Appendix 15A. Tables

Table 15-1. Deleted Per 1993 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. short term core cooling 2. long 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	
15.4.4	Startup of an Inactive Reactor 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. DNB analysis
15.6.5	Loss of Coolant Accident	1. DECLG C_D=1.0, Reference Transient 2. 1.5 inch SBLOCA 3. 2 inch SBLOCA 4. 3 inch SBLOCA 5. 4 inch SBLOCA

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
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, 15.1.5, 15.2.7(1), 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(1), 15.3.1, 15.3.2, 15.3.3, 15.4.1, 15.4.2, 15.4.3, 15.4.8, 15.6.1, 15.6.3
Deleted Per 2006 Update	
Deleted Per 2008 Update	
CASMO 3/SIMULATE-3P or 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
NOTRUMP	15.6.5
WCOBRA/TRAC	15.6.5
LOCTA-IV	15.6.5
LOTIC	15.6.5
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 ID	MTC (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm)	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (%)	Feedwater Temp. (°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 [16]	561	2208	16	Note [15]
15.1.5	1	Note [9]	NA	-3.5	0	371,796	561	2198	16	Note [15]
15.1.5	2	Note [9]	NA	-3.5	0	371,796	561	2198	16	Note [15]
15.1.5	3	NA	Note [21]	-1.2	3469	388,000	585.1	2250	46	443
15.1.5	4	NA	Note [21]	-1.2	3469	388,000	585.1	2250	46	443
15.2.3	2	NA	Note [6]	-0.9	3479	420,000	589.1	2310	64	440
15.2.3	1	NA	Note [6]	-0.9	3479	373,596	589.1	2280	64	440
15.2.6		Note [14]	NA	Note [14]	3479	373,596	589.1	2250	55	440
15.2.7	1	NA	Note [6]	-0.9	3469	388,000	585.1	2250	46	443
15.2.7	2	NA	Note [6]	-0.9	3479	379,464	589.1	2208	46	440
15.2.8	1	NA	Note [6]	-0.9	3479	Note [18]	589.1	2208	46	440
15.2.8	2	NA	Note [6]	-0.9	3469	388,000	585.1	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	379,464	589.1	2310	64	443
15.3.3	2&3	NA	Note [6]	-0.9	3469	388,000	585.1	2250	46	442

FSAR Section	Case ID	MTC (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm)	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (%)	Feedwater Temp. (°F)
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	336
15.4.2	2	NA	Note [6]	Note [4]	273	375,669	563.8	2250	37	333
15.4.2	3	NA	Note [6]	Note [4]	1734	384,120	571.0	2250	31	382
15.4.2	4	NA	Note [6]	Note [4]	3399.6	384,120	584.5	2250	45.4	438
15.4.2	5	NA	Note [6]	Note [4]	3469	388,000	585.1	2250	46	440
15.4.3a, b		NA	Note [6]	-0.9	3469	388,000	585.1	2250	46	443
Deleted Row per 2017 Update										
15.4.3c		NA	NA	NA	3411	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 ID	MTC (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm)	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (%)	Feedwater Temp. (°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 [6]	NA	-1.2	3479	373,596	581.1	2310	64	440
15.6.3	2	Note [6]	NA	-0.9	3469	388,000	585.1	2250	46	442
15.6.5	1	NA	Note 11	Note 11	3445[20]	Note [19]	587.5	2250	55	442
15.6.5	2	NA	Note 11	Note 11	3479	Note [19]	585.1	2250	55	442
15.6.5	3	NA	Note 11	Note 11	3479	Note [19]	585.1	2250	55	442
15.6.5	4	NA	Note 11	Note 11	3479	Note [19]	585.1	2250	55	442
15.6.5	5	NA	Note 11	Note 11	3479	Note [19]	585.1	2250	55	442

FSAR Section	Case ID	MTC (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm)	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (%)	Feedwater Temp. (°F)

Notes:

1. Deleted per 1998 update.
2. -0.9 pcm/°F at HFP to -1.20 pcm/°F at HZP
3. -1.04 pcm/°F at HFP to -1.325 pcm/°F at HZP
4. -1.20 pcm/°F at HFP to -1.50 pcm/°F at HZP.
5. Deleted per 1998 update.
6. The most positive MTC (implemented as a least positive or most negative MDC) allowed by the Technical Specifications was used.
7. Deleted per 1998 update.
8. The McGuire 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. Deleted per 1998 update.
13. Deleted per 1998 update.
14. The results of this transient are not sensitive to reactivity feedback assumptions.
15. Main feedwater temperature is 60°F. Auxilliary feedwater temperature is 32°F.
16. 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.
17. Deleted Per 2012 Update.
18. 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.
19. 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 reduced RCS flow.

FSAR Section	Case ID	MTC (pcm/°F)	Moderator Density Coefficient (%Δk/k/g/cc)	Doppler Coefficient (pcm/°F)	Initial Core Output (MWt)	RCS Flow (gpm)	Vessel T_{avg} (°F)	Pzr Press. (psia)	Pzr Liquid Inventory (%)	Feedwater Temp. (°F)

20. Analysis was originally performed at 3445 MWt (3411 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 15.61.
21. Based on MTC = -24 pcm/°F

Table 15-5. Deleted Per 1992 Update

Table 15-6. Deleted Per 1992 Update

Table 15-7. 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]
15.4.6	Note [1]
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, Note [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 FSAR Section 15.6.3, this note only applies to the dose analysis.
2. Reactor trip was not necessary to analyze transient.

