SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS

ACCEPTANCE CRITERIA	VALUE	REFERENCE
HTP DNB Correlation 95/95 Limit	1.141*	21
Fuel Centerline Melt	21.000 kw/ft	50
Modified Barnett DNB Correlation 95/95 Limit	1.135*	48

* A mixed core penalty is applied to this correlation limit.

TRIP SETPOINTS FOR ANALYSIS OF PALISADES REACTOR AT 2565.4MWt

TRIP	SETPOINT VALUES	DELAY TIME (sec)
Low Reactor Coolant Flow	93% of Tech Spec flow	0.8
High Pressurizer Pressure	2,277 psia	0.8
Low Pressurizer Pressure	1,750 psia	0.8
Low Steam Generator Pressure	485 psia	0.8
Low Steam Generator Level[1]	23.7%	0.8
Thermal Margin/Low Pressure[2]	$P=f(T_H,T_C)$	0.8
Variable High Power	Trip <23.1%[3] above power with a 113.4% maximum and a 36.35% minimum	0.6

[1] Narrow Range Level

[2] The TM/LP trip setpoint is based on pressurizer pressure (P) setpoint, varying as a function of the maximum cold leg temperature (T_C), the measured power, and the measured axial shape index.

[3] Used for fast transient. For slow transient, it is 20.2% above power.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM			
15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3	(b)
15.1.2	Increase in Feedwater Flow	Bounded	15.1.3	(b)
15.1.3	Increase in Steam Flow			14.10
	1. Transient Response	Bounded	Ref. 29	
	2. MDNBR	Reanalyzed	Ref. 60	
	3. FCM	Reanalyzed	Ref. 60	
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve			
	1. Power Operation (Mode 1)	Bounded	15.1.3	
	2. Startup (Mode 2)	Bounded	15.4.1	
15.1.5	Steam System Piping Failures Inside and Outside of Containment			14.14
	1. Transient Response	Bounded	Ref. 36	
	2. MDNBR	Reanalyzed	Ref. 60	
	3. LHR	Reanalyzed	Ref. 60	
	4. Reactivity Verification	Reanalyzed	Ref. 60	

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
15.2.1	Loss of External Load			14.12
	1. Primary Over-pressurization	Bounded	Ref. 38	
	2. Secondary Over-pressurization	Bounded	Ref. 30	
	3. MDNBR	Non-Limiting	Ref. 60	
15.2.2	Turbine Trip	Bounded	15.2.1	
15.2.3	Loss of Condenser Vacuum	Bounded	15.2.1	
15.2.4	Closure of the Main Steam Isolation Valves (MSIVs)	Bounded	15.2.1	
15.2.5	Steam Pressure Regulator Failure	Not Applicable; BWR Event		
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries			
	1. Short-term Consequences	Bounded	15.3.1	
	2. Long-term Consequences	Bounded	15.2.7	
15.2.7	Loss of Normal Feedwater Flow			14.13
	1. Primary Over-pressurization	Bounded	15.2.1	
	2. Minimum Steam Generator Inventory	Bounded	Ref. 9	
	3. MDNBR	Bounded	15.3.1	

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment			
	1. Cooldown	Bounded	15.1.5	
	2. Heatup	Bounded	15.2.1	
	3. Long-term Cooling	Bounded	15.2.7	
15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW				
15.3.1	Loss of Forced Reactor Coolant Flow			14.7
	1. Transient Response	Bounded	Ref. 11[a]	
	2. MDNBR	Reanalyzed	Ref. 60	
15.3.2	Flow Controller Malfunction	Not Applicable		
15.3.3	Reactor Coolant Pump Rotor Seizure			14.7
	1. Transient Response	Bounded	Ref. 11[a]	
	2. MDNBR	Reanalyzed	Ref. 60	
15.3.4	Reactor Coolant Pump Shaft Break	Bounded	15.3.3	14.7

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES			
15.4.1	Uncontrolled Control Rod Bank Withdrawal From a Subcritical or Low Power Startup Condition			14.2.1
	1. Transient Response (Mode 2 /Mode 3)	Bounded	Ref. 27[a]/Ref. 5	
	2. MDNBR	Non-Limiting	Ref. 60	
15.4.2	Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions			14.2.2
	1. Transient Response	Bounded	Ref. 13[a]	
	2. MDNBR	Reanalyzed	Ref. 60	
	3. FCM/LHR	Reanalyzed	Ref. 60	
15.4.3	Control Rod Misoperation			
15.4.3.1	Dropped Control Bank/Rod			14.4
	1. Transient Response	Bounded	Ref. 11[a]	
	2. Half-Scram Transient	Bounded	Ref. 7	
	3. MDNBR	Reanalyzed	Ref. 60	
	4. FCM/LHR	Reanalyzed	Ref. 60	
15.4.3.2	Dropped Part-Length Control Rod	Bounded	15.4.3.1	14.6
15.4.3.3	Malpositioning of the Part-Length Control Group	Not Applicable		

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.4.3.4	Statically Misaligned Control Rod/Bank			14.6
	1. MDNBR	Reanalyzed	Ref. 60	
	2. FCM/LHR	Reanalyzed	Ref. 60	
15.4.3.5	Single Control Rod Withdrawal			14.2.3
	1. Transient Response	Bounded	Ref. 15.4.2	
	2. MDNBR	Reanalyzed	Ref. 60	
	3. FCM/LHR	Reanalyzed	Ref. 60	
15.4.3.6	Core Barrel Failure	Bounded	15.4.8	14.5
15.4.4	Start-Up of an Inactive Loop	Not Credible		14.8
15.4.5	Flow Controller Malfunction	Not Applicable; No Flow Controller		
15.4.6	CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant	Reanalyzed	Ref. 60	14.3
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Not Part of Licensing Basis		

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.4.8	Spectrum of Control Rod Ejection Accidents			14.16
	1. Transient Response	Bounded	Ref. 38	
	2. MDNBR	Reanalyzed	Ref. 60	
	3. FCM/LHR	Reanalyzed	Ref. 60	
	4. Deposited Enthalpy	Reanalyzed	Ref. 60	
	5. Radiological Analysis	Bounded	Ref. 40 and 41	
15.4.9	Spectrum of Rod Drop Accidents (BWR)	Not Applicable; BWR Event		
15.5 INCREASES IN REACTOR COOLANT INVENTORY				
15.5.1	Inadvertent Operation of the ECCS That Increases Reactor Coolant Inventory			
	1. Primary Over-pressurization	Bounded	15.2.1	
	2. Reactivity	Bounded	15.4.6	
15.5.2	CVCS Malfunction That Increases Reactor Coolant Inventory			
	1. Primary Over-pressurization	Bounded	15.2.1	
	2. Reactivity	Bounded	15.4.6	

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.6 DECREASES IN REACTOR COOLANT INVENTORY				
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve			
	1. Short-term Consequences	Bounded	Ref. 11[a]	
	2. Long-term Consequences	Bounded	15.6.5	
	3. MDNBR	Reanalyzed	Ref. 60	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment	Bounded	15.6.5	14.23
15.6.3	Radiological Consequences of Steam Generator Tube Failure	Bounded	Ref. 41	14.15
15.6.4	Radiological Consequences of a Main Steam Line Failure Outside Containment	Not Applicable; BWR Event		
15.6.5	Loss of Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary			
	1. SBLOCA	Bounded	Ref. 53	14.17.2
	2. LBLOCA	Bounded	Ref. 51	14.17.1
	UO2 Rods	Bounded	Ref. 51	
	8% Gadolinia Rods/6% Gadolinia Rods/ 2% Gadolinia Rods/4% Gadolinia Rods	Reanalyzed	Ref. 51	
	3. Radiological Consequences	Bounded	Ref. 12	14.22

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

DISPOSITION OF EVENTS SUMMARY FOR PALISADES CYCLE 27

SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT				
15.7.1	Waste Gas System Failure	Deleted[c]	Ref. 41	14.21
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Deleted[c]	Ref. 41	14.20
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	Bounded	Ref. 41	14.20
15.7.4	Radiological Consequences of Fuel Handling Accidents	Bounded	Ref. 23	14.19
15.7.5	Spent Fuel Cask Drop Accidents	Bounded	Ref. 24	14.11

a. PTSPWR2 system response analysis is given reference for this event.
b. Deleted from the FSAR.
c. This section of the Standard Review Plan has been deleted.

CYCLE 27 SUMMARY OF RESULTS FOR STANDARD REVIEW PLAN CHAPTER 15 EVENTS
(Reference 60)

SRP	FSAR	EVENT	MDNBR	PEAK[2] LHR (kW/ft)	MAXIMUM PRESSURE (psia)
15.1.3	14.10	Increase in Steam Flow	1.282	17.39	Note[5]
15.1.5	14.14	Steam System Piping Failures Inside and Outside of Containment	1.641	16.77	Note[3]
15.2.1	14.12	Loss of External Load	Note[5]	Note[5]	2627[4]
15.2.7	14.13	Loss of Normal Feedwater	Note[5]	Note[5]	Note[5]
15.3.1	14.7.1	Loss of Forced Reactor Coolant Flow	1.179	Note[5]	Note[5]
15.3.3	14.7.2	Reactor Coolant Pump Rotor Seizure	1.209	Note[5]	Note[5]
15.4.1	14.2.1	Uncontrolled Control Bank Withdrawal	Note[5]	Note[5]	Note[5]
15.4.2	14.2.2	Uncontrolled Control Bank Withdrawal at Power	1.301	20.32	Note[5]

1 Not used.

2 The calculated FCM limit for Cycle 27 was bounded by the Technical Specification limit of 21,000 kW/ft, which was used as the actual Cycle 27 limit for UO_2 rods. The centerline melt temperature limit is 4,687°F. These values preclude melting of any fuel rod (with or without gadolina).

3 This is a depressurization event. Limit is not challenged by this event.

4 The maximum secondary side pressure for the secondary side over pressurization case is 1,063.3 psia.

5 Limit is not challenged by this event.

CYCLE 27 SUMMARY OF RESULTS FOR STANDARD REVIEW PLAN CHAPTER 15 EVENTS
(Reference 60)

SRP	FSAR	EVENT	MDNBR	PEAK[7] LHR (kW/ft)	MAXIMUM PRESSURE (psia)
15.4.3		Control Rod Misoperation			
	14.4.1	-Dropped Rod	1.240	17.43	Note[9]
	14.4.2	-Dropped Bank[8]			
	14.2.3	-Single Rod Withdrawal	1.320	18.26	Note[9]
	14.6	-Statically Misaligned Control Rod	1.219	20.73	Note[9]
15.4.6	14.3	CVCS Malfunction Resulting in Decreased Boron Concentration	Adequacy of Shutdown Margin is demonstrated		
15.4.8	14.16	Control Rod Ejection	Less than HTP limit <12% Fuel Failures Predicted	4306 ^oF Peak Fuel Centerline Temperature	Note[9]
15.6.1	Note[10]	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	1.454	Note[9]	Note[9]

[6] When the probability of DNB occurring falls below approximately one in a million, the effective MDNBR value becomes essentially meaningless. For this reason, the calculation of an effective MDNBR is limited to a value no greater than five standard deviations from the mean of the HTP DNB correlation limit with a mixed-core penalty applied.

[7] The calculated FCM limit for Cycle 27 was bounded by the Technical Specification limit of 21.000 kW/ft, which was used as the actual Cycle 27 limit for UO_2 rods. The centerline melt temperature limit is 4,702^{o}F. These values preclude melting of any fuel rod (with or without gadolina).

[8] Bounded by dropped rod.

[9] Limit is not challenged by this event.

[10] This event is not part of the Licensing Basis analyses for Palisades

CYCLE 27 SUMMARY OF RESULTS FOR STANDARD REVIEW PLAN CHAPTER 15 EVENTS
(Reference 60)

SRP	FSAR	EVENT	DISPOSITION/RESULTS
15.6.2	14.23	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Note[11]
15.6.3	14.15	Radiological Consequences of Steam Generator Tube Failure	Note[11]
15.6.5	14.17 14.22	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within a Reactor Coolant	Note[11], 10CFR50.46(b) acceptance criteria are met
15.7.3	14.20	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Note[11]
15.7.4	14.19	Radiological Consequences of Fuel Handling Accidents	Note[11]
15.7.5	14.11	Spent Fuel Cask Drop Accidents	Note[11]

[11] Radiological consequences acceptance criteria are met.

