Appendix 15A. Tables

Table 15-1. Deleted Per 1992 Update

Table 15-2. Summary of Accidents Analyzed With Computer Codes

FSAR Section	Description of Transient	Summary of Cases Analyzed
15.1.2	Increase in Feedwater Flow	1. Full power 2. Zero power
15.1.3	Increase in Steam Flow	1. Manual rod control, most negative moderator coefficient 2. Automatic rod control, most negative moderator coefficient
15.1.4	Accidental Depressurization of Main Steam System	
15.1.5	Steam Line Break	1. Offsite power maintained at hot zero power 2. Offsite power lost at hot zero power 3. CFM at hot full power 4. DNB at hot full power
15.2.3	Turbine Trip	1. Peak RCS pressure 2. Peak Main Steam System pressure
15.2.6	Loss of Offsite Power	
15.2.7	Loss of Normal Feedwater	1. Unit 1 long term core cooling 2. Unit 2 long term core cooling 3. Unit 1 short term core cooling
15.2.8	Feedwater Line Break	1. Long term core cooling 2. Short term core cooling
15.3.1	Partial Loss of Flow	
15.3.2	Complete Loss of Flow	
15.3.3	Locked Rotor	1. Peak RCS pressure 2. Core cooling with offsite power maintained 3. Core cooling with offsite power lost
15.4.1	Zero Power Rod Bank Withdrawal	1. Core cooling 2. Peak RCS pressure

FSAR Section	Description of Transient	Summary of Cases Analyzed
15.4.2	At Power Rod Bank Withdrawal	1. Bank withdrawal from 10% power core cooling 2. Bank withdrawal from 8% power peak RCS pressure 3. Bank withdrawal from 50% power core cooling 4. Bank withdrawal from 98% power core cooling 5. Bank withdrawal from 100% power core cooling
15.4.3	Control Rod Misoperation a. Dropped rod(s) b. Dropped rod bank c. Misaligned rod d. Single rod withdrawal	Deleted Per 2010 Update
15.4.4	Startup of an Inactive Coolant Pump at an Incorrect Temperature	
15.4.7	Misloaded Assembly	1. Region 1 ⟷ Region 3 2. Region 1 ⟷ Region 2 3. Region 2 in center 4. Region 2 in periphery
15.4.8	Rod Ejection	1. BOL, full power 2. BOL, zero power 3. EOL, full power 4. EOL, zero power 5. BOL, full power peak RCS pressure
15.6.1	Accidental RCS Depressurization	
15.6.3	Steam Generator Tube Rupture	1. Thermal hydraulic input to dose analysis 2. Steam generator overfill 3. DNB analysis

FSAR Section	Description of Transient	Summary of Cases Analyzed
15.6.5	Loss of Coolant Accident	1. DECLG C_D=1.0, Reference Transient 2. 1.5 inch SBLOCA (Unit 1) 3. 2 inch SBLOCA (Unit 1) 4. 3 inch SBLOCA (Unit 1) 5. 4 inch SBLOCA (Unit 1) 6. 1.5 inch SBLOCA (Unit 2) 7. 2 inch SBLOCA (Unit 2) 8. 3 inch SBLOCA (Unit 2) 9. 4 inch SBLOCA (Unit 2)

Table 15-3. Summary of Computer Codes and Methodologies Used in Accident Analyses

Computer Code or Methodology	Transient Numbers[1] Analyzed with that Computer Code or Methodology
NOTRUMP	15.6.5
LOCTA-IV	15.6.5
LOTIC	15.6.5
WCOBRA/TRAC	15.6.5
WLOP, W-3S	15.1.5
RETRAN-02	15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.2.3, 15.2.6, 15.2.7, 15.2.8, 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3a, b, d, 15.4.4, 15.4.8, 15.6.1, 15.6.3
VIPRE-01	15.1.2, 15.1.3, 15.1.5, 15.2.7, 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3a, b, d, 15.4.8, 15.6.1, 15.6.3
SCD	15.1.2, 15.1.3, 15.2.7, 15.2.8, 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3a, b, d, 15.6.1, 15.6.3
WRB-2M	15.1.2, 15.2.7 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3a,d, 15.4.8, 15.6.1, 15.6.3
Deleted Per 2006 Update	
BWU	15.1.3
CASMO-4/SIMULATE-3 MOX	15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.2.3, 15.2.6, 15.2.7, 15.2.8, 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3, 15.4.4, 15.4.6, 15.6.1, 15.6.3
Deleted Per 2006 Update	
SIMULATE-3K	15.4.8

Note:

1. Transients are numbered according to the cases listed in Table 15-2.

Table 15-4. Summary of Input Parameters for Accident Analyses Using Computer Codes

FSAR Section	Case Identifier (refer to Table 15-2)	Moderator Temperature Coefficient (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm) Note 19	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (ft^3 or %)	Feedwater Temperature (°F)
15.1.2	1	-51	NA	-1.2	3469	388,000	585.1	2250	64%	443
15.1.2	2	Note 9	NA	-3.5	0	382,000	557	2250	34%	70
15.1.3	1	-51	NA	-1.2	3469	388,000	585.1	2250	64%	443
15.1.3	2	-51	NA	-1.2	3469	388,000	585.1	2250	64%	443
15.1.4		Note 9	NA	-3.5	0	Note 5	561	2208	16%	Note 17
15.1.5	1	Note 9	NA	-3.5	0	371,796	561	2198	16%	Note 17
15.1.5	2	Note 9	NA	-3.5	0	371,796	561	2198	16%	Note 17
15.1.5	3	NA	Note 20	-1.2	3469	390,000	587.5	2250	46%	445
15.1.5	4	NA	Note 21	-1.2	3469	388,000	585.1	2250	46%	443
15.2.3	1	NA	Note 6	-0.9	3479	381,420	591.5	2280	64%	443
15.2.3	2	NA	Note 6	-0.9	3479	420,000	591.5	2310	64%	443
15.2.6		Note 14	NA	Note 14	3479	373,596	594.8	2250	61.5%	440
15.2.7	1	NA	Note 6	-0.9	3479	381,420	589.1	2208	46%	440
15.2.7	2	NA	Note 6	-0.9	3479	376,530	594.8	2208	51%	440
15.2.7	3	NA	Note 6	-0.9	3469	384,000	585.1	2250	46%	443
15.2.8	1	NA	Note 6	-0.9	3479	373,596(U2) Note 7(U1)	594.8(U2) 589.1(U1)	2208	52.5%(U2) 46%(U1)	445(U2) 440(U1)
15.2.8	2	NA	Note 6	-0.9	3469	388,000 (U1) 390,000 (U2)	585.1(U1) 587.5(U2)	2250	46%	443
15.3.1		NA	Note 6	-0.9	3469	388,000	585.1	2250	46%	440
15.3.2		NA	Note 6	-0.9	3469	388,000	585.1	2250	46%	443
15.3.3	1	NA	Note 6	-0.9	3479	375,552	589.1	2310	64%	443

FSAR Section	Case Identifier (refer to Table 15-2)	Moderator Temperature Coefficient (pcm/°F)	Moderator Density Coefficient (%k/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm) Note 19	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (ft^3 or %)	Feedwater Temperature (°F)
15.3.3	2	NA	Note 6	-0.9	3469	388,000	585.1	2250	46%	442
15.3.3	3	NA	Note 6	-0.9	3469	388,000	585.1	2250	46%	442
15.4.1	1	NA	Note 6	Note 4	0	299,613	557	2250	16%	NA
15.4.1	2	NA	Note 6	Note 4	0	371,796	557	2310	34%	NA
15.4.2	1	NA	Note 6	Note 4	347	384,120	559.8	2250	19%	335.7
15.4.2	2	NA	Note 6	Note 4	273	375,669	563.8	2250	37%	333.0
15.4.2	3	NA	Note 6	Note 4	1734.5	384,120	571	2250	31%	382.3
15.4.2	4	NA	Note 6	Note 4	3399.6	384,120	584.5	2250	45.4%	438.3
15.4.2	5	NA	Note 6	Note 4	3469	388,000	585.1	2250	46%	440.6
15.4.3a, b		NA	Note 6	-0.9	3469	384,000	585.1	2250	46%	443
15.4.3c		NA	NA	NA	3469	388,000	590.8	2250	NA	NA
15.4.3d		NA	Note 6	Note 4	3469	388,000	585.1	2250	46%	440
15.4.4		-51	NA	-1.2	1735	272,747	574.8	2208	30.4%	372
15.4.7	1	NA	NA	NA	3493	NA	NA	NA	NA	NA
15.4.7	2	NA	NA	NA	3493	NA	NA	NA	NA	NA
15.4.7	3	NA	NA	NA	3493	NA	NA	NA	NA	NA
15.4.7	4	NA	NA	NA	3493	NA	NA	NA	NA	NA
15.4.8	1	Note 10	Note 10	Note 10	3479	371,796	589.1	2203	46%	NA
15.4.8	2	Note 10	Note 10	Note 10	68	290,000	561	2203	16%	NA
15.4.8	3	Note 10	Note 10	Note 10	3479	371,796	589.1	2203	46%	NA
15.4.8	4	Note 10	Note 10	Note 10	68	290,000	561	2203	16%	NA

FSAR Section	Case Identifier (refer to Table 15-2)	Moderator Temperature Coefficient (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm) Note 19	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (ft^3 or %)	Feedwater Temperature (°F)
15.4.8	5	Note 10	Note 10	Note 10	3479	371,796	589.1	2310	64%	443
15.6.1		0.0	NA	-0.9	3469	388,000	587.5	2250	46%	445
15.6.3	1	Note 14	NA	Note 14	3479	373,599	581.1	2310	64%	440
15.6.3	2	Note 14	NA	Note 14	3479	Note 12	571.1	2310	53.3%	440
15.6.3	3	NA	Note 8	-0.90	3469	388,000	585.1	2250	46%	442
15.6.5	1	NA	Note 11	Note 11	3445[18]	Note 16	587.5	2250	55%	442
15.6.5 (Unit 1)	2	NA	Note 11	Note 11	3479	Note 16	585.1	2250	55%	442
15.6.5 (Unit 1)	3	NA	Note 11	Note 11	3479	Note 16	585.1	2250	55%	442
15.6.5 (Unit 1)	4	NA	Note 11	Note 11	3479	Note 16	585.1	2250	55%	442
15.6.5 (Unit 1)	5	NA	Note 11	Note 11	3479	Note 16	585.1	2250	55%	442
15.6.5 (Unit 2)	6	NA	Note 11	Note 11	3479	390,000	587.5	2250	55%	442
15.6.5 (Unit 2)	7	NA	Note 11	Note 11	3479	390,000	587.5	2250	55%	442
15.6.5 (Unit 2)	8	NA	Note 11	Note 11	3479	390,000	587.5	2250	55%	442
15.6.5 (Unit 2)	9	NA	Note 11	Note 11	3479	390,000	587.5	2250	55%	442

Notes:

1. The assumed feedwater temperature varies inversely proportional to the overfeed percentage.
2. -0.9 pcm/°F at HFP to -1.20 pcm/°F at HZP

FSAR Section	Case Identifier (refer to Table 15-2)	Moderator Temperature Coefficient (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm) Note 19	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (ft^3 or %)	Feedwater Temperature (°F)

3. -1.04 pcm/°F at HFP to -1.325 pcm/°F at HZP.
4. -1.2 pcm/°F at HFP to -1.5 pcm/°F at HZP.
5. An RCS flow of 390,000 gpm x 0.99 - 2.2% is assumed. The analysis results are always bounded by results in Section 15.1.5. Therefore, the analysis was not re-analyzed with 388,000 gpm flow.
6. The most positive MTC (most negative MDC) allowed by the Technical Specifications was used.
7. An RCS flow of 390,000 gpm - 2.2% is assumed. The analysis was evaluated and the reduced flow has negligible impact on the analysis.
8. The Catawba Technical Specification limit for the moderator temperature coefficient (MTC) is based on a +7 pcm/°F MTC from 0 to 70% of nominal power, ramping to 0 pcm/°F at full power. Sensitivity studies have shown that a 0 pcm/°F MTC at a full power condition conservatively bounds the combinations of power and MTC permitted by the Technical Specifications.
9. Refer to Figure 15-17.
10. Refer to Section 15.4.8.2.2.
11. The moderator density and Doppler effects on reactivity during LOCA transients are accounted for in the evaluation models as described in Section 15.6.5 and the associated references.
12. An RCS flow of 390,000 gpm is assumed. The results of this transient are not sensitive to RCS flow.
13. Deleted
14. The results of this transient are not sensitive to reactivity feedback assumptions.
15. Deleted Per 2013 Update.
16. An RCS flow of 390,000 gpm is assumed. An evaluation of a change to 388,000 gpm concluded that there would be no impact on meeting the relevant acceptance criteria due to the reduced RCS flow.
17. Main feedwater temperature is 60ºF. Auxiliary feedwater temperature is 32ºF.
18. Analysis was originally performed at 3445 MWt (3411 MWt plus 1% for conservatism). However, 1% for heat balance error was also added into the analysis, so it remains bounding for the MUR (3479 MWt). An MUR uprate evaluation was performed at 3469 MWt (101.7% of 3411 MWt) plus 0.3% uncertainty to derive the PCT penalty included in Table 15-82.
19. Evaluations of all accidents for RCS Flows of 384,000 gpm (U1) and 387,000 gpm (U2) concluded that there would be no impact on meeting the relevant acceptance criteria due to the reduced RCS flow.

FSAR Section	Case Identifier (refer to Table 15-2)	Moderator Temperature Coefficient (pcm/°F)	Moderator Density Coefficient (%□k/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm) Note 19	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (ft^3 or %)	Feedwater Temperature (°F)

20. Based on MTC = -17 pcm/°F

21. Based on MTC = - 24 pcm/°F

Table 15-5. Deleted Per 1992 Update

Table 15-6. Rod Drop Times Used in FSAR Analyses

FSAR Section	Drop Time to Dashpot (sec)
15.1.2	2.2
15.1.3	Note 2
15.1.4	Instantaneous
15.1.5	Instantaneous for hot zero power 2.2 for hot full power
15.2.3	2.2
15.2.6	2.2
15.2.7	2.2
15.2.8	2.2
15.3.1	2.2
15.3.2	2.2
15.3.3	2.2
15.4.1	2.2
15.4.2	2.2
15.4.3	2.2
15.4.4	Note 2
15.4.7	Note 2
15.4.8	2.2
15.6.1	2.2
15.6.2	Note 2
15.6.3	2.2,(1)
15.6.5(small break)	2.2
15.7 (all sections)	Note 2

Notes:

1. Results of transient are not sensitive to rod drop time for the dose and overfill analyses.
2. Reactor trip was not necessary to analyze transient.

