US-APWR DCD Revision 2
RAI Tracking Report

December 2009

Revision History

Revision	Page	Description
0	All	Original issued Including RAI responses that were submitted through October 31, 2009

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General Description

This report includes a table that identifies the impact of each response to the Request for Additional Information ("RAI") relative to the Design Control Document ("DCD") Revision 2 of US-APWR. This table shows the RAI responses which have been submitted since October 2009 and also should be incorporated into Tracking Report and DCD in future revision.

The report also includes the DCD Markups and Revision List for the RAI responses that impacted the DCD.

Contents

For ease of using this Tracking Report, each chapter is organized in a stand alone fashion that includes a cover sheet and the following relevant information:

- DCD Revision List – a list of the revision resulting from RAI responses and others changes
- DCD Markups – a copy of the DCD pages that have changes resulting from RAI responses or others change.

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.2.1	Seismic Classification											
3.2.2	System Quality Group								CP RAI 67	CP_03.02.02-3	0	3
	Classification											
3.3.1	Wind Loadings											
3.3.2	Tornado Loadings											
3.4.1	Internal Flood Protection for											
	Onsite Equipment Failures											
3.4.2	Analysis Procedures											
3.5.1.1	Internally Generated Missiles											
	(Outside Containment)											
3.5.1.2	Internally-Generated Missiles											
	(Inside Containment)											
3.5.1.3	Turbine Missiles											
3.5.1.4	Missiles Generated by											
	Tornadoes and Extreme Winds											
3.5.1.5	Site Proximity Missiles											
	(Except Aircraft)											
3.5.1.6	Aircraft Hazards											
	Structures, Systems, and											
	Components to be Protected											
	from Externally-Generated											
	Missiles											
3.5.3	Barrier Design Procedures	482	03.05.03-7	2009/12/9	N	N	N		-	-	N/A	N/A
		482	03.05.03-8	2009/12/9	Y	N	N		-	DCD_03.05.03-8	TBD	
3.6.1	Plant Design for Protection											
	Against Postulated Piping Failures											
	in Fluid Systems											
	Outside Containment											
3.6.2	Determination of Rupture Locations	459	03.06.02-20	2009/10/16	Y	N	N		-	DCD_03.06.02-20	-	2
	and Dynamic Effects Associated	459	03.06.02-21	2009/10/16	N	N	N		-	-	N\A	N\A
	with the Postulated Rupture	459	03.06.02-22	2009/10/16	Y	N	N		-	DCD_03.06.02-22	-	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
	of Piping	459	03.06.02-23	2009/10/16	Y	N	N		-	DCD_03.06.02-23	-	2
		459	03.06.02-24	2009/10/16	Y	N	N		-	DCD_03.06.02-24	-	2
		459	03.06.02-25	2009/10/16	Y	N	N		-	DCD_03.06.02-25	0	3
		459	03.06.02-26	2009/10/16	N	N	N		-	-	N\A	N\A
		459	03.06.02-27	2009/10/16	Y	N	N		-	DCD_03.06.02-27	-	2
		459	03.06.02-28	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-29	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-30	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-31	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-32	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-33	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-34	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-35	2009/12/1	N	N	N		-	-	N\A	N\A
		459	03.06.02-36	2009/10/16	N	N	N		-	-	N\A	N\A
		459	03.06.02-37	10/16/2009	Y	N	N		-	DCD_03.06.02-37	-	2
		459	03.06.02-38	10/16/2009	Y	N	N		-	DCD_03.06.02-38	-	2
		459	03.06.02-39	2009/12/1	Y	N	N		-	DCD_03.06.02-39	TBD	
3.6.3	Leak-Before-Break											
	Evaluation Procedures											
3.7.1	Seismic Design Parameters											
3.7.2	Seismic System Analysis	212	3.7.2-3	2009/5/7	Y	N	N		-	DCD_3.7.2-3	TBD	
		212	3.7.2-17	2009/5/7	Y	N	N		-	DCD_3.7.2-17	TBD	
		212	3.7.2-18	2009/5/7	Y	N	N		-	DCD_3.7.2-18	TBD	
		212	3.7.2-19	2009/5/7	Y	N	N		-	DCD_3.7.2-19	TBD	
3.7.3	Seismic Subsystem Analysis											
3.7.4	Seismic Instrumentation											
3.8.1	Concrete Containment											
		-	-	-	-	-	-	-	COL3.8(2) deleted	MAP-03-004	0	2, 3
3.8.3	Concrete and Steel											
	Internal Structures											
	of Steel or Concrete Containments											
3.8.4	Other Seismic Category I Structures											
3.8.5	Foundations											
3.9.1	Special Topics for											
	Mechanical Components											
3.9.2	Dynamic Testing and Analysis											
	of Systems, Structures,											
	and Components											
3.9.3	ASME Code Class 1, 2, and 3											
	Components,											
	and Component Supports,											
	and Core Support Structures											
3.9.4	Control Rod Drive Systems											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.9.5	Reactor Pressure Vessel Internals											
3.9.6	Functional Design, Qualification,											
	and Inservice Testing Programs											
	for Pumps, Valves,											
	and Dynamic Restraints											
3.10	Seismic/Dynamic Qual	486	03.10-10	2009/12/9	N	N	N		-	-	N/A	N/A
	of Mech/Elec Eqmt	486	03.10-11	2009/12/9	Y	N	N		-	DCD_03.10-11	TBD	
		486	03.10-12	2009/12/9	Y	N	N		-	DCD_03.10-12	TBD	
3.11	Environmental Qual	445	03.11-16	2009/9/29	Y	¥ 1.8	N		-	DCD_03.11-16	0	3
	of Mech/Elec Eqmt											
3.12	ASME Code Class 1, 2, and 3	465	03.12-17	2009/12/2	Y	N	N		-	DCD_O3.12-17	TBD	
	Piping Systems,	465	03.12-18	2009/11/18	N	N	N		-	-	N/A	N/A
	Piping Components	465	03.12-19	2009/11/18	Y	N	N		-	DCD_03.12-19	0	3
	and their Associated Supports	465	03.12-20	2009/11/18	Y	N	N		-	DCD_03.12-20	0	3
		465	03.12-21	2009/11/18	N	N	N		-	-	N/A	N/A
		465	03.12-22	2009/11/18	N	N	N		-	-	N/A	N/A
		465	03.12-23	2009/12/2	Y	N	N		-	DCD_03.12-24	TBD	
		465	03.12-24	2009/11/18	Y	N	N		-	DCD_03.12-24	0	3
3.13	Threaded Fasteners -											
	ASME Code Class 1, 2, and 3											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
4.2	Fuel System Design											
4.3	Nuclear Design											
4.4	Thermal and Hydraulic Design	377	04.04-4	2009/6/25	N	N	N		-	-	N/A	N/A
				2009/12/2	N	N	N		-	-	N/A	N/A
4.5.1	Control Rod Drive	457	04.5.01-8	10/29/2009	Y	N	N		-	DCD_04.5.01-8	0	3
	Structural Materials	457	04.5.01-9	10/29/2009	Y	N	N		-	DCD_04.5.01-9	0	3
		457	04.5.01-10	10/29/2009	Y	N	N		-	DCD_04.5.01-10	0	3
4.5.2	Reactor Internal and Core											
	Support Structure Materials											
4.6	Functional Design											
	of Control Rod Drive System											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
6.1.1	Engineered Safety Features	487	06.01.01-11	2009/12/3	Y	N	N		-	DCD_06.01.01-11	TBD	
	Materials	487	06.01.01-12	2009/12/3	N	N	N		-	-	N/A	N/A
6.1.2	Protective Coating Systems (Paints)											
	Organic Materials											
6.2.1	Containment Functional Design											
	Organic Materials											
6.2.1.2	Subcompartment Analysis											
6.2.1.3	Mass and Energy Release											
	Analysis for Postulated											
	Loss-of-Coolant Accidents											
6.2.1.4	Mass and Energy Release											
	Analysis for Postulated											
	Secondary System Pipe Ruptures											
	(LOCAs)											
6.2.1.5	Min. Containment Pressure											
	Analysis for											
	for Emergency Core Cooling Sys.											
	Performance Capability Studies											
6.2.2	Containment	85	06.02.02-10	2009/11/12	Y	N	N	fin.	-	DCD_06.02.02-10	1	2
	Heat Removal Systems	85	06.02.02-11	2009/11/12	N	N	N	fin.	-	-	N/A	N/A
		354	06.02.02-24	2009/7/7	N	N	N		-	-	N/A	N/A
				10/16/2009	Y	N	N		-	DCD_06.02.02-24	TBD	
		354	06.02.02-31	2009/7/7	Y	Y	N		-	DCD_06.02.02-31	4	2
				10/06/2009	Y	Y	N		-	DCD_06.02.02-31	-	2
		354	06.02.02-32	2009/7/7	Y	Y	N		-	DCD_06.02.02-32	-	2
				10/06/2009	Y	N	N		-	DCD_06.02.02-32	-	2
		354	06.02.02-33	2009/7/7	Y	Y	N		-	DCD_06.02.02-33	-	2
				10/06/2009	Y	N	N		-	DCD_06.02.02-33	-	2
		354	06.02.02-34	2009/7/7	Y	Y	N		-	DCD_06.02.02-34	-	2
				10/06/2009	Y	N	N		-	DCD_06.02.02-34	-	2
		354	06.02.02-35	2009/7/7	Y	Y	N		-	DCD_06.02.02-35	-	2
				10/06/2009	Y	N	N		-	DCD_06.02.02-35	-	2
		354	06.02.02-36	2009/7/7	Y	Y	N		-	DCD_06.02.02-36	-	2
				10/06/2009	Y	N	N		-	DCD_06.02.02-36	-	2
		354	06.02.02-44	2009/7/17	Y	N	N		-	DCD_06.02.02-44	TBD	
			06.02.02-52									
		466	06.02.02-53	2009/11/24	N	N	N		-	-	N/A	N/A
		466	06.02.02-54	2009/11/24	N	N	N		-	-	N/A	N/A
		466	06.02.02-55	2009/11/24	Y	N	N		-	DCD_06.02.02-55	TBD	
6.2.4	Containment Isolation System											
6.2.5	Combustible Gas Control	471	6.2.5-35	11/6/2009	Y	N	N		-	DCD_6.2.5-35	0	3
	in Containment											
6.2.6	Containment Leakage Testing	472	06.02.06-23	2009/11/13	Y	N	N		-	DCD_06.02.06-23	TBD	
		472	06.02.06-24	2009/11/13	Y	N	N		-	DCD_06.02.06-24	TBD	
		472	06.02.06-25	2009/11/13	N	N	N		-	-	N/A	N/A
		472	06.02.06-26	2009/11/27	Y	N	N		-	DCD_06.02.06-26	TBD	
		472	06.02.06-27	2009/11/27	Y	N	N		-	DCD_06.02.06-27	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
6.2.7	Fracture Prevention											
	of											
	Containment Pressure Boundary											
6.3	Emergency Core Cooling System											
6.4	Control Room Habitability System	49	06.04-9	2008/9/16	Y	N	N	fin.	-	DCD_06.04-9	(1)	2
		473		11/13/2009	Y	N	N		-	DCD_06.04-9	0	3
6.5.1	ESF Atmosphere Cleanup Systems		06.05.01-1									
6.5.2	Containment Spray	460	06.05.02-7	11/13/2009	N	N	N		-	-	N/A	N/A
	as a Fission Product											
	Cleanup System											
6.5.3	Fission Product											
	Control Systems and Structures											
6.5.5	Pressure Suppression Pool											
	as a											
	Fission Product Cleanup System											
6.6	Inservice Inspection and Testing											
	of Class 2 and 3 Components											
6.6.2	Inservice Inspection and Testing											
	of Class 2 and 3 Components											
6.6.3	Inservice Inspection and Testing											
	of Class 2 and 3 Components											
6.6.4	Inservice Inspection and Testing											
	of Class 2 and 3 Components											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
8.1	Electric Power - Introduction											
8.2	Offsite Power System	432	08.02-10	2009/9/18	Y	N	N		-	DCD_08.02-10	0	3
		432	08.02-12	2009/9/18	Y	N	N		-	DCD_08.02-12	0	3
		432	08.02-13	2009/9/18	Y	N	N		-	DCD_08.02-13	0	3
8.3.1	A-C Power Systems (Onsite)											
8.3.2	D-C Power Systems (Onsite)											
8.4	Station Blackout											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.1.1	Criticality Safety of Fresh and											
	Spent Fuel Storage and Handling											
9.1.2	New and Spent Fuel Storage											
9.1.3	Spent Fuel Pool Cooling and											
	Cleanup System											
9.1.4	Light Load Handling System											
	(Related to Refueling)											
9.1.5	Overhead Heavy Load											
	Handling Systems											
9.2.1	Station Service Water System											
9.2.2	Reactor Auxiliary											
	Cooling Water Systems											
9.2.4	Potable and Sanitary Water Systems											
9.2.5	Ultimate Heat Sink											
9.2.6	Condensate Storage Facilities											
9.3.1	Compressed Air System											
		294	09.03.02-6	2009/5/13	Y	N	N		-	DCD_09.03.02-6	0	3
9.3.2	Process and Post-accident	448	09.03.02-11	2009/9/28	Y	N	N		-	DCD_09.03.02-11	0	3
	Sampling Systems	461	09.03.02-12	2009/11/17	Y	N	N		-	DCD_09.03.02-12	TBD	
9.3.3	Equipment and Floor	426	09.03.03-15	2009/9/14	Y	N	N		-	DCD_09.03.03-15	TBD	
	Drainage System	426	09.03.03-16	2009/9/14	Y	N	N		-	DCD_09.03.03-16	0	3
		426	09.03.03-17	2009/9/14	Y	N	N		-	DCD_09.03.03-17	TBD	
9.3.4	Chemical and Volume Control System											
	(PWR)											
	(Including Boron Recovery System)											
9.4.1	Control Room Area	475	09.04.01-12A	2009/11/20	Y	Y	N		-	DCD_09.04.01-12A	TBD	
	Ventilation System	475	09.04.01-13A	2009/11/20	Y	N	N		-	DCD_09.04.01-13A	TBD	
		475	09.04.01-14A	2009/11/20	N	N	N		-	-	N/A	N/A
		484	09.04.01-15A	2009/12/9	N	N	N		-	-	N/A	N/A
9.4.2	Spent Fuel Pool Area											
	Ventilation System											
9.4.3	Auxiliary and Radwaste Area											
	Ventilation System											
9.4.4	Turbine Area Ventilation System											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.4.5	Engineered Safety Feature	474	09.04.05-10	11/13/2009	Y	N	N		-	DCD_09.04.05-10	0	3
	Ventilation System											
9.5.1	Fire Protection Program											
9.5.2	Communications Systems											
9.5.3	Lighting Systems											
9.5.4	Emergency Diesel Engine Fuel	467	09.05.04-43	11/10/2009	Y	Y	N		-	DCD_09.05.04-43	TBD	
	Oil Storage and Transfer System											
9.5.5	Emergency Diesel Engine											
	Cooling Water System											
9.5.6	Emergency Diesel Engine											
	Starting System											
9.5.7	Emergency Diesel Engine	469	09.05.07-18	11/6/2009	N	N	N		-	-	N/A	N/A
	Lubrication System	469	09.05.07-19	11/6/2009	N	N	N		-	-	N/A	N/A
9.5.8	Emergency Diesel Engine	470	09.05.08-18	2009/12/2	Y	N	N		-	DCD_09.05.08-18	TBD	
	Combustion Air Intake and	470	09.05.08-19	2009/12/2	N	N	N		-	-	N/A	N/A
	Exhaust System	470	09.05.08-20	2009/12/2	Y	N	N		-	DCD_09.05.08-20	TBD	
		470	09.05.08-21	2009/12/2	Y	N	N		-	DCD_09.05.08-21	TBD	
		470	09.05.08-22	2009/12/2	Y	N	N		-	DCD_09.05.08-22	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
10.2	Turbine Generator											
10.2.3	Turbine Rotor Integrity											
10.3	Main Steam Supply System											
10.3.6	Steam and Feedwater System Materials											
10.4.1	Main Condensers											
10.4.2	Main Condenser Evacuation System											
10.4.3	Turbine Gland Sealing System											
10.4.4	Turbine Bypass System											
10.4.5	Circulating Water System											
10.4.6	Condensate Cleanup System	441	10.04.06-8	2009/9/16	Y	N	N		-	DCD_10.04.06-8	0	3
10.4.7	Condensate and Feedwater System											
10.4.8	Steam Generator Blowdown System (PWR)											
10.4.9	Auxiliary Feedwater System (PWR)											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
11.1	Source Terms											
11.2	Liquid Waste Management System	458	11.02-21	2009/10/26	N	N	N		-	-	N/A	N/A
		462	11.02-22	2009/11/17	Y	N	N		-	DCD_11.02-22	TBD	
		462	11.02-23	2009/11/17	N	N	N		-	-	N/A	N/A
		462	11.02-24	2009/11/17	N	N	N		-	-	N/A	N/A
		462	11.02-25	2009/11/17	Y	N	N		-	DCD_11.02-25	0	3
		462	11.02-26	2009/11/17	N	N	N		-	-	N/A	N/A
		462	11.02-27	2009/11/17	Y	N	N		-	DCD_11.02-27	0	3
11.3	Gaseous Waste											
	Management System											
11.4	Solid Waste Management System											
11.5	Process and Effluent											
	Radiological Monitoring											
	Instrumentation and											
	Sampling Systems											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
12.1	Assuring that											
	Occupational Radiation Exposures											
	Are As Low As											
	Is Reasonably Achievable											
12.2	Radiation Sources	427	12.02-19	2009/9/28	Y	N	N		-	DCD_12.02-19	0	3
		427	12.02-21	2009/9/28	Y	N	N		-	DCD_12.02-21	0	3
		427	12.02-22	2009/9/28	Y	N	N		-	DCD_12.02-22	0	3
12.3-	Radiation Protection	425	12.03-12.04-21	2009/9/4	Y	N	N		-	DCD_12.03-12.04-21	0	3
12.4	Design Features	429	12.03-12.04-25	2009/9/28	Y	N	N		-	DCD_12.03-12.04-25	0	3
		429	12.03-12.04-26	2009/9/28	Y	Y	N		-	DCD_12.03-12.04-26	0	3
		429	12.03-12.04-27	2009/9/28	Y	N	N		-	DCD_12.03-12.04-27	0	3
		429	12.03-12.04-30	2009/9/28	Y	N	N		-	DCD_12.03-12.04-30	0	3
		429	12.03-12.04-31	2009/9/28	Y	N	N		-	DCD_12.03-12.04-31	0	3
12.5	Operational Radiation											
	Protection Program											