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF THE CHAPTER 14 EVENTS

FSAR SECTION	SCENARIO DESCRIPTION			OFFSITE DOSES AND LIMITS				CONTROL ROOM HABITABILITY			
Section 14.11: CASK DROP IN THE SPENT FUEL POOL (Section 14.24 for CRH)	**Cask Drop Scenario**	**Pre-Isolation % Filter Bypass**	**Post-Isolation % Filter Bypass**	**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	30 Day Decay w/Charcoal Filters	N/A	10	TEDE	6.3	2.04	0.25	0 min / 100 cfm	TEDE	5	1.37
	30 Day Decay w/Charcoal Filters	N/A	17.5	TEDE	6.3	2.78	0.35	0 min / 100 cfm	TEDE	5	1.99
	90 Day Decay No Charcoal Filters	100	100	TEDE	6.3	0.08	0.01	Not Required (3) / 100 cfm	TEDE	5	1.67
Section 14.14: STEAM LINE BREAK (Section 14.24 for CRH)	**Dose Calculation Assumptions**			**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	Break Outside the Containment			TEDE	25	0.67	0.20	20 min / 20 cfm	TEDE	5	3.34
Section 14.15: STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER (Section 14.24 for CRH)	**Iodine Spike Assumptions**			**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	Previous Iodine Spike			TEDE	25	0.99	0.22	20 min / 100 cfm	TEDE	5	3.79
	Generated Iodine Spike			TEDE	2.5	1.17	0.21	20 min / 100 cfm	TEDE	5	3.48
Section 14.16: CONTROL ROD EJECTION EVENT (Section 14.24 for CRH)	**Dose Calculation Assumptions**			**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	Containment Release			TEDE	6.3	2.60	0.68	Auto Switch (4) / 20 cfm	TEDE	5	1.27
	Steam Generator/ADV Release			TEDE	6.3	2.70	0.43	20 min / 20 cfm	TEDE	5	1.00

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF THE CHAPTER 14 EVENTS

FSAR SECTION	SCENARIO DESCRIPTION	OFFSITE DOSES AND LIMITS				CONTROL ROOM HABITABILITY			
Section 14.19: FUEL HANDLING INCIDENT (Section 14.24 for CRH)	**Dose Calculation Assumptions**	**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	One fuel bundle fails 2 days after shutdown, No Charcoal Filtration	TEDE	6.3	2.20	0.28	20 min / 100 cfm	TEDE	5	4.04
	One fuel bundle fails 2 days after shutdown, 10% Filtered Release	TEDE	6.3	2.02	0.25	20 min / 100 cfm	TEDE	5	3.68
	One fuel bundle fails 2 days after shutdown, 50% Filtered Release	TEDE	6.3	1.31	0.17	20 min / 100 cfm	TEDE	5	2.22
Section 14.21.1: GAS DECAY TANK RUPTURE (Section 14.24 for CRH)	**Dose Calculation Assumptions**	**Offsite Dose Limits (1) [rem]**		**Site Boundary (60 min) [rem]**	**Low Population Zone (60 min) [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (2) [rem]**		**Control Room Dose [rem]**
	22 Hr Decay Tank Inventory	Thyroid	75	0.000	0.000	20 min	Thyroid	30	0.000
		Whole Body	6	0.415	0.067	85 cfm	Whole Body	5	0.036
Section 14.21.2: VOLUME CONTROL TANK RUPTURE (Section 14.24 for CRH)	**Iodine Spike Assumptions**	**Offsite Dose Limits (1) [rem]**		**Site Boundary (60 min) [rem]**	**Low Population Zone (60 min) [rem]**	**Time to E-HVAC [min] / CRE Inleakage [cfm]**	**Control Room Dose Limits (2) [rem]**		**Control Room Dose [rem]**
	Equilibrium Iodine	Thyroid	75	1.59	0.258	20 min	Thyroid	30	0.277
		Whole Body	6	0.081	0.013	85 cfm	Whole Body	5	0.005
	Previous Iodine Spike	Thyroid	75	63.77	10.32	20 min	Thyroid	30	11.086
		Whole Body	6	0.081	0.013	85 cfm	Whole Body	5	0.005
	Generated Iodine Spike	Thyroid	75	3.69	0.597	20 min	Thyroid	30	0.922
		Whole Body	6	0.081	0.013	85 cfm	Whole Body	5	0.005

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF THE CHAPTER 14 EVENTS

FSAR SECTION	SCENARIO DESCRIPTION	OFFSITE DOSES AND LIMITS				CONTROL ROOM HABITABILITY			
Section 14.22: PALISADES MAXIMUM HYPOTHETICAL ACCIDENT (MHA)	**Offsite Dose Calculation Assumptions**	**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min]** **CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	Containment Leakage			10.48	1.53				1.31
	ESF Leakage			2.28	1.74				1.58
	SIRWT Leakage			0.00	0.00				0.00
	Non-SIRWT Shine			N/A	N/A				0.30
	SIRWT Shine			N/A	N/A				0.81
	Total Dose	TEDE	25	12.76	3.27	Auto Switch (4) 16 cfm	TEDE	5	4.00
Section 14.23: SMALL LINE BREAK OUTSIDE CONTAINMENT	**Iodine Spike Assumptions**	**Offsite Dose Limits (5) [rem]**		**Exclusion Area Boundary (0-2 hrs) [rem]**	**Low Population Zone [rem]**	**Time to E-HVAC [min]** **CRE Inleakage [cfm]**	**Control Room Dose Limits (6) [rem]**		**Control Room Dose [rem]**
	Generated Iodine Spike	TEDE	2.5	0.41	0.05	20 min 100 cfm	TEDE	5	0.53

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF THE CHAPTER 14 EVENTS

Notes:

(1) The acceptance criteria for the offsite dose limits are given in 10CFR Part 100 Section 11 (Reference 42). It states that the public 'would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine' following a stipulated fission product release. In addition, the terms 'well within' and 'small fraction of' are utilized in the acceptance criteria in Chapter 15 of the Standard Review Plan (SRP 15) (Reference 43). The two terms correspond to 25% (75 rem thyroid, 6 rem whole body) and 10% (30 rem thyroid, 2.5 rem whole body) of the radiation levels given in 10 CFR Part 100 respectively.

(2) The acceptance criteria for the control room dose limits are given in 10CFR Part 50 Appendix A, General Design Criterion 19 (GDC 19) (Reference 44). It states that the doses will be less than '5 rem whole body or its equivalent to any part of the body, for the duration of the accident.' SRP 6.4 (Reference 45) interprets the limits of GCD 19 of being 5 rem whole body, 30 rem thyroid, and 30 rem skin dose.

(3) 'Not Required' - Control Room CRHVAC emergency mode not needed to maintain control room doses below specified limits. Specifically, the 30 day integrated TEDE doses do not exceed limits assuming no emergency filtration.

(4) 'Auto Switch' - Automatic Switchover to CRHVAC emergency mode following CHP or CHR and Diesel sequencing due to a loss of Offsite Power. Total time to CRHVAC emergency mode is assumed to be 1.5 minutes (Reference 41)

(5) The acceptance criteria for the offsite doses are given in RG 1.183 (Reference 54), which are based upon 10CFR Part 50 Section 67. RG 1.183 and 10CFR 50.67 do not provide specific acceptance criteria for either the cask drop in the spent fuel pool or the small line break outside containment. The acceptance criteria for the cask drop in the spent fuel pool are taken as those of the fuel handling accident. The acceptance criteria for the small line break outside containment are taken to be a small fraction (i.e., 10%) of the 10CFR 50.67 limits.

(6) The acceptance criteria for the control room doses are given in 10CFR Part 50 Section 67.

EVENT SUMMARY FOR THE UNCONTROLLED BANK WITHDRAWAL FROM A LOW POWER EVENT

Event	Value	Time(sec)
Bank Withdrawal Begins		0.00
Letdown Valve Open		0.00
Pressurizer Spray Activates		0.01
Low VHP Trip Setting reached		51.0
Peak Power Level	1815.7 Mwt	51.61
Peak Core Average Heat Flux	56,630 Btu/hr-ft^2	52.59
Peak Pressurizer Pressure	see Table 14.1-5	65.95

EVENT SUMMARY FOR THE UNCONTROLLED ROD BANK WITHDRAWAL EVENT FROM POWER

EVENT	VALUE	TIME (sec)
Start Rod Withdrawal		0.00
Letdown Flow Valve Open		0.00
Reactor Scram (TM/LP Trip)		24.82
Turbine Stop Valve Closed		25.00
Peak Power Level	2900.9 MWt	25.38
Peak Pressurizer Pressure	2267.1 psia	27.00
Steam Line Safety Valves Open		29.50
Peak Steam Dome Pressure	1039.6	31.69

SAMPLED LBLOCA PARAMETERS

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties and burnup)
	Break type (guillotine versus split)
	Break size
	Critical flow discharge coefficients (break)
	Decay heat
	Critical flow discharge coefficients (surgeline)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
	T_{min} (intersection of film and transition boiling)
	Initial stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condensation interphase heat transfer
	Metal-water reaction
Plant[1]	
	Offsite power availability
	Core power and power distribution
	Pressurizer pressure
	Pressurizer liquid level
	SIT pressure
	SIT liquid level
	SIT temperature (based on containment temperature)
	Containment temperature
	Containment volume
	Initial flow rate
	Initial operating temperature
	Diesel start (for loss of offsite power only)

[1] Uncertainties for plant parameters are based on plant-specific values with the exception of “Offsite power availability,” which is a binary result that is specified by the analysis methodology.

EVENT SUMMARY FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE

EVENT	VALUE	TIME (sec)
Primary Coolant Pump Rotor Seizes		0.0
Pressurizer Spray Actuates		0.20
Reactor Scram (Low Flow)		1.13
Turbine Stop Valve Closed		1.40
Peak Power Level	2743.1 MWt	1.68
Minimum DNBR Predicted to Occur	see Table 14.1-5	1.76
Peak Core Average Temperature	579.3 °F	2.09
Peak Pressurizer Pressure	2145.19 psia	3.82
Peak Steam Dome Pressure	986.34 psia	6.13

SEQUENCE OF EVENTS FOR EXCESS LOAD LIMITING MDNBR CASE

Event	Time (sec)	Value
30% step increase in steam flow (event initiator)	0.0	- - -
Auctioneered core power signal reaches VHPT setpoint	22.99	110.4% of RTP
Scram insertion begins	24.08	- - -
MDNBR occurred	24.2	See Table 14.1-5

POSTULATED CASK DROP ACCIDENTS

Impact Location	Cask Weight(Tons)	Cask Drop Height Air(ft)	Water(ft)	Total(ft)	Velocity @ Impact(in/sec)	Penetration(in)	Impact on Structure
STRUCTUAL							
Cask Loading Area	97.5	4	35.6	39.6	498.2	Crush depth of foam=17.15	The Pressure transferred to the slab is less than P(Critical) for the slab
MTC drop into washdown pit	97.5	16	---	16	385.2	Crush depth of foam=22.96	The Pressure transmitted to the slab is less than P(Critical) for the slab
MTC drop onto VCC in Track Alley	93.5	5.83	---	5.83	232.6	---	MTC remains intact & LDS remains intact
MTC seismic overturn in pool loading area	---	---	---	---	---	---	The loaded MTC/MSB will not tip over & damage fuel
RADIOLOGICAL							
MTC Tip over onto West 11x7 fuel rack	97.5	---	---	---	---	---	Failure of all 73 assemblies & doses within applicable 10 CFR 50.67 limits
Drop of the loaded MTC on to VCC	93.5	5.83	---	5.83	232.6	---	The MTC/MSB will not fail. No radiological release.

Table 14.11-2 Spent Fuel Cask Drop Radiological Analysis – Source Terms*

*Listed source term is for a single assembly.

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
Co-58	0.0000E+00	0.0000E+00
Co-60	0.0000E+00	0.0000E+00
Kr-85	0.6563E+04	0.6493E+04
Kr-85m	0.0000E+00	0.0000E+00
Kr-87	0.0000E+00	0.0000E+00
Kr-88	0.0000E+00	0.0000E+00
Rb-86	0.3205E+03	0.3450E+02
Sr-89	0.5045E+06	0.2213E+06
Sr-90	0.5178E+05	0.5157E+05
Sr-91	0.1431E-16	0.0000E+00
Sr-92	0.0000E+00	0.0000E+00
Y-90	0.5184E+05	0.5159E+05
Y-91	0.6805E+06	0.3344E+06
Y-92	0.0000E+00	0.0000E+00
Y-93	0.3964E-15	0.0000E+00
Zr-95	0.8884E+06	0.4637E+06
Zr-97	0.1774E-06	0.0000E+00
Nb-95	0.1146E+07	0.7766E+06
Mo-99	0.6714E+03	0.1816E-03
Tc-99m	0.6469E+03	0.1749E-03
Ru-103	0.6102E+06	0.2118E+06
Ru-105	0.0000E+00	0.0000E+00
Ru-106	0.2925E+06	0.2611E+06
Rh-105	0.5545E+00	0.3056E-12
Sb-127	0.3350E+03	0.6807E-02
Sb-129	0.0000E+00	0.0000E+00
Te-127	0.8360E+04	0.5490E+04
Te-127m	0.8207E+04	0.5606E+04
Te-129	0.1138E+05	0.3301E+04
Te-129m	0.1748E+05	0.5071E+04
Te-131m	0.5959E-02	0.2118E-16
Te-132	0.1664E+04	0.4757E-02

Table 14.11-2 Spent Fuel Cask Drop Radiological Analysis – Source Terms*

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
I-131	0.5355E+05	0.3038E+03
I-132	0.1715E+04	0.4902E-02
I-133	0.5526E-04	0.7976E-25
I-134	0.0000E+00	0.0000E+00
I-135	0.0000E+00	0.0000E+00
Xe-133	0.3313E+05	0.1194E+02
Xe-135	0.0000E+00	0.0000E+00
Cs-134	0.1016E+06	0.9612E+05
Cs-136	0.6310E+04	0.2638E+03
Cs-137	0.6483E+05	0.6459E+05
Ba-139	0.0000E+00	0.0000E+00
Ba-140	0.2450E+06	0.9480E+04
La-140	0.2819E+06	0.1091E+05
La-141	0.0000E+00	0.0000E+00
La-142	0.0000E+00	0.0000E+00
Ce-141	0.6297E+06	0.1752E+06
Ce-143	0.3033E+00	0.2220E-13
Ce-144	0.8358E+06	0.7222E+06
Pr-143	0.2630E+06	0.1226E+05
Nd-147	0.7146E+05	0.1663E+04
Np-239	0.2042E+04	0.1581E+02
Pu-238	0.2174E+04	0.2199E+04
Pu-239	0.2968E+03	0.2968E+03
Pu-240	0.2972E+03	0.2972E+03
Pu-241	0.8941E+05	0.8870E+05
Am-241	0.1070E+03	0.1260E+03
Cm-242	0.2428E+05	0.1882E+05
Cm-244	0.2743E+04	0.2726E+04
I-130	0.5151E-13	0.0000E+00
Kr-83m	0.0000E+00	0.0000E+00
Xe-138	0.0000E+00	0.0000E+00
Xe-131m	0.2962E+04	0.1204E+03
Xe-133m	0.5377E+01	0.3038E-07
Xe-135m	0.0000E+00	0.0000E+00