Table 15-7. Trip Points and Time Delays to Trip Assumed in Accident Analyses

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delays (Seconds)
Power range high neutron flux, high setting	Note 2	0.5
Power range high neutron flux, low setting	116.1%	0.5
Overtemperature ΔT	Variable see Figure 15-1	1.5
Overpower ΔT	Variable see Figure 15-1	1.5
High pressurizer pressure	Note 2	2.0
Low pressurizer pressure	Note 2	2.0
Low reactor coolant flow (from loop flow detectors)	83.5% loop flow	1.0
Undervoltage trip	Note 3	1.5
Low-low steam generator level	Note 2	2.0
Safety injection	Not applicable	2.0

Notes:

1. Time delay from the indicated parameter satisfying the trip condition until the beginning of rod motion. The delays due to RTD response (ΔT trips only) and electronic signal filtering are accounted for by explicit modeling.
2. The numerical setpoint assumed for this trip function varies depending on the accident being analyzed. The values used are given in the descriptions of the various accidents.
3. A value for this trip setpoint is not explicitly modeled. However, an actual trip setpoint of less than 68% of nominal bus voltage, adjusted for uncertainty and margin, may invalidate the delay time to trip assumed in the analysis.

Table 15-8. Deleted Per 1992 Update

Table 15-9. Deleted Per 1992 Update

Table 15-10. Deleted Per 1992 Updaate

Table 15-11. Deleted Per 1992 Update

Table 15-12. Fission Product Radioactivity Levels in the Reactor Core*

Noble Gases		Tellurium Group	
Radioisotope	Core Activity (Curies)	Radioisotope	Core Activity (Curies)
Kr83m	1.56E+07	Se81	5.84E+06
Kr85m	3.40E+07	Se83	7.04E+06
Kr85	1.07E+06	Se83m	7.97E+06
Kr87	6.96E+07	Se84	2.91E+07
Kr88	9.79E+07	Se87	2.11E+07
Kr89	1.25E+08	Sb127	9.65E+06
Xe131m	1.43E+06	Sb128	1.65E+06
Xe133m	6.72E+06	Sb128m	1.59E+07
Xe133	2.08E+08	Sb129	3.43E+07
Xe135m	4.51E+07	Sb130	1.15E+07
Xe135	6.65E+07	Sb130m	4.76E+07
Xe137	1.98E+08	Sb131	8.38E+07
Xe138	1.98E+08	Sb132m	4.95E+07
		Te127m	1.58E+06
Halogens		Te127	9.51E+06
	Core Activity	Te129	3.27E+07
Radioisotope	**(Curies)**	Te129m	6.63E+06
Br83	1.55E+07	Te131	8.69E+07
Br85	3.41E+07	Te132	1.49E+08
Br87	5.56E+07	Te133	1.22E+08
I130	2.96E+06	Te133m	1.01E+08
I131	1.04E+08	Te134	2.12E+08
I132	1.52E+08		
I133	2.15E+08		
I134	2.47E+08		
I135	2.06E+08		

Noble Gases Alkali Metals		Tellurium Group Alkali Earth Metals	
	Core Activity		Core Activity
Radioisotope	(Curies)	Radioisotope	(Curies)
Rb86	2.08E+05	Sr89	1.03E+08
Rb88	1.00E+08	Sr90	9.31E+06
Rb89	1.33E+08	Sr91	1.66E+08
Rb90	1.25E+08	Sr92	1.69E+08
Cs134	2.09E+07	Sr93	1.83E+08
Cs136	5.60E+06	Ba139	2.00E+08
Cs137	1.26E+07	Ba140	1.88E+08
Cs138	2.09E+08	Ba141	1.82E+08
Cs139	1.96E+08	Ba142	1.78E+08

Noble Metals		Lanthanides	
	Core Activity		Core Activity
Radioisotope	**(Curies)**	**Radioisotope**	**(Curies)**
Mo99	1.97E+08	Y90	9.66E+06
Mo101	1.76E+08	Y91	1.34E+08
Mo102	1.70E+08	Y91m	9.72E+07
Tc99m	1.74E+08	Y92	1.51E+08
Tc101	1.76E+08	Y93	1.23E+08
Tc104	1.46E+08	Y94	1.90E+08
Ru103	1.72E+08	Y95	1.94E+08
Ru105	1.25E+08	Zr95	1.78E+08
Ru106	6.37E+07	Zr97	1.78E+08
Ru107	7.57E+07	Nb95	1.79E+08
Rh103m	1.72E+08	Nb95m	1.98E+06
Rh105	1.12E+08	Nb97	1.78E+08
Rh106m	4.02E+06	La140	1.98E+08
Rh107	7.59E+07	La141	1.81E+08
Pd109	4.67E+07	La142	1.82E+08
Pd111	7.22E+06	La143	1.79E+08
Pd112	3.24E+06	Nd147	6.93E+07
		Nd149	4.04E+07
Cerium Group		Nd151	2.16E+07
	Core Activity	Pm147	1.75E+07
Radioisotope	**(Curies)**	Pm148	1.88E+07
Ce141	1.73E+08	Pm148m	2.96E+06
Ce143	1.79E+08	Pm149	6.64E+07
Ce144	1.32E+08	Pm151	2.18E+07
Ce145	1.21E+08	Sm153	5.73E+07
Ce146	9.35E+07	Sm156	2.75E+06
Np237	4.23E+01	Eu154	9.87E+05
Np238	5.04E+07	Eu155	3.86E+05
Np239	2.32E+09	Eu156	3.17E+07

Cerium Group			**Core Activity**
	Core Activity	**Radioisotope**	**(Curies)**
Radioisotope	**(Curies)**	Eu157	3.66E+06
Np240	7.00+06	Pr142	7.74E+06
Pu236	7.32E+01	Pr143	1.56E+08
Pu238	4.29E+05	Pr144	1.33E+08
Pu239	3.74E+04	Pr144m	1.86E+06
Pu240	5.16E+04	Pr145	1.21E+08
Pu241	1.45E+07	Pr146	9.40E+07
Pu242	2.97E+02	Pr147	7.22E+07
Pu243	5.62E+07	Am241	1.75E+04
		Am242m	1.14E+03
		Am242	8.95E+06
		Am243	4.41E+03
		Am244	2.45E+07
		Cm242	5.13E+06
		Cm244	9.41E+05

Table 15-13. Reactor Coolant Specific Activities for Iodine and Noble Gas Isotopes

Nuclide	Specific Activity[1] (μCi/g)
I-131	2.5
I-132	0.9
I-133	4.0
I-134	0.6
I-135	2.2
Xe-131m	2.3
Xe-133m	17.5
Xe-133	278.2
Xe-135m	0.49
Xe-135	7.4
Xe-138	0.66
Kr-83m	0.47
Kr-85m	2.1
Kr-85	7.5
Kr-87	1.3
Kr-88	3.7
Kr-89	0.0

Note:

1. Reactor coolant concentrations at equilibrium assuming 1 percent failed fuel.

Table 15-14. Total Effective Dose Equivalents (TEDEs - Rem) Following Design Basis Events

Design Basis Event	UFSAR Section	Exclusion Area Boundary	Low Population Zone Boundary	Control Room
Main Steam Line Break	15.1.5.3			
Pre-existent iodine spike		0.29	0.07	1.76
Accident initiated iodine spike		0.33	0.26	3.01
Locked Rotor Accident	15.3.3.3	1.63	0.37	1.57
Rod Ejection Accident	15.4.8.3	4.75	2.82	2.70
Instrument Line Break	15.6.2			
Pre-existent iodine spike		0.60	0.09	0.80
Accident initiated iodine spike		0.17	0.02	0.18
Steam Generator Tube Rupture (1)	15.6.3.3			
Pre-existent iodine spike		1.19	0.28	1.32
Accident initiated iodine spike		0.61	0.19	0.81
Loss of Coolant Accident	15.6.5.3	8.79	3.78	3.31
Waste Gas Tank Rupture	15.7.1	0.27	0.04	0.10
Liquid Storage Tank Rupture	15.7.2	0.99	0.14	1.00
Fuel Handling Accident (3,4)	15.7.4	1.76		2.59
Weir Gate Drop (4)	15.7.4	2.68		4.24
Fuel Cask Drop (4)	15.7.5	0.006		0.001

Notes:

1) A supplemental analysis of the steam generator tube rupture is reported in Section 15.6.3.3. It is the basis for a set of conditions cited in Facility Operating License Amendment 159/151. The license conditions set limits of 0.46 µCi/gm Dose Equivalent Iodine-131 (DEI) for equilibrium iodine specific activity in the reactor coolant and 26 DEI for transient iodine specific activity in the reactor coolant. Thyroid radiation doses are calculated in this supplemental analysis and reported in Table 15-74.

2) This footnote is reserved for future use.
3) Section 15.7.4 reports the analysis of fuel handling accident in containment and a fuel handling accident in the fuel building. The radiation doses calculated for offsite locations and in the control room take the same values.
4) TEDEs at the boundary of the Low Population Zone (denoted as the LPZ) are not listed for the fuel handling accidents, weir gate drop, or fuel cask drop. The regulatory acceptance criteria in 10 CFR 50.67 and Regulatory Guide 1.183 for radiation doses at the Exclusion Area Boundary and LPZ are the same. In addition, for the fuel handling accidents and weir gate drop, the TEDEs at the LPZ are significantly lower than the TEDEs at the Exclusion Area Boundary.

Table 15-15. Time Sequence of Events for Incidents Which Cause an Increase In Heat Removal By The Secondary System

Accident	Event	Time (sec.)
Excessive Feedwater Flow at Full Power	All main feedwater control valves fail fully open	0
	Overpower ΔT setpoint reached	53.2
	Reactor trip occurs due to overpower ΔT	54.7
	Turbine trip occurs due to reactor trip	54.9
	Minimum DNBR occurs	55.0
Excessive Increase in Secondary Steam Flow		
Manual Reactor Control	10% step load increase	0
	Equilibrium conditions reached (approximate time only)	260
Inadvertent Opening of a Steam Generator Relief or Safety Valve	Inadvertent opening of one main steam safety valve	0
	Pressurizer empties	102
	Low pressurizer pressure setpoint reached	211
	Return to criticality	254
	Borated water reaches core	329
	Subcriticality achieved	418
Steam System Piping Failure		
1. With offsite power maintained at hot zero power	Break occurs	0
	Operator manually trips reactor	0
	Pressurizer level goes offscale low	12
	SI actuation on low pressurizer pressure	21
	Criticality occurs	22
	Steam line isolation on low steam line pressure	24
	Main feedwater flow ceases	33
	SI pumps begin to deliver unborated water to RCS	38
	Peak heat flux occurs	119
	NV injection lines purged of unborated water	119
	One train of SI fails	119

Accident	Event	Time (sec.)
	Subcriticality achieved	166
	Pressurizer level returns onscale	>200
2. With offsite power lost at hot zero power	Break occurs	0
	Operator manually trips reactor	0
	Pressurizer level goes offscale low	12
	SI actuation on low pressurizer pressure	21
	Offsite power lost	21
	Reactor coolant pumps begin to coast down	21
	Main feedwater pumps trip	21
	Criticality occurs	22
	Steam line isolation on low steam line pressure	24
	Main feedwater flow ceases	32
	SI pumps begin to deliver unborated water to RCS	53
	NV injection lines purged of unborated water	137
	One train of SI fails	137
	Pressurizer level returns onscale	182
	Peak heat flux occurs	224
	Subcriticality achieved	242
3. CFM at hot full power	Break occurs	0
	High flux trip setpoint reached	12.8
	Reactor trip occurs due to hight flux trip	13.3
	Peak reactor power occurs	13.5
	Turbine trip occurs due to reactor trip	13.6
	Loss of offsite power occurs on turbine trip	13.6
	RCPs trip due to loss of offsite power	13.6
4. DNB at hot full power	Break occurs	0
	OPΔT trip setpoint reached	11.6
	Reactor trip occurs due to OPΔT trip	12.1
	Peak reactor power occurs	12.3

Accident	Event	Time (sec.)
	Turbine trip occurs due to reactor trip	12.4
	Loss of offsite power occurs on trubine trip	12.4
	RCPs trip due to loss of offsite power	12.4
	MDNBR occurs	13.2

Table 15-16. Deleted Per 1992 Update

Table 15-17. Parameters for Postulated Main Steam Line Break Offsite Dose Analysis

1.	Data and assumptions pertaining to the radioactive source term	
a.	Equilibrium reactor coolant DEX specific activity (μCi/gm - Note 1)	670
b.	Equilibrium reactor coolant DEI specific activity - concurrent iodine spike (μCi/gm - Note 2)	1
c.	Concurrent iodine spike multiplier for the equilibrium reactor coolant iodine appearance rate	500
d.	Maximum reactor coolant DEI specific activity-pre-existent iodine spike (μCi/gm - Note 2)	60
e.	Steam generator secondary side DEI specific activity (μCi/gm)	0.1
f.	Main condenser iodine scrubbing efficiency (%)	100
g.	Iodine composition fractions (%)	
	Diatomic iodine	97
	Organic iodine compounds	3
2.	Data and assumptions pertaining to transport and release of radioactivity	
a.	Offsite power available?	No
b.	Initial steam release from the faulted steam generator (lbm)	
	Unit 1 (0-600 sec)	234,000
	Unit 2 (0-400 sec)	165,000
c.	Primary-to secondary leak rate (gpd per SG)	150
d.	Limiting time spans for tube bundle uncovery in the intact steam generators (hr)	
	(Unit 1)	0-2.5
	(Unit 2)	0-2.5
e.	Iodine partition fraction for steam releases	0.01
f.	Integrated steam release from the intact steam generators (lbm, 0-5.5 hr)	
	Unit 1	
	0- 2 hours	8.35 E+05
	0-5.5 hours	1.85E+06
	Unit 2	
	0- 2 hours	8.22E+05
	0- 5.5 hours	1.81E+06
3.	Dispersion data	
a.	χ/Q at exclusion area boundary (sec/m^3)	4.78×10^{-4}

	b.	χ/Q at low population zone (sec/m^3)	
		0- 8 hr	$6.85x10^{-5}$
		8- 24 hr	$4.00x10^{-5}$
		24- 96 hr	$2.00x10^{-5}$
		96-720 hr	$7.35x10^{-6}$
4.	Dose conversion data		
	a.	Source of dose coefficients	
		Deep dose equivalent coefficent	FGR 12
		Committed effective dose equivalent coefficient	FGR 11
5.		Total Effective Dose Equivalents (TEDEs, Rem)	Table 15-14

Notes:

1) DEX denotes Dose Equivalent Xenon-133. The isotpic activities corresponding to this value are presented in Table 15-83.

2) DEI denotes Dose Equivalent Iodine-131. The isotopic activities corresponding to this value are presented in Table 15-84.