Table 15-8. 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[1]
Overpower ΔT	Variable see Figure 15-1	1.5[1]
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[1]
Safety injection	Not applicable	2.0

Note:

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 15-9. Deleted Per 1992 Update

Table 15-10. Reactor Core Iodine and Noble Gas Source Terms Gap Release Fractions

Nuclide	Core Inventory[1] (Curies)	Locked Rotor[2] Gap Release Fractions	Rod Ejection[2] Gap Release Fractions
I-130	2.52E+04	5%	10%
I-131	7.52E+05	8%	10%
I-132	1.11E+06	5%	10%
I-133	1.60E+06	5%	10%
I-134	1.86E+06	5%	10%
I-135	1.52E+06	5%	10%
Kr-83m	1.27E+05	5%	10%
Kr-85m	2.85E+05	5%	10%
Kr-85	7.31E+03	10%	10%
Kr-87	5.86E+05	5%	10%
Kr-88	8.29E+05	5%	10%
Kr-89	1.07E+06	5%	10%
Xe-131m	9.63E+03	5%	10%
Xe-133m	4.88E+04	5%	10%
Xe-133	1.57E+06	5%	10%
Xe-135m	3.20E+05	5%	10%
Xe-135	4.14E+05	5%	10%
Xe-137	1.48E+06	5%	10%
Xe-138	1.52E+06	5%	10%
Rb-86	1.68E+03	12%	12%
Rb-88	8.48E+05	12%	12%
Rb-89	1.13E+06	12%	12%
Rb-90	1.07E+06	12%	12%
Cs-134	1.91E+05	12%	12%
Cs-136	4.16E+04	12%	12%
Cs-137	9.15E+04	12%	12%
Cs-138	1.59E+06	12%	12%
Cs-139	1.51E+06	12%	12%
Br-83	1.27E+05	5%	10%
Br-85	2.85E+05	5%	10%
Br-87	4.72E+05	5%	10%

Note:

1. Based on fuel assembly burnup to 62,000 MWD/MTU
2. Based upon Regulatory Guide 1.183

Table 15-11. Normal Reactor Coolant Specific Activities for Iodine and Noble Gas Isotopes

Nuclide	Specific Activity[1] (μCi/g)
I-131	0.66
I-132	0.24
I-133	1.1
I-134	0.16
I-135	0.58
Xe-131m	1.9
Xe-133m	3.1
Xe-133	281.0
Xe-135m	0.7
Xe-135	6.3
Xe-138	0.7
Kr-83m	0.0.
Kr-85m	2.1
Kr-85	8.8
Kr-87	1.2
Kr-88	3.7
Kr-89	0.0

Note:

1. Reactor coolant concentrations at equilibrium assuming Technical Specification Iodine activity.

Table 15-12. Environmental Consequences

		Rem Total Effective Dose Equivalent (TEDE)		
Accident	**FSAR Section**	**Exclusion Area Boundary**	**Low Population Zone**	**Control Room**
Main Steam Line Break	15.1.5			
Pre-Existing Iodine Spike		0.23 (25.0)	0.03 (25.0)	2.12 (5.0)
Coincident Iodine Spike		0.25 (2.5)	0.08 (2.5)	3.74 (5.0)
Locked Rotor	15.3.3			
Loss of Offsite Power		1.73 (2.5)	0.20 (2.5)	2.91 (5.0)
Offsite Power Available		1.68 (2.5)	0.17 (2.5)	1.69 (5.0)

		Rem Total Effective Dose Equivalent (TEDE)		
Accident	**FSAR Section**	**Exclusion Area Boundary**	**Low Population Zone**	**Control Room**
Rod Ejection	15.4.8	2.52 (6.3)	0.67 (6.3)	4.05 (5.0)

		Rem Total Effective Dose Equivalent (TEDE)		
Accident	**FSAR Section**	**Exclusion Area Boundary**	**Low Population Zone**	**Control Room**
Instrument Line Break	15.6.2			
Pre-Existing Iodine Spike		1.12	0.10	0.45
		(2.5)	(2.5)	(5.0)
Coincident Iodine Spike		0.41	0.04	0.14
		(2.5)	(2.5)	(5.0)
Steam Generator Tube Rupture	15.6.3			
Pre-Existing Iodine Spike		4.78	0.67	2.97
		(25.0)	(25.0)	(5.0)
Coincident Iodine Spike		2.08	0.42	1.61
		(2.5)	(2.5)	(5.0)

		Rem Total Effective Dose Equivalent (TEDE)		
Accident	FSAR Section	Exclusion Area Boundary	Low Population Zone	Control Room
Loss of Coolant Accident	15.6.5	12.25	2.23	4.86
		(25.0)	(25.0)	(5.0)

Accident	FSAR Section	Rem Total Effective Dose Equivalent (TEDE)

		Exclusion Area Boundary	Low Population Zone	Control Room
Waste Gas Decay Tank Failure	15.7.1	0.25	0.02	0.01
		(0.5)	(0.5)	(5.0)
Liquid Storage Tank Failure	15.7.2	1.89	0.17	0.61
		(2.5)	(2.5)	(5.0)

		Rem Total Effective Dose Equivalent (TEDE)		
Accident	FSAR Section	Exclusion Area Boundary	Low Population Zone	Control Room
Cask Drop in Pit	15.7.4	0.01	0.0009	0.0006
		(6.3)	(6.3)	(5.0)

Accident	**FSAR Section**	**Rem Total Effective Dose Equivalent (TEDE)**		
		Exclusion Area Boundary	**Low Population Zone**	**Control Room**
Fuel Handling Accident				
Inside Containment	15.7.4.1	3.25	0.29	3.86
		(6.3)	(6.3)	(5.0)
Deleted row per 2017 update				
Dropped Weir Gate Inside SFP Building	15.7.4.3	6.16	0.56	3.25
		(6.3)	(6.3)	(5.0)

		2-hr Dose at 2500 ft. Exclusion Area Boundary	30 day Dose at 29000 ft. Low Population Zone	
Accident	FSAR Section	Exclusion Area Boundary	Low Population Zone	Control Room
Tornado Generated Missile Accident	15.10.3	0.72	0.71	1.85
		(25.0)	(25.0)	(5.0)