Table 14.11-2 Spent Fuel Cask Drop Radiological Analysis – Source Terms*

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
Cs-138	0.0000E+00	0.0000E+00
Cs-134m	0.0000E+00	0.0000E+00
Rb-88	0.0000E+00	0.0000E+00
Rb-89	0.0000E+00	0.0000E+00
Sb-124	0.5599E+03	0.2806E+03
Sb-125	0.8790E+04	0.8446E+04
Sb-126	0.1278E+03	0.4525E+01
Te-131	0.1341E-02	0.4765E-17
Te-133	0.0000E+00	0.0000E+00
Te-134	0.0000E+00	0.0000E+00
Te-125m	0.1902E+04	0.1940E+04
Te-133m	0.0000E+00	0.0000E+00
Ba-141	0.0000E+00	0.0000E+00
Ba-137m	0.6132E+05	0.6110E+05
Pd-109	0.1499E-10	0.0000E+00
Rh-106	0.2925E+06	0.2611E+06
Rh-103m	0.5502E+06	0.1908E+06
Tc-101	0.0000E+00	0.0000E+00
Eu-154	0.6312E+04	0.6229E+04
Eu-155	0.4225E+04	0.4129E+04
Eu-156	0.2550E+05	0.1649E+04
La-143	0.0000E+00	0.0000E+00
Nb-97	0.1911E-06	0.0000E+00
Nb-95m	0.6589E+04	0.3441E+04
Pm-147	0.1206E+06	0.1162E+06
Pm-148	0.4604E+04	0.3221E+03
Pm-149	0.3193E+02	0.2179E-06
Pm-151	0.3097E-02	0.1665E-17
Pm-148m	0.1557E+05	0.5685E+04
Pr-144	0.8358E+06	0.7222E+06
Pr-144m	0.1003E+05	0.8666E+04
Sm-153	0.5430E+01	0.2827E-08
Y-94	0.0000E+00	0.0000E+00
Y-95	0.0000E+00	0.0000E+00

Table 14.11-2 Spent Fuel Cask Drop Radiological Analysis – Source Terms*

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
Y-91m	0.9094E-17	0.0000E+00
Br-82	0.1919E-02	0.1012E-14
Br-83	0.0000E+00	0.0000E+00
Br-84	0.0000E+00	0.0000E+00
Am-242	0.1237E+02	0.1236E+02
Np-238	0.1195E+02	0.6211E-01
Pu-243	0.2743E-06	0.2743E-06

Spent Fuel Cask Drop Radiological Analysis – Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Discharged Fuel Assembly Burnup	39,300 – 58,900 MWD/MTU
Fuel Enrichment	3.0 – 5 w/o
Number of Fuel Assemblies Damaged	73
Delay Before Cask Drop	Cases 1 & 2 – 30 days Case 3 – 90 days
Source Terms	See Table 14.11-2
Water Level Above Damaged Fuel Assembly	23.4 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1 Overall - 200
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Time of Control Room Ventilation System Isolation	Cases 1 & 2 – 0 seconds Case 3 - ∞
Time of Control Room Filtered Makeup Flow	Cases 1 & 2 – 0 seconds Case 3 - ∞
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

EVENT SUMMARY FOR LOSS OF LOAD

Primary Side Pressurization

EVENT	VALUE	TIME (Sec)
Turbine Trip		0.00
Reactor scramed on high pressure trip		7.0
Pressurizer safety valves open		9.7
Peak PCS pressure	2626.9 psia	10.2
Pressurizer safety valves close		11.1
Lowest bank of main steam line safety valves opened		11.2

Secondary Side Pressurization

EVENT	VALUE	TIME (Sec)
Turbine Trip		0.0
Loss of Main Feedwater		0.1
Steam Line Safety Valves Open		3.2
Reactor Scram (High Pressurizer Pressure)		7.6
Peak Power Level	108%	8.1
Pressurizer PORVs Open		9.0
Peak Secondary Side Pressure	1063.25 psia	13.7

INITIAL CONDITIONS FOR THE LOSS OF NORMAL FEEDWATER ANALYSIS

Parameter	Value
Core Power, MW_t (with uncertainty)	2580.6
Core Inlet Temperature, °F	544
Pressurizer Pressure, psia	2060
Pressurizer Liquid Level, %	62
PCS Loop Flow Rate, gpm	341,400
Primary Coolant Pump Heat, MW	16.3
Steam Generator Pressure, psia	770.7
Steam Generator Liquid Level, % Narrow Range	63.9
Steam Generator (A) Secondary Total Mass, lb_m	133,961
Steam Generator (B) Secondary Total Mass, lb_m	133,971
Main Steam Flow, lb_m/hr	11.354×10^6
MFW Temperature, °F	440.7
Steam Generator Blowdown per Steam Generator, lb_m/hr	30,000

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER FLOW ANALYSIS WITH OFFSITE POWER AVAILABLE AND STEAM DUMP SYSTEM DISABLED

TIME (sec)	EVENT
0	Total loss of main feedwater
22.8	Auxiliary feedwater actuation signal on low steam generator water level
23.6	Reactor trip signal on low steam generator water level
23.7	Main turbine trip
24.1	Control rods begin to drop
26.0	Early maximum post-trip PCS average temperature (571.5°F)
28.0	Early maximum pressurizer level (65.7%)
142.8	Motor-driven AFW pump starts (120 seconds after AFAS)
1200.0	AFW flow controller increased to maximum setting by operator
1500.0	PCPs tripped by operator
3120.0	Late maximum PCS average temperature (570.2°F)
3148.0	Late maximum pressurizer level (67.2%)
4238.0	Minimum steam generator liquid inventory occurs in Steam Generator “A” (8,515 lbm)
8000.0	End of calculation

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER FLOW ANALYSIS WITH OFFSITE POWER AVAILABLE AND STEAM DUMP SYSTEM AVAILABLE

TIME (sec)	EVENT
0	Total loss of main feedwater
22.8	Auxiliary feedwater actuation signal on low steam generator water level
23.6	Reactor trip signal on low steam generator water level
23.7	Main turbine trip
24.1	Control rods begin to drop
22.0	Maximum post-trip PCS average temperature (571.4°F)
26.0	Maximum pressurizer level (64.7%)
142.8	Motor-driven AFW pump starts (120 seconds after AFAS)
1200.0	AFW flow controller increased to maximum setting by operator
1500.0	PCPs tripped by operator
1712.0	Minimum steam generator liquid inventory occurs in Steam Generator "A" (8,268.4 lbm)
5000.0	End of calculation

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER FLOW ANALYSIS WITHOUT OFFSITE POWER AVAILABLE AND STEAM DUMP SYSTEM DISABLED

TIME (sec)	EVENT
0	Total loss of main feedwater
22.8	Auxiliary feedwater actuation signal on low steam generator water level
23.6	Reactor trip signal on low steam generator water level
23.7	Main turbine trip
24.1	Control rods begin to drop
26.0	Maximum post-trip PCS average temperature (571.5°F)
28.0	Maximum pressurizer level (65.6%)
34.6	PCPs automatically tripped (11 seconds after reactor trip)
142.8	Motor-driven AFW pump starts (120 seconds after AFAS)
7342.0	Minimum steam generator liquid inventory occurs in Steam Generator “A” (10,618 lbm)
12000.0	End of calculation

RESULTS SUMMARY FOR LOSS OF NORMAL FEEDWATER

CASE	MINIMUM STEAM GENERATOR LIQUID MASS (lbm) SG A / SG B / Total	MAXIMUM PRESSURIZER LEVEL (%)
Off-site Power Available (Steam Dump System Disabled)	8,515.0 / 8,549.0 / 17,064.0	67.2
Offsite Power Available (Steam Dump System Available)	8,268.0 / 8,374.0 / 16,642.0	64.7
Offsite Power Unavailable (Steam Dump System Disabled)	10,618.0 / 10,644.0 / 21,262.0	65.6

MAIN STEAM LINE BREAK INPUT PARAMETERS AND ASSUMPTIONS

Parameter	Biasing	Value
Break location	Limiting	Main steam line, at steam generator outlet
Break flow area (choked at steam generator integral flow restrictor)	Maximum	$\frac{\pi}{4}\left(18\,{}^{5}\!/_{8}\ in\right)^{2}$
Break flow model	Conservative	Moody critical flow (at steam generator integral flow restrictor)
Break fluid conditions	Conservative	Steam-only restriction
Steam generator tube plugging	Minimum	No plugged tubes
Full-power moderator temperature coefficient	Most-negative analysis limit	-35 pcm/°F
Scram worth	Cycle 17 minimum analysis value for full-power cases	5022 pcm
	Technical Specification minimum shutdown margin for hot zero power cases	2000 pcm
Shutdown control rod configuration	Maximum peaking	Most reactive control rod stuck out of the core
Low Steam Generator Pressure Main Steam Isolation Signal and scram setpoint	Minimum analysis value	485.0 psia
Containment High Pressure Main Steam Isolation Signal and scram setpoints	Conservative	Not credited
Main Steam Isolation Signal signal processing	Maximum delay	1.0 second
Main Steam Isolation Valve closure	Maximum delay	5.0 seconds after Main Steam Isolation Signal
Main feedwater termination	Maximum delay	22.0 seconds after Main Steam Isolation Signal
Main feedwater temperature	Minimum	Reduced to 32.1°F when turbine trips
Main feedwater flow	Maximum	Increased 20% when break occurs (full-power cases)
Main feedwater delivery	Limiting	All to affected steam generator when break occurs
Auxiliary feedwater actuation	Minimum delay	When break occurs
Auxiliary feedwater temperature	Minimum	32.1°F
Auxiliary feedwater flow rate	Maximum analysis value	200 gpm / Steam Generator

MAIN STEAM LINE BREAK INPUT PARAMETERS AND ASSUMPTIONS

Parameter	Biasing	Value
Auxiliary feedwater delivery	Limiting	All to affected steam generator
Auxiliary feedwater termination	Maximum delay for operator action	1800.0 seconds after break occurs
Single active failure	Worst	1 of 2 High Pressure Safety Injection pumps required to be in service fails
Initial boron concentration in safety injection lines	Minimum	0 ppm
Total safety injection line purge volume	Maximum	78 ft^3
Safety Injection and Refueling Water Tank boron concentration	Technical Specification Minimum	1720 ppm
Safety Injection and Refueling Water Tank temperature	Technical Specification Minimum	40.0°F
Low Pressurizer Pressure Safety Injection Signal setpoint	Minimum analysis value	1450.0 psia
Safety injection actuation (if primary coolant system pressure is less than High Pressure Safety Injection Pump shutoff head)	Maximum delay	30.0 seconds after Safety Injection Signal with offsite power available
		40.0 seconds after Safety Injection Signal with loss of offsite power
High Pressure Safety Injection pump flow	Minimum	Degraded flow curve for single pump
High Pressure Safety Injection pump shutoff head	Analysis value	1200.7 psia
Turbine control valves position	Limiting	Increased to fully open when break occurs (full-power case)
Scram signal processing	Maximum delay	0.8 second
Turbine trip	Maximum delay	At scram signal
Initial reactor power	Bounding	2565.4 MW at full power
		1.0 W at hot zero power
Initial pressurizer pressure	Nominal	2060.0 psia
Initial pressurizer level	Programmed	57.0% of span at full power
		42% of span at hot zero power
Initial primary coolant system cold leg temperature	Maximum at full power	544.0°F
		532.0°F

MAIN STEAM LINE BREAK INPUT PARAMETERS AND ASSUMPTIONS

Parameter	Biasing	Value
Initial primary coolant system total flow rate	Technical Specification minimum value minimum measurement uncertainty	341,400 gpm
Core bypass flow rate	Maximum analysis value	3% of primary coolant system total
Initial reactor vessel upper head temperature	Analysis assumption	Close to initial primary coolant system hot leg temperature
Initial steam generator inventory	Analysis value	141,065 lb_m / steam generator at full power
		210,759 lb_m / steam generator at hot zero power
Initial steam generator pressure	Analysis value	820.4 psia at full power
		889.1 psia at hot zero power

STEAM LINE BREAK SEQUENCE OF EVENTS DURING LHR-LIMITING TRANSIENT (HZP, OFFSITE POWER AVAILABLE)