Table 15-18. Time Sequence Of Events For Incidents Which Cause A Decrease In Heat Removal By The Secondary System

Accident	Event	Time (Sec)
Turbine Trip		
1. Maximum Secondary System Pressure Case	Turbine Trip, loss of main feed flow	0.0
	Pressurizer PORVs lift	3.3
	Steam Safety Valves lift	6.3
	Overtemperature ΔT setpoint reached	12.9
	Control rod insertion begins	14.4
	Peak secondary system pressure occurs	14.8
2. Maximum Primary System Pressure Case	Turbine Trip, loss of main feed flow	0.0
	High pressurizer pressure setpoint reached	4.8
	Control rod insertion begins	6.8
	Pressurizer Safety Valves lift	7.0
	Peak primary system pressure occurs	7.5
	Steam Safety Valves lift	8.1
3. Deleted		
4. Deleted		
Loss of Non-Emergency AC Power	Main feedwater flow stops	0.1
	Power lost to control rod gripper coils	0.1
	Reactor coolant pumps begin to coast down	0.1
	Rods begin to drop	0.6
	Peak water level in pressurizer occurs	4
	Flow from two motor driven auxiliary feedwater pumps is started	60
	Feedwater lines are purged and cold auxiliary feedwater is delivered to four steam generators	195
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~ 1200
Loss of Normal Feedwater Flow		
1. Unit 1 Long-Term Core Cooling Case	Main feedwater flow stops	1.0
	Pressurizer PORVs begin cycling	40.2

Accident	Event	Time (Sec)
	Low-Low steam generator level reactor trip reached	57.7
	Rods begin to drop	59.7
	Steam safety valves lift	61.1
	Auxiliary feedwater flow on	117.7
	Core decay heat plus pump heat decreases to auxililary feedwater heat removal capacity	~1290
2. Unit 2 Long-Term Core Cooling Case	Main feedwater flow stops	0.0
	Pressurizer sprays on	10.69
	Pressurizer PORVs begin cycling	27.92
	Low-low steam generator level reactor trip reached	44.89
	Rods begin to drop	46.89
	Steam safety valves lift	49.47
	Pressurizer sprays off	50.80
	Auxiliary feedwater flow on	104.89
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~230
3. Unit 1 Short-Term Core Cooling Case	Main feedwater flow stops	0.0
	PORVs begin cycling	24.6
	Low-low steam generator level reactor trip reached	57.0
	Minimum DNBR occurs	58.9
	Rods begin to drop	59.0
Feedwater System Pipe Break (Unit 2 Analysis)		
	Feedwater line break to SG B	0
	Safety injection on high containment pressure	10.10
	Reactor trip on high containment pressure SI	10.10
	Turbine trip on reactor trip	10.3
	Steam line isolation on hi-hi containment pressure	15

Accident	Event	Time (Sec)
	Reactor coolant pumps tripped	15
	Safety injection terminated	70
	Motor-driven auxiliary feedwater pump delivers flow	75.3
	Auxiliary feedwater to faulted generator isolated	120
	SG B boiled dry	130
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~1655
	End of simulation	2000
Feedwater System Pipe Break (Unit 1 Analysis)	Feedwater line break to SG B	0
	Safety injection on high containment pressure	10.05
	Reactor trip on high containment pressure SI	10.05
	Reactor coolant pumps tripped	10.05
	Turbine trip on reactor trip	10.2
	Steam line isolation on turbine trip	10.2
	Safety injection terminated	70
	Motor-driven auxiliary feedwater pumps deliver flow	70
	SG-B boiled dry	100
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1750
	Auxiliary feedwater to faulted generator isolated	1800
	End of simulation	3000

Table 15-19. Deleted Per 2010 Update

Table 15-20. Time Sequence of Events for Incidents Which Cause a Decrease in Reactor Coolant System Flow

Accident	Event	Time (Sec)
Partial Loss of Forced Reactor Coolant Flow	Coastdown begins	0.0
	Low flow reactor trip setpoint reached	1.47
	Rods begin to drop	2.47
	Minimum DNBR occurs	3.3
Complete Loss of Forced Reactor Coolant Flow	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump undervoltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.4
Reactor Coolant Pump Shaft Seizure (Core Cooling Capability for Offsite Power Maintained)	Rotor on one pump locks	1.0
	Low flow reactor trip setpoint reached	1.08
	Rods begin to drop	2.08
	Minimum DNBR occurs	3.5
Reactor Coolant Pump Shaft Seizure (Core Cooling Capability for Offsite Power Lost) (U1)	Rotor on one pump locks	1.0
	Low flow reactor trip setpoint reached	1.08
	Rods begin to drop	2.08
	Minimum DNBR occurs	3.9
Reactor Coolant Pump Shaft Seizure (Core Cooling Capability for Offsite Power Lost) (U2)	Rotor on one pump locks	0.0
	Low flow reactor trip setpoint reached	0.04
	Rods begin to drop	1.04
	Minimum DNBR occurs	2.9
Reactor Coolant Pump Shaft Seizure (Peak RCS Pressure)	Rotor on one pump locks	0.0
	Low flow reactor trip setpoint reached	0.07
	Rods begin to drop	1.07
	Maximum RCS pressure occurs	4.7

Table 15-21. Deleted Per 1992 Update

Table 15-22. Parameters for the Postulated Locked Rotor Accident

1.	Data pertaining to the radioactive source term	
	a. Percent fuel with clad failure (%)	
	Deleted Per 2006 Update	
	Deleted Per 2006 Update	
	Offsite power available	1
	Offsite power lost	5.4
	b. Offsite power available?	No
	c. Core isotopic inventory	Table 15-73
	d. Iodine composition fraction (%)	
	Elemental iodine	97
	Particulate iodine	0
	Organic iodine compounds	3
	e. Fission product gap fraction (%)	Section 15.0
2.	Data and assumptions pertaining to transport and release of radioactivity	
	a. Fraction of gap inventory released to the reactor coolant (%)	100
	b. Initial primary-to-secondary leak rate (gpd per SG)	150
	c. Time spans of SG tube bundle uncovery (seconds)	
	Offsite power available	
	S/G 1	895
	S/G 2	46
	S/Gs 3 & 4 a piece	9087
	Total for Unit 1	19,115
	Offsite power lost	
	S/G 1	617
	S/G 2	0
	S/Gs 3 & 4 a piece	3073
	Total for Unit 2	6763
	d. SG iodine partition fraction	0.01
	e. Duration of post accident unit cooldown (hours)	6
	f. Total steam released (lbm)	
	Unit 1	
	0-2 hours	748,330

	2-6 hours	989,069
	Unit 2	
	0-2 hours	736,434
	2-6 hours	966,906
g.	Power level (MWt)	3479.
3.	Dispersion data	
a.	χ/Q at exclusion area boundary (sec/m^3)	4.78×10^{-4}
b.	χ/Q at low population zone (sec/m^3)	
	0 - 8 hr	6.85×10^{-5}
	8 - 24 hr	4.00×10^{-5}
	24 - 96 hr	2.00×10^{-5}
	96 - 120 hr	7.35×10^{-6}
c.	χ/Q for the control room (sec/m^3, Note 1)	
	Releases from the outboard SG doghouse vents	
	0 - 2 hr	7.14×10^{-3}
	2 - 8 hr	4.05×10^{-3}
	8 - 10 hr	2.24×10^{-3}
	10 - 24 hr	1.81×10^{-3}
	24 - 96 hr	1.24×10^{-3}
	96 - 720 hr	7.26×10^{-4}
	Releases from the inboard SG doghouse steam vents	
	0 - 2 hr	2.27×10^{-3}
	2 - 8 hr	1.74×10^{-3}
	8 - 10 hr	1.02×10^{-3}
	10 - 24 hr	8.70×10^{-4}
	24 - 96 hr	7.14×10^{-4}
	96 - 720 hr	5.74×10^{-4}
4.	Dose conversion data	
a.	Method of conversion from activity to dose	R.G. 1.183
b.	Source of dose conversion factors	
	Whole body radiation dose coefficient	FGR 12
	Origin radiation dose coefficient	FGR 11

5.	Data pertaining to the control room and Control Room Area Ventilation System	Table 15-41
6.	Total effective dose equivalents (Rem)	Table 15-14

Note on Table 15-22

1) The values for the control *X/Q*s shown here correspond to both control room outside air intakes being open during the event.

Table 15-23. Time Sequence of Events for Incidents which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (Core Cooling Capability)	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	11.2
	Peak nuclear power occurs	11.3
	Rods begin to fall into core	11.7
	Peak heat flux occurs	12.0
	Minimum DNBR occurs	12.0
	Peak average fuel temperature occurs	12.2
Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (Peak RCS Pressure)	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	11.2
	Peak nuclear power occurs	11.3
	Rods begin to fall into core	11.7
	Peak RCS Pressure	13.9
Uncontrolled RCCA Bank Withdrawal at Power (Core Cooling Capability)	Initiate Bank Withdrawal	0.0
	Pressurizer Sprays Full On	7.3
	Pressurizer PORVs Full Open	24.4
	High Flux Trip Setpoint Reached	42.6
	Pressurizer Safety Valve Lifts	42.9
	Control Rod Insertion Begins	43.1
Uncontrolled RCCA Bank Withdrawal at Power (Peak RCS Pressure)	Initiate Bank Withdrawal	0.0
	High Pressure Reactor Trip Setpoint Reached	12.3
	Pressurizer Safety Valves Lift	14.0
	Control Rod Insertion Begins	14.3
	Peak Pressure Occurs	14.8
Single RCCA Withdrawal	Initiate RCCA Withdrawal	0.0
	Pressurizer Sprays Full On	2.2

Accident	Event	Time (sec.)
	RCCA Completely Withdrawn	4.2
	OTΔT Reactor Trip Setpoint Reached	39.2
	Control Rod Insertion Begins	40.7
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	Initiation of pump startup	0.1
	Pump reaches full speed	10.1
	Peak heat flux occurs	15.5
CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant		
1.		
a) Dilution during power operation (manual rod control)	Dilution begins	-
	Reactor trip setpoint reached	0
	Operator terminates dilution	<1034
b) Dilution during power operation (automatic rod control)	Dilution begins	-
	Rod insertion limit alarm setpoint reached	0
	Operator terminates dilution	<1611
2. Dilution during startup	Dilution begins	-
	Reactor trip setpoint reached	0
	Operator terminates dilution	<1034
3.		
a) Dilution during hot standby (BDMS operable)	Dilution begins	0
	BDMS setpoint reached	2355
	Dilution source isolated	2380
	Borated water reaches core	2657
b) Dilution during hot standby (BDMS inoperable)	Dilution begins	0
	High-flux-at-shutdown alarm setpoint reached	5771
	Operator terminates dilution	6671
4.		
a) Dilution during hot shutdown without an RCP in operation (BDMS operable)	Dilution begins	0
	BDMS setpoint reached	2021
	Dilution source isolated	2046

Accident	Event	Time (sec.)
	Borated water reaches core	<2262
b) Dilution during hot shutdown without an RCP in operation (BDMS inoperable)	Dilution begins	0
	High-flux-at-shutdown alarm setpoint reached	5594
	Operator terminates dilution	<6494
c) Dilution during cold shutdown with an RCP in operation (BDMS operable)	Dilution begins	0
	BDMS setpoint reached	2134
	Dilution source isolated	2159
	Borated water reaches core	<2397
d) Dilution during cold shutdown witth an RCP in operation (BDMS inoperable)	Dilution begins	0
	High-flux-at-shutdown alarm setpoint reached	5923
	Operator terminates dilution	<6823
5.		
a) Dilution during cold shutdown without an RCP in operation (BDMS operable)	Dilution begins	0
	BDMS setpoint reached	2029
	Dilution source isolated	2054
	Borated water reaches core	2278
b) Dilution during cold shutdown without an RCP in operation (BDMS inoperable)	Dilution begins	0
	High-flux-at-shutdown alarm setpoint reached	5634
	Operator terminates dilution	<6534
c) Dilution during cold shutdown with an RCP in operation (BDMS operable)	Dilution begins	0
	BDMS setpoint reached	2143
	Dilution source isolated	2168
	Borated water reaches core	<2418
d) Dilution during cold shutdown witth an RCP in operation (BDMS inoperable)	Dilution begins	0
	High-flux-at-shutdown alarm setpoint reached	5969
	Operator terminates dilution	<6869
Rod Cluster Control Assembly Ejection		
1. Beginning of Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux high setpoint reached	0.056

Accident	Event	Time (sec.)
	Peak nuclear power occurs	0.083
	Rods begin to fall into core	0.556
2. End of Cycle, Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.272
	Peak nuclear power occurs	0.323
	Rods begin to fall into core	0.772

Table 15-24. Deleted Per 1992 Update

Table 15-25. Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident

Time in Cycle	Beginning	Beginning	End	End
Power Level, %	102	0	102	0
Ejected rod worth, $	0.19	1.32	0.26	1.45
Delayed neutron fraction, %	0.56	0.56	0.47	0.47
F_q after rod ejection	4.75	19.60	4.84	20.78
Number of operational pumps	4	3	4	3
Max. fuel pellet average temperature, °F	3313	1626	2850	1316
Max. fuel center temperature, °F	5021	2064	4353	1646
Max. clad temperature, °F	794	738	1296	755
Max. fuel stored energy, cal/gm	146	61	133	48
% Failed fuel	<50	<50	<50	<50

Table 15-26. Parameters for the Postulated Rod Ejection Accident

1.	Data pertaining to the radioactive sources term	
a.	Percent fuel with clad failure	40
b.	Percent fuel melted	0
c.	Core isotopic inventory	Table 15-73
d.	Iodine composition fraction (%)	
	Elemental iodine	97
	Particulate iodine	0
	Organic iodine compounds	3
e.	Fission product gap fraction (%)	
	Alkali metals	12
	All other fission products	10
f.	Fractions for source term release to be containment (%)	
	Noble gas containment release fraction	100
	Iodine containment release fraction	100
g.	Fractions for source term release to the containment sump (%)	
	Noble gas containment release fraction	0
	Iodine containment release fraction	100
h.	Fractions for source term release to the reactor coolant (%)	
	Noble gas containment release fraction	100
	Iodine containment release fraction	100
2.	Data and assumptions pertaining to transport and release of radioactivity	
a.	Containment volume (cu. ft. – Note 1)	1,016,454
b.	Containment compartment volume (cu. ft.)	
	Lower compartment (Note 2)	346,895
	Upper compartment	669,559
c.	Containment leak rate (% air mass per day)	
	0 hr $\leq$t $\leq$24 hr	0.3
	t > 24 hr	0.15
d.	Bypass leakage fraction (% containment leak rate)	7
e.	Annulus volume (cu. ft. – Note 3)	484,090
f.	Annulus ventilation total airflow rate (cfm))	8,100

g.	Annulus ventilation response time (sec)	23
h.	Annulus ventilation filter efficiencies and bypass fraction (%)	
	Elemental iodine	92
	Particulate iodine forms	95
	Organic iodine compounds	92
	Bypass airflow fraction	1
i.	Characteristics of natural deposition of aerosols in containment	
	Natural deposition time constants (hr^{-1})	
	0.00 – 0.50 hr	0.02801
	0.50 – 1.80 hr	0.05713
	1.80 – 3.80 hr	0.06502
	3.80 – 11.80 hr	0.09151
	11.80 – 13.80 hr	0.09146
	13.80 – 22.22 hr	0.09146
	22.22 – 27.78 hr	0.03770
	27.78 – 33.33 hr	0.02770
	33.33 – 720.00 hr	0.00000
	Natural deposition decontamination factor limits	
	0.00 – 0.50 hr	1.0134
	0.50 – 1.80 hr	1.0944
	1.80 – 3.80 hr	1.3220
	3.80 – 11.80 hr	1.3220
	11.80 – 13.80 hr	3.9270
	13.80 – 22.22 hr	3.9270
	22.22 – 27.78 hr	8.2920
	27.78 – 33.33 hr	8.2920
	33.33 – 720.00 hr	1.0000
j.	Offsite power available?	Yes (Note 4)
k.	Initial primary to secondary leak rate (gpd per SG – Note 5)	150
l.	Time spans of SG tube bundle uncovery (seconds)	
	Unit 1	
	S/G 1	751
	S/G 2	662