		2-hr Dose at 2500 ft. Exclusion Area Boundary	
Accident	FSAR Section	Whole Body	Thyroid
Cask Drop Accident	15.7.4.5	0.01	0.2
		(2.5)	(30.0)

Table 15-13. 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
	Over power 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
	Low steam line pressure setpoint reached	NA
	Steam Line Isolation	NA
	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

Accident	Event	Time (sec.)
	Peak heat flux occurs	119
	NV injection lines purged of unborated water	119
	One train of SI fails	119
	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
	Deleted per 2015 update	
3. CFM at hot full power	Break occurs	0
	High flux trip setpoint reached	12.7
	Reactor trip occurs due to high flux trip	13.2
	Peak reactor power occurs	13.3
	Turbine Trip occurs due to reactor trip	13.5
	Loss of offsite power occurs on turbine trip	13.5
	RCPs trip due to loss of offsite power	13.5
4. DNB at hot full power	Break occurs	0

Accident	Event	Time (sec.)
	OPΔT trip setpoint reached	11.6
	Reactor trip occurs due to OPΔT trip	12.1
	Peak reactor power occurs	12.3
	Turbine trip occurs due to reactor trip	12.4
	Loss of offsite power occurs on turbine trip	12.4
	RCPs trip due to loss of offsite power	12.4
	MDNBR occurs	13.2

Table 15-14. Parameters for Main Steam Line Break Dose Analysis

1. Failed fuel (%)	0
2. Iodine spike values for each case	
a. Pre-existing spike	60
b. Coincident spike	500
3. Control Room Data	
a. Control room volume (ft^3)	107,000
b. Control room pressurization (cfm)	1800
c. In-leakage before pressurization (cfm)	625
d. In-leakage after pressurization (cfm)	210
e. Control room filter efficiencies (% particulates, elementals/organic)	99, 98
4. Partitioning fraction	0.01
5. Iodine fractions (% elemental, organic)	97, 3
6. Maximum primary to secondary leak rate (gpd)	389
7. Letdown flow (gpm)	125
8. Reactor coolant system leakage (gpm)	11
9. Total steam release from the faulted steam generator (lbm)	2.34E+05
10. Total steam release from the intact steam generators (lbm)	1.85E+06

Table 15-15. Deleted per 2014 Update

Table 15-16. 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	0.0
	Pressurizer PORVs lift	4.3
	Steam Safety Valves lift	6.7
	Overtemperature ΔT setpoint reached	13.8
	Control rod insertion begins	15.3
	Peak secondary system pressure occurs	18.4
2. Maximum Primary System Pressure Case	Turbine Trip, loss of main feed flow	0.0
	High pressurizer pressure setpoint reached	5.6
	Control rod insertion begins	7.6
	Steam Safety Valves lift	8.0
	Pressurizer Safety Valves lift	8.2
	Peak primary system pressure occurs	8.7
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 coastdown	0.1
	Rods begin to drop	0.6
	Peak water level in pressurizer occurs	3
	Flow from two motor driven auxiliary feedwater pumps is started	60
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~ 600
Loss of Main Feedwater		
1. Short-Term Core Cooling Case	Main feedwater flow stops	1.0
	Pressurizer PORVs begin cycling	24.6
	Low-low steam generator level reactor trip reached	56.4
	Minimum DNBR occurs	58.0
	Rods begin to drop	58.4
2. Long-Term Core Cooling Case	Main feedwater flow stops	0.01

Accident	Event	Time (Sec)
	Pressurizer PORVs begin cycling	38.2
	Low-low steam generator level reactor trip reached	56.6
	Rods begin to drop	58.6
	Steam safety valves lift	60.1
	Auxiliary feedwater flow on	116.6
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~1749
Feedwater System Pipe Break		
	Feedwater line break to SG B	0
	Safety injection on high containment pressure	10.1
	Reactor trip on high containment pressure SI	10.1
	Reactor coolant pumps tripped	10.1
	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 15-17. 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	1.5
	Rods begin to drop	2.5
	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)	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 (Peak RCS Pressure)	Rotor on one pump locks	1.0
	Low flow reactor trip setpoint reached	1.06
	Rods begin to drop	2.06
	Maximum RCS pressure occurs	5.6

Table 15-18. Parameters for Locked Rotor Dose Analysis

1.	Data and assumptions used to estimate radioactive sources from postulated accidents.	
	a. Failed fuel for Loss of Offsite Power (LOOP) scenario (%)	6
	b. Failed fuel for Offiste Power Available (OPA) scenario (%)	1.5
	c. Reactor core inventory	Table 15-10
	d. Concurrent Iodine Spiking Factor	335
	e. Iodine fractions (% elemental, organic)	97, 3
	f. Reactor turbine and trip (minutes)	0
2.	Data and assumptions used to estimate activity released	
	a. Total Steam Release from the Faulted Steam Generator (LOOP) (lbm)	2.95E+05
	b. Total Steam Release from the Intact Steam Generators (LOOP) (lbm)	8.85E+05
	c. Total Steam Release from the Faulted Steam Generator (OPA) (lbm)	3.42E+05
	d. Total Steam Release from the Intact Steam Generator (OPA) (lbm)	1.02E+06
	e. Control room volume (ft^3)	107,000
	f. Control room pressurization (cfm)	1800
	g. Control room in-leakage before pressurization (cfm)	500
	h. Control room in-leakage after pressurization (cfm)	210
	i. Control room filter efficiencies (% particulates, elemental/organics)	99, 98
	j. Steam generator partitioning fraction	0.01
3.	Dispersion data	
	a. Distance to exclusion area boundary (m)	762
	b. Distance to low population zone (m)	8850
	c. X/Q at exclusion area boundary (sec/m^3)	9.0E-04
	d. X/Q at low population zone (sec/m^3)	8.0E-05
4.	Dose data	Table 15-12

Table 15-19. 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 Valves Lift	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