Chapter:14

SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
14.2	Initial Plant Test Program -	455	14.02-119	2009/10/1	Y	N	N		-	DCD_14.02-119	-	2
	Design Certification and											
	New License Applicants											
14.3	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.2	Structural and Systems Engineering	452	14.03.02-9	2009/10/1	Y	N	N		-	DCD_14.03.02-9	-	2
	Inspections, Tests, Analyses,	452	14.03.02-10	2009/10/1	Y	N	N		-	DCD_14.03.02-10	-	2
	and Acceptance Criteria	452	14.03.02-11	2009/10/1	Y	N	N		-	DCD_14.03.02-11	-	2
		452	14.03.02-12	2009/10/1	Y	N	N		-	DCD_14.03.02-12	-	2
		452	14.03.02-13	2009/10/8	Y	N	N		-	DCD_14.03.02-13	-	2
		452	14.03.02-14	2009/10/1	Y	N	N		-	DCD_14.03.02-14	-	2
14.3.3	Piping Systems and Components											
	and Acceptance Criteria											
14.3.4	Reactor Systems											
14.3.5	Instrumentation and Controls -											
14.3.6	Electrical Systems -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.7	Plant Systems -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.8	Radiation Protection -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.9	Human Factors Engineering -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.10	Emergency Planning -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.11	Containment Systems -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
		396	14.03.12-20	2009/7/17	Y	N	N		-	DCD_14.03.12-20	TBD	
14.3.12	Physical Security Hardware -	481	14.03.12-25	11/10/2009	N	N	N		-	-	N\A	N/A
	Inspections, Tests, Analyses,	481	14.03.12-26	11/10/2009	Y	N	N		-	DCD_14.03.12-26	0	3
	and Acceptance Criteria	481	14.03.12-27	11/10/2009	Y	N	N		-	DCD_14.03.12-27	0	3
		481	14.03.12-28	11/10/2009	N	N	N		-	-	N\A	N/A
		481	14.03.12-29	11/10/2009	Y	N	N		-	DCD_14.03.12-29	0	3
		481	14.03.12-30	11/10/2009	Y	N	N		-	DCD_14.03.12-30	0	3