	S/Gs 3 & 4 a piece	2458
	Total for Unit 1	6329
	Unit 2	
	S/G 1	2425
	S/G 2	2605
	S/Gs 3 & 4 a piece	810
	Total for Unit 2	6650
m.	SG iodine partition fraction	0.01
n.	Duration of post accident cooldown (hr)	6
o.	Total steam released (lbm)	
	Unit 1	
	0-2 hours	748,330
	2-6 hours	989,069
	Unit 2	
	0-2 hours	736,434
	2-6 hours	966,906
p.	Power level (MWt)	3479.
3.	Dispersion data	
a.	χ/Q at exclusion area boundary (sec/m^3)	4.78x10^{-4}
b.	χ/Q at low population zone (sec/m^3)	
	0- 8 hr	6.85x10^{-5}
	8- 24 hr	4.00x10^{-5}
	24- 96 hr	2.00x10^{-5}
	96-720 hr	7.35x10^{-6}
c.	Control Room χ/Q values (sec/m^3, Note 8)	
	Releases from the steam generator doghouse steam vents (Note 9)	
	0 – 2 hr	7.14x10^{-3}
	2 – 8 hr	4.05x10^{-3}
	8 – 10 hr	2.24x10^{-3}
	10 – 24 hr	1.81x10^{-3}
	24 – 96 hr	1.24x10^{-3}
	96 – 720 hr	7.26x10^{-4}
	Containment leakage	

	0 – 2 hr	1.04×10^{-3}
	2 – 8 hr	8.82×10^{-4}
	8 – 10 hr	4.14×10^{-4}
	10 – 24 hr	3.68×10^{-4}
	24 – 96 hr	2.67×10^{-4}
	96 – 720 hr	1.87×10^{-4}
4.	Data pretaining to the control room and Control Room Area Ventilation System	Table 15-41
5.	Dose conversion data	
a.	Method of conversion to dose	R.G. 1.183
b.	Source of dose conversion factors	
	External dose	FGR 12
	Inhaled dose	FGR 11
6.	Total Effective Dose Equivalents (Rem)	Table 15-14

Notes on Table 15-26

1) The containment volume and compartment volumes do not include the volume of the ice condenser.
2) The lower compartment volume includes the volume of the dead ended compartments.
3) It is assumed that the airflow returned to the annulus by the Annulus Ventilation System mixes with the air in half of the volume of the annulus or 242,045 cu. ft.
4) For the purposes of calculation of steam releases, offsite power is assumed to be available and the main coolant pumps in operation. Steam and radioactivity are assumed to be released to the environment via the steam generator power operated relief valves. This is consistent with assumed unavailability of the condenser dump valves and the modulated atmospheric dump valve which is consistent with assumed loss of offsite power.
5) Time dependent primary to secondary leak rates were calculated with the use of RETRAN02.
6) The radiation doses for a rod ejection accident postulated at Unit 2 are presented given that they are higher than the radiation doses for a rod ejection accident postulated at Unit 1.
7) Reserved for future use.
8) Both control room outside air intakes are assumed to be open with a 60/40 split in airflow.
9) These values correspond to the limiting release point on a outboard steam generator doghouse.

Table 15-27. Deleted Per 1995 Update

Table 15-28. Time Sequence of Events For Inadvertent Opening of a Pressurizer Safety Valve

Event	Time (sec)
Safety valve opens	0.1
Low pressurizer pressure reactor trip setpoint reached	22.9
Rods begin to drop	24.9
Minimum DNBR occurs	25.4

Table 15-29. Parameters for Postulated Instrument Line Break Offsite Dose Analysis

1.	Data and assumptions pertaining to the radioactive source term		
	a.	Reactor coolant DEX specific activity (μCi/gm - Note 1)	670
	b.	Equilibrium Reactor coolant DEI specific activity - concurrent iodine spike (μCi/gm - Note 2)	1
	c.	Concurrent iodine spike multiplier for the equilibrium reactor coolant iodine appearance rate	335
	d.	Maximum reactor coolant DEI specific activity pre-existent iodine spike (μCi/gm - Note 2)	60
	e.	Iodine composition fractions (%)	
		Diatomic iodine	97
		Organic iodine compounds	3
2.	Data and assumptions pertaining to acttivity release		
	a.	Break flow rate (gpm referenced at standard conditions)	150
	b.	Break flow iodine partition fraction	0.1
	c.	Time to isolate break (minutes)	30
3.	Dispersion data		
	a.	χ/Q at Exclusion Area Boundary (sec/m^3)	4.78×10^{-4}
	b.	χ/Q at the boundary of the Low Population Zone (sec/m^3)	
		0 hr – 8 hr	6.85×10^{-5}
		8 hr - 24 hr	4.00×10^{-5}
		24 hr - 96 hr	2.00×10^{-5}
		96 hr - 720 hr	7.35×10^{-6}
	c.	Control room χ/Q (sec/m^3)	
		0 hr - 2hr (Note 3)	1.74×10^{-3}
		0 hr - 2hr (Note 4)	1.04×10^{-3}
		2 hr - 8hr (Note 3)	1.47×10^{-3}
		2 hr - 8hr (Note 4)	8.82×10^{-4}
		8 hr - 10hr (Note 3)	6.90×10^{-4}
		8 hr - 10hr (Note 4)	4.14×10^{-4}
		10 hr - 24 hr (Note 5)	3.74×10^{-4}
		24 hr - 96 hr	2.67×10^{-4}
		96 hr - 720 hr	1.87×10^{-4}

4.	Data pertaining to the control room and Control Room Area Ventilation System		Table 15-41
	a.	Time to start redundant pressurized filter train - failure of the on-line pressurized filter train (minutes)	30
5.	Dose Data		
	a.	Source of dose coefficients	
		Deep Dose Equivalent coefficients	FGR 11
		Committed Dose Equivalent and Committed Effective Dose Equivalent (CEDE) coefficients	FGR 12
	b.	Source of CEDE coefficients for defining the DEI source term	FGR 11
	c.	Total effective dose coefficients (TEDEs, Rem)	
		Exclusion Area Boundary	
		Pre-existent iodine spike	0.60
		Concurrent iodine spike	0.17
		Boundary of the Low Population Zone	
		Pre-existent iodine spike	0.09
		Concurrent iodine spike	0.02
		Control room (Note 6)	
		Pre-existent iodine spike	0.80
		Concurrent iodine spike	0.18

Notes:

1) The term DEX denotes Dose Equivalent Xenon-133. The isotopic activities corresponding to this value are presented in Table 15-83.

2) The term DEI denotes Dose Equivalent Iodine-131. The isotopic activities corresponding to this value are presented in Table 15-84.

3) This control room X/Q value is used for the instrument line break scenarios with one control room outside air intake initially closed.

4) This control room X/Q value is used for the instrument line break scenarios with failure of the on-line Control Room Area Ventilation pressurized filter train. Both control room outside air intakes initially are open.

5) Regardless of the scenario, both control room outside air intakes are open beginning at 10 hours after the initiating event.

6) The limiting scenario includes failure of the on-line Control Room Area Ventilation pressurized filter train.

Table 15-30. Deleted Per 1990 Update

Table 15-31. Parameters for Postulated Steam Generator Tube Rupture Dose Analysis

1.	Data and assumptions pertaining to the radioactive source term		
	a.	Reactor coolant DEX specific activity (μCi/gm - Note 1)	670
	b.	Equilibrium Reactor coolant DEI specific activity - concurrent iodine spike (μCi/gm - Note 2)	1
	c.	Concurrent iodine spike multiplier for the equilibrium reactor coolant iodine appearance rate	335
	d.	Maximum reactor coolant DEI specific activity pre-existent iodine spike (μCi/gm)	60
	e.	Steam generator secondary side DEI specific activity (μCi/gm)	0.1
	f.	Preaccident main condenser iodine scrubbing efficiency (%, Note 3)	100
	g.	Iodine composition fractions (%)	
		Diatomic iodine	97
		Organic iodine compounds	3
2.	Data and assumptions pertaining to activity released		
	a.	Post accident condenser scrubbing efficiency (%, Note 4)	85
	b.	Main Feedwater isolation	At unit trip
	c.	Steam generator iodine partition factor	100
	d.	Integrated break flow (lbm, Note 5)	355,000
	e.	Integrated steam releases from the ruptured SG (lbm, Note 5)	
		To the condenser (before trip)	1,110,000
		To the atmosphere (after trip)	259,000
	f.	Integrated steam releases from the intact SGs (lbm, Note 5)	
		To the condenser (before trip)	3,330,000
		To the atmosphere (after trip)	1,850,000
	g.	Intact steam generator tube leak rate (gpm per steam generator)	0.104
3.	Dispersion data		
	a.	χ/Q at Exclusion Area Boundary (sec/m^3)	$4.78x10^{-4}$
	b.	χ/Q at the boundary of the Low Population Zone (sec/m^3.)	
		0 hr – 8 hr	$6.85x10^{-5}$
		8 hr - 24 hr	$4.00x10^{-5}$
		24 hr - 96 hr	$2.00x10^{-5}$
		96 hr - 720 hr	$7.35x10^{-6}$

	c.	Control room χ/Q (sec/m^3	
		Releases from the ruptured steam generator (Note 6)	
		0 hr - trip (Note 7)	$1.74x10^{-3}$
		0 hr - trip (Note 8)	$1.04x10^{-3}$
		trip - 2 hr Note 7)	$1.19x10^{-2}$
		trip - 2 hr (Note 8)	$7.14x10^{-3}$
		2 hr - 8 hr (Note 7)	$6.75x10^{-3}$
		2 hr - 8 hr (Note 8)	$4.05x10^{-3}$
		8 hr - 10 hr (Note 7)	$3.74x10^{-3}$
		8 hr - 10 hr (Note 8)	$2.24x10^{-3}$
		10 hr - 24 hr (Note 9)	$1.81x10^{-3}$
		24 hr - 96 hr	$1.24x10^{-3}$
		96 hr - 720 hr	$7.26x10^{-4}$
		Releases from the intact steam generators (Note 6)	
		0 hr - trip (Note 7)	$1.74x10^{-3}$
		0 hr - trip (Note 8)	$1.04X10^{-3}$
		trip - 2 hr (Note 7)	$6.49x10^{-3}$
		trip - 2 hr (Note 8)	$3.89x10^{-3}$
		2 hr - 8 hr (Note 7)	$4.18X10^{-3}$
		2 hr - 8 hr (Note 8)	$2.51x10^{-3}$
		8 hr - 10 hr (Note 7)	$2.38x10^{-3}$
		8 hr - 10 hr (Note 8)	$1.43x10^{-3}$
		10 hr - 24 hr (Note 7)	$1.18x10^{-3}$
		24 hr - 96 hr	$8.89x10^{-4}$
		96 hr - 720 hr	$6.25x10^{-4}$
4.	Data pertaining to the control room and Control Room Area Ventilation System		Table 15-41
5.	Dose Data		
	a.	Source of dose coefficients	
		Deep Dose Equivalent coefficients	FGR 12
		Committed Dose Equivalent and Committed Effective Dose Equivalent (CEDE) coefficients	FGR 11
	b.	Source of CEDE coefficients for defining the DEI source term	FGR 11
	c.	Total effective dose coefficients (TEDEs, Note 11)	

Exclusion Area Boundary	
Pre-existent iodine spike	1.19
Concurrent iodine spike	0.61
Boundary of the Low Population Zone	
Pre-existent iodine spike	0.28
Concurrent iodine spike	0.19
Control room	
Pre-existent iodine spike	1.32
Concurrent iodine spike	0.81

Notes:

1) DEX denotes Dose Equivalent Xenon-133. The isotopic activities corresponding to this value are presented in Table 15-83.

2) DEI denotes Dose Equivalent Iodine-131. The isotopic activities corresponding to this value are presented in Table 15-84.

3) The efficiency of the condenser in scrubbing iodine from the steam flow into it is set to this value (100%) before the initiating event to increase to a maximum the iodine specific activity in the Main and Auxiliary Feedwater Systems and associated condensate grade sources.

4) The value is taken in the calculations of fission product releases between the initiating event and trip of the affected nuclear unit.

5) This value is taken for the steam generator tube rupture scenario yielding the limiting offsite TEDE (the TEDE at thr Exclusion Area Boundary following a steam generator tube rupture at Unit 2 with the power operated relief valve for the ruptured steam generator failed open).

6) Before unit trip, fission products are released from the unit vent stack. After trip, fission products are released from the steam vents on the roofs of the steam generator doghouses.

7) One control room air intake is open.

8) Two control room air intakes are open.

9) From this time onwards, both control room intakes are open.

10) The ruptured steam generator and one intact steam generator presumably vent through steam vents on the outboard steam generator doghouse. The remaining two intact steam generators presumably vent through steam vents on the inboard steam generator doghouse.

11) The limiting TEDEs at offsite locations are associated with a steam generator tube rupture scenarios at Unit 2 with the power operated relief valve for the ruptured steam generator failed open. The limiting TEDEs in the control room for a steam generator tube rupture are associated with a steam generator tube rupture at Unit 2 with a closed control room outside air intake.