Accident	Event	Time (sec.)
	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
	RCCA Completely Withdrawn	4.2
	OT Δ T 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	< 998
b) Dilution during power operation (automatic rod control)	Dilution begins	-
	Rod insertion limit alarm setpoint reached	0
	Operator terminates dilution	<1554
2. Dilution during startup	Dilution begins	-
	Reactor trip setpoint reached	0
	Operator terminates dilution	< 998
Deleted Per 2008 Update.		
Rod Cluster Control Assembly Ejection		
1. Beginning of Cycle, 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 15-20. 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	4.90	19.60	4.84	20.78
Number of operational pumps	4	3	4	3
Max. fuel pellet average temperature, °F	3327	1621	2818	1299
Max. fuel center temperature, °F	5068	2062	4326	1640
Max. clad temperature, °F	798	741	1296	768
Max. fuel stored energy, cal/gm	147	61	132	47
% Failed fuel	<22	<22	<22	<22

Table 15-21. Parameters for Rod Ejection Accident Dose Analysis

1.	Data and assumptions used to estimate radioactive sources from postulated accidents	
	a. Failed fuel (%)	22
	b. Reactor core inventory	Table 15-10
	c. Iodine fractions (% elemental, organic)	97, 3
	d. Reactor and turbine trip (minutes)	0
2.	Data and assumptions used to estimate activity released	
	a. Control Room volume (ft^3)	107,000
	b. Control Room pressurization (cfm)	1800
	c. Control room in-leakage before pressurization (cfm)	500
	d. Control room in-leakage after pressurization (cfm)	210
	e. Control room filter efficiency (% particulates, elemental/organics)	99, 98.05
	f. Steam generator iodine partitioning fraction	0.01
3.	Dispersion data	
	a. Distance to exclusion area boundary (m)	762
	b. Distance to low population zone (m)	8850
	c. χ/Q at exclusion area boundary (sec/m^3)	9.0E-04
	d. χ/Q at exclusion area boundary (sec/m^3)	
	0-8 hours	8.0E-05
	8-24 hours	5.2E-06
	1-4 days	1.7E-06
	4-30 days	3.7E-07
4.	Dose data	Table 15-12

Table 15-22. Deleted Per 1998 Update.

Table 15-23. Time Sequence of Events For Incidents Which Cause A Decrease In Reactor Coolant Inventory

Accident	Event	Time (sec)
Inadvertent Opening of a Pressurizer Safety Valve	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
Steam Generator Tube Rupture (Dose Analysis)	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 isolate CA flow to the ruptured SG	1290
	Operators identify ruptured SG and close ruptured SG MSIV	2100
	Operators close failed open steam line PORV	3362
	Operators begin RCS cooldown with operable SG PORVs	5754
	Operators close operable steam line PORVs	6325
	Operators open pressurizer PORV to depressurize RCS	6850
	Break flow terminated	6931
(DNB Analysis)	Double ended tube rupture occurs	1.0
	Reactor trip/turbine trip on oTΔT	319.0
	Reactor coolant pumps lost	319.0
	MDNBR occurs	320.9

Table 15-24. Parameters for Steam Generator Tube Rupture Dose Analysis

1. Failed fuel (%)	0
2. Reactor and turbine trip (minutes)	20
3. Iodine spike values for each case	
a. Pre-existing spike	60
b. Coincident spike	335
4. Control Room Data	
a. Control room volume (ft^3)	107,000
b. Control room pressurization (cfm)	1800
c. In-leakage before pressurization (cfm)	500
d. In-leakage after pressurization (cfm)	210
e. Control room filter efficiencies (% particulates, elementals/organic)	99, 98
5. Partitioning fraction (steam generator/condenser)	0.01,0.15
6. Iodine fractions (% elemental, organic)	97, 3
7. Maximum primary to secondary leak rate (gpd)	389
8. Letdown flow (gpm)	125
9. Reactor coolant system leakage (gpm)	11
10. Total steam release from the ruptured steam generator (lbm)	2.40E+05
11. Total steam release from the intact steam generator (ibm)	1.75E+05

Table 15-25. Deleted Per 1998 Update

Table 15-26. Deleted Per 1998 Update

Table 15-27. Deleted Per 1996 Update

Table 15-28. Deleted Per 1996 Update

Table 15-29. Deleted Per 1996 Update

Table 15-30. Deleted Per 1996 Update

Table 15-31. Deleted Per 1998 Update

Table 15-32. Deleted Per 1998 Update

Table 15-33. Parameters for Minimum Safeguards (Design Basis) LOCA Dose Analysis

1.	Data and assumptions used to estimate radioactive source from postulated accidents	
	a. Power Level (MW_{th})	3479
	b. Failed fuel (%)	100
2.	Iodine Species Breakdown (% particulate, elemental, organic)	
	a. Containment Model	95, 4.85, 0.15
	b. ECCS Model	0, 97, 3
3.	Data and assumptions used to estimate activity released	
	a. Containment Free Volume (including ice condensers)	
	1. Upper containment volume (ft^3)	827,000
	2. Lower containment volume (ft^3)	391,000
	3. Total containment free volume (ft^3)	1,120,000
	b. Annulus Volume - half of the volume credited (427,000 ft^3)	213,000
	c. Control Room Volume (ft^3)	107,000
	d. Containment Leak Rate (percent of containment volume per day)	
	1. 0 ≤ t 24 hrs	0.3
	2. t > 24 hrs	0.15
	e. Bypass Leakage Fraction	0.07
	f. VE start time (seconds)	39
	g. Annulus vacuum established (seconds)	71
4.	Equipment Hatch Release (sscm)	500
5.	ECCS back-leakage to the Auxiliary Building (gpm)	0.9
6.	ECCS back-leakage to the FWST (gpm)	10
7.	Control Room In-leakage Data (One train of VC)	
	a. Time of control room pressurization (seconds)	30
	b. Control room in-leakage before pressurization (cfm)	625
	c. Control room in-leakage after pressurization (cfm)	210
8.	Control Room Ventilation Data	
	a. VC fan flow (cfm)	1800
	b. VC iodine filter efficiency (% particulates, elemental and organic)	99, 98
9.	Annulus Ventilation Data	
	a. VE fan flow (cfm)	7200
	b. VE Iodine filter efficiency (% particulates, elemental and organic)	98, 91
10.	Spray Removal Data	
	a. NS Start Time (minutes)	80
	b. Auxiliary Spray Start Time (minutes)	N/A
	c. Spray credit ceases (hours)	24
	d. Spray Decontamination Factors	
	1. Particulate	50
	2. Elemental	200
	3. Organic (Spray credit not taken for organic Iodine)	N/A
11.	Doses	Table 15-12