Chapter 16

SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	cc	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
16.1	General, Plant Sys,	463	16-299	10/28/2009	N	N	N		-	-	N/A	N/A
	Refueling, & Adm Ctrls:											
	Technical Specifications											
16.2	SLs, Reactivity,											
	Core Op Limits, & Special Ops:											
	Technical Specifications											
16.3	Instrumentation:											
	Technical Specifications											
16.4	CS & ECCS: Technical Specificatio	OI	16-146-1804/79	10/14/2009	N	N	N		-	-	N/A	N/A
		OI	16-135-1818/51	10/14/2009	Y	Y	N		-	DCD_16-135-1818/51	0	3
		OI	16-135-1818/53	10/14/2009	Y	Y	N		-	DCD_16-135-1818/53	0	3
		OI	16-2.4-50	10/16/2009	N	N	N		-		N/A	N/A
		OI	16-9.2.1-26	10/14/2009	N	N	N		-		N/A	N/A
		OI	16-133-1827/136	10/16/2009	N	N	N		-		N/A	N/A
		OI	16-133-1827/15	2009/10/28	Y	Y	N		-	DCD_16-133-1827/15	0	3
		OI	16-133-1827/20	2009/10/28	N	N	N		-	-	N/A	N/A
		OI	16-1769/284	10/28/2009	N	N	N		-	-	N/A	N/A
		OI	16-1784/172	11/10/2009	Y	Y	N		-	DCD_16-1784/172	TBD	
		OI	16-1784/174	11/10/2009	Y	Y	N		-	DCD_16-1784/174	TBD	
		OI	16-1784/186	11/10/2009	Y	Y	N		-	DCD_16-1784/186	TBD	
		OI	16-1784/188	11/10/2009	Y	Y	N		-	DCD_16-1784/188	TBD	
		OI	16-1784/192	11/10/2009	Y	Y	N		-	DCD_16-1784/192	TBD	
		OI	16-1769/209	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/220	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/228	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/230	11/10/2010	N	N	N		-	-	N/A	N/A
		OI	16-1769/231	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/232	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/233	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/238	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/241	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/242	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/270	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/271	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/272	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/273	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/274	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/275	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-1769/282	11/10/2009	Y	Y	N		-	DCD_16-1769/282	TBD	
		OI	16-1769/290	11/10/2009	N	N	N		-	-	N/A	N/A
		OI	16-134-1825/26	10/30/2009	Y	Y	N		-	DCD_16-134-1825/26	0	3
		OI	16-134-1825/27	10/30/2009	N	N	N		-	-	N/A	N/A
		OI	16-72-853	10/30/2009	Y	Y	N		-	DCD_16-72-853	0	3
16.5	Containment Systems:											
	Technical Specifications											
16.6	Electrical Power Sys:											
	Technical Specifications											

Chapter:19

SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
19	Probabilistic Risk Assessment	88	19-150	2008/11/27	Y	N	N	fin.	-	DCD_19-150	TBD	
	and Severe Accident Evaluation	423	19-362	2009/9/7	Y	N	N		-	DCD_19-362	0	3
	for New Reactors	423	19-363	2009/9/7	Y	N	N		-	DCD_19-363	0	3
		423	19-368	2009/9/7	Y	N	N		-	DCD_19-368	0	3
		423	19-371	2009/9/7	Y	N	N		-	-	0	3
		423	19-373	2009/9/7	Y	N	Y		-	DCD_19-373	0	3
		423	19-374	2009/9/7	Y	N	N		-	DCD_19-374	TBD	
		423	19-375	2009/9/7	Y	N	N		-	DCD_19-375	TBD	
		423	19-376	2009/9/7	Y	N	N		-	DCD_19-376	0	3
		423	19-387	2009/9/7	Y	N	N		-	DCD_19-387	0	3
		443	19-391	2009/10/1	N	N	N		-	-	N/A	N/A
		443	19-392	2009/10/1	N	N	N		-	-	N/A	N/A
		443	19-393	2009/10/1	Y	N	N		-	DCD_19-393	-	2
		443	19-394	2009/10/1	N	N	N		-	-	N/A	N/A
		443	19-395	2009/10/1	N	N	N		-	-	N/A	N/A
		443	19-396	2009/10/1	Y	N	N		-	DCD_19-396	0	3
		443	19-397	2009/10/1	Y	N	N		-	DCD_19-397	0	3
		454	19-398	2009/10/9	N	N	Y		-	-	N/A	N/A
		454	19-399	2009/10/9	N	N	Y		-	-	N/A	N/A
		454	19-400	2009/10/9	N	N	Y		-	-	N/A	N/A
		454	19-401	2009/10/9	Y	N	Y		-	DCD_19-401	-	2
		479	19-402	2009/11/25	Y	N	N		-	DCD_19-402	TBD	
		479	19-403	2009/11/25	Y	N	N		-	DCD_19-403	TBD	
		479	19-404	2009/11/25	Y	N	N		-	DCD_19-404	TBD	
		479	19-405	2009/11/25	N	N	N		-	-	N/A	N/A
		479	19-406	2009/11/25	N	N	N		-	-	N/A	N/A
19.1	Determining the Technical Adequacy											
	of Probabilistic Risk Assessment											
	Results for Risk-Informed Activities											
		-	-	-	-	-	-	-	COL 19.3(5) deleted	MAP-19-001	TBD	
19.2	Review of Risk Information											
	Used to Support Permanent Plant -											
	Specific Changes											
	to the Licensing Basis:											
	General Guidance											
		-	-	-	-	-	-	-	COL 19.3(6) deleted	MAP-19-002	TBD	

Chapter 1

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
1.8-19	Table 1.8-2 Sheet 15 *COL 3.11(6)*	Change: "*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an equivalent qualification process to that delineated for the US-APWR Standard Plant.*" to "*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).*" Reason: Grammatically corrected statement to indicate "a qualification process that is equivalent" to the US-APWR standard plant. [RAI 445-2759 Question 3.11-16]
1.9-4	Table 1.9.1-1 RG 1.21	RAI: No.294, 9.3.2-6 Inserted the corresponding Subsection "6.1.1".

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 15 of 44)

COL ITEM NO.	COL ITEM
COL 3.11(4)	*The COL Applicant is to describe periodic tests, calibrations, and inspections to be performed during the life of the plant, which verify the identified equipment remains capable of fulfilling its intended function.*
COL 3.11(5)	*The COL Applicant is to identify the site-specific equipment to be addressed in the EQ Program, including locations and environmental conditions.*
COL 3.11(6)	*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a~~n equivalent~~ qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).*
COL 3.11(7)	*The COL Applicant is to identify chemical and radiation environmental requirements for site-specific qualification of electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment).*
COL 3.11(8)	*The COL Applicant is to provide the site-specific mechanical equipment requirements.*
COL 3.11(9)	*Optionally, the COL Applicant may revise the parameters based on site-specific considerations.*
COL 3.12(1)	*Deleted*
COL 3.12(2)	*If any piping is routed in tunnels or trenches in the yard, the COL Applicant is to generate site-specific seismic response spectra, which may be used for the design of these piping systems.*
COL 3.12(3)	*If the COL Applicant finds it necessary to lay ASME Code, Section III (Reference 3.12-2), Class 2 or 3 piping exposed to wind or tornado loads, then such piping must be designed to the plant design basis loads.*
COL 3.12(4)	*The COL Applicant is to screen piping systems that are sensitive to high frequency modes for further evaluation.*
COL 3.13(1)	*Deleted*

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 2 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.16	Reporting of Operating Information – Appendix A Technical Specifications (Rev. 4, August 1975)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	Chapter 16, 14.2.6, 14.2.7
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	Conformance with exceptions. The measurement at startup test for SG's internals is not planned.	3.9.2.3, 3.9.2.4, 3.9.2.6, 5.4.2.1.2.10, 14.2,
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	Conformance with exceptions. To be conformed by COL Applicant with site-specific information.	3.1.6, 9.3.2, 11.5.1, 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 1972)	Conformance with no exceptions identified.	7.1.3.11, 7.1.3.14, 8.1.5.3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Rev. 1, March 2007)	Not applicable. To be conformed by COL Applicant with site-specific characterization information.	N/A
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 1972)	Conformance with exceptions. To be conformed by COL Applicant with site-specific characterization information.	11.3.3
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 1972)	Not applicable. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" is applied instead of Regulatory Guide 1.25.	N/A
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	Conformance with no exceptions identified.	3.2.2, 5.2.1.1, 5.2.2.1, 5.2.4.1
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	Conformance with exceptions. US-APWR is designed in accordance with the functional requirements for a UHS as described in this RG, however design of the UHS is site-specific and will be the responsibility of the COL Applicant.	9.2.1.3, 9.2.5
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	Conformance with no exceptions identified.	14.2.7, 17.5

Chapter 3

US-APWR DCD Revision 2 Chapter 3 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.2-16	Table 3.2-1 Sheet 2	Add two rows for CWS components “Cooling towers” and “Circulating water pumps” to the table. Reason: The components of the CWS system are required for normal shutdown. The cooling towers and the circulating water pumps were erroneously omitted. [RAI 2757 (CP RAI #67) Question 03.02.02-3]
3.6-14	Subsection 3.6.2.1.3.1 Last Paragraph	Add new 2nd Sentence: “Piping stiffness is used only when a plastic hinge is not developed in the piping.” Reason: Provided clarification. [RAI 459-3331 Question 3.6.2-25]
3.7-36	Subsection 3.7.2.7	Add as new last sentence: “The 10% grouping method is used for piping as described in Subsection 3.12.3.2.4.” Reason: Clarify that the more conservative modal combination methods contained in Revision 1 of RG 1.92 are used for piping. [RAI 465-3382 Question 3.12-19]
3.8-20	Subsection 3.8.1.5.2.2 Last Paragraph Last sentence	Delete: “It is the responsibility of the COL Applicant to assure that wobble and curvature coefficients used in computing prestressing losses due to friction are consistent with the tendon system corrosion protection coatings present at the time of prestressing.” Reason: COL 3.8(2) was deleted in Revision 2. [MAP-03-004]
3.11-10	Subsection 3.11.4 2nd Paragraph 1st Sentence	Change: “The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an equivalent qualification process to that delineated for the US-APWR Standard Plant.” to “The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a qualification process that is equivalent to that delineated for the US-APWR standard plant, as described in Technical Report MUAP-08015(R1).” Reason: Grammatically corrected statement to indicate “a qualification process that is equivalent” to the US-APWR standard plant. [RAI 445-2759 Question 3.11-16]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.11-10	Subsection 3.11.5 1st Paragraph 2nd Sentence	Change: "This equipment is to be qualified using an equivalent qualification process to that delineated for the US-APWR Standard Plant" to "This equipment is to be qualified using a qualification process that is equivalent to that delineated for the US-APWR standard plant, as described in Technical Report MUAP-08015(R1)." Reason: Grammatically corrected statement to indicate "a qualification process that is equivalent" to the US-APWR standard plant. [RAI 445-2759 Question 3.11-16]
3.11-11	Subsection 3.11.6 2nd Paragraph 2nd Sentence	Change: "This equipment is to be qualified using an equivalent qualification process to that delineated for the US-APWR Standard Plant" to "This equipment is to be qualified using a qualification process that is equivalent to that delineated for the US-APWR standard plant, as described in Technical Report MUAP-08015(R1)." Reason: Grammatically corrected statement to indicate "a qualification process that is equivalent" to the US-APWR standard plant. [RAI 445-2759 Question 3.11-16]
3.11-11	Subsection 3.11.7 *COL 3.11(6)*	Change: "*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an equivalent qualification process to that delineated for the US-APWR Standard Plant.*" to "*The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).*" Reason: Grammatically corrected statement to indicate "a qualification process that is equivalent" to the US-APWR standard plant. [RAI 445-2759 Question 3.11-16]
3.12-4	Subsection 3.12.3.2.4 **Low frequency (non-rigid) modes** 1st Paragraph	Add new last Sentence: "The response of low frequency (non-rigid) modes is obtained from all the low frequency modes with frequencies at least up to ZPA cutoff frequency." Reason: Clarify statement [RAI 465-3382 Question 3.12-24]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3.12-4	Subsection 3.12.3.2.4 **High frequency (rigid) modes** 1st Paragraph 2nd Sentence	Change: "The response from high frequency modes must be included in the response of the piping system, if it results in an increase in the dynamic response of more than 10%." to "The response from high frequency modes must be included in the response of the piping system." Reason: Delete the qualifying statement [RAI 465-3382 Question 3.12-20]