Event	Time After Break (sec)
Reactor is at EOC all-rods-in-except-most-reactive-rod HZP condition. Double-ended guillotine break in main steam line at steam generator outlet occurs.	0.0
Low Steam Generator Pressure ESF signal (485 psia) initiates steam generator isolation.	13.8
MSIVs are fully closed (6.0 sec delay).	19.8
Low Pressurizer Pressure ESF signal (1450 psia) initiates SI.	20.4
Shutdown worth has been fully overcome by moderator and Doppler feedback (0.0 $ total reactivity).	32.2
Credited HPSI pump is running at rated speed (30.0 sec delay) and begins filling SI lines with borated water.	50.4
Peak post-scram power (26.92% of rated) and peak LHR occur.	158.0
Borated water has filled SI lines and begins to enter PCS cold legs.	158.7
Borated water front has passed through core. Power begins to decrease noticeably.	160.0
Affected steam generator begins to dry out.	243.0
Affected-sector core inlet temperature begins to increase. Power begins to drop to decay-heat level.	248.0

OVERALL CORE CONDITIONS AT TIME OF PEAK LHR

Parameter	Value
Time, sec	158.0
Reactor Power, MW	690.70
Core Outlet Pressure, psia	863.16
Core Inlet Temperature, °F	
Affected Sector	380.21
Unaffected Sector	489.31
Core Inlet Flow Rate, lb_m/sec	
Stuck Rod Region	5,489.9
Rest of Affected Sector	14,564
Unaffected Sector	18,316

STEAM LINE BREAK SEQUENCE OF EVENTS DURING DNBR-LIMITING TRANSIENT (HZP, LOSS OF OFFSITE POWER)

Event	Time After Break (sec)
Reactor is at EOC all-rods-in-except-most-reactive-rod HZP condition. Double-ended guillotine break in main steam line at steam generator outlet occurs. Offsite power is lost.	0.0
Low Steam Generator Pressure ESF signal (485 psia) initiates steam generator isolation.	12.4
MSIVs are fully closed (6.0 sec delay).	18.4
Low Pressurizer Pressure ESF signal (1450 psia) initiates SI.	26.3
Shutdown worth has been fully overcome by moderator and Doppler feedback (0.0 $ total reactivity).	49.7
Credited HPSI pump is running at rated speed (40.0 sec delay) and begins filling SI lines with borated water.	66.3
Borated water has filled SI lines and begins to enter PCS cold legs.	170.1
Borated water front has passed through core.	172.0
Peak post-scram power (13.23% of rated) and MDNBR occur. Power begins to decrease.	190.0
Affected steam generator begins to dry out.	517.0
Affected-sector core inlet temperature begins to increase. Power begins to drop to decay-heat level.	592.0

OVERALL CORE CONDITIONS AT TIME OF MDNBR

Parameter	Value
Time, sec	190.0
Reactor Power, MW	339.35
Core Outlet Pressure, psia	821.76
Core Inlet Temperature, °F	
Affected Sector	314.56
Unaffected Sector	466.81
Core Inlet Flow Rate, lb_m/sec	
Stuck Rod Region	768.61
Rest of Affected Sector	992.17
Unaffected Sector	661.63

Main Steam Line Break (MSLB) Radiological Analysis – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2703 MW_{th}
Core Average Burnup	39,300 MWD/MTU
Radial Peaking Factor	2.04
Fuel Damage	0.5% DNB 0% Fuel Centerline Melt
Steam Generator Tube Leakage Rate	0.3 gpm per SG
Time to establish shutdown cooling and terminate steam release	8 hours
Time for PCS to reach 212°F and terminate SG tube leakage	12 hours
PCS Mass	432,977 lb_m
SG Secondary Side Mass	Maximum (Hot Zero Power) – 210,759 lb_m (used for faulted SG to maximize release) Minimum (Hot Full Power) – 141,065 lb_m (used for intact SG to maximize concentration)
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 14.14-7
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode	Table 14.24-1 20 minutes
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183 Section 4.2.6

MAIN STEAM LINE BREAK (MSLB) RADIOLOGICAL ANALYSIS – INTACT SG STEAM RELEASE RATE

Time (hours)	Intact SG Steam Release (lb_m)
0 – 8	800,000
8 – 720.0	0

INITIAL CONDITIONS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

PARAMETER	ASSUMED VALUE
Initial core power level, MWt	2600.6
Core inlet coolant temperature, °F	550.65
Core mass flow rate, 10^6 lbm/hr	138.*
Reactor coolant system pressure, psia	2,110
Steam generator pressure, psia	770
Initial pressurizer liquid volume, ft^3	800
Steam generator level, ft above tube sheet	31.74

* Lower core flowrate dispositioned in Reference 6.

SETPOINTS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

Parameter	Setpoint
Steam generator MSSV setpoint, psia	1000
AFW actuation on steam generator level	
AFAS signal generation, % NR	23.7
SIAS setpoint, psia	1605
Shutdown cooling entry conditions:	
Hot leg temperature, °F	300
Pressurizer pressure, psia	270

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

Time (second)	Event	Setpoint or Value
1.0	Tube rupture occurs	----
32.9	Proportional heaters are fully energized, psia	2085
105.7	Backup heaters are energized, psia	2035
211.3	Heaters are de-energized on low level in the pressurizer, ft^3	558.6
703.7	Pressurizer pressure reaches low pressurizer pressure setpoint (TM/LP floor), psia	1700.
704.8	Trip signal is generated	
705.2	Trip breakers open	
706.1	Turbine Valves begin to close	
707.1	Turbine valves are completely closed	
708.2	Loss of offsite power	
714.8	Feedwater flow begins ramping down at a rate of 5%/second	
715.9	SIAS setpoint is reached, psia	1605
720.3	MSSVs begin to open, psia	1000
725.8	Pressurizer empties	
733.9	Safety Injection pumps reach full speed	
735.0	Upper head void begins to appear	
811.5	Safety Injection flow to RCS begins, psia	1237.7
995.0	Maximum upper head void fraction	0.271
1107.0	Minimum PCS pressure, psia	1107.8
1370.5	Upper head void disappears	

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

Time (second)	Event	Setpoint or Value
1372.0	Pressurizer begins to refill	
1466.6	Low steam generator level signal for Auxiliary feedwater actuation, ft	25.7
1586.6	Auxiliary feedwater reaches the steam generators, lbm/sec/SG	27.0
1800.0	Operator takes action, opens ADVs to initiate cooldown	
3000.0	Operator isolates the affected SG, below setpoint loop temperatures, °F	525.0
13000.0	Operator initiates steaming the affected generator to avoid overfilling, percent SG wide range span	90
23300.0	Shutdown Cooling entry condition is reached, PCS pressure, psia/temperature, °F	270/300
28800.0	PCS pressure and temperature demonstrated to be stabilized, transient terminated.	

INTEGRATED PARAMETERS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

Parameter	0-2 hr	0-8 hr
Integrated primary to secondary leak, lbm	183,202	605,101
Integrated Steam release, lbm		
a. Through affected SG ADV	37,382	313,736
b. Through affected SG MSSV	44,654	44,654
c. Through intact SG ADV	185,000	719,448
d. Through intact SG MSSV	44,645	44,645

STEAM GENERATOR TUBE RUPTURE (SGTR) RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	2703 MW_{th}
Initial PCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 100/E-bar gross activity
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131
Maximum pre-accident spike iodine concentration	40 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	0.3 gpm per SG
Time to establish shutdown cooling and terminate steam release	8 hours
PCS Mass	529,706 lb_m for pre-accident iodine spike case 459,445 lb_m for concurrent iodine spike case
SG Secondary Side Mass	141,065 lb_m per SG (minimum mass used to maximize concentration from tube leakage)
Integrated Mass Release	Table 14.15-6
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG (flashed tube flow) – Table 14.15-11 Faulted SG (non-flashed tube flow) – 100 Intact SG – 100
Break Flow Flash Fraction	Table 14.15-7
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode	20 minutes
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

SGTR RADIOLOGICAL ANALYSIS – INTEGRATED MASS RELEASES [1]

Time (hours)	Break Flow in Ruptured SG (lb_m)	Steam Release from Ruptured SG (lb_m)	Steam Release from Unaffected SG (lb_m)
0 – 0.196417	24,011.15	0	0
0.196417 - 0.5	37,111.85	44,654	53,574
0.5 - 1.388889	81,281	22,152.3	109,629.6
1.388889 - 2	40,798	15,229.7	75,370.4
2 - 3.638889	64,773	75,485.6	145,983.5
3.638889 - 8	357,126	200,868.4	388,464.5
8 - 720	0	0	0

[1] Flowrate assumed to be constant within time period

SGTR RADIOLOGICAL ANALYSIS – FLASHING FRACTION FOR FLOW FROM BROKEN TUBE

Time (seconds)	Flashing Fraction
0	0.110
707.1	0.065
736	0.031
859	0.023
1090	0.006
1800	0.006

SGTR RADIOLOGICAL ANALYSIS – 40 μCI/GM D.E. I-131 ACTIVITIES

Isotope	Activity (μCi/gm)
Iodine-131	33.2194
Iodine-132	7.6660
Iodine-133	34.4971
Iodine-134	3.0025
Iodine-135	14.6932

SGTR RADIOLOGICAL ANALYSIS – IODINE EQUILIBRIUM APPEARANCE ASSUMPTIONS

Input Assumption	Value
Maximum Letdown Flow	40 gpm
Assumed Letdown Flow *	44 gpm at 120°F, 2060 psia
Maximum Identified PCS Leakage	10 gpm
Maximum Unidentified PCS Leakage	1 gpm
PCS Mass	459,445 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus 10% uncertainty

SGTR **RADIOLOGICAL ANALYSIS – CONCURRENT (335 X) IODINE SPIKE APPEARANCE RATE**

Isotope	Appearance Rate (Ci/min)	Time of Depletion (hours)
Iodine-131	58.0966961	> 8
Iodine-132	79.8319317	> 8
Iodine-133	90.1310904	> 8
Iodine-134	74.0318685	> 8
Iodine-135	68.9790622	> 8

SGTR RADIOLOGICAL ANALYSIS – AFFECTED STEAM GENERATOR WATER LEVEL AND DECONTAMINATION FACTORS FOR FLASHED FLOW

Time (seconds)	Water Level Above U-Tubes (feet)	Calculated Decontamination Factor	Decontamination Factor Used in Analysis
0	0.0 (assumed)*	1.0	1.0
707.1	0.0 (assumed)*	1.0	1.0
736	0.11	1.002299	1.002299
859	0.55	1.045037	1.045037
1090	1.39	1.452436	1.452436
1800	3.97	1.467378	1.467378
5000	6.79	60.03443	1.467378
7200	9.43	38.01867	1.467378
13100	12.34	553073.5	58.16008
28800	15.16	58.16008	58.16008

*It is conservatively assumed that no scrubbing occurs until after the reactor trip at 707.1 seconds. Since the U-tubes remain covered throughout the event, it is also conservatively assumed that at the time of trip the water level is just above the top of the U-tubes. The time-dependent water level after the trip is a function of the allowable primary to secondary leakage, broken tube flow, and MSSV/ADV releases from the affected steam generator. To minimize the water level available for scrubbing, the location of the tube break is assumed to be at the top of the U-tubes.

EVENT SUMMARY FOR THE EOC HZP CONTROL ROD EJECTION

EVENT	VALUE	TIME (sec)
Ejection of a Single Control Rod	-------	0.0
Core Power Reached VHP Trip Setpoint	36.86% RTP	0.309
Core Power Peaked	1,903%RTP	0.410
Core Average Rod Surface Heat Flux Peaked	101.9% RTP	0.507
Minimum DNBR Occurred	see Table 14.1-5	0.507
Scram Rod Insertion Begins	-------	1.409

CONTROL ROD EJECTION RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Maximum Radial Peaking Factor	2.04
% DNB Fuel	14.7%
% Fuel Centerline Melt	0.5%
LOCA Source Term	Table 14.22-3
Initial PCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 100/E-bar gross activity
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131
Release From DNB Fuel	Section 1 of Appendix H to RG 1.183
Release From Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	100
Steam Generator Tube Leakage	0.3 gpm per SG
Time to establish shutdown cooling	8 hours
PCS Mass	432,976.8 lb_m
SG Secondary Side Mass	minimum – 141,065 lb_m (per SG) Minimum mass used for SGs to maximize steam release nuclide concentration.
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97 % Organic – 3%
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Time of Control Room Ventilation System Isolation	20 minutes
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

CONTROL ROD EJECTION RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Containment Volume	1.64E+06 ft^3
Containment Leakage Rate	
0 to 24 hours	0.10% (by weight)/day
after 24 hours	0.05% (by weight)/day
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 1.3 hr^{-1} Organic Iodine – None

CONTROL ROD EJECTION RADIOLOGICAL ANALYSIS – STEAM RELEASE

Time	SG Steam Release (lb_m)
0 – 1100 sec	107,158.8
1100 sec – 0.5 hours	31,336.8
0.5 hr – 8 hr	1,007,100
>8 hr	0

SAMPLED LBLOCA PARAMETERS

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties and burnup)
	Break type (guillotine versus split)
	Break size
	Critical flow discharge coefficients (break)
	Decay heat
	Critical flow discharge coefficients (surgeline)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
	T_{min} (intersection of film and transition boiling)
	Initial stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condensation interphase heat transfer
	Metal-water reaction
Plant[1]	
	Offsite power availability
	Core power and power distribution
	Pressurizer pressure
	Pressurizer liquid level
	SIT pressure
	SIT liquid level
	SIT temperature (based on containment temperature)
	Containment temperature
	Containment volume
	Initial flow rate
	Initial operating temperature
	Diesel start (for loss of offsite power only)

[1] Uncertainties for plant parameters are based on plant-specific values with the exception of “Offsite power availability,” which is a binary result that is specified by the analysis methodology.