Table 15-34. Deleted Per 2009 Update

Table 15-35. Source Term Inventory and Gap Fractions Assumed for Fuel Handling and Tornado Missile Accidents

Nuclide	Assembly Inventory (curies)	Gap Release Fractions[1]	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+03
I-131	8.09E+05	0.08	6.47E+04
I-132	1.18E+06	0.05	5.90E+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	0.05	1.49E+04
Kr-85	7.48E+03	0.10	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	7.95E+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 Release Fractions[1]	Gap Inventory (curies)
Cs-139	1.58E+06	0.12	1.90E+05

Note:

NRC Assumption in Regulatory Guide 1.183

For 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 (References 1 and 2). A maximum of 25 fuel rods, per assembly, shall be allowed to exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, in accordance with the license amendment request submitted by letter dated July 15, 2015.

Table 15-36. Deleted Per 1992 Update

Table 15-37. Parameters for Postulated Instrument Line Break Accident Analysis

1. Failed fuel (%)	0
2. Isolation of Instrument Line (minutes)	30
3. Iodine spike values for each case	
a. Pre-existing spike	60
b. Coincident spike	335
4. Control Room Data	
a. Control room volume (ft^3)	107,000
b. Control room pressurization (cfm)	1800
c. In-leakage before pressurization (cfm)	625
d. In-leakage after pressurization (cfm)	210
e. Control room filter efficiencies (% particulates, elementals/organic)	99, 98
5. Partitioning fraction	0.1
6. Iodine fractions (% elemental, organic)	97, 3
7. Assumed ILB flow rate (gpm)	150
8. Reactor coolant system leakage (gpm)	11

Table 15-38. Deleted Per 1998 Update.

Table 15-39. Parameters Used to Evaluate Tornado Missile Impact On Spent Fuel

1. Meteorology	
a. Offsite atmospheric dilution for tornado conditions	8.1E-5 s/m^3
2. Spent Fuel Radioactivity Bases	
a. Number of fuel assemblies damaged	38
b. Conservative case maximum assembly inventory	See Table 15-35
c. Decay period	
1) 8 Assemblies	16 days
2) 30 Assemblies	295 days
3. Iodine Partition Factor	200
4. Effective Iodine Composition Fractions	
a. Elemental Iodine	57%
b. Organic Iodine	43%
5. Ventilation Credit Assumed	
a. Duration VF is in filter mode or secured	27 days
b. Time required to start VC	30 minutes
c. Rate of Unfiltered Control Room Inleakage	
1. Pre-pressurization	500 cfm
2. Pressurization	210 cfm
d. VC Air Flow Rate	1800 cfm

Table 15-40. Parameters Used to Evaluate LOCA During Lower Containment Pressure Relief

1. REACTOR COOLANT RADIOACTIVITY INVENTORY BASES	
a. Iodine concentrations	60 μCi/gm Thyroid Dose Equivalents of I-131
b. Noble gas concentrations	100/ĒμCi/gm
c. Reactor coolant mass	447,274 lbm
2. RADIOACTIVITY RELEASE BASES	
a. Lower containment air mass (for dilution)	37,360 lbm
Basis: Active volume = 368,000 ft3	
Temperature = 250ºF	
b. Lower containment mass release	19 lbm
Basis: LOCA overpressure = 12 psig	
VQ valve isolation = 4 sec	
c. Filtration	None

Table 15-41. Deleted Per 2015 Update

Table 15-42. Input Parameters Used in the SBLOCA Analyses

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 15-137
Fuel assembly array	17x17 RFA
Nominal cold leg accumulator water volume (ft^3/accumulator)	950
Nominal cold leg accumulator tank volume (ft^3/acumulator)	1363
Minimum cold leg accumulator gas pressure (psia)	570
Cold leg accumulator temperature (°F)	125
Pumped safety injection flow	See Table 15-50
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 15-43. Deleted Per 2001 Update

Table 15-44. Deleted Per 2001 Update

Table 15-45. Minimum Injected ECCS Flows Assumed in LBLOCA Analyses.

One Train Operational

RCS Pressure (psia)	High-Head SI (gpm)	Intermediate-Head (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 15-46. 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)	RWST minimum[3]
Refueling Water Storage Tank (RWST) to Valve FW28	RWST minimum[3]
RWST to IHSI suction	RWST minimum[3]
RWST to Valve NV223	RWST minimum[3]
Normal Containment Spray Discharge	RWST minimum[3]
Containment Spray Suction from RWST	RWST minimum[3]
LHSI Discharge to Aux. Cont. Spray (downstream of isolation MOVs)	350[1]
LHSI Suction from Sump	350[1]
LHSI Suction from Loop C Hot Leg	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 ND58 to Valves NI332A & NI333B) (LHSI Discharge to Valves ND58 & NI136B)	variable[2]
LHSI Discharge to B and C Hot Legs	variable[2]
Valve FW28 to LHSI Suction	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]
Valve NV223 to HHSI Suction	variable[2]
LHSI Discharge to Aux. Cont. Spray (upstream of isolation MOVs)	variable[2]

Volume Grouping	Boron Concentration (ppm)

Note:

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 15-47. Deleted Per 1998 Update

Table 15-48. Deleted Per 1998 Update

Table 15-49. Small Break LOCA Results Fuel Cladding Data

	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

Note:

1. There is no core uncovery for the 1.5 inch case.

Table 15-50. Minimum ECCS Flow Assumed in SBLOCA Analyses

(One Train Operational, Break Backpressure Equal to RCS Presure)

	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 2008 Update.