Table 3.2-1 Non-Safety Components Required for Normal Shutdown (Sheet 2 of 2)

Instrument Air System	Instrument air compressors
Secondary System	Condenser
	Condensate pump
	Deaerator
	Main feedwater pump
	Cooling towers
	Circulating water pumps
Heating, Ventilation, and Air Conditioning	Containment fan cooler unit fan
	Reactor cavity cooling fan
	Control rod drive mechanism cooling fan
	Non-Class 1E electrical room air handling unit fan
	Non-essential chiller units
	Non-essential chilled water pumps

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. The line restrictions, flow limiters, positive pump-controlled flow and the absence of energy reservoirs may be taken into account, as applicable.

Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or pipe stiffness. Piping stiffness is used only when a plastic hinge is not developed in the piping. The effective cross sectional (inside diameter) flow area of the pipe is used in the jet discharge evaluation. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to initiate pipe movement in the direction of the jet reaction.

3.6.2.1.3.2 Longitudinal Pipe Breaks

Longitudinal breaks are postulated in high-energy fluid system piping and branch runs in nominal pipe sizes 4 inches and larger. Longitudinal breaks are postulated in high- energy fluid system piping at locations of circumferential breaks as described in Subsection 3.6.2.1.3.1.

If the maximum stress range exceeds the limits specified in Subsection 3.6.2.1.1.2 and the axial stress range is greater than 1.5 times the circumferential stress range, no longitudinal break is postulated, only a circumferential break (Subsection 3.6.2.1.3.1) is postulated.

Longitudinal breaks need not be postulated at terminal ends.

Longitudinal breaks in the form of axial split without pipe severance are postulated in the center of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of piping configuration and produces out-of-plane bending. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location, where D is the effective inner diameter of the pipe. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or pipe stiffness as demonstrated by inelastic limit analysis.

3.6.2.1.3.3 Leakage Cracks

Leakage cracks are postulated in high-energy fluid system piping at locations identified in Subsection 3.6.2.1.1.3. Leakage cracks are also postulated in moderate-energy fluid

$$R^2 = \sum_{k=1}^{N} R_k^{\ 2} + 2\sum \left| R_i R_j \right| \qquad i \neq j$$

The second summation is to be done on all i and j modes whose frequencies are closely spaced to each other.

All terms for the modal combination remain the same as defined above.

The 10% grouping method is more conservative than the grouping method because the same mode can appear in more than one group. The 10% grouping method is used for piping as described in Subsection 3.12.3.2.4.

For the seismic response spectra analysis, the ZPA cut-off frequency is 50 Hz. High frequency or rigid modes must be considered using the static ZPA method, the left-out force method as described in Subsection 3.7.2.7 below, or the Kennedy Missing Mass method contained in Revision 2 of RG 1.92 (Reference 3.7-27).

3.7.2.7.1 Left-Out-Force Method (or Missing Mass Correction for High Frequency Modes)

The left-out-force method is based on the Left-Out-Force Theorem. This theorem states that for every time history load, there is a frequency, f_r, called the "rigid mode cutoff frequency" above which the response in modes with natural frequencies above f_r will very closely resemble the applied load at each instant of time. These modes are called "rigid modes." The formulation follows and is based on the method used in the computer program PIPESTRESS (Reference 3.7-29). The left-out-force method is not used for seismic analysis of the major seismic category I structures; however, it may be used for other seismic category I and II systems and subsystems.

The left-out-force vector for time history analyses, { *Fr* }, is calculated based on lower modes:

$$\{ Fr \} = [\, 1 - \sum M\, e_j\, e_j^{T} \,]\, f(t)$$

where

f (t) = the applied load vector

M = the mass matrix

e_j = the eigenvector

Note that ∑ only represents the flexible modes, not including the rigid modes.

In the response spectra analysis, the total inertia force contribution of higher modes can be interpreted as:

$$\{ Fr \} = Am\, [M]\, [\, \{ r \} - \sum P_j\, e_j \,]$$

where

Am = the maximum spectral acceleration beyond the flexible modes

Losses

The losses considered in the tendons are based on the items defined in ASME Code, Section III (Reference 3.8-2), Subarticle CC-3542. In addition, RG 1.35.1 (Reference 3.8-6) is used as guidance in the determination of prestressing losses. Prestressing losses are computed on the basis of the US-APWR 60 year design life. ~~It is the responsibility of the COL Applicant to assure that wobble and curvature coefficients used in computing prestressing losses due to friction are consistent with the tendon system corrosion protection coatings present at the time of prestressing.~~

3.8.1.5.2.3 Reinforcing Steel Systems

Tension

In accordance with ASME Code, Section III, Subarticle CC-3432.1 (Reference 3.8-2), the average tensile stress is limited to $0.5f_y$; however, provisions are included for increases under certain conditions.

Compression

In accordance with ASME Code, Section III, Subarticle CC-3432.2 (Reference 3.8-2), the compressive stress is limited to $0.5f_y$; however, provisions are included for increases under certain conditions.

General Shear

See discussion above for qualification of general shear capacity with service loads.

Radial Shear

The radial shear provisions for the US-APWR are in accordance with the ASME Code, Section III, Subarticle CC-3431.3 “Shear, Torsion, and Bearing” and Subarticle CC-3522 “Service Load Design” (Reference 3.8-2).

Tangential Shear

The US-APWR design is in accordance with ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2). Since only wind generates tangential shear in the service load category, wind does not govern the design.

Peripheral Shear

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2).

Torsional Shear

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2).

The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a~~n equivalent~~ qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1). This includes equipment that is subject to environmental control systems including heat tracing and air conditioning.

3.11.5 Estimated Chemical and Radiation Environment

The COL Applicant is to identify chemical and radiation environmental requirements for site-specific qualification of electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment). This equipment is to be qualified using a~~n equivalent~~ qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).

3.11.5.1 Chemical Environment

The adverse effects of various chemicals used within the plant are normally contained by design. However, during a DBA, various chemicals can be released into the equipment's environment, which could impair the ability of the equipment to operate. The impact of the various chemicals used in the plant is factored into the design and EQ process. Chemical exposure also includes the potential for exposure to hydrogen based on a 100% fuel-clad metal-water reaction (10 CFR 50.44(c), [Reference 3.11-19]). Equipment subject to chemical exposure, including submergence, are qualified with concentrations of chemicals equivalent or more severe than those resulting from the most limiting mode of the plant operations associated with DBAs. Mechanical equipment is evaluated during the design (analysis) and procurement phases for exposure to various chemicals applicable to normal and off normal conditions. Requirements to be able to withstand the effects of chemical exposure are factored into the procurement process. Equipment subject to chemical exposure following a DBA is identified and qualified, pursuant to the implementation of the US-APWR EQ Program.

3.11.5.2 Radiation Environment

Areas in the plant are subject to varying levels of radiation exposure during normal and accident conditions. Radiation environments also exist in those plant areas where there is the potential for airborne contamination, for those process and effluent streams where contamination is possible, and in accessible areas as a result of unusual radiological events. The normal operational radiation exposure is based on the radiation sources (source term) provided in Chapter 12.

The radiation sources presented in this DCD are developed from the DBA available data and in accordance with NUREG-1465 (Reference 3.11-20). The radiation dose rates and integrated doses of neutrons, beta, and gamma radiation which are associated with normal, accident, post-accident, test, and harsh environmental conditions for various plant areas and systems, are presented in Appendix 3D. Their parameters are presented in time-based units, wherever applicable.

The expected levels of radiation exposure factored into the design process are based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe DBA during or following an accident in which the equipment is required to remain functional, including the radiation resulting from re-circulating fluids for equipment located near the re-circulating lines and including dose-rate effects. Equipment that will not be exposed to total integrated doses of 10^4 Rads as a result of a DBA is not qualified for radiation exposure in most cases. In all cases, each piece of equipment in a potential harsh environment is evaluated for the need for qualification due to radiation exposure.

Electrical and mechanical equipment subject to radiation exposure is qualified for use in the US-APWR pursuant to the implementation of the US-APWR EQ Program. For equipment that is only located in areas considered harsh by the potential presence of radiation, this equipment is qualified by analysis and partial test data with the appropriate considerations for margins and aging effects.

3.11.6 Qualification of Mechanical Equipment

The qualification of mechanical equipment is included in Subsection 3.11.2.1.

The COL Applicant is to provide the site-specific mechanical equipment requirements. This equipment is to be qualified using a~~n equivalent~~ qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).

3.11.7 Combined License Information

COL 3.11(1) *The COL Applicant is responsible for assembling and maintaining the environmental qualification document, which summarizes the qualification results for all equipment identified in Appendix 3D, for the life of the plant.*

COL 3.11(2) *The COL Applicant is to describe how the results of the qualification tests are to be recorded in an auditable file in accordance with requirements of 10 CFR 50.49 (j).*

COL 3.11(3) *The COL Applicant is to provide a schedule showing the EQ Program proposed implementation milestones.*

COL 3.11(4) *The COL Applicant is to describe periodic tests, calibrations, and inspections to be performed during the life of the plant, which verify the identified equipment remains capable of fulfilling its intended function.*

COL 3.11(5) *The COL Applicant is to identify the site-specific equipment to be addressed in the EQ Program, including locations and environmental conditions.*

COL 3.11(6) *The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using a~~n equivalent~~ qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in Technical Report MUAP-08015(R1).*

3.12.3.2.4 Modal Combination

Guidance on combining the individual modal results due to each response spectrum in a dynamic analysis is provided in RG 1.92, Rev.1 (Reference 3.12-11).

Low frequency (non-rigid) modes: For piping systems with no closely spaced modes, the SRSS method is applied to obtain the representative maximum response of each element, for each direction of excitation as delineated in Regulatory Position C1.1 of RG 1.92, Rev.1 (Reference 3.12-11). A 10% grouping method is used for combining the responses of closely spaced modes as delineated in Regulatory Position C1.1 of RG 1.92, Rev.1 (Reference 3.12-11). The response of low frequency (non-rigid) modes is obtained from all the low frequency modes with frequencies at least up to ZPA cutoff frequency.

