLE HAS BEEN DELETED

TABLE 15.7-18

FUEL HANDLING ACCIDENT - (INSIDE CONTAINMENT DURING TYPE C LEAK RATE TESTING) RADIOLOGICAL EFFECTS

THIS TABLE HAS BEEN DELETED

TABLE 15.7-19

FUEL HANDLING ACCIDENT - (INSIDE CONTAINMENT) PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS WITH CONTAINMENT AIR LOCKS OPEN

RIVER BEND STATION
UPDATED SAFETY ANALYSIS REPORT

THIS TABLE HAS BEEN DELETED

TABLE 15.7-20

FUEL HANDLING ACCIDENT - (INSIDE CONTAINMENT WITH CONTAINMENT AIR LOCKS OPEN) RADIOLOGICAL EFFECTS

●→10

Table 15.8-1

Initial Conditions for ATWS Analysis

Parameter	Units	Value
●→14		
Core Power	Mwt (% Rated)	3039
Dome Pressure	psig	1055
Core Flow	Mlbm/hr (% Rated)	68.4 (81)
Steam Flow	Mlbm/hr	13.199
Feedwater Flow	Mlbm/hr	13.169
RPV Water Level	ft above separator skirt	4.2
Core Average Void Fraction	---	47
Feedwater Enthalpy	BTU/lbm	402.6
Initial Void Reactivity Coefficient	cents/%	-11.2
Initial Suppression Pool Temperature	°F	100
●→11 Initial Suppression Pool Volume At Minimum Water Level 11←●	ft^3	135,500

10←●

•→10

Table 15.8-2

ATWS - Equipment Performance Characteristics

Name	Unit	Value
Nominal Closure Time of MSIV	sec	4.0
•→14 Relief Valve System Capacity (Relief mode, all 16 SRVs assumed to be in service)	% Rated Steam Flow/ Number of Valves	101/16
Relief Valve Setpoint Range	psia	1178-1198
Relief Valve and Sensor Time Delay	sec	0.4
Relief Valve Opening Time	sec	0.15
Relief Valve Closure Time Delay	sec	0.3
SLCS Injection Location	---	Standpipe
Sodium Pentaborate Solution Concentration in the Storage Tank	% by weight	9.5
•→12 Nominal Boron 10 Enrichment	atom %	60
Nominal SLCS Boron Injection rate 12←•	gpm	41.2
SLCS Initiation Method	---	Manual
RCIC Flow Rate	gpm	600
RCIC Start/Stop Levels	---	L2/L8
ATWS High Pressure RPT Setpoint, Upper Analytical Limit (UAL)	psig	1180

10←• 14←•

•→10

Table 15.8-2 (Cont)

ATWS - Equipment Performance Characteristics

Name	Unit	Value
•→14 ATWS Dome Pressure Sensor and Logic Time Delay 14←•	sec	0.1
ATWS Low Water RPT Setpoint	---	L2
Recirculation Pump System Inertia Constant	sec	7
RHR Pool Cooling Capacity (each)	BTU/sec	390
Setpoint for Low Water Level Closure of MSIV	---	L1
Setpoint for Low Steamline Pressure Closure of MSIV	psig	860
Service Water Temperature	°F	95

10←• •→14 •→12 12←• 14←•

•→10

Table 15.8-3

Typical Sequence of Events of ATWS Main Steamline Isolation Valve Closure Event

Event	Time (sec)
Main steam line isolation valves begin to close. Control rods do not insert in response to reactor protection system logic.	0
•→14 Main steam line isolation valves closed.	4
Peak Neutron Flux.	4.0
ATWS high pressure setpoint reached.	4.25
Recirculation pumps trip.	4.39
Main steam safety/relief valves begin to lift in relief mode.	4.46
Peak vessel pressure reached.	4.71
Peak Heat Flux.	4.81
Suppression pool temperature reaches 110°F. Operator initiates level reduction and inhibits the automatic depressurization system logic.	29
Operator initiates the standby liquid control system.	124
Boron from Standby liquid control system reaches the reactor core.	230
Operator initiates RHR in the suppression pool cooling mode.	600
Peak suppression pool temperature achieved.	1480
Hot shutdown achieved (neutron flux < 1% of Nuclear Boiler Rated).	1513

10←• 14←•

•→10

Table 15.8-4
Summary of Peak Results[1]

•→14 •→11

Event	Neutron Flux		Surface Heat Flux		Bottom Head Pressure		Fuel cladding Temperature[2]		Suppression Pool Temperature	
11←•	% Rated	@ Time (sec)	% Rated	@ Time (sec)	psig	@ Time (sec)	°F	@ Time (sec)	°F	@ Time (sec)
MSIV Closure	293	4	131	4.8	1293.9	4.7			177.3	1513
GE-11							736	5		
GE-8							1185	35		
Pressure Regulator Failures - Maximum Demand[3]	358	21.9	146	25.3	1295.7	25.2			174.6	1220
GE-11							1586	66		
GE-8							1431	66		
Inadvertent Open Relief Valve	100	0.0	100	0.0	1082.2	0.0			145.2	5380
GE-11							---	---		
GE-8							---	---		

[1] Results are representative of a typical reload cycle.

[2] Cladding temperature only calculated for the MSIV closure and pressure regulator – failure downscale cases as these cases result in the highest surface heat flux.

•→11 10←• 11←• 14←•

[3] The pressure regulator failure-open is no longer a postulated single failure scenario; no Ovation TCPS single failure scenarios are postulated that could result in all valves open. This specific USAR section is maintained for historical purposes.