Table 15-51. Small Break LOCA Time Sequence of Events

	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 15-52. 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 15-53. Key Large Break LOCA Parameters and Initial Transient Assumptions

		Parameter	Initial Transient	Uncertainty or Bias
1.0	**Plant Physical Description**			
a.	Dimensions		Nominal	ΔPCT_{MOD}[1]
b.	Flow resistance		Nominal	ΔPCT_{MOD}
c.	Pressurizer location		Opposite broken loop	Bounded
d.	Hot assembly location		Under limiting location	Bounded
e.	Hot assembly type		17x17 RFA with IFM	Bounded
f.	SG tube plugging level		D5, maximum (10%)	Bounded[4]
2.0	**Plant Initial Operating Conditions**			
	2.1	**Reactor Power**		
	a.	Core average linear heat rate (AFLUX)	Nominal power (3445 MWt)[6]	ΔPCT_{PD}[2]
	b.	Peak linear heat rate (PLHR)	Derived from desired Tech Spec (TS) limit and maximum baseload FQ	ΔPCT_{PD}
	c.	Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$	ΔPCT_{PD}
	d.	Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD}
	e.	Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD}
	f.	Axial power distribution (PBOT, PMID)	Figure 15-244	ΔPCT_{PD}
	g.	Low power region relative power (PLOW)	Minimum (0.2)	Bounded[4]
	h.	Hot assembly burnup	BOL	Bounded
	i.	Prior operating history	Equilibrium decay heat	Bounded
	j.	Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
	k.	HFP boron	800 ppm	Typical

			Parameter	Initial Transient	Uncertainty or Bias
	2.2	Fluid Conditions			
		a.	T_{avg}	Nominal T_{avg} = 587.5°F (Catawba Unit 2)	ΔPCT_{IC}[3]
		b.	Pressurizer pressure	Nominal (2250 psia)	ΔPCT_{IC}
		c.	Loop flow	Minimum (97500 gpm)	ΔPCT_{MOD}[5]
		d.	T_{UH}	Best Estimate	0
		e.	Pressurizer level	Nominal (55% of volume)	0
		f.	Accumulator temperature	Nominal (115°F)	ΔPCT_{IC}
		g.	Accumulator pressure	Nominal (631.5 psig, Catawba units)	ΔPCT_{IC}
		h.	Accumulator liquid volume	Nominal (7106 gal, McGuire units)	ΔPCT_{IC}
		i.	Accumulator line resistance	Nominal (McGuire Unit 2)	ΔPCT_{IC}
		j.	Accumulator boron	Minimum (McGuire units)	Bounded
3.0	**Accident Boundary Conditions**				
	a.	Break location		Cold leg	Bounded
	b.	Break type		Guillotine	ΔPCT_{MOD}
	c.	Break size		Nominal (cold leg area)	ΔPCT_{MOD}
	d.	Offsite power		On (RCS pumps running)	Bounded[4]
	e.	Safety injection flow		Minimum	Bounded
	f.	Safety injection temperature		Nominal (85°F)	ΔPCT_{IC}
	g.	Safety injection delay		Max delay (17 sec)	Bounded

		Parameter	Initial Transient	Uncertainty or Bias
	h.	Containment pressure	Minimum based on WC/T M&E	Bounded
	i.	Single failure	ECCS: Loss of 1 SI train	Bounded
	j.	Control rod drop time	No control rods	Bounded
4.0	**Model Parameters**			
	a.	Critical Flow	Nominal (as coded)	ΔPCT_{MOD}
	b.	Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}
	c.	Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}
	d.	Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}
	e.	Delivery and bypassing of ECCS	Nominal (as coded)	Conservative
	f.	Steam binding/entrainment	Nominal (as coded)	Conservative
	g.	Noncondensable gases/accumulator nitrogen	Nominal (as coded)	Conservative
	h.	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. Analysis was originally performed at 3445 MWt (3411 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 15-61.

Sensitivity analysis concluded loss of offsite power is more limiting than assuming offsite power on (RCS pumps running)

Table 15-54. 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 15-55. Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis

			Parameter	Operating Range
1.0	Plant Physical Description			
	a)	Dimensions		No in-board assembly grid deformation during LOCA + SSE
	b)	Flow resistance		N/A
	c)	Pressurizer location		N/A
	d)	Hot assembly location		Anywhere in core
	e)	Hot assembly type		Fresh 17X17 RFA
	f)	SG tube plugging level		≤ 10% (Catawba 2) and ≤ 5% (McGuire and Catawba 1)
2.0	Plant Initial Operating Conditions			
	2.1	Reactor Power		
		a)	Core avg linear heat rate	Core power ≤ 3445 MWt[4]
		b)	Peak linear heat rate	$F_Q \leq 2.70$ (≤ 4 ft), $F_Q \leq 2.50$ (> 4 ft) [see Note 1]
		c)	Hot rod average linear heat rate	$F_{\Delta H} \leq 1.67$ [see Note 2]
		d)	Hot assembly average linear heat rate	$\overline{P}_{HA} \leq 1.67/1.04$ [see Note 3]
		e)	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]
		f)	Axial power dist (PBOT, PMID)	Figure 15-243
		g)	Low power region relative power (PLOW)	0.2 ≤ PLOW ≤ 0.8
		h)	Hot assembly burnup	≤ 75000 MWD/MTU, lead rod
		i)	Prior operating history	All normal operating histories
		j)	MTC	≤ 0 at HFP
		k)	HFP boron	Normal letdown
		l)	Rod power census	See Table 15-56