High frequency (rigid) modes: Piping system modes with frequencies greater than the ZPA cutoff frequency are considered as high frequency or rigid range modes. The response from high frequency modes must be included in the response of the piping system~~, if it results in an increase in the dynamic response of more than 10%~~. The guidance for including the missing mass effects is provided in SRP 3.7.2 (Reference 3.12-12), as well as in RG 1.92, Rev.2 (Reference 3.12-13).

The PIPESTRESS computer program is used for analyzing most of the piping systems. This program uses the left-out-force (LOF) method in order to calculate the effect of the high frequency rigid modes. The LOF method is described in the “PIPESTRESS Theory Manual” (Reference 3.12-14) and the “Outline of Dynamic Analysis for Piping Systems” (Reference 3.12-15).

3.12.3.2.5 Directional Combination

The responses due to each of the three spatial input components of motion are combined using the SRSS method as provided in Regulatory Position C2.1 of RG 1.92, Rev.1 (Reference 3.12-11).

3.12.3.2.6 Seismic Anchor Motions

The analysis of seismic anchor motions (SAMs) is a static analysis and is performed using the same piping model used to analyze the inertial effects in a dynamic analysis.

When piping is analyzed using the USM method, effects of SAM in each of the three different spatial directions are analyzed separately considering all dynamic supports to be active. The three resulting solutions are combined by the SRSS rule to obtain cumulative effect of support displacements.

For piping supported by a single concrete building, the SAM at all elevations above the foundation basemat are considered to be in phase. Support movements relative to the foundation basemat are used in the analysis.

Chapter 4

US-APWR DCD Revision 2 Chapter 4 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
4.5-1	Subsection 4.5.1 2nd paragraph	RAI No.457, 4.5.1-8 Insert the following sentence at the end of he 2nd paragraph of subsection 4.5.1. • GDC26, as it relates to reactivity control system redundancy and capability
4.5-2	Subsection 4.5.1.1 2nd paragraph	RAI No.457, 4.5.1-9 Insert the following sentence at the end of he 2nd paragraph of subsection 4.5.1. “Strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.”
4.5-3	Subsection4.5.1.2 1st.paragraph.	RAI No.457, 4.5.1-8 Replace the first sentence of subsection 4.5.1.2 from “Discussions of fabrication and processing of austenitic stainless steel are provided in Subsection 5.2.3. The processes for control of welding described in Subsection 5.2.3 are applicable to the pressure housing of the CRDM.” to “Austenitic stainless steel base materials for CRDM applications are the solution heat that is treated to prevent sensitization and stress corrosion cracking (SCC).”
4.5-8	Table 4.5-1	RAI No.457, 4.5.1-10 1st column: Material specification for the CRDM pressure housing, ”LN” is add after the “SA-182 Grade F316”. RAI No.457, 4.5.1-8 Last column: Environment of the welding material for the CRDM pressure housing is changed from “Not exposed to reactor coolant water” to “Inside surface exposed to reactor coolant water”.

4.5 Reactor Materials

This section demonstrates the adequacy of the materials selected for the US-APWR control rod drive system (CRDS) and the reactor Internals and core support structures. All such materials used in the US-APWR have been used successfully for these applications in the United Sates and Japanese PWRs.

Subsection 4.5.1 describes the structural materials used in the CRDS. Subsection 4.5.2 describes the materials used in the reactor internals and the core support structures.

To avoid duplication, Section 4.5 refers to Subsection 5.2.3 for additional information on materials specifications and details on matters such as compatibility of the materials with the reactor coolant and control of welding processes.

4.5.1 Control Rod Drive System Structural Materials

This subsection begins with a description of the material specifications. It then addresses austenitic stainless steel components, other materials, and cleaning and cleanliness control. Table 4.5-1 lists the materials of interest.

Information in this subsection addresses relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A (Reference 4.5-1):

- GDC 1, as it relates to quality standards for structures, systems, and components important to safety
- GDC 14, as it relates to low probability of abnormal leakage, rapidly propagating failure, or gross rupture
- GDC26, as it relates to reactivity control system redundancy and capability

This subsection also demonstrates that CRDS structural materials are designed, fabricated, tested, and inspected to quality standards commensurate with the importance of their safety function consistent with 10 CFR 50.55a (Reference 4.5-2).

The CRDS, for purposes of this subsection, is considered to be comprised of the control rod drive mechanisms (CRDMs). It does not include the electrical and hydraulic systems necessary to actuate the CRDMs.

The rod control cluster assembly (RCCA) is described in Section 4.2. Type 403 martensitic stainless steel used for the coupling of the drive rod assembly undergoes a proprietary heat treatment process that achieves the desired coupling stiffness and toughness.

4.5.1.1 Material Specifications

Austenitic stainless steel, nickel based alloys, and cobalt based alloys are selected for CRDM components that are in contact with the reactor coolant water because of their corrosion resistance. The material specifications are listed in Table 4.5-1.

The properties of the materials selected for the CRDM are found in Section III, Appendix I, Division 1 of the ASME Boiler and Pressure Vessel Code (ASME Code) (Reference 4.5-3) or Section II, Parts D of the ASME Code. Strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.

All other materials for use in this system are selected for their compatibility with the reactor coolant water, as described in NB-2160 and NB-3120 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels are specified for assurance that these materials will not deteriorate from stress corrosion cracking in service. Acceptable heat treatment temperatures include aging at 1050°F for Type 410 stainless steel.

The metallic and nonmetallic materials used in the CRDMs are the same as those used in operating plants in the United States and Japan.

4.5.1.1.1 Pressure Housing

The pressure housing material in contact with the reactor coolant water is type 316 austenitic stainless steel, which meets the requirements of ASME code Section III. Detailed description of the austenitic stainless steel for pressure housing material is given in Subsection 5.2.3. The material of the CRDM pressure housing is identified in Tables 4.5-1.

Flux rings made from ferrite material are attached around the latch housing to provide a magnetic flux route.

4.5.1.1.2 Latch Assembly

The material for the latch assembly, magnetic poles, plungers and keys is type 410 martensitic stainless steel. Where strength is not an issue, annealed type 410 stainless steel is used because of its superior magnetic properties. Springs are made of Alloy X-750. Link pins are made of cobalt alloy. Tip and pin holes of the latch arms are clad with the cobalt alloy, Stellite. Cobalt cladding is used to improve resistance for wear. Other parts are made of type 304 austenitic stainless steel. Hard chrome plating is applied for sliding surfaces. Chrome carbide coating is applied on the tips of the latch arms to improve resistance to wear.

4.5.1.1.3 Drive Rod Assembly

The material for the drive rod assembly, drive rod, unlatch button and protection sleeve is type 410 martensitic stainless steel. The coupling is fabricated from type 403 martensitic stainless steel. Springs are made of Alloy X-750 and the locking button is fabricated from cobalt alloy. Other drive rod assembly parts are made from type 304 austenitic stainless steel.

4.5.1.1.4 Coil Assembly

The coil housings in the coil assembly are ductile iron castings, selected for their suitable magnetic properties. Coils are wound on bobbins of molded glass silicon resin material, with double glass insulated copper wire. Coils are impregnated with silicon varnish under vacuum condition. A wrapping of mica sheet is secured to the coil outside diameter. The coil assembly is a proven design used in many operating plants in the United States and Japan.

4.5.1.2 Austenitic Stainless Steel Components

~~Discussions of fabrication and processing of austenitic stainless steel are provided in Subsection 5.2.3. The processes for control of welding described in Subsection 5.2.3 are applicable to the pressure housing of the CRDM~~. Austenitic stainless steel base materials for CRDM applications are the solution heat that is treated to prevent sensitization and stress corrosion cracking (SCC).

The welding materials used for joining the austenitic stainless steels meet the requirements of the welding material specification SFA 5.9 in ASME Code Section II. In addition, the above welding materials meet the requirements of ASME Code Section III.

For design temperatures up to and including 800 °F, the minimum acceptance delta ferrite is 5FN (Ferrite Number).

Manufacturing process controls for preventing intergranular corrosion of stainless steel components are used in accordance with the guidance in RG 1.44 (Reference. 4.5-5). Furnace sensitized material is allowed, and methods described in the guide are followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding.

The process controls during manufacturing for abrasive work on austenitic stainless steel surfaces are designed to prevent contamination that may result in stress corrosion cracking. These controls are consistent with Regulatory Position C of RG 1.37 (Reference 4.5-6).

The final surfaces are required to meet the acceptance standards specified in ASME NQA-1 (Reference 4.5-7). Tools used on austenitic stainless steel surfaces are controlled to prevent materials that could contribute to stress corrosion cracking from contaminating these surfaces.

4.5.1.3 Other Materials

MIL-S23192 (Reference 4.5-8) is the standard that will be used for spring material made from X-750. This standard requires solution heat treatment and aging heat treatment to preclude SCC. This material has not observed stress corrosion cracking which is based on experience in operating plants.

Cobalt alloy for pins is ordered in the solution treatment and strain hardened condition. This material is used in the link pins, and has not observed stress corrosion cracking which is based on experience in operating plants.

Table 4.5-1 Summary of Control Rod Drive System Structural Materials

Component	Material Specification[1]	Environment
CRDM pressure housing material in contact with reactor coolant on the inside surface	SA-182 Grade F316 LN	Inside surface exposed to reactor coolant water
Flux Ring	ASTM A519 Gr.1015	Not exposed to reactor coolant water
Latch assembly - magnetic poles, plungers, and keys	SA-479 Type 410	Exposed to reactor coolant water
Latch assembly - springs	Alloy X-750 (ASME SB637 N07750)[2]	Exposed to reactor coolant water
Latch assembly - link pins	Cobalt alloy (HAYNES No. 25 or equivalent material[3])	Exposed to reactor coolant water
Latch assembly - other parts	SA-479 Type 304 SA-213 Grade TP 304	Exposed to reactor coolant water
Latch assembly - cladding on latch arm tips and pin holes	Cobalt alloy (Stellite No.6 or equivalent material[3])	Exposed to reactor coolant water
Latch assembly - plating on sliding surfaces	Chrome plate	Exposed to reactor coolant water
Latch assembly - coating on tips of latch arms	Chrome carbide	Exposed to reactor coolant water
Drive rod assembly - drive rod,	SA-268 TP410	Exposed to reactor coolant water
Drive rod assembly - unlatch button, protection sleeve	SA-479 Type 410	Exposed to reactor coolant water
Drive rod assembly - coupling	SA-479 Type 403	Exposed to reactor coolant water
Drive rod assembly - springs	Alloy X-750 (ASME SB637 N07750)[2]	Exposed to reactor coolant water
Locking button in the drive rod assembly and pins in the latch assembly	Cobalt alloy (HAYNES No. 25 or equivalent material[3])	Exposed to reactor coolant water
Drive rod assembly other parts	SA-479 Type 304	Exposed to reactor coolant water
Coil assembly - housing	ASTM A536 Grade 60-40-18	Not exposed to reactor coolant water
Coil assembly - coil bobbins	Glass silicone resin	Not exposed to reactor coolant water
Coil assembly - wire	Double glass insulated copper	Not exposed to reactor coolant water
Welding material used in CRDMs housing	SFA-5.9 ER316L EC316L	~~Not exposed to reactor coolant water~~ Inside surface exposed to reactor coolant water

Notes: (1) Additional information appears in the text of Section 4.5 and Subsection 5.2.3.
(2) Additional stringent specification, MIL-S-23192, is applied.