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

	Event	Operating Range
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.417 in
	b) Cladding inside diameter	0.367 in
	c) Cladding thickness	0.025 in
	d) Pellet outside diameter	0.360 in
	e) Pellet density	96.0% of theoretical
	f) Active fuel length	132.6 in
	g) Resinter densification	[2%]
	h) Gd_2O_3 concentrations	2, 4, 6 and 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis considers plant-specific form and friction losses
	b) Pressurizer location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	15x15 Framatome
	e) SG tube plugging	15%
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	2,565.4 MWt
	b) LHR	≤ 15.28 kW/ft[1]
	c) F_r^T	≤ 2.04[2]
	2.2 Fluid Conditions	
	a) Loop flow	130 Mlbm/hr ≤ M ≤ 145 Mlbm/hr
	b) PCS inlet core temperature	537 ≤ T ≤ 544 °F[3]
	c) Upper head temperature	< core outlet temperature
	d) Pressurizer pressure	2,010 ≤ P ≤ 2,100 psia[4]
	e) Pressurizer liquid level	46.25% ≤ L ≤ 67.8%
	f) SIT pressure	214.7 ≤ P ≤ 239.7 psia
	g) SIT liquid volume	1,040 ≤ V ≤ 1,176 ft^3
	h) SIT temperature	80 ≤ T ≤ 140 °F (coupled to containment temperature)
	i) SIT resistance (fL/D)	As-built piping configuration
	j) Minimum ECCS boron	≥ 1,720 ppm

[1] Includes a 5% local LHR measurement uncertainty, a 3% engineering uncertainty and a 0.5925% thermal power measurement uncertainty.
[2] Includes a 4.25% measurement uncertainty.
[3] Sampled range of +7 °F includes both operational tolerance and measurement uncertainty.
[4] Based on representative plant values, including measurement uncertainty.

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

	Event	Operating Range
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge piping
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to CL pipe)	$0.05 \leq A \leq 0.5$ full pipe area (split) $0.5 \leq A \leq 1.0$ full pipe area (guillotine)
	d) Worst single-failure	Loss of one ECCS pumped injection train
	e) Offsite power	On or Off
	f) LPSI flow	Minimum flow
	g) HPSI flow	Minimum flow
	h) ECCS pumped injection temperature	100 °F
	i) HPSI delay time	30 (w/ offsite power) 40 seconds (w/o offsite power)
	j) LPSI delay time	30 (w/ offsite power) 40 seconds (w/o offsite power)
	k) Containment pressure	14.7 psia, nominal value
	l) Containment temperature	$80 \leq T \leq 140$ °F
	m) Containment spray/fan cooler delays	0/0 seconds

STATISTICAL DISTRIBUTION USED FOR PROCESS PARAMETERS

Parameter	Operational Uncertainty Distribution	Parameter Range
Core Power Operation (%)	Uniform	100.0 - 100.5
Pressurizer Pressure (psia)	Uniform	2,010 – 2,100
Pressurizer Liquid Level (%)	Uniform	46.25 – 67.8
SIT Liquid Volume (ft^3)	Uniform	1,040 – 1,176
SIT Pressure (psia)	Uniform	214.7 – 239.7
Containment/SIT Temperature (°F)	Uniform	80 – 140
Containment Volume[1] ($x10^6$ ft^3)	Uniform	1.64 – 1.80
Initial Flow Rate (Mlbm/hr)	Uniform	130 – 145
Initial Operating Temperature (°F)	Uniform	537 – 544
SIRWT Temperature (°F)	Point	100
Offsite Power Availability[2]	Binary	0,1
Delay for Containment Sprays (s)	Point	0
Delay for Containment Fan Coolers (s)	Point	0
HPSI Delay (s)	Point	30 (w/ offsite power) 40 (w/o offsite power)
LPSI Delay (s)	Point	30 (w/ offsite power) 40 (w/o offsite power)

[1] Uniform distribution for parameter with demonstrated PCT importance conservatively produces a wider variation of PCT results relative to a normal distribution. Treatment consistent with approved RLBLOCA evaluation model (Reference 5).

[2] No data are available to quantify the availability of offsite power. During normal operation, offsite power is available. Since the loss of offsite power is typically more conservative (loss in coolant pump capacity), it is assumed that there is a 50 percent probability the offsite power is unavailable.

SUMMARY OF MAJOR PARAMETERS FOR THE LIMITING PCT CASE

	6.0 % Gad Rod
Core Average Burnup (EFPH)	**7,381.22**
Core Power (MWt)	**2,572.79**
Hot Rod LHR, kW/ft	**14.60**
Total Hot Rod Radial Peak (F_r^T)	**2.040**
Axial Shape Index (ASI)	**0.1602**
Break Type	**Guillotine**
Break Size (ft^2/side)	**3.339**
Offsite Power Availability	**Not Available**
Decay Heat Multiplier	**1.01073**

SUMMARY OF HOT ROD LIMITING PCT RESULTS

	15 x 15 Framatome
Fuel Type	w/o Gd_2O_3
Case Number	22
PCT	
Temperature	1,740 °F
Time	27.2 s
Elevation	2.151 ft
Metal-Water Reaction	
Oxidation Maximum	0.59%
Total Oxidation	< 0.01%

CALCULATED EVENT TIMES FOR THE LIMITING PCT CASE

Event	Time (sec)
Break Opened	0
PCP Trip	0
SIAS Issued	0.6
Start of Broken Loop SIT Injection	14.9
Start of Intact Loop SIT Injection (loops 1B, 2A and 2B, respectively)	17.1, 17.1 and 17.1
Beginning of Core Recovery (Beginning of Reflood)	27.2
PCT Occurred	27.2
Start of HPSI	40.6
LPSI Available	40.6
Broken Loop LPSI Delivery Began	40.6
Intact Loop LPSI Delivery Began (loops 1B, 2A and 2B, respectively)	40.6, 40.6 and 40.6
Broken Loop HPSI Delivery Began	40.6
Intact Loop HPSI Delivery Began (loops 1B, 2A and 2B, respectively)	40.6, 40.6, 40.6
Broken Loop SIT Emptied	50.7
Intact Loop SIT Emptied (loops 1B, 2A and 2B, respectively)	50.8, 54.6 and 53.1
Transient Calculation Terminated	300

CONTAINMENT HEAT SINK DATA

Heat Sink		Surface Area (ft^2)	Total thickness (ft)	Material
1	Containment Dome and Upper Wall	69,630.20	0.0208	Carbon steel liner; no coatings
			4.2625	Concrete; no coating
2	Containment Wainscot	2,200.20	0.0208	Carbon steel liner; no coating
			4.2625	Concrete; no coatings
3	Containment Floor Slab	7,567.80	1.5	Concrete; no paint
			0.0208	Carbon steel; no paint
			15.971	Concrete; no paint
4	Containment Sump Slab	380.10	0.0156	Stainless steel
			1.5	Concrete; no coating
			0.0208	Carbon steel; no paint
			28.3	Concrete; no coating
5	Reactor Cavity Slab	380.10	0.0208	Stainless steel
			1.4792	Concrete; no coating
6	Lower Biological Shield	243.4 (Inner surface of cylindrical shape)	0.015625	Stainless steel; no paint
			7.9167	Concrete; no coating
7	Internal Concrete with Carbon Steel Liner Plate	2,048.40	0.0208	Carbon steel
			3.8958	Concrete; no coating
8	Internal Concrete with Stainless Steel Liner Plate	4,712.70	0.0417	Stainless steel
			2.4083	Concrete; no coating
9	Internal Concrete with Decking	2,672.90	0.004	Carbon steel liner; no coating
			2.4833	Concrete; no coatings
10	Internal Concrete	62,870.90	1.708	Concrete; no coating
11	Gravel Pit	384.50	4.208	Concrete; no coating
12	Equipment Tanks and Heat Exchangers	18,011.00	0.0364	Carbon steel; no paint
13	Miscellaneous Equipment	18,344.80	0.0112	Carbon steel; no coating
14	Polar Crane	8,241.50	0.1258	Carbon steel; no coating
15	Ductwork plus Electrical Panels	31,127.50	0.0026	Carbon steel; no coating
16	Grating	16,812.20	0.00692	Carbon steel; no coating
17	quarter inch Structural Steel	35,812.90	0.0217	Carbon steel; no coating
18	half inch Structural Steel	48,705.20	0.0433	Carbon steel; no coating
19	Sump Strainer and Piping	3,750.00	0.00645	Stainless steel

CONTAINMENT INITIAL AND BOUNDARY CONDITIONS

Parameter	Parameter Value	
Containment free volume range, ft^3	1.64E+06 to 1.80E+06	
Initial relative humidity	100.0 %	
Initial compartment pressure, psia	14.7, nominal value	
Initial compartment temperature, °F	80 ≤ T ≤ 140	
Containment spray time of delivery, sec	0.0	
Containment spray flow rate, lb/sec	576.7	
Containment spray temperature, ºF	40.0	
Fan cooler heat removal as a function of temperature	Temp (ºF)	Heat Removal (BTU/sec)
	284	-196242.0
	264	-157899.0
	244	-118137.0
	224	-82197.0
	204	-53190.0
	184	-32475.0
	164	-19533.0
	144	-11559.0
	124	-6735.0
	104	-3831.0
	35	0.0

SYSTEM PARAMETERS AND INITIAL CONDITIONS USED IN THE PALISADES SBLOCA ANALYSIS

Parameter	Palisades Analysis Value
Primary Heat Output, Mwth	2580.6
Primary Coolant Flow, gpm	341,400
Operating Pressure, psia	2060
Inlet Coolant Temperature, °F	544
SIT Pressure, psia	215
SIT Fluid Temperature, °F	100
Steam Generator Tube Plugging, %	15
SG Secondary Pressure, psia	763
SG Main Feedwater Temperature, °F	439.5
SG Auxiliary Feedwater Temperature, °F	120
HPSI Fluid Temperature, °F	100
Reactor Scram Low Pressure Setpoint (TM/LP floor), psia	1585
Reactor Scram Delay Time on TM/LP, s	0.8
Scram CEA Holding Coil Release Delay Time, s	0.5
SIAS Activation Setpoint Pressure, psia	1450
HPSI Pump Delay Time on SIAS, s	40
Main Steam Safety Valve Setpoint Pressure, psia	
MSSV-1	1029.3
MSSV-2	1049.9
MSSV-3	1070.5

PCT RESULTS OF THE PALISADES SBLOCA ANALYSIS

Break Size (ft^2)	PCT (°F)
0.04	1296
0.05	1451
0.06	1479
0.08	1734
0.10	1654
0.15	1356

SEQUENCE OF EVENTS FOR THE PALISADES SBLOCA EVENT

Event	Time (s)
Break in Cold Leg 2B opened	0.0
Pressurizer Pressure reached TM/LP setpoint	16.98
Reactor scram	18.28
Loss of off-site power	18.28
MFW terminated	18.28
Turbine tripped	18.28
Pressurizer pressure reaches SIAS setpoint (1450 psia)	24.86
Minimum SG level reaches AFAS setpoint (23.7% span)	25
HPSI pump ready for delivery	64.86
Cold Leg pressure reaches HPSI shutoff head (1200.7 psia)	96
Motor-driven AFW delivery begins	145
Loop seal in Cold Leg 1B cleared	282
Break uncovered	300
PCT occurs	1690
SIT discharge begins	1690
Reactor vessel mass inventory reaches minimum value	1698

SBLOCA ANALYSIS CALCULATION RESULTS

Peak Cladding Temperature	
Temperature (°F)	1734
Time (s)	1690
Elevation (ft)	10.2
Metal-Water Reaction	
Local Maximum (%)	2.0
Elevation of Local Maximum (ft)	10.2
Total Core Wide (%)	<1.0

MAXIMUM STRESSES, PRESSURES AND DEFLECTIONS IN CRITICAL REACTOR INTERNALS FOLLOWING A MAJOR LOSS OF COOLANT ACCIDENT

Structural Component	Failure Mode and Loading Condition	Location of Failure	Failure Condition(a)	Allowable Condition(b)	Calculated Condition
Core Barrel	Tension - Axial Load	Middle Section of Core Barrel	54,000 psi	29,300 psi	3,200 psi
	Buckling - External Pressure	Upper Portion of Core Barrel (Arch)	Δp = 572 psi	Δp = 381 psi	Δp = 380 psi
	Tension - Internal Pressure	Middle Section of Core Barrel	54,000 psi	29,300 psi	26,750 psi
Lower Core Support	Bending - Transverse Load	Beam Flange	54,000 psi	43,950 psi	22,510 psi
	Shear - Transverse Load	Junction of Flange to Web	32,400 psi	17,580 psi	7,710 psi
Control Rod Shrouds 1st Row (Near Nozzle)	Bending - Axial and Transverse Load	Lower End of Shroud	54,000 psi	32,230 psi	70,310 psi
	Deformation - Axial and Transverse Load	Center of Shroud	Defl = 0.76"	Defl = 0.51"	Defl > 0.51"

(a) The figures in this column represent the estimated stress, pressure or deflection limits at which the component will no longer perform its function.
(b) The figures in this column represent the allowable stress, pressure or deflection limits in accordance with the design bases established in Chapter 3 of this FSAR.