Parameter			Operating Range
	2.2	Fluid Conditions	
		a) Tavg	$581.1 \le T_{avg} \le 593.9$°F
		b) Pressurizer pressure	$2190 \le P_{RCS} \le 2310$ psia
		c) Loop flow	≥ 97,500 gpm/loop
		d) T_{UH}	Current upper internals, T_{cold} UH
		e) Pressurizer level	Normal level, automatic control
		f) Accumulator temperature	$105 \le T_{ACC} \le 125$°F
		g) Accumulator pressure	$555 \le P_{ACC} \le 708$ psig
		h) Accumulator volume	$6790 \le V_{ACC} \le 7422$ gal. (McGuire), $7550 \le V_{ACC} \le 8159$ gal. (Catawba)
		i) Accumulator fL/D	Current line configuration
		j) Minimum accumulator boron	≥ 2275 ppm
3.0	**Accident Boundary Conditions**		
	a)	Break location	N/A
	b)	Break type	N/A
	c)	Break size	N/A
	d)	Offsite power	Available or LOOP
	e)	Safety injection flow	Table 15-45
	f)	Safety injection temperature	58°F ≤ SI Temp ≤ 90°F, Reference 60 (covers a RWST temperature range of 70-100°F and component cooling water temperature down to 45°F)
	g)	Safety injection delay	≤ 17 seconds (with offsite power) ≤ 32 seconds (with LOOP)
	h)	Containment pressure	Bounded -- see Figure 15-233

	Parameter	Operating Range
i)	Single failure	Loss of one train
j)	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. 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 withing 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

4. Analysis was originally performed at 3445 MWt (3411 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 15-61.

Table 15-56. 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 15-57. Deleted Per 2012 Update

Table 15-58. Reactor Core Inventory and Release Fractions for LOCA

Noble Gases		Halogens		Alkali Metals		Tellurium Metals		Ba, Sr	
Release Fractions		Release Fractions		Release Fractions		Release Fractions		Release Fractions	
Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel
5%	95%	5%	35%	5%	25%	0%	5%	0%	2%
Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)
Kr83m	1.56E+07	Br83	1.55E+07	Rb86	2.08E+05	Sb127	9.65E+06	Sr89	1.03E+08
Kr85m	3.40E+07	Br85	3.41E+07	Rb88	1.00E+08	Sb129	3.43E+07	Sr90	9.31E+06
Kr85	1.07E+06	Br87	5.56E+07	Rb89	1.33E+08	Te127m	1.58E+06	Sr91	1.66E+08
Kr87	6.96E+07	I130	2.96E+06	Rb90	1.25E+08	Te127	9.51E+06	Sr92	1.69E+08
Kr88	9.79E+07	I131	1.04E+08	Cs134	2.09E+07	Te129	3.27E+07	Sr93	1.83E+08
Kr89	1.25E+08	I132	1.52E+08	Cs136	5.60E+06	Te129m	6.63E+06	Ba139	2.00E+08
Xe131m	1.43E+06	I133	2.15E+08	Cs137	1.26E+07	Te131	8.69E+07	Ba140	1.88E+08
Xe133m	6.72E+06	I134	2.47E+08	Cs138	2.09E+08	Te132	1.49E+08	Ba141	1.82E+08
Xe133	2.08E+08	I135	2.06E+08	Cs139	1.96E+08	Te133	1.22E+08		
Xe135m	4.51E+07					Te133m	1.01E+08		
Xe135	6.65E+07					Te134	2.12E+08		
Xe137	1.98E+08								
Xe138	1.98E+08								

Noble Metals		Cerium Group		Lanthanides					
Release Fractions		Release Fractions		Release Fractions		Release Fractions		Release Fractions	
Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel	Gap	Early In-Vessel
0%	0.25%	0%	0.05%	0%	0.02%	0%	0.02%	0%	0.02%
Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)
Mo99	1.97E+08	Ce141	1.73E+08	Y90	9.66E+06	La140	1.98E+08	Eu155	3.86E+05
Tc99m	1.74E+08	Ce143	1.79E+08	Y91	1.34E+08	La141	1.81E+08	Eu156	3.17E+07
Tc101	1.76E+08	Ce144	1.32E+08	Y91m	9.72E+02	La142	1.82E+08	Pr143	1.56E+08
Ru103	1.72E+08	Np237	4.23E+01	Y92	1.51E+08	La143	1.79E+08	Pr144	1.33E+08
Ru105	1.25E+08	Np238	5.04E+07	Y93	1.23E+08	Nd147	6.93E+07	Pr144m	1.86E+06
Ru106	6.37E+07	Np239	2.32E+09	Y94	1.90E+08	Pm147	1.75E+07	Am241	1.75E+04
Rh103m	1.72E+08	Pu236	7.32E+01	Y95	1.94E+08	Pm148	1.88E+07	Am242m	1.14E+03
Rh105	1.12E+08	Pu238	4.29E+05	Zr95	1.78E+08	Pm148m	2.96E+06	Am242	8.95E+06
Pd109	4.67E+07	Pu239	3.74E+04	Zr97	1.78E+08	Pm149	6.64E+07	Am243	4.41E+03
		Pu240	5.16E+04	Nb95	1.79E+08	Pm151	2.18E+07	Cm242	5.13E+06
		Pu241	1.45E+07	Nb95m	1.98E+06	Sm153	5.73E+07	Cm242	9.41E+05
		Pu242	2.97E+02	Nb97	1.78E+08	Eu154	9.87E+05		
		Pu243	5.62E+07						