Chapter 6

US-APWR DCD Revision 2 Chapter 6 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
6.2-60	6.2.5.2	Added “Detailed evaluation of the equipment survivability is provided in the technical report “US-APWR Probabilistic Risk Assessment” Section 15.7 (Ref. 6.2-37). The hydrogen igniters can perform its function during and after exposure to the environmental conditions created by hydrogen burn. Through the equipment survivability study, it is evaluated that the peak temperature of containment atmosphere becomes as high as approximately 1200°F, and the temperature rise from 400°F and reduced back to 400°F due to hydrogen burn takes approximately 10 minutes. The amount of hydrogen burnt in this analysis is conservatively assumed to be 100% active fuel length cladding reaction, hence this analysis broadly covers various uncertainties involved in the hydrogen generation and burn. Therefore, in terms of the equipment survivability, it is required that the hydrogen ignition system must keep its function longer than 10 minutes under the condition of containment atmosphere with higher than 400°F and its peak temperature to be as high as 1200°F.” RAI No.471, 06.02.05-35
6.4-6, 7	6.4.2.3	Added “, including 10 ft3/min for egress and ingress” Deleted “total system” Added “test value of CRE” Replaced “120” with “110” RAI No.473, 06.04-9

A diagram of the containment hydrogen monitoring and control system is presented in Figure 6.2.5-1. Containment hydrogen monitoring and control design parameters are found in Table 6.2.5-1.

Hydrogen monitoring system provides an ability to monitor and record the containment hydrogen concentration continuously at least 24 hours in the MCR. Service testing and calibration of the hydrogen monitoring system is always available because this system is located at outside of the containment. Monitoring and recoding are functional within 90 minutes after the initiation of safety injection with satisfying the requirements described in Revision 3 of RG 1.7 C.2.1 (Ref. 6.2-29).

The hydrogen monitoring and control system is supplied by the non-Class 1E P1 and P2 power system, with alternate power capability. P1 and P2 buses are capable of cross-connection, providing power to both motor control centers (MCCs). Both P1 and P2 buses are backed by non-Class 1E alternate ac gas turbine generators. The power distribution to the monitor and igniters is designed to minimize the impact of the loss of any single power source. As noted above, the containment hydrogen concentration is indicated in the MCR. This system may also be actuated manually.

The containment hydrogen monitoring and control system is not designed for seismic category I requirements since this system is required for plant protection for beyond design-basis accident. However, in considering the importance of the containment hydrogen monitoring and control system in order to maintain the containment integrity during postulated severe accidents, it is designed satisfying the plant HCLPF (high confidence of low probability failure) is evaluated more than 0.5G.

The containment hydrogen monitor and igniters are designed to function in a severe accident environment. Chapter 19, Subsection 19.2.3.3.7 describes equipment survivability in severe accident conditions inside the containment. Detailed evaluation of the equipment survivability is provided in the technical report “US-APWR Probabilistic Risk Assessment” Section 15.7 (Ref. 6.2-37). The hydrogen igniters can perform its function during and after exposure to the environmental conditions created by hydrogen burn. Through the equipment survivability study, it is evaluated that the peak temperature of containment atmosphere becomes as high as approximately 1200°F, and the temperature rise from 400°F and reduced back to 400°F due to hydrogen burn takes approximately 10 minutes. The amount of hydrogen burnt in this analysis is conservatively assumed to be 100% active fuel length cladding reaction, hence this analysis broadly covers various uncertainties involved in the hydrogen generation and burn.

Therefore, in terms of the equipment survivability, it is required that the hydrogen ignition system must keep its function longer than 10 minutes under the condition of containment atmosphere with higher than 400°F and its peak temperature to be as high as 1200°F.

The twenty hydrogen igniters are strategically located around the containment: one near the PRT, one in the upper area of the pressurizer subcompartment, one in the lower area of the pressurizer subcompartment, four in the SG/reactor coolant loop subcompartment (one in each subcompartment), four in the 2nd floor of the containment, four in the 3rd floor of the containment and five in the containment dome (near the top of

6.4.2.2.3 Isolation Dampers

MCR Air Intake Isolation Dampers:

- Two motor-operated air-tight dampers are installed in series in the outside air intake of the MCR HVAC system. These dampers are isolated in isolation mode. The two dampers are in series for single failure considerations.

MCR Toilet/Kitchen Exhaust Line Isolation Dampers:

- Two air-operated air-tight dampers are interlocked with the MCR toilet/kitchen exhaust fans and are installed at the inlet side of the MCR toilet/kitchen exhaust fans. These dampers are isolated in pressurization mode and isolation mode. The two dampers are in series for single failure considerations.

MCR Smoke Purge Line Isolation Dampers:

- Two air-operated air-tight dampers are interlocked with the MCR smoke purge fan and are installed at the inlet side of the MCR smoke purge fan. These dampers are isolated in pressurization mode and isolation mode. The two dampers are in series for single failure considerations.

MCR Emergency Filtration Unit Air Intake Damper:

- One motor-operated damper is installed in the duct between the outside air intake and the inlet side of each MCR emergency filtration unit. This damper sets the makeup air flow rate during pressurization mode.

MCR Emergency Filtration Unit Air Return Damper:

- One motor-operated damper is installed in the duct between the recirculation duct and the inlet side of each MCR emergency filtration unit. This damper sets the return air flow rate directed to the emergency filtration unit during pressurization mode.

The above mentioned isolation dampers are Equipment Class 3, seismic category I components.

6.4.2.3 Leaktightness

The potential leak paths (out-leakage) of the CRE are cable, pipe, and ductwork penetrations, doors, and HVAC equipment. The extent of out-leakage (and therefore pressurization) is dependent on the sealing characteristics, and integrity, at penetrations and doors. Total system inleakage in emergency pressurization mode is equal to or less than 120 ft^3/min, including 10 ft^3/min for egress and ingress. The makeup (outside air ventilation) flow rate during emergency pressurization mode is equal to or less than 1,200 ft^3/min. Exfiltration, required to create (and maintain) the differential pressure across the CRE boundary, is expected to equal the amount of makeup air and occur at the potential leak paths mentioned above.

System flow balancing and leakage tests are performed during the initial test program, as described in Chapter 14. The leakage tests establish exfiltration and infiltration rates to determine the MCR and emergency CRE flow balance necessary to achieve design pressure with respect to surrounding areas, in accordance with ASTM E741-00 (Ref. 6.4-3). The ASTM E741 tests confirm ~~total system~~ inleakage test value of CRE (~~~120~~110 ft^3/min) in the emergency pressurization mode and makeup flow rate (~1,200 ft^3/min) in the emergency pressurization mode.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

Positive pressure, due to exfiltration, is maintained inside CRE when the main control room HVAC system is in the emergency pressurization mode. This positive pressure reduces the infiltration of airborne radioactive contamination into the CRE during a Design Basis Accident. The positive pressure results in airflow in the outward direction from the CRE. In addition, the Class 1E electrical room HVAC system services rooms above, below and adjacent to the CRE. The auxiliary building HVAC system services the access corridor to CRE. These ventilation systems are configured and balanced to preclude airflow into the CRE, which harmonizes with the main control room HVAC system.

Other HVAC systems service areas adjacent to, above and below the CRE, however, no portion of these systems are connected to or pass through the CRE. The MCR toilet/kitchen exhaust fans and the smoke purge fan provide service to the CRE. Any adverse interaction from these two systems is prevented since the fan motors are de-energized and associated CRE isolation boundary dampers are closed, when emergency CRE ventilation flow is automatically initiated. Any potential leak paths are addressed in Subsection 6.4.2.3. There are no pressure-containing tanks or piping systems in the CRE that could, on failure, transfer or introduce hazardous material(s) into the CRE.

6.4.2.5 Shielding Design

The MCR shielding design requirements are based on the design basis accident analyses. Chapter 15 analyzes a broad array of accidents, including source term determinations and dose evaluations for the control room operators. The associated shielding requirements and designs are discussed in Chapter 12, Section 12.3, which also includes applicable plant arrangement drawings.

The design of the control room envelope shielding is based on the sources identified in Table 6.4-3. The distribution on the LOCA sources outside the control room is shown in Figure 6.4-7. Shielding thicknesses for the control room are described in Chapter 12, Subsection 12.3.

6.4.3 System Operational Procedures

In the normal operation mode, the MCR HVAC system maintains the proper environment in the MCR and other area within the CRE. The normal operation mode is described in Subsection 6.4.2 and Subsection 9.4.1.

Chapter 8

US-APWR DCD Revision 2 Chapter 8 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.2-2	Subsection 8.2.1.2 4th paragraph 9th sentence	RAI No.432 Question No.08.02-13 Added: “Performance of these transfers is permitted when the bus faulted signal is not initiated.”
8.2-9	Subsection 8.2.1.2 3rd paragraph 7th sentence	RAI No.432 Question No.08.02-10 Added: “The interface requirement for offsite power is maintaining a transmission system operating voltage of ±10% and a frequency of ±5%.”
8.2-10	COL 8.2(11)	RAI No.432 Question No.08.02-12 Replaced: “The COL applicant is to address the stability and reliability study of the offsite power system. Stability study is to be addressed in accordance with BTP 8-3 (Reference 8.2-17). The study addresses the loss of the unit, loss of the largest unit, loss of the largest load, or loss of the most critical transmission line including operating range, for maintaining transient stability. A failure modes and effects analysis (FMEA) is to be provided.” with “The COL applicant is to address the stability and reliability study of the offsite power system. The stability study is to be conducted in accordance with BTP 8-3 (Reference 8.2-17). The study should address the loss of the unit, loss of the largest unit, loss of the largest load, or loss of the most critical transmission line including the operating range, for maintaining transient stability. A failure modes and effects analysis (FMEA) is to be provided.”