MAXIMUM STRESSES, PRESSURES AND DEFLECTIONS IN CRITICAL REACTOR INTERNALS FOLLOWING A MAJOR LOSS OF COOLANT ACCIDENT

Structural Component	Failure Mode and Loading Condition	Location of Failure	Failure Condition(a)	Allowable Condition(b)	Calculated Condition
Control Rod Shrouds 2nd Row	Bending - Axial and Transverse Load	Lower End of Shroud	54,000 psi	32,230 psi	28,090 psi
	Deformation - Axial and Transverse Load	Center of Shroud	Defl = 0.76"	Defl = 0.51"	Defl - 0.279"
Upper Grid Beam	Bending - Transverse Load	Center of Beam	54,000 psi	43,950 psi	12,980 psi
Upper Structure Flange	Bending - Axial Load	Junction of Flange and Barrel Cylinder	54,000 psi	43,950 psi	40,630 psi

(a) The figures in this column represent the estimated stress, pressure or deflection limits at which the component will no longer perform its function.
(b) The figures in this column represent the allowable stress, pressure or deflection limits in accordance with the design bases established in Chapter 3 of this FSAR.

ASYMMETRIC LOADS ANALYSIS - REACTOR VESSEL INTERNAL COMPONENT STRESS MARGINS

Component	Location	Percent Margin (%)*
Core Support Barrel	Upper Flange	6
	Upper Cylinder	7
	Center Cylinder	11
Lower Support Structure	Support Columns	2
	Beams	3
	Core Support Plate	13
Upper Guide Structure	Grid Beams	1

*Percent margin is computed as (S_{allow} - S_{calc}) (100%) / S_{allow}, where S_{calc} is the calculated component stress and S_{allow} is the ASME Code allowable stress.

LOCA ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
1.	Containment Wall and Dome Carboline 3912 Carbo Zinc 11 Carbon Steel Liner Air Gap Concrete	69,630.2
2.	Containment Wainscot Phenoline 305 Carbo Zinc 11 Carbon Steel Liner Air Gap Concrete	2,200.2
3.	Containment Floor Slab Phenoline 305 Carboline 195 Concrete Air Gap Carbon Steel Air Gap Concrete	7,567.8
4.	Containment Sump Slab Stainless Steel Air Gap Concrete Air Gap Carbon Steel Liner Air Gap Concrete	380.1
5.	Reactor Cavity Slab (Note 1) Stainless Steel Air Gap Concrete Air Gap Unibestos Stainless Steel	380.1

LOCA ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
6.	Lower Biological Shield (Note 2) Stainless Steel Air Gap Concrete	417.8
7.	Internal Concrete Phenoline 305 Carboline 195 Concrete	61,337.5
8.	Internal Concrete with Carbon Steel Liner Plate Stainless Steel Wool Carbon Steel Air Gap Concrete	2,048.4
9.	Internal Concrete with Stainless Steel Liner Plate Stainless Steel Air Gap Concrete	4,712.7
10.	Internal Concrete with Decking (Note 3) Carbon Steel Air Gap Concrete Carboline 195 Carboline 305	2,672.9
11.	Gravel Pit Phenoline 305 Carboline 195 Concrete/Gravel Mixture	375.1
12.	Structural Steel Adjacent to the Liner Plate Carboline 3912 Carbo Zinc 11 Carbon Steel	30,609.3
13.	Structural Steel Carbo Zinc 11 Carbon Steel	41,628.4

LOCA ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
14.	Polar Crane Carboline 3912 Carbo Zinc 11 Carbon Steel	7,044
15.	Pressurizer Quench Tank (Note 4) Carbon Steel Carbo Zinc 11	679
16.	Safety Injection Tanks (Note 5) Stainless Steel Carbon Steel Carbo Zinc 11	4,098.4
17.	Clean Waste Receiver Tanks (Note 6) Carbon Steel Carbo Zinc	9,255.6
18.	Clean Waste Receiver Tank Skirts (Note 7) Carbon Steel Carbo Zinc 11	3,577.2
19.	Shield Cooling Surge Tank (Note 8) Carbon Steel Carbo Zinc 11	112.2
20.	Deleted	
21.	Letdown Heat Exchanger Phenoline 305 Carbo Zinc 11 Carbon Steel	101.8
22.	Shield Cooling Heat Exchanger Carbo Zinc 11 Carbon Steel Water	25

LOCA ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
23.	Head Lift Rig and Containment Air Coolers Phenoline 305 Carbon Steel	14,308.2
24.	Electrical Panels Carbo Zinc 11 Carbon Steel	2,141.4
25.	Refueling Mast and Grapple Stainless Steel	1,371.1
26.	Grating Carbon Steel	14,369.4
27.	Ductwork Carbon Steel	24,463.3
28.	PCS Metal Wall #1 Reactor Vessel and Internals	35,539.2
29.	PCS Metal Wall #2 Reactor Vessel and Internals	12,441.8
30.	PCS Metal Wall #3 Reactor Core	10,378.8

LOCA ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

Notes:

1 The reactor cavity slab heat conductor is in contact with the containment atmosphere on both sides.

2 The lower biological shield heat conductor is a tube. While the surface area specified above represents the outside surface area, only the inside surface area is in contact with the containment atmosphere.

3 The internal concrete with decking heat conductor is in contact with the containment atmosphere on both sides.

4 The pressurizer quench tank heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

5 The safety injection tanks heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

6 The clean waste receiver tanks heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

7 The clean waste receiver tank skirts heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

8 The shield cooling surge tank heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

LOCA ANALYSIS ENGINEERED SAFEGUARDS EQUIPMENT ALIGNMENT

	D/G 1-2 Failure Equipment Operated	D/G 1-1 Failure Equipment Operated
Containment Sprays	P-54B & P54C	P-54A
LPSI	P-67B	P-67A
HPSI	P-66B	P-66A
Containment Air Coolers		VHX-1, VHX-2 & VHX-3
Component Cooling Water	P-52A & P-52C	P-52B
Service Water	P-7B	P-7A & P-7C

LOCA INITIAL CONDITIONS

Containment Free Volume	1.64×10^6 ft^3
Containment Temperature	145°F
Containment Pressure	15.7 Psia
Relative Humidity	30%
SIRW Tank Temperature	100°F

CONTAINMENT BUILDING RESPONSE TO LOCA DOUBLE ENDED GUILLOTINE BREAK IN A HOT LEG

Case	Peak Pressure (Psig)	Time (Sec)
D/G 1-2 Failure	54.2	13.2
D/G 1-1 Failure	54.2	13.2

The peaks for both cases are the same because they occurred so early in the transient that the differences in safeguards equipment used had not yet taken effect.

LOCA ANALYSIS PARAMETER ASSUMPTIONS

Initial Containment Air Temp	145°F
Initial Containment Pressure	15.7 Psia (1.0 Psig)
Relative Humidity	30%
CCWHX Tube Fouling Coefficient	.001 hr-ft^2-°F/BTU
Service Water Temperature	85°F
D/G 1-2 (RCF) Failure Data	
ECCS Injection Flow pre-RAS (1 HPSI, 1 LPSI pump)	3,471 gpm
ECCS Injection Flow post-RAS (1 HPSI pump)	705 gpm
1 SW Pump Flow Rate to CCWHXs	4,214 gpm
1 CCW Pump Flow Rate to SDCHXs	4,480 gpm
2 CS Pump Flow Rate to Containment (pre RAS)	2,472 gpm
2 CS Pump Flow Rate to Containment (post RAS-HLI)	1,684 gpm
Post-RAS Spillage after Initiation of Hot Leg Injection	328 gpm
ECCS Injection Flow after Initiation of Hot Leg Injection	273 gpm
D/G 1-1 (LCF) Failure Data	
ECCS Injection Flow pre-RAS (1 HPSI, 1 LPSI pump)	3,443 gpm
ECCS Injection Flow post-RAS (1 HPSI pump)	703 gpm
2 SW Pump Flow Rate to CCWHXs	4,286 gpm
2 SW Pump Flow Rate to 3 Containment Air Coolers	1, 600 gpm/Air Cooler
1 CCW Pump Flow Rate to SDCHXs	4,480 gpm
1 CS Pump Flow Rate to Containment (pre RAS)	1,781 gpm
1 CS Pump Flow Rate to Containment, 1 header (pre RAS)	1,233 gpm
1 CS Pump Flow Rate to Containment (post RAS-HLI)	788 gpm
Post-RAS Spillage after Initiation of Hot Leg Injection	308 gpm
ECCS Injection Flow after Initiation of Hot Leg Injection	279 gpm

INITIAL CONDITIONS FOR THE MSLB CONTAINMENT ANALYSIS

Parameter	Assumed Value
Containment Free Volume, ft^3	1.64×10^6
Initial Containment Temperature, °F	145.0
Initial Containment Pressure, psig	1.0*
Initial Containment Humidity, %	30
Containment Spray Water Temperature, °F	100.0
Main Feedwater Regulating Valve Closure Time, sec	22
Main Steam Isolation Valve Closure Time, sec	2

* Zero power cases assumed 1.5 psig

INITIAL CONDITIONS FOR THE MSLB CONTAINMENT ANALYSIS
Power- and Case-Dependent Parameters for CONTRANS Code

Case	Power %	Power MWTh*	Cold Leg Temp, °F	S/G Pressure psia	PCS Flow Rate, lbm/hr#
102%	102	2600.6	550.65	770.0	$144.6x10^6$
75%	75	1917.5	548.70	784.0	$144.6x10^6$
0%	0	20.0	539.00	900.0	$144.6x10^6$
EEQ	102	2600.6	550.65	770.0	$144.6x10^6$

* This power level includes an assumed contribution of 20 MWTh from the primary coolant pumps.

\# Lower PCS flowrate dispositioned in Reference 25.

MSLB CONTAINMENT ANALYSIS RESULTS

Case Description	Power Level	Peak Pressure (psig)
Limiting Pressure - Relay 5P-7 Failure w/Open MSIV Bypass Valves	0%	53.5

MSLB ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
1.	Containment Wall and Dome Carboline 3912 Carbo Zinc 11 Carbon Steel Liner Air Gap Concrete	69,630.2
2.	Containment Wainscot Phenoline 305 Carbo Zinc 11 Carbon Steel Liner Air Gap Concrete	2,200.2
3.	Containment Floor Slab Phenoline 305 Carboline 195 Concrete Air Gap Carbon Steel Air Gap Concrete	7,567.8

MSLB ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
4.	Containment Sump Slab Stainless Steel Air Gap Concrete Air Gap Carbon Steel Liner Air Gap Concrete	380.1
5.	Reactor Cavity Slab (Note 1) Stainless Steel Air Gap Concrete Air Gap Unibestos Stainless Steel	380.1
6.	Lower Biological Shield (Note 2) Stainless Steel Air Gap Concrete	417.8
7.	Internal Concrete Phenoline 305 Carboline 195 Concrete	61,337.5
8.	Internal Concrete with Carbon Steel Liner Plate Stainless Steel Wool Carbon Steel Air Gap Concrete	2,048.4
9.	Internal Concrete with Stainless Steel Liner Plate Stainless Steel Air Gap Concrete	4,712.7

MSLB ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
10.	Internal Concrete with Decking (Note 3) Carbon Steel Air Gap Concrete Carboline 195 Carboline 305	2,672.9
11.	Gravel Pit Phenoline 305 Carboline 195 Concrete/Gravel Mixture	375.1
12.	Structural Steel Adjacent to the Liner Plate Carboline 3912 Carbo Zinc 11 Carbon Steel	30,609.3
13.	Structural Steel Carbo Zinc 11 Carbon Steel	41,628.4
14.	Polar Crane Carboline 3912 Carbo Zinc 11 Carbon Steel	7,044
15.	Pressurizer Quench Tank (Note 4) Carbon Steel Carbo Zinc 11	679
16.	Safety Injection Tanks (Note 5) Stainless Steel Carbon Steel Carbo Zinc 11	4,098.4
17.	Clean Waste Receiver Tanks (Note 6) Carbon Steel Carbo Zinc	9,255.6
18.	Clean Waste Receiver Tank Skirts (Note 7) Carbon Steel Carbo Zinc 11	3,577.2

MSLB ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

	HEAT SINK	SURFACE AREA (ft^2)
19.	Shield Cooling Surge Tank (Note 8) Carbon Steel Carbo Zinc 11	112.2
20.	Deleted	
21.	Letdown Heat Exchanger Phenoline 305 Carbo Zinc 11 Carbon Steel	101.8
22.	Shield Cooling Heat Exchanger Carbo Zinc 11 Carbon Steel Water	25
23.	Head Lift Rig and Containment Air Coolers Phenoline 305 Carbon Steel	14,308.2
24.	Electrical Panels Carbo Zinc 11 Carbon Steel	2,141.4
25.	Refueling Mast and Grapple Stainless Steel	1,371.1
26.	Grating Carbon Steel	14,369.4
27.	Ductwork Carbon Steel	24,463.3

MSLB ANALYSIS CONTAINMENT BUILDING HEAT SINKS/SOURCES

Notes:

1 The reactor cavity slab heat conductor is in contact with the containment atmosphere on both sides.

2 The lower biological shield heat conductor is a tube. While the surface area specified above represents the outside surface area, only the inside surface area is in contact with the containment atmosphere.

3 The internal concrete with decking heat conductor is in contact with the containment atmosphere on both sides.

4 The pressurizer quench tank heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

5 The safety injection tanks heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

6 The clean waste receiver tanks heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

7 The clean waste receiver skirts heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

8 The shield cooling surge tank heat conductor is a tube. The surface area specified above represents the outside surface area, which is the carbo zinc 11 side and which is in contact with the containment atmosphere.