Table 15-59. Assumptions Used for the Cask Drop Accident

1. Data and assumptions used to estimate radioactive source from postulated accidents	
a. Number of assemblies ruptured	32
b. Percentage of pins breached (%)	100
c. Cask Free Volume (m^3)	5.39
d. Respirable Fraction (%)	5
e. Percentage of particulate CRUD released to the Fuel Building (%)	30
2. Fuel Building Filtration Assumptions	
a. HEPA filter particulate removal efficiency (%)	95
b. Charcoal filter volatile and I-129 removal efficiency (%)	90
3. χ/Q at Exclusion Area Boundary (sec/m^3)	9.0E-04
4. Doses	Table 15-12

Table 15-60. Isotopic Inventory of the Dry Cask Drop Accident

Isotope	Ci/Assembly	Ci/Cask	Release Chemical Form	ISG-5 Release Fraction to Cask	Cask Release Fraction	Release from Pool Area
Mn-54	2.85	9.12E+01	Act. Prod.	7.83E-03	2.40E-01	1.71E-01
Fe-55	19.03	6.09E+02	Act. Prod.	7.83E-03	2.40E-01	1.14E+00
Co-60	19.03	6.09E+02	Act. Prod.	1.00E+00	2.40E-01	1.46E+02
Ni-63	373	1.19E+04	Act. Prod.	7.83E-03	2.40E-01	2.24E+01
Pu-238	2320	7.42E+04	Fines	3.00E-05	4.00E-02	8.91E-02
Pu-239	168	5.38E+03	Fines	3.00E-05	4.00E-02	6.45E-03
Pu-240	261	8.35E+03	Fines	3.00E-05	4.00E-02	1.00E-02
Pu-241	60700	1.94E+06	Fines	3.00E-05	4.00E-02	2.33E+00
Am-241	876	2.80E+04	Fines	3.00E-05	4.00E-02	3.36E-02
Cm-244	2300	7.36E+04	Fines	3.00E-05	4.00E-02	8.83E-02
H-3	206	6.59E+03	Gas	3.00E-01	8.00E-01	1.58E+03
Kr-85	3390	1.08E+05	Gas	3.00E-05	8.00E-01	2.60E+04
Sr-90	38400	1.23E+06	Volatile	3.00E-05	8.00E-02	2.95E+00
Y-90	38400	1.23E+06	Fines	3.00E-05	4.00E-02	1.47E+00
Ru-106	3140	1.00E+05	Volatile	2.00E-04	8.00E-02	1.61E+00
Rh-106	3140	1.00E+05	Volatile	2.00E-04	8.00E-02	1.61E+00
Sb-125	866	2.77E+04	Fines	3.00E-05	4.00E-02	3.33E-02
Te-125m	212	6.78E+03	Fines	3.00E-05	4.00E-02	8.14E-03
I-129	0.02	6.46E-01	Gas	3.00E-01	8.00E-01	1.55E-01
Cs-134	11200	3.58E+05	Volatile	2.00E-04	8.00E-02	5.73E+00
Cs-137	57600	1.84E+06	Volatile	2.00E-04	8.00E-02	2.95E+01
Ba-137m	54400	1.74E+06	Volatile	2.00E-04	8.00E-02	2.79E+01
Ce-144	1310	4.19E+04	Fines	3.00E-05	4.00E-02	5.03E-02
Pr-144	1310	4.19E+04	Fines	3.00E-05	4.00E-02	5.03E-02
Pm-147	13600	4.35E+05	Fines	3.00E-05	4.00E-02	5.22E-01
Eu-154	4560	1.46E+05	Fines	3.00E-05	4.00E-02	1.75E-01
Eu-155	2000	6.40E+04	Fines	3.00E-05	4.00E-02	7.68E-02

Table 15-61. 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 15-54]	2028	52
PCT Assessments		
Decay heat in Monte Carlo calculations	8	69
MONTECF power uncertainty correction	20	70
Safety Injection temperature range	59	60
Input error resulting in an incomplete solution matrix	25	71
Revised blowdown heatup uncertainty distribution	5	72
Vessel unheated conductor noding	0	73
Revised algorithm for average fuel temperature	0	73
Peak transient FQ = 2.7 in bottom third of core	0	74
Change from PAD 3.4 to PAD 4.0	-75	74
Fuel Thermal Conductivity Degradation with Peaking Factor Burndown	15	74
MUR Uprate to 101.7% of 3411 MWt	16	74
Revised Heat Transfer Multiplier Distribution	-85	76
HOTSPOT Clad Burst Strain Error	70	77
Current Licensing Basis LBLOCA PCT Including Assessments	2086	77
Small Break LOCA; NOTRUMP		
Analysis of Record PCT (2-inch break) [See Table 15-49]	1323	75
PCT Assessments		
None	0	74
Current Licensing Basis SBLOCA PCT Including Assessments	1323	74

Table 15-62. Dose Equivalent Iodine-131 (DEI-131)

Isotope	Concentration (µCi/gm)	FGR No. 11, Table 2.1 DCFs (Sv/Bq)	DEI (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
		DEI	**1.00E+00**

Table 15-63. Dose Equivalent Xenon-133 (DEX-133)

Isotope	Concentration (µCi/gm)	FGR No. 12, Table III.1 DCFs (Sv-s/Bq-m3)	DEX (µCi/gm)
KR-85M	2.10E+00	7.48E-15	1.01E+01
KR-85	8.80E+00	1.19E-16	6.71E-01
KR-87	1.20E+00	4.12E-14	3.17E+01
KR-88	3.70E+00	1.02E-13	2.42E+02
XE-131M	1.90E+00	3.89E-16	4.74E-01
XE-133M	3.10E+00	1.37E-15	2.72E+00
XE-133	2.81E+02	1.56E-15	2.81E+02
XE-135M	7.00E-01	2.04E-14	9.15E+00
XE-135	6.30E+00	1.19E-14	4.81E+01
XE-138	7.00E-01	5.77E-14	2.59E+01
		DEX	**6.52E+02**

Table 15-64. Deleted Per 2014 Update