Table 15-32. Input Parameters Used in the SBLOCA LOCA Analyses (Unit 2)

Parameter	Value Used
Core Power (mwt)	3479
Total Peaking Factor, F_Q	2.7 (≤ 4 ft), 2.5 (> 4 ft)
Hot rod enthalpy rise peaking factor ($F_{\Delta H}$)	1.67
K(z) limit	1.0 (≤ 4 ft), 0.9259 (> 4 ft)
Power shape	See Figure 15-282
Fuel assembly array	17x17 RFA
Nominal cold leg accumulator water volume (ft^3/accumulator)	1050
Nominal cold leg accumulator tank volume (ft^3/accumulator)	1356
Minimum cold leg accumulator gas pressure (psia)	570
Cold leg accumulator temperature (°F)	125
Pumped safety injection flow	See Table 15-57
Pumped safety injection temperature (°F)	110
Nominal vessel average temperature (°F)	587.5
Pressurizer pressure (psia)	2250
RCS flow (gpm/loop)	97,500
Steam generator tube plugging (%)	10
Pressurizer low pressure safety injection setpoint (psia)	1715

Table 15-33. Deleted Per 1994 Update

Table 15-34. Deleted Per 1994 Update

Table 15-35. Deleted Per 1995 Update

Table 15-36. Deleted Per 1994 Update

Table 15-37. Deleted Per 1995 Update

Table 15-38. Small Break LOCA Time Sequence of Events (Unit 2)

	1.5 inch (sec)	**2 inch (sec)**	**3 inch (sec)**	**4 inch (sec)**
Start	0	0	0	0
Reactor Trip signal	249	91	24	14
ESFAS signal	261	101	34	24
ECC delivery	293	133	66	56
Loop seal cleared	N/A	N/A	489	298
Core uncovery	N/A	2019	900	669
Cold leg accumulator injection	N/A	N/A	N/A	919
RWST low level	1214	1207	1198	1180
Peak cladding temperature occurs	N/A	2736	1724	1032
Core recovery	N/A	4095	2768	>2000

Table 15-39. Small Break LOCA Results Fuel Cladding Data (Unit 2)

	1.5 inch	**2 inch**	**3 inch**	**4 inch**
Peak cladding temperature [1] °F	N/A	1054	1164	1243
Time of PCT (sec)	N/A	2736	1724	1032
PCT Location, (ft)	N/A	11.25	11.50	11.25
Maximum local ZrO_2 (%)	N/A	0.04	0.10	0.08
Maximum local ZrO_2 Location, (ft)	N/A	11.25	11.25	11.25
Total core-wide average ZrO_2 (%)	N/A	0.00	0.01	0.01
Hot rod burst time (sec)	N/A	N/A	N/A	N/A
Hot rod burst location, (ft)	N/A	N/A	N/A	N/A
Deleted Per 2006 Update				

Notes:

[1] There is no core uncovery for the 1.5 inch case

Table 15-40. Radiological Consequences of the Design Basis Loss of Coolant Accident – Data and Results

Parameter	Value	Notes
Data and assumptions pertaining to the source term (References 19 and 20)		
Power level (MWth)	3479	1
Fuel pins assumed to fail (%)	100	
Activity release time span		
Gap release phase (min)	0.5 – 30	
Early in-vessel release phase (hours)	0.5 – 1.8	
Activity release fraction in the gap phase (%)		3
Noble gases	5	2
Halogens	5	
Alkali metals	5	
Activity release fraction in the early-in vessel phase (%)		3
Noble gases	95	2
Halogens	35	
Alkali metals	25	
Tellurium metals	5	
Alkali earth metals (barium and strontium)	2	
Noble metals	0.25	
Cerium group	0.05	
Lanthanides	0.02	
Composition of iodine in containment atmosphere (%)		
Diatomic iodine	4.85	
Organic iodine compounds	0.15	
Particulates (iodide salts)	95	
Composition of iodine released from all ESF leakage (%)		
Diatmoic iodine	97	
Organic iodine compounds	3	
Particulates (iodide salts)	0	
Core Isotopic Inventory	Cf. Table 15-12	

Parameter			Value		Notes
Data and assumptions pertaining to activity transport in containment and release					
Lower compartment volume (cu.ft.)			346,895		4, 5
Upper compartment volume (cu.ft.)			669,559		5
Containment leak rate (La, mass % per day)			0.3		6
Containment bypass leak rate (% La)			7		6
Containment Air Return (VX) fan start time (minutes)			10		
VX fan airflow rate (cfm per fan)			40,000		
Time to begin credit for Spray washout (minutes)			80		7
CSS washout time constant					
Time Span (sec)			CSS time constant (sec^{-1})		
Start		End	I2	CsI (8)	9
0.0	-	4800	0.00	0.00	
4800	-	7800	0.23	9.51	
7800	-	30000	0.23	0.95	
30000	-	40000	0.23	0.95	
40000		50000	0.23	0.95	
50000	-	60000	0.23	0.95	
60000	-	70000	0.22	0.95	
70000	-	80000	0.20	0.95	
80000	-	86400	0.20	0.95	
86400	-	2592000	0.00	00.0	10
Data pertaining to the annulus and Annulus Ventilation (VE) System					
Annulus volume (cu.ft.)			484,090		11
AVS filter efficiency (%)					
Diatomic iodine			92		
Organic iodine compounds			92		
Particulate fission products			95		
AVS airflow bypass fraction (%)			1		
Time at which the annulus pressure falls to –0.25” w.g.					12

Parameter			Value	Notes
One AVS train in operation			41.4 sec	13
Both AVS trains in operation			30.5 sec	14
AVS Exhaust and Recirculation airflow rates			Table 6-75	
Data and assumptions pertaining to ESF System Leakage in the Auxiliary Building				
Containment sump volume (cu.ft.)			79,000	
ESF Leak Rate in the Auxiliary Building (gpm)				
Filtered			0.5	
Initially unfiltered			0.4	15
Iodine partition fractions for ESF leak in the Aux Bldg				
Time Span (hours)			Partition fraction	
Filtered leak, one ABFVES train unavailable				16
0	-	2	0.100	
2	-	72	0.025	
72	-	720	0.010	
Filtered leak, all ABFVES trains in operation				17
0	-	2	0.100	
2	-	72	0.031	
72	-	720	0.010	
Initially unfiltered leak, one ABFVES train unavailable				18
0	-	2	0.010	
2	-	72	0.010	
72	-	720	0.010	
Initially unfiltered leak, all ABFVES trains in operation				19
0	-	2	0.013	
2	-	72	0.010	
72	-	720	0.010	
Initially unfiltered leak, all ABFVES trains in operation, RHRS or CSS HX failure				20
0	-	2	0.100	
2	-	72	0.027	
72	-	720	0.010	

Parameter			Value	Notes
ABFVES carbon bed filter efficiency for iodine absorption (%)				21
Diatomic iodine			92	
Organic iodine compounds			92	
ABFVES airflow bypass percent			1	
Data pertaining to ESF backleakage to the Refueling Water Storage Tank (RWST)				
Rate of ESF backleakage to the RWST (gpm)			10	35
Iodine partition fraction for ESF back leakage to the RWST				
Time span (seconds)			Iodine partition fraction	36
Design basis LOCA with no heat exchanger failure				
0	-	2160	0	
2160		3000	1.022×10^{-6}	
3000	-	3600	1.022×10^{-6}	
3600	-	4200	1.071×10^{-6}	
4200	-	4800	1.071×10^{-6}	
4800	-	6000	1.001×10^{-6}	
6000	-	7200	9.280×10^{-7}	
7200	-	28800	9.139×10^{-7}	
28800	-	36000	1.082×10^{-6}	
36000	-	86400	6.056×10^{-7}	
86400	-	345600	2.154×10^{-6}	
345600	-	2592000	2.238×10^{-6}	
Design basis LOCA with RHRS or CSS Heat Exchanger failure				
0	-	2160	0	
2160	-	3000	1.049×10^{-6}	
3000	-	3600	1.049×10^{-6}	
3600	-	4200	1.111×10^{-6}	
4200	-	4800	1.111×10^{-6}	
4800	-	6000	1.044×10^{-6}	
6000	-	7200	9.723×10^{-7}	
7200	-	28800	1.099×10^{-6}	
28800	-	36000	1.586×10^{-6}	
36000	-	86400	1.179×10^{-6}	

Parameter			Value	Notes
86400	-	345600	8.273×10^{-6}	
345600	-	2592000	1.230×10^{-5}	
Atmospheric dispersion factors (χ/Qs) for transport of fission products to offsite locations				
χ/Q at the Exclusion Area Boundary (sec/m3)			4.78x10-4	22
χ/Q at the boundary of the Low Population Zone				
0 – 8 hr			6.85x10-5	
8 – 24 hr			4.00x10-5	
24 – 96 hr			2.00x10-5	
96 – 720 hr			7.35x10-6	
χ/Q for transport of fission products to the control room outside air intakes (23, 24)				
Release from the unit vent stack (sec/m3)				25
0 – 2 hr			1.74x10-3	26, 27
0 – 2 hr			1.04x10-3	26, 28
2 – 8 hr			1.47x10-3	26, 27
2 – 8 hr			8.82x10-4	26, 28
8 – 10 hr			6.90x10-4	27
8 – 10 hr			4.14x10-4	28
10 – 24 hr			3.68x10-4	
24 – 96 hr			2.67x10-4	
96 – 720 hr			1.87x10-4	
Release from the RWST vent (sec/m3)				29
0 – 2 hr			1.92x10-3	26, 27
0 – 2 hr			1.26x10-3	26, 28
2 – 8 hr			1.48x10-3	26, 27
2 – 8 hr			9.78x10-4	26, 28
8 – 10 hr			7.40x10-4	27
8 – 10 hr			4.86x10-4	28
10 – 24 hr			4.10x10-4	
24 – 96 hr			2.86x10-4	
96 – 720 hr			1.87x10-4	
Airflow imbalance in the VC outside air intakes			60/40	

Parameter	Value	Notes
Control Room and CRAVS PFT Data	Table 15-41	
Coefficients for conversion from activity to radiation dose		
Inhaled CDE coefficients	FGR-11	30
DDE coefficients	FGR-12	31
Offsite breathing rates (m3/sec)		
0 – 8 hr	3.5x10-4	
8 – 24 hr	1.8x10-4	
24 – 720 hr	2.3x10-4	
Control room breathing rate (0 – 720 hr)	3.5x10-4	
Control room occupancy factors		
0 – 24 hr	1.0	
24 – 96 hr	0.6	
96 – 720 hr	0.4	
Limiting Offsite Radiation Doses (Rem, 32)		
Exclusion Area Boundary TEDE	8.79	
TEDE at the boundary of the Low Population Zone (LPZ)	3.78	
Limiting Control Room Radiation Doses (Rem, 32)		
Control room TEDE		34
Fission Product buildup in the Control Room	2.56	
Direction radiation from external sources	0.75	
Total	3.31	

Notes:

1) The assumed power level (3479 MWt) is 102% of original licensed rated thermal for the reactor (3411 MWt).

2) All noble gases are assumed to be retained in the containment atmosphere, none are assumed to be in the containment sump.

3) Release of all fission products in the radioactive source term to the containment atmosphere pursuant to RG 1.183 is simulated. Release of tellurium and iodine isotopes to the containment sump only is simulated. This is equivalent to the Staff position that only iodine may be released from ESF leakage to the environment.

4) The ice condenser is not modeled as a containment volume.

5) The volume of the lower compartment includes the dead-end compartments.

6) The containment leak rate and the containment bypass leak rate are partitioned into two constituents: one for the lower compartment and one for the upper compartment. The partitioning is in proportion with compartment volume.

7) The control room operators start the Containment Spray System (CSS) after they begin the transfer of the Emergency Core Cooling System to cold leg to recirculation. It is assumed that they start the CSS at 80 minutes after the initiating LOCA.

8) CsI (cesium iodide) is the notation for all fission products in particulate form. The NRC Staff takes the position that CSS washout time constants for fission products in particulate form should be reduced to 1/10 of the calculated value if the decontamination factor reaches 50.

9) The spray washout time constant for organic iodine compounds (e.g., CH_3I) is 0.

10) It is assumed that the CSS System is turned off at 1 day after the initiating event.

11) Credit is taken for mixing of Annulus Ventilation System (AVS) recirculation airflow with air in half the volume of the annulus (242,045 cu.ft.). No credit is taken for mixing of containment leakage to the annulus directly in the annulus airspace. Rather, it is assumed that containment leakage to the annulus flows directly to the AVS Exhaust louvers.

12) The calculation of the annulus drawdown time takes into account the effect of the differences in hydrostatic gradients inside the annulus and outside the reactor building. The outside air temperature is set to the 99^{th} percentile low temperature (1 percentile temperature).

13) The limiting single failure with the effect of loss of one AVS train is the Minimum Safeguards failure.

14) Scenarios in which both AVS trains are assumed to be in operation include the design basis LOCA with failure of a AVS pressure transmitter – the transmitter gives a false high indication of annulus pressure. The effected AVS train continues to operate but in the Exhaust Mode at times when it should operate in the Recirculation Mode.

15) It is assumed that the operators will align the Class 1E Exhaust filter trains of the Auxiliary Building Ventilation System ABFVES within 3 days following the initiating event.

16) The limiting scenario associated with these iodine partition fractions are the design basis LOCA with Minimum Safeguards failure.

17) This scenario encompasses all design basis LOCA scenarios for which the failure does not affect the ABFVES System.

18) The limiting scenario associated with these iodine partition fractions are the design basis LOCA with Minimum Safeguards failure.

19) The limiting scenarios include the design basis LOCA with either a AVS pressure transmitter failure or a closed control room air intake isolation valve.

20) The limiting failure is the design basis LOCA with failure of flow of cold water flow to either a RHRS or CSS Heat Exchanger.

21) No particulate fission products are assumed to be entrained in ESF Systems leakage.

22) This χ/Q is taken for the 2 hour period of maximum release of radioactivity to the environment.

23) The control room outside air intakes are the “receptors” for outside airflow drawn by the pressurized filter trains of the Control Room Area Ventilation System (CRAVS). The control

room χ/Qs for unfiltered inleakage to the control room are based on the intake penetrations being the limiting entry point for unfiltered inleakage to the control room.

24) The values for 10-720 hr are based on transport of fission products with dispersion to two CRAVS outside air intakes.

25) The unit vent stack is taken to be the release point for (1) containment leakage that is filtered by the Annulus Ventilation System, (2) containment bypass leakage, and (3) ESF leakage in the Auxiliary Building.

26) The 0-2 hr control room χ/Q is taken for the 2 hr period of maximum releases. The 2-8 hr control room χ/Q is taken for the remainder of the 0-8 hr time span. All releases were predicted to be at a maximum over a 2 hour time span within 0-8 hr.

27) One CRAVS outside air intake is taken to be open.

28) Both CRAVS outside air intakes are taken to be open.

29) The vent of the RWST is the release point for iodine released from the airspace following a design basis LOCA with assumed ESF System backleakage to the RWST.

30) Federal Guidance Report 11, cf. Reference 98.

31) Federal Guidance Report 12, cf. Reference 99.

32) All radiation doses given include the combined effects of containment leakage and ESF leakage both inside the Auxiliary Building and to the Refueling Water Storage Tank.

33) The DF for spray washout of particulates was calculated to reach 50 at the beginning of this interval.

34) The constituents to the control room TEDE from external sources are associated specifically with fission products accumulating on ventilation filters and fission products inside containment (cf. Section 1.8.1.19).

35) The backleakage may be routed either (1) completely to the bottom of the RWST, (2) completely to the top, or (3) partly to the bottom and partly to the top.

36) These values are applied to the backleakage mixed with inventory in the RWST. For backleakage to the top of the RWST, the iodine partition fraction via the airspace is set to 2E-5.

Table 15-41. Data Pertaining to the Control Room and Control Room Area Ventilation (VC) System.

Parameter	Design Basis Value
Control room volume (cu.ft.)	117,920
Lower bound VC PFT total airflow to the control room (cfm)	3,500
Lower bound VC PFT recirculation airflow through the control room (cfm)	1,500
Rate of unfiltered inleakage to the control room (cfm)	100
VC PFT efficiency for removal of fission products and bypass airflow fraction (%)	
Diatomic iodine	98.1
Organic iodine compounds	98.1
Particulate fission products	99
Bypass airflow fraction	0.05
Flow split in the VC outside air intakes (%/%)	60/40

Note: Unless otherwise noted, this data is used in the calculations of radiation doses for all design basis accidents with postulated releases of radioactivity releases to the environment.