REACTOR CAVITY GEOMETRIC FACTORS

Volume of Cavity	6,653 ft^3
Volume of Sump	1,364 ft^3
Mass of Upper Seal	3,000 lb
Refueling Pool Seal Breaks and Begins To Lift at	5.8 Psi

	Total Flow Area (ft^2)	Forward Loss Coefficient (ft^2)
Refueling Pool Seal		
Before Breaking Away	4.77	0.57
After Broken Away	82.23	1.42
Annulus Around Coolant Pipes	24.2	1.45
30-Inch Access Tube	4.75	2.37
6 Pipes Into Sump	10.1	1.19

GEOMETRY AND PEAK PRESSURES IN STEAM GENERATOR COMPARTMENTS

Steam Generator Compartment	Volume (ft^3)	Vent Area (ft^2)	Peak Pressure (Psi)
North	55,210	1,043.3	24.8
South	62,090	1,091.3	22.4

DIFFERENTIAL PRESSURES AT VARIOUS LOCATIONS

		Calculated Pressure (Psi)	Design Pressure (Psi)
1.	Maximum Uplift Differential Pressure Across the Reactor Cavity Floor for a 42-Inch Pipe Double-Ended Rupture Outside the Reactor Cavity	0.4	7.3
2.	Maximum Differential Pressure Across the Primary Shield Walls Due To a Break of a 42-Inch Pipe Within the Reactor Cavity	52.4	72
3.	Maximum Differential Pressure Across the Primary Shield Walls Due To a Break of a 30-Inch Pipe Within the Reactor Cavity	67.7	72
4.	Maximum Differential Pressure Across Secondary Shield Walls of the North Steam Generator Compartment Due To a 42-Inch Pipe Double-Ended Rupture Within the Compartment	24.8	31
5.	Maximum Differential Pressure Across the Secondary Shield Walls of the South Steam Generator Compartment Due To a 42-Inch Pipe Double-Ended Rupture Within the Compartment	22.4	27

FUEL HANDLING ACCIDENT (FHA) RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level Before Shutdown	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Discharged Fuel Assembly Burnup	39,300 – 58,900 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Maximum Radial Peaking Factor	2.04
Number of Fuel Assemblies Damaged	1 fuel assembly
Delay Before Spent Fuel Movement	48 hours
FHA Source Term for a Single Assembly	Table 14.19-2
Water Level Above Damaged Fuel Assembly	22.5 feet minimum
Iodine Decontamination Factors	Elemental – 252 Organic – 1 Overall – 183.07
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Time of Control Room Ventilation System Isolation	20 minutes
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6
FHB Ventilation Filter Efficiencies	Elemental iodine – 94% Organic iodine – 94% Noble gas – n/a

FUEL HANDLING ACCIDENT RADIOLOGICAL ANALYSIS – SOURCE TERM

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
Co-58	0.0000E+00	I-135	0.8949E+04	Sb-126	0.9900E+03
Co-60	0.0000E+00	Xe-133	0.1298E+07	Te-131	0.8307E+04
Kr-85	0.1052E+05	Xe-135	0.8201E+05	Te-133	0.2034E-10
Kr-85m	0.1174E+03	Cs-134	0.2034E+06	Te-134	0.2217E-14
Kr-87	0.1647E-05	Cs-136	0.5284E+05	Te-125m	0.3417E+04
Kr-88	0.4302E+01	Cs-137	0.1100E+06	Te-133m	0.1213E-09
Rb-86	0.1819E+04	Ba-139	0.4861E-04	Ba-141	0.0000E+00
Sr-89	0.7020E+06	Ba-140	0.1130E+07	Ba-137m	0.1041E+06
Sr-90	0.8456E+05	La-140	0.1235E+07	Pd-109	0.2825E+05
Sr-91	0.2679E+05	La-141	0.2730E+03	Rh-106	0.5771E+06
Sr-92	0.4453E+01	La-142	0.5794E-03	Rh-103m	0.1097E+07
Y-90	0.8623E+05	Ce-141	0.1168E+07	Tc-101	0.0000E+00
Y-91	0.9107E+06	Ce-143	0.4100E+06	Eu-154	0.1246E+05
Y-92	0.3229E+03	Ce-144	0.1014E+07	Eu-155	0.8442E+04
Y-93	0.4137E+05	Pr-143	0.1071E+07	Eu-156	0.1935E+06
Zr-95	0.1210E+07	Nd-147	0.4211E+06	La-143	0.0000E+00
Zr-97	0.1684E+06	Np-239	0.1023E+08	Nb-97	0.1692E+06
Nb-95	0.1248E+07	Pu-238	0.4494E+04	Nb-95m	0.8748E+04
Mo-99	0.8264E+06	Pu-239	0.3578E+03	Pm-147	0.1296E+06
Tc-99m	0.7956E+06	Pu-240	0.5406E+03	Pm-148	0.1659E+06
Ru-103	0.1216E+07	Pu-241	0.1522E+06	Pm-149	0.2481E+06
Ru-105	0.5426E+03	Am-241	0.1897E+03	Pm-151	0.5012E+05
Ru-106	0.5771E+06	Cm-242	0.5649E+05	Pm-148m	0.2899E+05
Rh-105	0.3958E+06	Cm-244	0.1339E+05	Pr-144	0.1015E+07
Sb-127	0.6450E+05	I-130	0.2546E+04	Pr-144m	0.1217E+05
Sb-129	0.1176E+03	Kr-83m	0.3727E+00	Sm-153	0.2171E+06
Te-127	0.7344E+05	Xe-138	0.0000E+00	Y-94	0.0000E+00
Te-127m	0.1222E+05	Xe-131m	0.8276E+04	Y-95	0.0000E+00
Te-129	0.2383E+05	Xe-133m	0.3403E+05	Y-91m	0.1702E+05
Te-129m	0.3637E+05	Xe-135m	0.1434E+04	Br-82	0.2060E+04
Te-131m	0.3690E+05	Cs-138	0.0000E+00	Br-83	0.8833E-01
Te-132	0.6852E+06	Cs-134m	0.5122E+00	Br-84	0.0000E+00
I-131	0.6424E+06	Rb-88	0.4804E+01	Am-242	0.1138E+05
I-132	0.7060E+06	Rb-89	0.0000E+00	Np-238	0.2238E+06
I-133	0.3019E+06	Sb-124	0.1663E+04	Pu-243	0.5681E+03
I-134	0.2087E-09	Sb-125	0.1566E+05		

MHA SEQUENCE OF EVENTS FOR THE DOSE CONSEQUENCE ANALYSIS

Time (minutes)	Event/Action
t = 0.0	Release of radionuclides to the containment atmosphere starts and the containment atmosphere begins leaking at the T.S. leak rate limit. Loss of Off-Site Power occurs. CHP and CHR signals are generated. The control room is depressurized. Control room inleakage occurs at the base infiltration rate.
t = 1.0	Full spray flow is delivered to the containment atmosphere by the Containment Spray System. Removal of particulate and elemental iodine species begins at this time. No credit is taken for the removal of organic iodine species.
t = 1.5	The control room is pressurized to > 1/8 " H_2O and running in the E-HVAC mode with one train operational due to the loss of one safety train. Control room unfiltered inleakage past the normal intake isolation dampers and the smoke purge dampers begins.
t = 19.0	The initial SIRWT inventory is depleted and containment spray suction is aligned to the containment sump. Leakage from ESF components and via the SIRWT begins. This assumes runout flows on 2 HPSI's, 2 LPSI's, 3 Containment Spray Pumps, minimum inventory of the SIRWT, and a containment backpressure of 55 psig.
t = 150.9	The elemental iodine decontamination factor reaches 200 at this time.
t = 203.1	The aerosol iodine decontamination factor reaches 50 at this time.
t = 600.0	Containment spray flow is conservatively assumed to be terminated. However SIRWT leakage is assumed to continue as if the CSS pumps continued to operate.
t = 1440.0	The containment design leak rate is assumed to decrease to one-half (t = 24 hours) the T.S. leakrate.
t = 43200	Low Population Zone (LPZ) doses are integrated over the interval (t = 30 days) from the initiation of the incident to 30 days. Site Boundary (SB) doses are integrated over the worst 2 hour period. Control Room doses are integrated over the interval (t = 30 days) from the initiation of the incident to 30 days.

MAXIMUM HYPOTHETICAL ACCIDENT / LOSS OF COOLANT ACCIDENT (MHA/LOCA) RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Release Inputs:	
Core Power Level	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Initial PCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 100/E-bar gross activity
Core Fission Product Inventory	Table 14.22-3
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.10% (by weight)/day 0.05% (by weight)/day
MHA release phase timing and duration	Table 14.22-4
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	RG 1.183, Sections 3.1, 3.2, and Table 2
<u>ECCS Systems Leakage (from 19 minutes to 30 days)</u>	
Sump Volume (minimum)	39,054 $ft.^3$
ECCS Leakage (2 times allowed value)	0.053472 ft^3/min
Flashing Fraction	Calculated – 0.03 to 0.06 Used for dose determination – 0.10
Chemical form of the iodine released from the ECCS leakage	97% elemental, 3% organic
Iodine Decontamination Factor	2 (current design basis)
No credit taken for dilution or holdup	

MAXIMUM HYPOTHETICAL ACCIDENT / LOSS OF COOLANT ACCIDENT (MHA/LOCA) RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
<u>SIRWT Back-leakage (from 19 minutes to 30 days)</u>	
Sump Volume	292,143 gallons (minimum valve for ECCS leakage, maximizes sump iodine concentration) 430,708 gallons (maximum value for SIRWT backleakage to be consist with assumption of minimum water level in SIRWT)
ECCS Leakage to SIRWT (2 times allowed value)	7.2 gpm until 1 hours after RAS, then 0.0125 gpm
Flashing Fraction (elemental iodine assumed to be released into tank space based upon partition factor)	0% based on temperature of fluid reaching SIRWT
SIRWT liquid/vapor elemental iodine partition factor	Table 14.22-9
Elemental Iodine fraction in SIRWT	Table 14.22-8
Initial SIRWT Liquid Inventory (minimum at time of recirculation)	4,144 gallons
Release from SIRWT Vapor Space	Table 14.22-10
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Wall Deposition	2.3/hour
Containment Spray Coverage	>90%
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Aerosol Time to reach DF of 50	 4.8/hour 2.515 hours 1.8/hour (reduced to 0.18 at 3.385 hours) 3.385 hours

MAXIMUM HYPOTHETICAL ACCIDENT / LOSS OF COOLANT ACCIDENT (MHA/LOCA) RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Spray Initiation Time	60 seconds (0.016667 hours)
Control Room Ventilation System Time of automatic control room isolation and switch to emergency mode Control Room Unfiltered Inleakage	Table 14.24-1 90 seconds 16 cfm
Transport Inputs:	
Containment Leakage Release	Containment closest point
ECCS Leakage	Plant stack
SIRWT Backleakage	SIRWT vent
Personnel Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

MHA/LOCA SOURCE TERM

Nuclide	Curies	Nuclide	Curies
Co-58	0.0000E+00	Pu-239	0.3558E+05
Co-60	0.0000E+00	Pu-240	0.5406E+05
Kr-85	0.1052E+07	Pu-241	0.1522E+08
Kr-85m	0.1948E+08	Am-241	0.1884E+05
Kr-87	0.3756E+08	Cm-242	0.5669E+07
Kr-88	0.5286E+08	Cm-244	0.5943E+06
Rb-86	0.1959E+06	I-130	0.3743E+07
Sr-89	0.7213E+08	Kr-83m	0.9119E+07
Sr-90	0.8458E+07	Xe-138	0.1211E+09
Sr-91	0.8874E+08	Xe-131m	0.8346E+06
Sr-92	0.9557E+08	Xe-133m	0.4659E+07
Y-90	0.8737E+07	Xe-135m	0.2999E+08
Y-91	0.9264E+08	Cs-138	0.1340E+09
Y-92	0.9596E+08	Cs-134m	0.4920E+07
Y-93	0.1101E+09	Rb-88	0.5369E+08
Zr-95	0.1236E+09	Rb-89	0.6895E+08
Zr-97	0.1206E+09	Sb-124	0.1702E+06
Nb-95	0.1249E+09	Sb-125	0.1567E+07
Mo-99	0.1368E+09	Sb-126	0.1107E+06
Tc-99m	0.1198E+09	Te-131	0.6601E+08
Ru-103	0.1260E+09	Te-133	0.8639E+08
Ru-105	0.9451E+08	Te-134	0.1220E+09
Ru-106	0.5794E+08	Te-125m	0.3413E+06
Rh-105	0.8741E+08	Te-133m	0.5406E+08
Sb-127	0.9111E+07	Ba-141	0.1188E+09
Sb-129	0.2568E+08	Ba-137m	0.1043E+08
Te-127	0.9047E+07	Pd-109	0.3327E+08
Te-127m	0.1223E+07	Rh-106	0.6285E+08
Te-129	0.2528E+08	Rh-103m	0.1135E+09
Te-129m	0.3772E+07	Tc-101	0.1261E+09
Te-131m	0.1113E+08	Eu-154	0.1247E+07
Te-132	0.1048E+09	Eu-155	0.8448E+06
I-131	0.7483E+08	Eu-156	0.2023E+08
I-132	0.1068E+09	La-143	0.1108E+09
I-133	0.1462E+09	Nb-97	0.1216E+09
I-134	0.1602E+09	Nb-95m	0.8835E+06
I-135	0.1372E+09	Pm-147	0.1292E+08
Xe-133	0.1466E+09	Pm-148	0.2144E+08