UAT1 or RAT1. MV bus N2 can be fed from UAT2 or RAT2. MV buses N3, N4, A, B and P1 can be fed from UAT3 or RAT3. MV buses N5, N6, C, D and P2 can be fed from UAT4 or RAT4. For all these MV buses, if power is lost from one source, the buses are automatically transferred to the other source by fast or slow transfer scheme. At that time, if bus voltage is adequate, fast transfer is initiated. If this is not the case, slow transfer is initiated. Performance of these transfers is permitted when the bus faulted signal is not initiated. Detailed explanation of bus transfer scheme is described in Subsection 8.3.1.1.2.4. All low voltage buses are provided power from the MV buses. Each of the 6.9 kV Class 1E MV buses has its own onsite Class 1E standby emergency power source. Similarly, each of the 6.9 kV non-Class 1E MV permanent buses has its own onsite non-Class 1E standby emergency power source, designated as AAC power source. All MV buses can be powered from their associated UAT or RAT.

There are four, two-winding, UATs, namely UAT1, UAT2, UAT3, and UAT4. The high-side of these transformers is connected to the main generator isolated phase busduct down-stream of the GLBS. During normal power operation, with the GLBS closed, the MG provides power to the plant MV buses N1, N2, N3, N4, N5 and N6 through the UATs. During all other modes of plant operation, including PAs, with the GLBS open, these MV buses are powered through the UATs by back-feeding the MT from the offsite power sources. During all modes of plant operation including startup, normal and emergency shutdown and PAs, the MV Class 1E buses A, B, C and D are powered through the RATs from offsite power sources. Secondary voltages of UAT and RAT are displayed in the MCR.

There are four, three-winding RATs, namely RAT1, RAT2, RAT3, and RAT4. The high-side of these transformers is connected to the high voltage transmission tie line from the switchyard. The transmission tie line voltage level is site-specific. This is the normal preferred power source for all plant safety-related auxiliary and service loads. RAT1 and RAT2 can feed the 13.8 kV non-Class 1E buses N1 and N2, respectively. RAT3 can feed the 6.9 kV Class 1E buses A and B, and non-Class 1E buses N3, N4 and P1. RAT4 can feed the 6.9 kV Class 1E buses C and D, and non-Class 1E buses N5, N6 and P2.

Each of the safety-related and non safety-related MV buses (13.8 kV non-Class 1E buses N1 and N2; 6.9 kV Class 1E buses A, B, C, and D; and 6.9 kV non-Class 1E buses N3, N4, N5, N6, P1 and P2) is connected to a UAT and an RAT. For all Class 1E (A, B, C and D) MV buses, power from the RAT is the normal preferred source and power from the UAT is the alternate preferred source. Each safety MV bus also has its own backup emergency power supply from a safety-related Class 1E GTG. MV permanent buses P1 and P2 also have their own backup emergency power supply from a dedicated non-Class 1E GTG.

During all modes of plant operation, including normal and emergency shutdown and postulated accident conditions, all safety-related unit auxiliary and safety-related plant service loads are powered from offsite power sources through the RATs. This is the normal preferred offsite power source for the plant safety-related loads. The alternate preferred offsite power source to the plant safety-related loads is from the UATs, which are powered from offsite power sources by back feeding the MT. All plant MV buses, both safety-related and non safety-related, are connected to the UATs and RATs

- The switchyard buses to which the main offsite circuits are connected shall be arranged as follows:
  - Any incoming or outgoing transmission line for one circuit can be switched without affecting the other circuit.
  - Any circuit breaker can be isolated for maintenance without interrupting service to these circuits.

Transmission system stability is consistent with the condition of the transient and accident analysis in Chapter 15. It is assumed that the power supply to RCPs following a reactor/turbine trip is maintained at least 3 seconds by the main generator (turbine generator coast down) or the offsite power in Chapter 15. Following a reactor/turbine trip, stability of the offsite power is expected to be maintained, including the power supply to the RCPs. In addition, when the offsite power is lost concurrent with a reactor/turbine trip, the turbine-generator is still connected to the UATs and RCPs are powered by turbine-generator. The large inertia of the turbine-generator will maintain voltage and frequency more than 3 seconds. In case of a unit trip due to an electrical fault, the main transformer circuit breaker opens and the non-Class 1E buses are powered continuously via RAT. The interface requirement for offsite power is maintaining a transmission system operating voltage of ±10% and a frequency of ±5%. The COL applicant is to perform grid stability analysis to confirm the assumption in Chapter 15.

Transmission system reliability is consistent with the condition of the probability risk analysis of Chapter 19. The COL applicant is to confirm transmission system reliability.

8.2.4 Combined License Information

COL 8.2(1)	*The COL applicant is to address transmission system of the utility power grid and its interconnection to other grids.*
COL 8.2(2)	*Deleted*
COL 8.2(3)	*The COL applicant is to address the plant switchyard which includes layout, control system and characteristics of circuit breakers and buses, and lightning and grounding protection equipment.*
COL 8.2(4)	*The COL applicant is to provide detail description of normal preferred power.*
COL 8.2(5)	*The COL applicant is to provide detail description of alternate preferred power.*
COL 8.2(6)	*Deleted*
COL 8.2(7)	*The COL applicant is to address protective relaying for each circuit such as lines and buses.*
COL 8.2(8)	*The COL applicant is to address switchyard dc power as part of switchyard design description.*

COL 8.2(9) *The COL applicant is to address switchyard ac power as part of switchyard design description.*

COL 8.2(10) *The COL applicant is to address transformer protection corresponded to site-specific scheme.*

COL 8.2(11) *The COL applicant is to address the stability and reliability study of the offsite power system. The ~~S~~stability study is to be ~~addressed~~conducted in accordance with BTP 8-3 (Reference 8.2-17). The study should address~~es~~ the loss of the unit, loss of the largest unit, loss of the largest load, or loss of the most critical transmission line including the operating range, for maintaining transient stability. A failure modes and effects analysis (FMEA) is to be provided.*

COL 8.2(12) *Deleted*

8.2.5 References

8.2-1 IEEE Standard General Requirements for Liquid-Immersed Distribution, Power, and Regulating Transformers, IEEE Std C57.12.00, 2000.

8.2-2 IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations (NPGS), IEEE Std 765, 2006.

8.2-3 Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants, Regulatory Guide 1.81 Revision 1, January 1975.

8.2-4 IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, IEEE Std 308, 2001.

8.2-5 Loss of all alternating current power, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50.63.

8.2-6 Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, NUMARC 93-01, 2000.

8.2-7 Requirements for monitoring the effectiveness of maintenance at nuclear power plants, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50.65.

8.2-8 IEEE Standard for Generating Station Grounding, IEEE Std 665, 1995.

8.2-9 IEEE Design Guide for Electric Power Service Systems for Generating Stations, IEEE Std 666, 1991.

8.2-10 IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations, IEEE Std 1050, 2004.

8.2-11 IEEE Application Guide for Surge Protection of Electric Generating Plants, IEEE Std C62.23, 1995.

Chapter 9

US-APWR DCD Revision 2 Chapter 9 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.3-12	Subsection 9.3.2.2.2	RAI No.448, 09.03.02-11: Revise Section 9.3.2.2.2 and Figure 9.3.2-1 (Sheet 4 of 5) to reflect the change of moving C/V Atmosphere Gas Sample Cooler and C/V Atmosphere Gas Sample Moisture Separator to downstream of C/V Atmosphere Gas Sample Hood. The change for Figure 9.3.2-1 (Sheet 4 of 5) has already been incorporated in DCD revision 2.
9.3-20 9.3-79, 80	Subsection 9.3.3.2.3 Figure 9.3.3-1	RAI No.426, 09.03.03-16: Revise Section 9.3.3.2.3 and Figure 9.3.3-1 based on DCD revision 2. Also, change is made to correct the connection from the Turbine building sump as the radioactive drain from the Turbine building sump is sent to the LWMS waste holdup tanks for processing.

- Collects gaseous samples from auxiliary systems.
- Provides protection against exposure and contamination during collection of samples, and send a residual dew condensation water to the LWMS.

Sampling points of each system are specified in Table 9.3.2-1.

The PGSS is designed to collect representative samples for analysis by the plant operating staff from the containment atmosphere during normal operation. The sampling point is located at upper compartment area in the containment so that the point is not too close to the containment fan and at where containment atmosphere is well mixed. The gas sample is routed to CV atmosphere gas sample equipment outside CV through sample piping. This sample equipment is located outside the containment penetration as close as possible in order to shorten the sample piping length. This point also is used as post-accident sample point after an accident. Chemical and radiochemical analyses are performed by the plant operating staff to monitor gas compositions in the containment. The results of these analyses are used to detect radioactive material leakage.
+
In addition, a portion of the GWMS collects gaseous samples from the auxiliary systems. The PGSS is located in the ~~A~~R/B complex. The gas sampling station of the PGSS is inline type, which returns purge gas to containment. Sample line heat tracing and insulation are used on high temperature sample lines to preclude plate out. The gaseous sample vessels are positioned inside a filtered vent hood. The gas sampling station of the PGSS has manual-operated valves with extended handle to minimize radiation exposure to the plant operating staff. Heat tracing and on-line sampling minimize ~~Residual~~ dew condensation ~~liquid collected in the gas sample vessel of the PGSS is routed to the holdup tanks~~. The lines are purged before sampling to ensure that samples are representative. The purged gas is routed back to the containment atmosphere.

9.3.2.2.3 Post-Accident Sampling

The US-APWR has specific post-accident sampling lines, which have the capability to obtain and analyze highly radioactive samples of the reactor coolant, refueling water storage pit water (equivalent to containment sump water for conventional PWR), and containment atmosphere.

The PASS is required to maintain the capability to draw highly radioactive samples following an accident. Analysis of these samples can provide information regarding the cause of the accident, to quantify certain radionuclides that are indicators of the degree of core damage and to measure the post-accident sample activities during the accident recovery phase to determine the degree of core damage and general plant contamination.

The PASS consists of two lines, a post-accident liquid sampling line and a post-accident containment atmosphere sampling line.

or stopped by a level preset by the local instrumentation in the sump or a sump tank. The T/B sump pumps are not required to operate during design base accident.

Sumps are provided with duplex pumps or with simplex pumps. The T/B sump pumps are aligned to discharge to the waste water system for treatment prior to discharge to the environment. If the radiation level detected in the fluid by the radiation monitor is above a predetermined set point, the discharge from the sump is sent, following operator initiation to the A/B floor drain sump to be sent to the LWMS for processing.

The subsystems and their operation are described in subsequent paragraphs according to their classification as non-radioactive or potentially radioactive.

All liquid wastes drained from potentially radioactive drainage piping are conveyed by gravity to the respective buildings radioactive sump or tanks. The liquid is then discharged to the LWMS for processing.

A. Oily waste

Potentially radioactive oily waste is drained into the radioactive sump tanks within the respective building. The separated oil is collected for offsite disposal and the clarified effluent is discharged to the LWMS for processing. Refer to Chapter 11.

B. Chemical waste

The chemical wastes, containing chemicals and corrosive substances are discharged to the chemical drain tank.