Table 15-42. Parameters for Postulated Liquid Storage Tank Rupture Evaluation (UFSAR 15.7.2)

1.	Data and assumptions pertaining to the radioactive source term		
	a.	Tank ruptured	Recycle holdup tank
	b.	Tank volume	112,000 gallons
	c.	Tank specific activities	
		Noble gases	Table 15-83
		Iodine	Table 15-84
	d.	Iodine chemical composition fractions	
		Diatomic iodine	0.97
		Organic iodine compounds	0.03
2.	Data and assumptions used to estimate activity released		
	a.	Iodine partition fraction	0.1
	b.	Release location	Unit vent stack
3.	Dispersion data		
	a.	χ/Q for the EAB (0-2 hr value)	4.78E-04 sec/m^3
	b.	χ/Q for the LPZ (0-8 hr value)	6.85E-05 sec/m^3
	c.	χ/Q for the control room (0-2 hr value)	1.74E-03 sec/m^3
4.	Data and assumptions pertaining to the control room		
	a.	Number of intakes open	1
	b.	Control room volume	117,920 cu. ft
	c.	Rate of unfiltered inleakage into the control room	2,100 cfm
	d.	Status of ventilation pressurized filter trains (UFSAR 15.7.2.2)	Off
5.	Data and assumptions pertaining to the control room		
	a.	Method of calculation	Regulatory Guide 1.183
	b.	Source of dose coefficients	
		External	Federal Guidance Report 12
		Inhaled	Federal Guidance Report 11
	c.	Post accident radiation doses	Table 15-14

Table 15-43. Parameters for Postulated Refueling Water Storage Tank Rupture Accident

		Conservative	Realistic
1.	Data and assumptions used to estimate radioactive source from refueling water storage tank rupture		Accident is not evaluated for realistic case
	a. Power level (MWt)	3565.	
	b. Percent of fuel defected (%)	1.	
	c. Release of activity by nuclide	Table 15-44	
2.	Pertinent data and assumptions used to estimate activity released	1. 100% of maximum tank activities released directly to discharge canal 2. No radiological decay credit is taken from tank to nearest surface water intake	
3.	Concentration data		
	a. Distance to nearest surface water intake	5.6 km	
	b. Dilution factor	2.0E+4	

Table 15-44. Activities for Postulated Refueling Water Storage Tank Rupture Accident

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Isotope	***Tank Activity (μCi/ml)***	***Resultant concentration at nearest surface water intake (μCi/ml)***	***Fraction of 10 CFR 20 limit***
Sr 89	*4.1E-10*	*8.0E-14*	*2.7E-8*
Sr 90	*2.7E-11*	*5.3E-15*	*1.8E-8*
Y 90	*2.1E-11*	*4.1E-15*	*2.0E-12*
Y 91	*7.7E-10*	*1.5E-13*	*5.0E-9*
Zr 95	*8.9E-11*	*1.7E-14*	*2.8E-10*
Nb 95	*7.9E-11*	*1.5E-14*	*1.5E-10*
Tc 99M	*2.1E-8*	*4.1E-12*	*1.4E-9*
Mo 99	*2.2E-8*	*4.3E-12*	*1.1E-7*
I 131	*1.1E-7*	*2.1E-11*	*7.0E-5*
I 132	*1.8E-9*	*3.5E-13*	*4.4E-8*
I 133	*5.9E-12*	*1.2E-15*	*1.2E-9*
Te 132	*1.7E-9*	*3.3E-13*	*1.6E-8*
Cs 134	*1.1E-3*	*2.1E-7*	*2.3E-2*
Cs 136	*1.3E-3*	*2.5E-7*	*4.2E-3*
Cs 137	*7.8E-4*	*1.5E-7*	*7.5E-3*
Ba 137M	*7.3E-4*	*1.4E-7*	*4.7E-2*
Ba 140	*2.9E-10*	*5.7E-14*	*2.8E-9*
La 140	*2.9E-10*	*5.7E-14*	*2.8E-9*
Ce 144	*6.5E-11*	*1.3E-14*	*1.3E-9*
Pr 144	*6.5E-11*	*1.3E-14*	*4.3E-9*
Xe 131M	*6.7E-11*	*1.3E-14*	*6.5E-11*
Xe 133	*6.5E-13*	*1.3E-16*	*6.5E-13*
Mn 54	*5.0E-11*	*9.8E-15*	*9.8E-11*
Co 58	*1.4E-9*	*2.7E-13*	*3.0E-9*
Co 60	*1.8E-10*	*3.5E-14*	*1.2E-9*
Fe 59	*4.8E-11*	*9.4E-15*	*1.9E-10*
Cr 51	*4.4E-10*	*8.6E-14*	*4.3E-11*
		Total FMPC	***8.2E-2***

Table 15-45. Activities in Highest Inventory Discharged Assembly for Postulated Fuel Handling Accidents

Nuclide	Assembly Inventory (Curies)	Gap Fraction	Gap Inventory (Curies)
Br-83	1.31E+05	0.05	6.55E+03
Br-85	2.99E+05	0.05	1.50E+04
Br-87	4.95E+05	0.05	2.48E+04
I-130	3.95E+04	0.05	1.98E+04
I-131	8.09E+05	0.05	6.47E+04
I-132	1.18E+06	0.05	5.09E+04
I-133	1.67E+06	0.05	8.35E+04
I-134	1.95E+06	0.05	9.75E+04
I-135	1.60E+06	0.05	8.00E+04
Kr-83m	1.32E+05	0.05	6.60E+03
Kr-85m	2.98E+05	0.10	1.49E+04
Kr-85	7.48E+03	0.05	7.48E+02
Kr-87	6.15E+05	0.05	3.08E+04
Kr-88	8.69E+05	0.05	4.35E+04
Kr-89	1.12E+06	0.05	5.60E+04
Xe-131m	1.24E+04	0.05	6.20E+02
Xe-133m	5.20E+04	0.05	2.60E+03
Xe-133	1.65E+06	0.05	8.25E+04
Xe-135m	3.62E+05	0.05	1.81E+04
Xe-135	4.12E+05	0.05	2.06E+04
Xe-137	1.55E+06	0.05	7.75E+04
Xe-138	1.59E+06	0.05	7.05E+04
Rb-86	2.54E+03	0.12	3.05E+02
Rb-88	8.89E+05	0.12	1.07E+05
Rb-89	1.18E+06	0.12	1.42E+05
Rb-90	1.12E+06	0.12	1.34E+05
Cs-134	2.06E+05	0.12	2.47E+04
Cs-136	5.92E+04	0.12	7.10E+03
Cs-137	9.23E+04	0.12	1.11E+04
Cs-138	1.66E+06	0.12	1.99E+05

Nuclide	Assembly Inventory (Curies)	Gap Fraction	Gap Inventory (Curies)
Cs-139	1.58E+06	0.12	1.90E+05

Note:

1. The gap fractions are from Table 3 of NRC Regulatory Guide RG 1.183
2. For those fuel rods which exceed the rod power/ burnup criteria of Footnote 11 in RG 1.183, the gap fractions shown above for Kr-85, Xe-133, Cs-134, and Cs-137 are tripled while the gap fractions shown above for all other noble gases and alkali metals, and for all halogens are doubled. A maximum of 25 fuel rods per fuel assembly shall be allowed to exceed the rod power/ burnup limit of Footnote 11 in RG 1.183 in accordance with the license amendment request submitted July 15, 2015.

Table 15-46. Parameters for Postulated Fuel Handling Accident (Note 1)

1.	Data and assumptions used to estimate radioative source from postulated accident			
	a.	Power Level (MWt)	3479	
	b.	Decay time before the initiating event (days)	3	
	c.	Gap fractions (%) Kr-85 Other noble gases I-131 Other iodine radioisotopes	 10 5 8 5	Note 2
	d.	Number of fuel assemblies impacted	1	
	e.	Damage to each fuel assembly impacted	All rods ruptured	
	f.	Other pertinent assumptions	R.G. 1.183	
2.	Data and assumptions used to estimate release of radioactivity from the pool			Note 3
	a.	Effective pool decontamination factor Noble gases Iodines	 1 200	
	b.	Composition fractions of iodine forms leaving the pool (%) Diatomic iodine Organic iodine compounds	 57 43	
	c.	Time profile for release to the environment	Decaying exponential	
		Release profile time constant (hr^{-1})	2	Note 4
3.	Offsite dispersion data			
	a.	EAB (0-2 hr) *X/Q* (sec/m^3	4.78×10^{-4}	
	b.	LPZ (0-8 hr) *X/Q* (sec/m^3	$6.85 X 10^{-5}$	Note 5
4.	Data for transport of radioactivity with dispersion to the control room outside air intakes			
	a.	Number of intakes assumed openn	1	
	b.	Control room *X/Q* (sec/m^3	1.74×10^{-3}	Note 5
5.	Data pertaining to the control room and Control Room Area Ventilation (VC) System			
	a.	Control room volume (cu. ft.)	117,920	

	b.	VC Pressure Filter Train airflow rates to the control room	3,500	
		Total airflow rate	1,500	
		Recirculation airflow rate		
		VC PFT efficiencies	99	
		Diatomic iodine	95	
		Organic iodine compounds		
6.	Data for conversion of activity to dose			
	a.	Coefficients for committed effective dose equivalents	FGR-11	
	b.	Coefficients for deep dose equivalent	FGR-12	
	c.	Other data for conversion of activity to dose	R.G. 1.183	
7.	Total Effective Dose Equivalent (TEDE's – Rem)			Note 6
	a.	EAB TEDE	1.76	
	b.	Control room TEDE	2.59	

Notes:

1. The data and assumptions listed in Table 15-46 are applied to the fuel handling accident inside containment and the fuel handling accident outside containment.
2. For those fuel pins which exceed the rod power/ burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals. A maximum of 25 fuel rods per fuel assembly shall be allowed to exceed the rod power/ burnup criteria of Footnote 11 in RG [Regulatory Guide] 1.183 in accordance with the license amendment request submitted by letter dated July 15, 2015.
3. The term "pool" denotes the spent fuel pool for fuel handling accidents outside containment and the reactor cavity for fuel handling accidents inside containment.
4. This time constant was selected to ensure that at 98% of the iodine taken to be released from the pool would be released to the environment within 2 hours.
5. Only the values for the first time period (0-8 hr for the LPZ *X/Q* and 0-2 hr for the control room *X/Q* are listed as the release of radioactivity to the environment is essentially complete in 2 hours. Cf. Note 4.
6. The LPZ TEDE is not listed as it is less than the EAB TEDE and compared to the same acceptance criterion (6.3 Rem).

Table 15-47. Deleted Per 2004 Update

Table 15-48. Parameters for Post-LOCA Subcriticality Analysis

Volume Grouping	Boron Concentration (ppm)
Low Head Safety Injection (LHSI) Discharge to Intermediate Head Safety Injection (IHSI) and High Head Safety Injection (HHSI) suction (Valve NI136B to Valves NI332A & NI333B) (Valve ND28A to Valve NV813)	RWST minimum[3]
Refueling Water Storage Tank (RWST) to Valves FW28 & FW56	RWST minimum[3]
RWST to IHSI suction	RWST minimum[3]
RWST to Valves NV252A & NV253B	RWST minimum[3]
Normal Containment Spray Discharge	RWST minimum[3]
Containment Spray Suction from RWST	RWST minimum[3]
Valves NS43A & NS38B to Aux. Cont. Spray	350[1]
Valves FW28 and FW56 to LHSI Suction	variable[2]
LHSI Suction from Sump	350[1]
LHSI Suction from Hot Legs	350[1]
Containment Spray Suction from Sump	350[1]
RCS	variable[2]
LHSI Discharge to Cold Legs	variable[2]
LHSI Discharge to IHSI and HHSI Suction (Valve NV813 to Valves NI332A & NI333B) (LHSI Discharge to Valve ND28A) (LHSI Discharge to Valve NI136B)	variable[2]
LHSI Discharge to B and C Hot Legs	variable[2]
LHSI Discharge to Valves NS43A & NS38B	variable[2]
LHSI Mini-Flow	variable[2]
IHSI Discharge to LHSI Discharge	variable[2]
IHSI Discharge to Hot Legs	variable[2]
IHSI Mini-Flow	variable[2]
HHSI Discharge to Cold Legs	variable[2]
Valves NV252A & NV253B to HHSI Suction	variable[2]

Volume Grouping	Boron Concentration (ppm)

Notes:

1. EOC Mode 4 RCS boron concentration
2. "variable" indicates that the associated volume concentration is assumed equal to the RCS boron concentration, which is a function of burnup.
3. This boron concentration is equal to the cycle specific RWST minimum boron concentration specified in the Core Operating Limits Report. The analysis assumes RWST boron concentrations between 2475 and 2875 ppm.

Table 15-49. Time Sequence of Events For Steam Generator Tube Rupture

Event **(Dose Analysis)**	**Time** **(sec)**
Double ended tube rupture occurs	0.1
Manual reactor trip	1200
Loss of offsite power occurs	1200
Steamline PORV on ruptured SG fails open	1201
2 pump/2 train maximum safety injection begins	1212
Operators identify ruptured SG and close ruptured SG MSIV	2100
Operators close failed open steam line PORV	2271
Operators begin RCS cooldown with operable SG PORV	3000
Operators close operable steam line PORV	3820
Operators open pressurizer PORV to depressurize the RCS	4410
Break flow terminated	4607

(DNB Analysis)	
Double-ended tube reputere occurs	1.0
Reactor trip/turbine trip on OTΔT	319.0
Reactor coolant pumps lost	319.0
MDNBR occurs	320.9

Table 15-50. Deleted Per 2001 Update

Table 15-51. Deleted Per 2001 Update

Table 15-52. Deleted Per 2001 Update

Table 15-53. Deleted Per 2001 Update

Table 15-54. Deleted Per 2001 Update

Table 15-55. Deleted Per 2000 Update

Table 15-56. Deleted Per 2000 Update

Table 15-57. Minimum ECCS Flow Assumed in SBLOCA Analyses One Train Operational, Break Backpressure Equal to RCS Pressure

	High-Head SI		Intermediate-Head SI	
RCS Pressure (psia)	**3 Injecting Lines (gpm)**	**1 Spilling Line (gpm)**	**3 Injecting Lines (gpm)**	**1 Spilling Line (gpm)**
14.7	275	105	405	150
50	275	100	400	145
75	270	100	395	145
100	270	100	390	145
125	270	100	385	145
150	265	100	385	140
200	265	100	375	140
250	260	100	365	135
300	255	95	360	135
500	245	90	320	120
700	230	85	280	105
900	210	80	235	90
1100	195	75	175	65
1300	175	65	85	35
1450	160	60	0	0
1500	155	60	0	0
2310	0	0	0	0

Deleted Per 2007 Update.

Table 15-58. Parameters for Postulated Weir Gate Drop

1. Data and Assumption	
a. Decay time before the initiating event (days)	19.5
b. Number of fuel assemblies impacted	7
c. All other data and assumptions	Table 15-46
2. Total Effective Dose Equivalent	Note 1
a. EAB TEDE	2.68
b. Control room TEDE	4.24

Note:

1) The LPZ TEDE is not listed as it is less than the EAB TEDE and compared to the same acceptance criterion (6.3 Rem).