MHA/LOCA SOURCE TERM

Nuclide	Curies	Nuclide	Curies
Xe-135	0.4692E+08	Pm-149	0.4541E+08
Cs-134	0.2037E+08	Pm-151	0.1606E+08
Cs-136	0.5873E+07	Pm-148m	0.2999E+07
Cs-137	0.1100E+08	Pr-144	0.1025E+09
Ba-139	0.1307E+09	Pr-144m	0.1224E+07
Ba-140	0.1260E+09	Sm-153	0.4423E+08
La-140	0.1299E+09	Y-94	0.1105E+09
La-141	0.1193E+09	Y-95	0.1183E+09
La-142	0.1156E+09	Y-91m	0.5151E+08
Ce-141	0.1212E+09	Br-82	0.5282E+06
Ce-143	0.1115E+09	Br-83	0.9102E+07
Ce-144	0.1020E+09	Br-84	0.1591E+08
Pr-143	0.1111E+09	Am-242	0.9062E+07
Nd-147	0.4770E+08	Np-238	0.4306E+08
Np-239	0.1830E+10	Pu-243	0.4690E+08
Pu-238	0.3927E+06		

MHA/LOCA RELEASE PHASES

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From Regulatory Guide 1.183, Table 4

MHA/LOCA TIME DEPENDENT SIRWT PH

Time (hours)	SIRWT pH
0.3167	4.500
0.50	4.508
1.3167	4.544
1.3167	4.544
2.00	4.544
4.00	4.545
8.00	4.546
16.00	4.548
24.00	4.550
48.00	4.557
72.00	4.563
96.00	4.570
120.00	4.576
144.00	4.583
168.00	4.589
192.00	4.595
240.00	4.607
288.00	4.618
336.00	4.630
384.00	4.641
432.00	4.651
528.00	4.672
624.00	4.692
720.00	4.711

MHA/LOCA TIME DEPENDENT SIRWT TOTAL IODINE CONCENTRATION

Time (hours)	SIRWT Iodine Concentration (gm-atom/liter)
0.3167	0.00E+00
0.50	9.60E-07
1.3167	4.82E-06
1.3167	4.82E-06
2.00	4.84E-06
4.00	4.90E-06
8.00	5.02E-06
16.00	5.25E-06
24.00	5.48E-06
48.00	6.16E-06
72.00	6.82E-06
96.00	7.46E-06
120.00	8.08E-06
144.00	8.68E-06
168.00	9.26E-06
192.00	9.83E-06
240.00	1.09E-05
288.00	1.20E-05
336.00	1.29E-05
384.00	1.39E-05
432.00	1.48E-05
528.00	1.64E-05
624.00	1.79E-05
720.00	1.93E-05

MHA/LOCA TIME DEPENDENT SIRWT LIQUID TEMPERATURE

Time (hr)	Temperature (^oF)
0.3167	100.0
0.50	100.0
1.3167	100.0
1.3167	100.0
2.00	100.0
4.00	100.5
8.00	101.3
16.00	102.4
24.00	103.2
48.00	104.7
72.00	105.0
96.00	105.0
120.00	104.9
144.00	104.8
168.00	104.8
192.00	104.7
240.00	104.6
288.00	104.6
336.00	104.5
384.00	104.5
432.00	104.5
528.00	104.4
624.00	104.4
720.00	104.4

MHA/LOCA Time Dependent SIRWT Elemental Iodine Fraction

Time (hr)	Elemental Iodine Fraction
0.3167	0.00E+00
0.50	2.02E-02
1.3167	7.93E-02
1.3167	7.93E-02
2.00	7.95E-02
4.00	8.02E-02
8.00	8.16E-02
16.00	8.42E-02
24.00	8.68E-02
48.00	9.38E-02
72.00	1.00E-01
96.00	1.06E-01
120.00	1.11E-01
144.00	1.15E-01
168.00	1.19E-01
192.00	1.23E-01
240.00	1.29E-01
288.00	1.34E-01
336.00	1.38E-01
384.00	1.41E-01
432.00	1.44E-01
528.00	1.47E-01
624.00	1.49E-01
720.00	1.49E-01

MHA/LOCA TIME DEPENDENT SIRWT PARTITION COEFFICIENT

Time (hr)	Elemental Iodine Partition Coefficient
0.3167	45.65
0.50	45.65
1.3167	45.65
1.3167	45.65
2.00	45.65
4.00	45.21
8.00	44.53
16.00	43.61
24.00	42.95
48.00	41.74
72.00	41.50
96.00	41.50
120.00	41.58
144.00	41.66
168.00	41.66
192.00	41.74
240.00	41.82
288.00	41.82
336.00	41.89
384.00	41.89
432.00	41.89
528.00	41.97
624.00	41.97
720.00	41.97

MHA/LOCA ADJUSTED RELEASE RATE FROM SIRWT

Time (hours)	Adjusted Iodine Release Rate (cfm)
0.3167	9.1718E-04
1.3167	1.1922E-05
8.00	1.2895E-05
24.00	1.4921E-05
72.00	1.7737E-05
168.00	1.9907E-05
240.00	2.1376E-05
336.00	2.2501E-05
432.00	2.3366E-05
624.00	2.3737E-05

SMALL LINE BREAK OUTSIDE OF CONTAINMENT RADIOLOGICAL ANALYSIS – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
PCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 100/E-bar gross activity
Break Flow Rate	160 gpm
Break Temperature	135°F
Break Pressure	35 psia
Time required to isolate break	60 minutes
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Iodine appearance rate for concurrent iodine spike (500x)	Table 14.23-2
Iodine fraction released from break flow	10%
Auxiliary building ventilation system filtration	None
Atmospheric Dispersion Factors Offsite Onsite	 Section 2.5.5.2 Tables 14.24-2 and 14.24-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode	20 minutes
Breathing Rates Offsite Onsite	 RG 1.183 Section 4.1.3 RG 1.183 Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

SMALL LINE BREAK OUTSIDE OF CONTAINMENT RADIOLOGICAL ANALYSIS – CONCURRENT (500 X) IODINE SPIKE APPEARANCE RATE

Isotope	Appearance Rate (Ci/min)
Iodine-131	86.7114868
Iodine-132	119.152137
Iodine-133	134.524016
Iodine-134	110.495326
Iodine-135	102.953824

TIME DEPENDENT CONTROL ROOM PARAMETERS

(For TID-14844 based analyses.)

			X/Q Containment Releases (Ventilation Stack/Aux Bldg)		X/Q SIRWT Releases	
Time Interval	Breathing Rates [m^3/s]	Occupancy Factors	Normal Intake [s/m^3]	Emergency Intake [s/m^3]	Normal Intake [s/m^3]	Emergency Intake [s/m^3]
0 - 8 hr	3.470×10^{-4}	1.0	7.72×10^{-4}	2.56×10^{-4}	1.32×10^{-2}	6.35×10^{-4}
8 - 24 hr	1.750×10^{-4}	1.0	4.55×10^{-4}	1.51×10^{-4}	7.78×10^{-3}	3.74×10^{-4}
1 - 4 days	2.320×10^{-4}	0.6	2.90×10^{-4}	9.60×10^{-5}	4.95×10^{-3}	2.38×10^{-4}
4 - 30 days	2.320×10^{-4}	0.4	1.27×10^{-4}	4.22×10^{-5}	2.18×10^{-3}	1.05×10^{-4}

Atmospheric Dispersion Coefficient for Unfiltered Air Inleakage = same as normal intake

BOUNDING CR-HVAC FLOWS

Emergency Mode Total Filtered Flow	= 2827.2 cfm
Emergency Mode Fresh Air Make-up Flow	= 1413.6 cfm
Emergency Mode Recirculation Flow	= 1413.6 cfm
Emergency Mode Unfiltered Inleakage Flow	= 16 cfm [1]
Normal Mode Fresh Air Make-up Flow	= 660.0 cfm
Base Infiltration Leak Rate (Depressurized)	= 384.2 cfm

CR-HVAC FILTER EFFICIENCIES

CR-HVAC Emergency Mode Charcoal Filter Efficiencies	= 99% for iodine and particulates = 0% for noble gas

[1] See specific events for actual Control Room envelope unfiltered inleakage assumed.

TIME DEPENDENT CONTROL ROOM PARAMETERS

(For TID-14844 based analyses.)

ACCIDENT TIMING SCENARIOS

Event	Abbreviation	FSAR Section	SRP Section	Accident Scenario †
Cask Drop Accident	SFCD	14.11	15.7.5	1
Main Steam Line Break	MSLB	14.14	15.1.5	2
Steam Generator Tube Rupture	SGTR	14.15	15.6.3	2
Control Rod Ejection	CRE	14.16	15.4.8	3,2‡
Loss of Coolant Accident	LOCA	14.17	15.6.5	3
Fuel Handling Accident	FHA	14.19	15.7.4	1
Liquid Waste Incident	LWI	14.20	15.7.2*	1
Gas Decay Tank Rupture	GDTR	14.21.1	15.7.1*	1
Volume Control Tank Rupture	VCTR	14.21.2	15.7.3	1
Small Line Break Outside Containment	SLBOC	14.23	15.6.2	1
Maximum Hypothetical Accident	MHA	14.22	15.6.5	3

† The four types of accident scenarios (1-4) are described below.

‡ The Control Rod Ejection has two release scenarios, an induced LOCA and a S/G-ADV release. The accident scenario type for these release scenarios are listed respectively, in the table above.

* The section has been deleted from the Standard Review Plan, however, it remains part of the licensing basis for Palisades.

TIME DEPENDENT CONTROL ROOM PARAMETERS

(For TID-14844 based analyses.)

The CR-HVAC flow mode, flow rates, and the time that these items change following accident initiation, are important parameters for determining control room radiological consequences. The time to CR-HVAC emergency mode of operation is particularly important, and depends mainly on whether a Loss of Offsite Power (LOOP) occurs coincident with an accident and whether a Containment High Pressure (CHP) or Containment High Radiation signal (CHR) is generated at accident initiation. Events that do not generate a CHP or CHR are collectively referred to as "Non-CHP/CHR Events;" whereas those that do, are referred to as "CHP/CHR Events." Four different accident scenarios result from the combination of these two items and encompass most FSAR Chapter 14 events:

1. Non-CHP/CHR Events Without a LOOP
2. Non-CHP/CHR Events With a LOOP
3. CHP/CHR Events With a LOOP
4. CHP/CHR Events Without a LOOP

Note: No FSAR Chapter 14 events utilize scenario 4.

CONTROL ROOM ATMOSPHERIC DISPERSION (*X/Q*) FACTORS FOR ALTERNATE SOURCE TERM ANALYSIS EVENTS

Release – Receptor Pair	Release Point	Receptor Point	0-2 hr X/Q	2-8 hr X/Q	8-24 hr X/Q	1-4 days X/Q	4-30 days X/Q
A	Containment Closest Point	Normal Intake 'B'	9.16E-03	7.17E-03	2.68E-03	2.07E-03	1.57E-03
B	Containment Closest Point	Emergency Intake	7.26E-04	6.18E-04	2.47E-04	1.77E-04	1.30E-04
C	SIRWT Vent	Normal Intake 'B'	9.57E-02	7.59E-02	2.87E-02	2.19E-02	1.65E-02
D	SIRWT Vent	Emergency Intake	9.66E-04	7.92E-04	3.13E-04	2.20E-04	1.64E-04
E	Plant Stack	Normal Intake 'B'	5.29E-03(1)	3.89E-03(1)	1.51E-03(1)	1.13E-03(1)	8.41E-04(1)
F	Plant Stack	Emergency Intake	8.32E-04	7.69E-04	2.83E-04	2.15E-04	1.57E-04
G	Closest ADV	Normal Intake 'A'	9.95E-03(2)	7.96E-03(2)	3.27E-03(2)	2.39E-03(2)	1.80E-03(2)
H	Closest ADV	Emergency Intake	7.36E-04	6.42E-04	2.43E-04	1.75E-04	1.28E-04
I	Closest SSRV	Normal Intake 'A'	1.24E-02(2)	-	-	-	-
J	Closest SSRV	Emergency Intake	7.96E-04	-	-	-	-
K	Containment Equipment Door	Normal Intake 'B'	1.25E-02	9.83E-03	3.62E-03	2.86E-03	2.28E-03
L	Containment Equipment Door	Emergency Intake	7.32E-04	6.13E-04	2.45E-04	1.75E-04	1.29E-04
M	Feedwater Area Exhauster V-22A	Normal Intake 'A'	2.20E-02	1.75E-02	7.10E-03	5.24E-03	3.87E-03
N	Feedwater Area Exhauster V-22A	Emergency Intake	8.65E-04	7.56E-04	2.81E-04	2.04E-04	1.47E-04

(1) bounding X/Q values used for FHA, SLBOC and SFCD
(2) bounding X/Q values used for SGTR

RELEASE-RECEPTOR POINT PAIRS ASSUMED FOR AST ANALYSIS EVENTS

Event		
MHA	Normal Intake & Unfiltered Inleakage	Emergency Intake
Containment Leakage	A	B
ECCS Leakage	E	F
SIRWT Backleakage	C	D
FHA		
Containment Release	K	L
FHB Release	E	F
SFCD		
Filtered Release	E	F
Unfiltered Release	K	L
MSLB		
Break Release	M	N
MSSV/ADV Release	G	H
SGTR	I & G Initial release via SSRVs switching to ADVs	J & H Initial release via SSRVs switching to ADVs
CRE		
Containment Leakage	A	B
Secondary Side Release	I & G Initial release via SSRVs switching to ADVs	J & H Initial release via SSRVs switching to ADVs
SLBOC	E	F