Table 15-59. Input Parameters Used in the SBLOCA LOCA Analyses (Unit 1)

Parameter	Value Used
Core power (mwt)	3479
Total peaking factor, F_Q	2.7 (≤ 4 ft), 2.5 (> 4 ft)
Hot rod enthalpy rise peaking factor ($F_{\Delta H}$)	1.67
K(z) limit	1.0 (≤ 4 ft), 0.9259 (> 4 ft)
Power shape	See Figure 15-282
Fuel assembly array	17 x 17 RFA
Nominal cold leg accumlator water volume (ft^3/accumulator)	950
Nominal cold leg accumulator tank volume (ft^3/accumulator)	1363
Minimum cold leg accumulator gas pressure (psig)	570
Cold leg accumulator temperature (°F)	125
Pumped safety injection flow	see Table 15-57
Pumped safety injection temperature (°F)	110
Nominal vessel average temperature (°F)	585.1
Pressurizer pressure (psia)	2250
RCS flow (gpm/loop)	97,500
Steam generator tube plugging (%)	5
Pressurizer low pressure safety injection setpoint (psia)	1715

Table 15-60. Small Break LOCA Time Sequence of Events (Unit 1)

	1.5 inch (sec)	2 inch (sec)	3 inch (sec)	4 inch (sec)
Start	0	0	0	0
Reactor trip signal	114	57	23	13
ESFAS signal	135	73	32	21
ECC delivery	167	105	64	53
Loop seal cleared	N/A	N/A	628	333
Core uncovery	N/A	2378	993	703
Cold leg accumulator injection	N/A	N/A	N/A	997
RWST low level	1211	1206	1199	1183
Peak cladding temperature occurs	N/A	3449	1986	1092
Core recovery	N/A	5122	2933	1971

Table 15-61. Small Break LOCA Results Fuel Cladding Data (Unit 1)

	1.5 inch	**2 inch**	**3 inch**	**4 inch**
Peak cladding temperature[1] (°F)	N/A	1323	1153	1208
Time of PCT (sec)	N/A	3449	1986	1092
PCT location (ft)	N/A	11.50	11.25	11.25
Maximum local ZrO_2 (%)	N/A	0.24	0.09	0.06
Maximum local ZrO_2 location(ft)	N/A	11.50	11.25	11.25
Total core-wide average ZrO_2 (%)	N/A	0.03	0.01	0.01
Hot rod burst time (sec)	N/A	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A	N/A
Deleted Per 2006 Update				

Notes:

[1] There is no core uncovery for the 1.5 inch case

Table 15-62. Deleted Per 2001 Update

Table 15-63. Deleted Per 2001 Update

Table 15-64. Deleted Per 2001 Update

Table 15-65. Minimum Injected ECCS Flows Assumed in LBLOCA Analyses - One Train Operational

RCS Pressure (psia)	High-Head SI (gpm)	Intermediate-Head SI (gpm)	Low-Head SI (gpm)
14.7	285	420	2600
50	280	410	1800
75	280	410	1225
100	275	405	500
125	275	400	0

Table 15-66. Deleted Per 2001 Update

Table 15-67. Deleted Per 2001 Update

Table 15-68. Large Break LOCA – Time Sequence of Events for Reference Transient

Event	Time (seconds)
Break opening time	20
Safety injection signal	24
Accumulator injection begins	31
Pumped safety injection begins	56
Bottom of core recovery	58
Accumulators empty	62
Time of peak cladding temperature	286

Table 15-69. Key Large Break LOCA Parameters and Initial Transient Assumptions.

Parameter	Initial Transient	Uncertainty or Bias
1.0 Plant Physical Description		
Dimensions	Nominal	ΔPCT_{MOD}[1]
Flow resistance	Nominal	ΔPCT_{MOD}
Pressurizer location	Opposite broken loop	Bounded
Hot assembly location	Under limiting location	Bounded
Hot assembly type	17x17 RFA with IFM	Bounded
SG tube plugging level	D5, maximum (10%)	Bounded[4]
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
Core average linear heat rate (AFLUX)	Nominal – power (3445 MWt)[7]	ΔPCT_{PD}[2]
Peak linear heat rate (PLHR)	Derived from desired Tech Spec (TS) limit and maximum baseload FQ	ΔPCT_{PD}
Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$	ΔPCT_{PD}
Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD}
Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD}
Axial power distribution (PBOT, PMID)	Figure 15-313	ΔPCT_{PD}
Low power region relative power (PLOW)	Minimum (0.2)	Bounded[4]
Hot assembly burnup	BOL	Bounded
Prior operating history	Equilibrium decay heat	Bounded
Moderate Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
HFP boron	800 ppm	Typical

Parameter		Initial Transient	Uncertainty or Bias
2.2	Fluid Conditions		
	T_{avg}	Nominal $T_{avg=}$ 587.5°F (Catawba Unit 2)	ΔPCT_{IC}[3]
	Pressurizer pressure	Nominal (2250 psia)	ΔPCT_{IC}
	Loop flow	Minimum (97500 gpm)	ΔPCT_{MOD}[5]
	T_{UH}	Best Estimate	0
	Pressurizer level	Nominal (55% of volume)	0
	Accumulator temperature	Nominal (115°F)	ΔPCT_{IC}
	Accumulator pressure	Nominal (631.5 psig, Catawba units)	ΔPCT_{IC}
	Accumulator liquid volume	Nominal (7106 gal, McGuire units)	ΔPCT_{IC}
	Accumulator line resistance	Nominal (McGuire Unit 2)	ΔPCT_{IC}
	Accumulator boron	Minimum (McGuire units)	Bounded
3.0	Accident Boundary Conditions		
	Break location	Cold leg	Bounded
	Break type	Guillotine	ΔPCT_{MOD}
	Break size	Nominal (cold leg area)	ΔPCT_{MOD}
	Offsite power	On (RCS pumps running)	Bounded[6]
	Safety injection flow	Minimum	Bounded
	Safety injection temperature	Nominal (85°F)	ΔPCT_{IC}
	Safety injection delay	Max delay (17 sec)	Bounded
	Containment pressure	Minimum based on <u>W</u>C/T M&E	Bounded
	Single failure	ECCS: Loss of 1 SI train	Bounded
	Control rod drop time	No control rods	Bounded

Parameter		Initial Transient	Uncertainty or Bias
4.0	Model Parameters		
	Critical Flow	Nominal (as coded)	ΔPCT_{MOD}
	Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}
	Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}
	Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}
	Delivery and bypassing of ECC	Nominal (as coded)	Conservative
	Steam binding/entrainment	Nominal (as coded)	Conservative
	Noncondensable gases/accumulator nitrogen	Nominal (as coded)	Conservative
	Condensation	Nominal (as coded)	ΔPCT_{MOD}

Notes:

1. ΔPCT_{MOD} indicates this uncertainty is part of code and global model uncertainty.
2. ΔPCT_{PD} indicates this uncertainty is part of power distribution uncertainty.
3. ΔPCT_{IC} indicates this uncertainty is part of initial condition uncertainty.
4. Confirmed by analysis.
5. Assumed to be result of loop resistance uncertainty.
6. Sensitivity analysis concluded loss of offsite power is more limiting than assuming offsite power on (RCS pumps running).
7. Analysis was originally performed at 3445 MWt (3411 MWt plus 1% for conservatism). However, 1% for heat balance error was also added into the analysis, so it remains bounding for the MUR (3479 MWt). An MUR uprate evaluation was performed at 3469 MWt (101.7% of 3411 MWt) plus 0.3% uncertainty to derive the PCT penalty included in Table 15-82.

Table 15-70. Best-Estimate Large Break LOCA – Overall Results

Component	Blowdown Peak (°F)	First Reflood Peak (°F)	Second Reflood Peak (°F)
$PCT^{50\%}$	<1256	<1384	<1512
$PCT^{95\%}$	<1548	<1692	<2028

Table 15-71. Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis

Parameter	Operating Range
1.0 Plant Physical Description	
Dimensions	No in-board assembly grid deformation during LOCA + SSE
Flow resistance	N/A
Pressurizer location	N/A
Hot assembly location	Anywhere in core
Hot assembly type	Fresh 17x17 RFA
SG tube plugging level	≤10% (Catawba 2) and ≤5% (McGuire and Catawba 1)
2.0 Plant Initial Operating Conditions	
2.1 Reactor Power	
Core avg linear heat rate	Core power ≤ 101% of 3445 MW_t[3]
Peak linear heat rate	$F_Q \leq 2.70$ (≤ 4 ft), $F_Q \leq 2.50$ (> 4 ft) [see Note 1]
Hot rod average linear heat rate	$F_{\Delta H} \leq 1.67$ [see Note 2]
Hot assembly average linear heat rate	$\overline{P}_{HA} \leq 1.67/1.04$ [see Note 4]
Hot assembly peak linear heat rate	$F_{QHA} \leq 2.7/1.04$ (≤ 4 ft), $F_Q \leq 2.50/1.04$ (> 4 ft) [see Note 1]
Axial power dist (PBOT, PMID)	Figure 15-312
Low power region relative power (PLOW)	0.2 ≤ PLOW ≤ 0.8
Hot assembly burnup	≤ 75000 MWD/MTU, lead rod
Prior operating history	All normal operating histories
MTC	≤ 0 at HFP
HFP boron	Normal letdown
Rod power census	See Table 15-72

Parameter		Operating Range
2.2	Fluid Conditions	
	Tavg	$581.1 \leq T_{avg} \leq 593.9°F$
	Pressurizer pressure	$2190 \leq P_{RCS} \leq 2310$ psia
	Loop flow	≥ 97,500 gpm/loop
	T_{UH}	Current upper internals, T_{cold} UH
	Pressurizer level	Normal level, automatic control
	Accumulator temperature	$105 \leq T_{ACC} \leq 125°F$
	Accumulator pressure	$555 \leq P_{ACC} \leq 708$ psig
	Accumulator volume	$6790 \leq V_{ACC} \leq 7422$ gal. (McGuire), $7550 \leq V_{ACC} \leq 8159$ gal. (Catawba)
	Accumulator fL/D	Current line configuration
	Minimum accumulator boron	≥ 2275 ppm
3.0	Accident Boundary Conditions	
	Break location	N/A
	Break type	N/A
	Break size	N/A
	Offsite power	Available or LOOP
	Safety injection flow	Table 15-65
	Safety injection temperature	58° ≤ SI Temp ≤ 90°F, Reference 82 (covers a RWST temperature range of 70-100°F and component cooling water temperatures down to 45°F)
	Safety injection delay	≤ 17 seconds (with offsite power) ≤ 32 seconds (with LOOP)
	Containment pressure	Bounded - - see Figure 15-302

Parameter	Operating Range
Single failure	Loss of one train
Control rod drop time	N/A

Notes:

1. To account for fuel pellet thermal conductivity degradation, the allowed F_Q peaking factor is subject to these normalization factors (interpolation allowed):

   Hot Rod Average Burnup = 0 GWD/MTU, F_Q normalization factor = 1.0

   Hot Rod Average Burnup = 35 GWD/MTU, F_Q normalization factor = 1.0

   Hot Rod Average Burnup = 55 GWD/MTU, F_Q normalization factor = 0.9

   Hot Rod Average Burnup = 62 GWD/MTU, F_Q normalization factor = 0.8

2. To account for fuel pellet thermal conductivity degradation, the allowed $F_{\Delta H}$ peaking factors are subject to these normalization factors (interpolation allowed):

   Hot Rod Average Burnup = 0 GWD/MTU, $F_{\Delta H}$ normalization factor = 1.0

   Hot Rod Average Burnup = 35 GWD/MTU, $F_{\Delta H}$ normalization factor = 1.0

   Hot Rod Average Burnup = 55 GWD/MTU, $F_{\Delta H}$ normalization factor = 0.95

   Hot Rod Average Burnup = 62 GWD/MTU, $F_{\Delta H}$ normalization factor = 0.9

3. Analysis was originally performed at 3445 MWt (3411 MWt plus 1% for conservatism). However, 1% for heat balance error was also added into the analysis, so it remains bounding for the MUR (3479 MWt). An MUR uprate evaluation was performed at 3469 MWt (101.7% of 3411 MWt) plus 0.3% uncertainty to derive the PCT penalty included in Table 15-82.

4. To account for fuel pellet thermal conductivity degradation, the allowed P_{HA} peaking factors are subject to these normalization factors (interpolation allowed; extrapolation beyond 59,615 MWD/MTU is acceptable, provided the individual fuel rod burnups remain within the licensed limit of 62,000 MWD/MTU):

   Assembly Average Burnup = 0 MWD/MTU, P_{HA} normalization factor = 1.0

   Assembly Average Burnup = 33,654 MWD/MTU, P_{HA} normalization factor = 1.0

   Assembly Average Burnup = 52,885 MWD/MTU, P_{HA} normalization factor = 0.95

   Assembly Average Burnup = 59,615 MWD/MTU, P_{HA} normalization factor = 0.9

Table 15-72. Rod Census Used in Best-Estimate Large Break LOCA Analysis

Rod Group	Power Ratio (Relative to HA Rod Power)	% of Core
1	1.0	10
2	0.912	10
3	0.853	10
4	0.794	30
5	0.726	40

Table 15-73. Fuel Assembly Isotopic Radioactivity Levels (Alternative Source Term Analysis of the Catawba Design Basis Locked Rotor and Rod Ejection Accidents)

Noble Gases		Halogens		Alkali Metals	
Radioisotope	**Core Activity (Curies)**	**Radioisotope**	**Core Activity (Curies)**	**Radioisotope**	**Core Activity (Curies)**
Kr83m	1.27E+05	Br83	1.27E+05	Rb86	1.68E+03
Kr85m	2.85E+05	Br85	2.85E+05	Rb88	8.48E+05
Kr85	7.31E+03	Br87	4.72E+05	Rb89	1.13E+06
Kr87	5.86E+05	I130	2.52E+04	Rb90	1.07E+06
Kr88	8.29E+05	I131	7.52E+05	Cs134	1.91E+05
Kr89	1.07E+06	I132	1.11E+06	Cs136	4.16E+04
Xe131m	9.63E+03	I133	1.60E+06	Cs137	9.15E+04
Xe133m	4.88E+04	I134	1.86E+06	Cs138	1.59E+06
Xe133	1.57E+06	I135	1.52E+06	Cs139	1.51E+06
Xe135m	3.20E+05				
Xe135	4.14E+05				
Xe137	1.48E+06				
Xe138	1.52E+06				

Table 15-74. Parameters for the Steam Generator Tube Rupture Supplemental Offsite Dose Analysis

1.	Data pertaining to the radioactive source term		
	a.	Equilibrium reactor coolant specific activity (μCi/gm DEI)	0.46
	b.	Transient reactor coolant specific activity(μCi/gm DEI)	26
2.	Data and assumptions pertaining to transport and release of radioactivity		
	a.	Power level (MWt)	3479.
	b.	Condenser iodine scrubbing efficiency before unit trip (%)	85
	c.	Time of unit trip (minutes after initiating event)	20
	d.	Offsite power	Lost at trip
	e.	Time span of primary bypass (minutes after initiating event)	20-25
	f.	Primary bypass fraction	0.12
	g.	Maximum flash fraction	0.14
	h.	Integrated break flow (lbm)	275,000
	i.	Ruptured steam generator steam release after unit trip (lbm)	589,000
	j.	Iodine partition fraction for steam releases	0.01
	k.	Intact steam generator tube leak rate (gpd per SG)	150
	l.	Intact steam generator steam release after unit trip (lbm – all three SGs)	3,230,000
	m.	Steam release rate before trip (lbm/min/SG)	64,110
3.	Dispersion data		
	a.	χ/Q at exclusion area boundary (sec/m^3)	4.78×10^{-4}
	b.	χ/Q at low population zone (sec/m^3)	6.85×10^{-5}
4.	Dose conversion data		
	a.	Method of conversion from activity to dose	R.G. 1.4
	b.	Source of dose conversion factors	
		Whole body radiation dose	R.G. 1.109
		Thyroid radiation dose	ICRP 30
5.	Thyroid radiation doses at the exclusion area boundary (Rem, Note 1)		
	a.	Pre-existent iodine spike	57.0
	b.	Concurrent iodine spike	22.0

Note:

1) Thyroid radiation doses at the exclusion area only are reported. They are limiting with respect to relative margin to the germane regulatory acceptance limits.

Table 15-75. Fission Product Radioactivity Levels in a Mixed Oxide Lead Test Assembly

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

Noble Gases	
Radioisotope	**LTA Activity (Curies)**
Kr83m	6.25E+04
Kr85m	1.17E+05
Kr85	4.36E+03
Kr87	2.18E+05
Kr88	2.93E+05
Kr89	3.33E+05
Xe131m	8.93E+03
Xe133m	4.71E+04
Xe133	1.35E+06
Xe135m	3.38E+05
Xe135	7.73E+05
Xe137	1.23E+06
Xe138	1.06E+06

Halogens	
Radioisotope	**LTA Activity (Curies)**
Br83	6.25E+04
Br85	1.17E+05
BR87	1.61E+05
I130	2.82E+04
I131	7.21E+05
I132	1.08E+06
I133	1.39E+06
I134	1.48E+06
I135	1.31E+06

Alkali Metals	
Radioisotope	**LTA Activity (Curies)**
Rb86	1.15E+03
Rb88	3.02E+05
Rb89	3.82E+05
Rb90	3.21E+05
Cs134	1.93E+05
Cs136	6.89E+04
Cs137	9.35E+04
Cs138	1.19E+06
Cs139	1.09E+06

Noble Metals	
Radioisotope	**LTA Activity (Curies)**
Mo99	1.24E+06
Mo101	1.19E+06
Mo102	1.20E+06
Tc99m	1.11E+06
Tc101	1.19E+06

Tellurium Group	
Radioisotope	**LTA Activity (Curies)**
Se81	3.36E+04
Se83	3.32E+04
Se83m	2.79E+04
Se84	1.01E+05
Se87	4.97E+04
Sb127	9.00E+04
Sb128	1.63E+04
Sb128m	1.32E+05
Sb129	2.82E+05
Sb130	1.06E+05
Sb130m	2.97E+05
Sb131	5.11E+05
Sb132m	2.74E+05
Te127m	1.30E+04
Te127	8.21E+06
Te129	2.51E+05
Te129m	4.78E+04
Te131	6.00E+05
Te132	1.04E+06
Te133	6.93E+05
Te133m	5.82E+05
Te134	1.04E+06

Alkali Earth Metals	
Radioisotope	**LTA Activity (Curies)**
Sr89	3.37E+05
Sr90	3.25E+04
Sr91	5.51E+05
Sr92	6.45E+05
Sr93	7.90E+05
Ba139	1.14E+06
Ba140	1.09E+06
Ba141	1.05E+06
Ba142	9.44E+05

Lanthanides	
Radioisotope	**LTA Activity (Curies)**
Y90	3.42E+04
Y91	4.86E+05
Y91m	3.20E+05
Y92	6.49E+05
Y93	5.44E+05

Noble Metals

Radioisotope	LTA Activity (Curies)
Tc104	1.16E+06
Ru103	1.24E+06
Ru105	1.04E+06
Ru106	6.94E+05
Ru107	6.71E+05
Rh103m	1.24E+06
Rh105	9.62E+05
Rh106m	2.53E+04
Rh107	6.73E+05
Pd109	4.53E+05
Pd111	6.34E+04
Pd112	2.75E+04

Cerium Group

Radioisotope	LTA Activity (Curies)
Ce141	9.79E+05
Ce143	9.06E+05
Ce144	6.43E+05
Ce145	6.22E+05
Ce146	5.13E+05
Np237	6.95E-02
Np238	1.03E+05
Np239	1.36E+07
Np240	4.60E+04
Pu236	2.40E-01
Pu238	2.08E+03
Pu239	1.29E+03
Pu240	9.77E+02
Pu241	2.69E+05
Pu242	4.96E+00
Pu243	9.63E+05

Lanthanides

Radioisotope	LTA Activity (Curies)
Y94	9.11E+05
Y95	9.78E+05
Zr95	8.90E+05
Zr97	1.05E+06
Nb95	8.81E+05
Nb95m	9.88E+03
Nb97	1.06E+06
La140	1.11E+06
La141	1.07E+06
La142	1.01E+06
La143	9.00E+05
Nd147	4.07E+05
Nd149	2.64E+05
Nd151	1.56E+05
Pm147	9.21E+04
Pm148	1.15E+05
Pm148m	1.85E+04
Pm149	3.90E+05
Pm151	1.56E+05
Sm153	4.66E+00
Sm156	2.36E+04
Eu154	1.26E+04
Eu155	3.99E+03
Eu156	3.45E+05
Eu157	3.48E+04
Pr142	5.81E+04
Pr143	8.55E+05
Pr144	6.49E+05
Pr144m	9.03E+03
Pr145	6.23E+05
Pr146	5.20E+05
Pr147	4.24E+05
Am241	4.40E+02
Am242m	3.56E+01
Am242	2.10E+05
Am243	8.28E+01
Am244	2.03E+05
Cm242	8.48E+04
Cm244	5.41E+03

Table 15-76. Fission Product Radioactivity Levels in a Mixed Oxide Lead Test Assembly (Fuel Handling Accident and Weir Gate Drop

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

Noble Gases

Radioisotope	LTA Activity (Curies)
Kr83m	7.25E+04
Kr85m	1.34E+05
Kr85	1.39E+03
Kr87	2.52E+05
Kr88	3.38E+05
Kr89	3.84E+05
Xe131m	1.07E+04
Xe133m	5.51E+04
Xe133	1.65E+06
Xe135m	3.92E+05
Xe135	6.80E+05
Xe137	1.48E+06
Xe138	1.26E+06

Halogens

Radioisotope	LTA Activity (Curies)
Br83	7.22E+04
Br85	1.33E+05
Br87	1.88E+05
I130	7.51E+03
I131	8.81E+05
I132	1.28E+06
I133	1.64E+06
I134	1.76E+06
I135	1.57E+06

Alkali Metals

Radioisotope	LTA Activity (Curies)
Rb86	3.13E+02
Rb88	3.48E+05
Rb89	4.39E+05
Rb90	3.72E+05
Cs134	1.94E+04
Cs136	3.41E+04
Cs137	2.67E+04
Cs138	1.43E+06
Cs139	1.31E+06

Table 15-77. LOCA Release Fractions

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

	Release Fraction (%)	
	LEU Fuel	**MOX LTA**
Group (Elements)		
Gap Release Phase		
Noble gases (Kr, Xe)	5	7.5
Halogens (Br, I)	5	7.5
Alkali metals (Rb, Cs)	5	7.5
Early In-vessel Release Phase		
Noble gases (Kr, Xe)	95	92.5
Halogens (Br, I)	35	52.5
Alkali metals (Rb, Cs	25	37.5
Tellurium group (Se, Sb, Te)	5	7.5
Alkali earth metals (Ba, Sr)	2	3
Noble metals (Mo, Tc, Ru, Rh, Pd)	0.25	0.375
Cerium group (Ce, Np, Pu)	0.05	0.075
Lanthanides (Y, Zr, Nb, La, Nd, Pm, Sm, Eu, Pr, Am, Cm)	0.02	0.03

Table 15-78. Non LOCA Gap Fractions

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

	Release Fraction (%)	
	LEU Fuel	**MOX LTA**
Group (Elements)		
Locked Rotor Accident, Fuel Handling Accident, and Weir Gate Drop		
Kr-85	10	15
I-131	8	12
Other Noble gases (Kr, Xe)	5	7.5
Other Halogens (Br, I)	5	7.5
Alkali metals (Rb, Cs)	12	18
Rod Ejection Accident		
Noble gases (Kr, Xe)	10	15
Halogens (Br, I)	10	15
Alkali metals (Rb, Cs)	12	18

Table 15-79. Iodine Partition Fractions for Post LOCA ESF Leakage in the Auxiliary Building

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

	Iodine Partition Fraction (%)	
Time span (hours)	**All LEU Fuel**	**With MOX LTAs**
Filtered leak, one VA train unavailable		
0.0 - 2.5	0.100	0.100
2.5 – 72.0	0.022	0.022
72.0 – 720.0	0.010	0.010
Filtered leak, all VA trains available		
0.0 – 2.5	0.100	0.100
2.5 – 72.0	0.028	0.028
72.0 – 720.0	0.010	0.010
Initially unfiltered leak, one VA train unavailable		
0.0 – 2.9	0.010	0.010
2.9 – 72.0	0.010	0.010
72.0 – 720.0	0.010	0.010
Initially unfiltered leak, all VA trains available		
0.0 – 2.9	0.013	0.014
2.9 – 72.0	0.010	0.010
72.0 – 720.0	0.010	0.010
Initially unfiltered leak, ND or NX NX failure, all VA trains available		
0.0 – 2.9	0.100	0.100
2.9 – 72.0	0.024	0.024
72.0 – 720.0	0.010	0.010

Table 15-80. Iodine Partition Fractions for ESF Backleakage to the Refueling Water Storage Tank Following a Design Basis Rod Ejection Accident

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

	Iodine Partition Fraction	
Time span (hours)	**All LEU Fuel**	**With MOX LTAs**
0 – 2	0.000E+00	0.000E+00
2 – 8	3.135E-06	3.200E-06
8 – 10	1.266E-05	1.292E-05
10 - 24	2.332-05	2.379E-05
24 – 96	8.910E-04	9.069E-04
96 – 720	2.415E-02	2.433-02

Table 15-81. Effect of Operation of Unit 1 with four MOX LTAs on Post Accident Radiation Doses

Currently, mixed oxide (MOX) fuel has been retired from use at Catawba Nuclear Station and any further reactor operation of MOX fuel may require a reanalysis of these values.

	TEDE (Rem)		
Design Basis Accident Scenario	**EAB**	**LPZ**	**CR**
Design basis locked rotor accident With all LEU fuel	1.52	0.30	1.19
Unit 1 in operation with four MOX LTAs (Note 1)	1.02	0.26	0.80
Regulatory limit	2.5	2.5	5
Design basis rod ejection accident With all LEU fuel	4.75	2.82	2.70
Unit 1 in operation with four MOX LTAs (Note 1)	3.96	2.75	2.32
Regulatory limit	6.3	6.3	5
	6.05	3.08	2.22
Design basis LOCA With all LEU fuel	Deleted Per 2007 Update		
Unit 1 in operation with four MOX LTAs (Note 2)	6.12	3.12	2.23
Regulatory limit	25	25	5
Design basis fuel handling accident With all LEU fuel	1.6	Note 3	2.3
Unit 1 in operation with four MOX LTAs (Notes 2, 4)	2.3	Note 3	2.1
Regulatory limit	6.3	6.3	5
Design basis weir gate drop With all LEU fuel	2.9	Note 3	3.5
Unit 1 in operation with four MOX LTAs (Notes 2, 4)	3.5	Note 3	3.3
Regulatory limit	6.3	6.3	5

Notes on Table 15-81

1. A separate set of analyses of radiological consequences of the design basis locked rotor and rod ejection accident was completed for each of the two nuclear units at Catawba.

Only Unit 1 at Catawba is in operation with the four MOX LTAs. Therefore, radiation doses following the design basis locked rotor and rod ejection accidents involving the MOX LTAs are reported for Unit 1 only. For the locked rotor and rod ejection accidents involving an all LEU core, the limiting radiation doses are associated with Unit 2.

2. One set of analyses of radiological consequences of the design basis LOCA, fuel handling accident, and weir gate drop that are bounding for both nuclear units at Catawba was completed.
3. TEDEs at the LPZ are not reported for the design basis fuel handling accident and weir gate drop. In every case, they are bounded by the TEDEs at the EAB.
4. For the fuel handling accidents and weir gate drop involving MOX LTAs, it was assumed that both control room outside air intakes were open. For the fuel handling accidents and weir gate drop involving all LEU fuel, it was assumed that only one control room intake was open.

Table 15-82. Summary of Licensing Basis LOCA PCT Results, Including PCT Assessments

Description	PCT (°F)	Reference
Best Estimate Large Break LOCA; CQD		
Analysis of Record PCT (Reflood 2) [See Table 15-70]	2028	81
PCT Assessments		
Decay heat in Monte Carlo calculations	8	104
MONTECF power uncertainty correction	20	105
Safety Injection temperature range	59	82
Input error resulting in an incomplete solution matrix	25	106
Revised blowdown heatup uncertainty distribution	5	107
Vessel unheated conductor noding	0	108
Revised algorithm for average fuel temperature	0	108
Peak transient FQ = 2.7 in bottom third of core	0	109
Change from PAD 3.4 to PAD 4.0	-75	109
Fuel Thermal Conductivity Degradation with Peaking Factor Burndown	15	109
Revised Heat Transfer Multiplier Distribution	-85	111
HOTSPOT Clad Burst Strain Error	70	112
Unit 1 MUR Uprate to 101.7% of 3411MWt	16	113
Unit 1 Current Licensing Basis LBLOCA PCT Including Assessments	2086	113
Unit 2 Current Licensing Basis LBLOCA PCT Including Assessments	2070	112
Small Break LOCA; NOTRUMP		
Unit 1 Analysis of Record PCT (2-inch break) [See Table 15-61]	1323	110
PCT Assessments		
None	0	110
Unit 1 Current Licensing Basis SBLOCA PCT Including Assessments	1323	109
Unit 2 Analysis of Record PCT (4-inch break) [See Table 15-39]	1243	110

PCT Assessments		
None	0	110
Unit 2 Current Licensing Basis SBLOCA PCT Including Assessments	1243	109

Table 15-83. Dose Equivalent Xenon-133 Noble Gas Specific Activities in the Reactor Coolant

Radioisotope	Specific Activity (µCi/gm)	FGR No. 12, Table III.1 DCFs (Sv-s/(Bq-m^3))	DEX Specific Activity (µCi/gm)
KR-85M	2.06E+00	7.48E-15	9.88E+00
KR-85	7.52E+00	1.19E-16	5.74E-01
KR-87	1.34E+00	4.12E-14	3.54E+01
KR-88	3.71E+00	1.02E-13	2.43E+02
XE-131M	2.27E+00	3.89E-16	5.65E-01
XE-133M	1.75E+01	1.37E-15	1.54E+01
XE-133	2.78E+02	1.56E-15	2.78E+02
XE-135M	4.95E-01	2.04E-14	6.47E+00
XE-135	7.42E+00	1.19E-14	5.66E+01
XE-138	6.59E-01	5.77E-14	2.44E+01
Total DEX			6.70E+02

Table 15-84. Dose Equivalent Iodine-131 Noble Gas Specific Activities in the Reactor Coolant

Radioisotope	Specific Activity (µCi/gm)	FGR No. 11, Table 2.1 DCFs (Sv/(Bq)	DEI Specific Activity (µCi/gm)
I-131	7.56E-01	8.89E-09	7.56E-01
I-132	2.72E-01	1.03E-10	3.15E-03
I-133	1.21E+00	1.58E-09	2.15E-01
I-134	1.81E-01	3.55E-11	7.25E-04
I-135	6.65E-01	3.32E-10	2.49E-02
Total DEI			1.00E+00