C. Liner plate leakage detection

The leakage from the spent fuel pit, fuel transfer canal, cask pit and the fuel inspection canal is drained to the R/B sump tank.

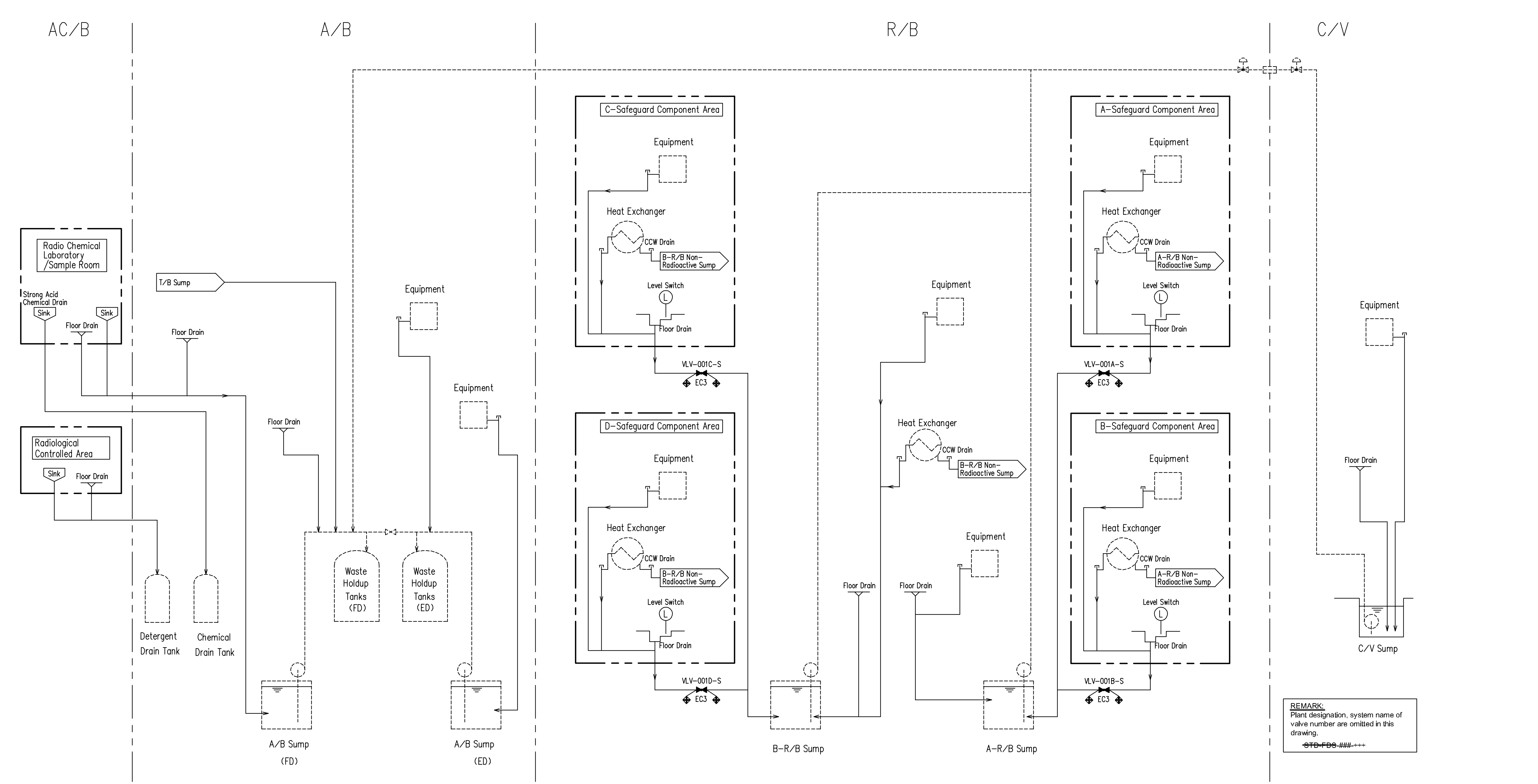
D. Non-radioactive drain collection points

- Reactor building non-radioactive drain sumps

  These sumps are located in the R/B and collects all non-radioactive equipment and floor drainage by gravity. The sump pumps normally discharge to the T/B sump. Both sump pumps are operated by the level instrumentation in each sump.

- Turbine building sump

  The T/B drain sump collects drain from all equipment and floor drainage in the T/B and non-radioactive drain sump. This sump normally discharges to the WWS for treatment. However, if the liquid draingage should become contaminated, the radiation monitor will detect a concentration exceeding the predetermined setpoint which will activate an alarm in the MCR for operator actions and will also activate the closure of the transfer valves. Following operator initiation, the contaminated waste is then sent to ~~the A/B floor drain sump to be transferred to~~ the LWMS~~.~~.



NOTE
A part stated as EC3 is safety-related on this figure.

Figure 9.3.3-1 Equipment and Floor Drain System Flow Schematic Radiological Controlled Area (Sheet 1 of 2)

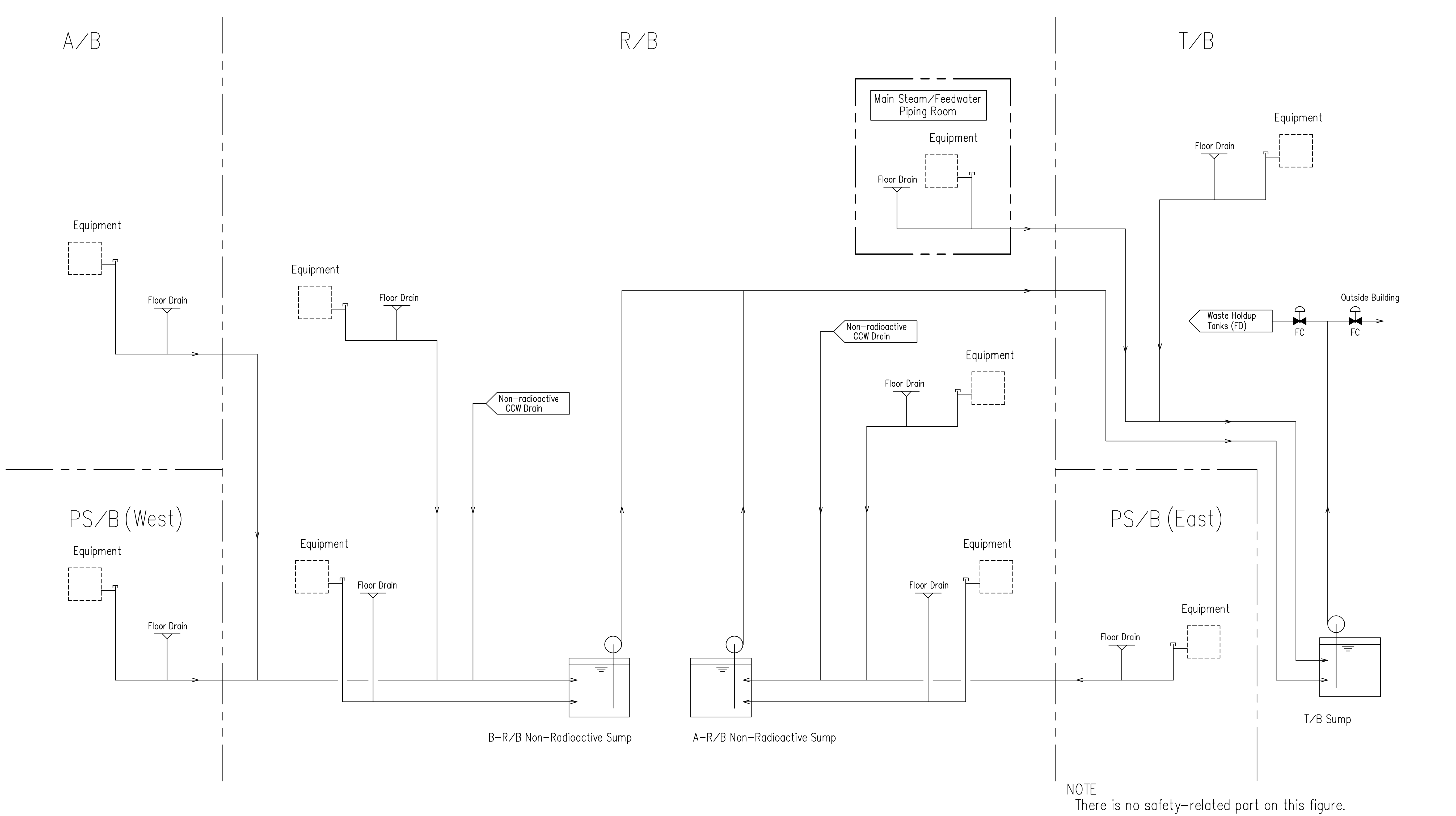


Figure 9.3.3-1 Equipment and Floor Drain System Flow Schematic Non-radiological Controlled Area (Sheet 2 of 2)

Chapter 10

US-APWR DCD Revision 2 Chapter 10 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.3-39	Section 10.3.5 Table 10.3.5-1 Crevice pHt	RAI 441-3461, Question No.10.04.06-8 Replaced “5-11” with “5-10”.
10.3-39	Section 10.3.5 Table 10.3.5-1 Notes 4	RAI 441-3461, Question No.10.04.06-8 Replaced “5-11” with “5-10”.

Table 10.3.5-1 Guidelines for Secondary Side Water Chemistry during Power Operation (Sheet 4 of 4)

– Steam Generator Blowdown –

Parameter	Control Value
Control	
Sodium, ppb	≤ 5.0(1)
Chloride, ppb	≤ 10.0(2)
Sulfate, ppb	≤ 10.0(3)
Diagnostic	
Crevice pHt	5-~~11~~10(4)

Notes:

1. There are several environments that can cause damage of SG tube, based on the SG corrosion susceptibility diagram for alloys 600MA, 600TT and 690TT (Reference 10.3-21). According to evaluation results for the SG crevice environment estimation code, when the sodium concentration increases and crevice concentration factor is 10^7, the SG tube crevice pH gradually increases and exceeds 10 at sodium concentrations of more than 5 ppb
2. Chloride causes pitting of SG tube material when present with oxidizing materials. The control value is set at 10 ppb and is 1/10 of the 100ppb standard dose and does not have significant influence on the SG tube material.
3. Sulfate causes an acidic environment of crevice pH.
4. SG tube materials are damaged at alkaline environments of pH 10 and over and acidic environments of pH 4 and below and temperatures at 300°C. The electric potential of the SG tube material increases due to oxidizing materials and may damage susceptibility range. Surveillance and management of the environment of the oxidizing agent carried over in the SG tube crevice is difficult. Therefore crevice pH during operation is evaluated by calculating the concentration of impurities to determine that they are within the recommended range of 5 to ~~11~~10.

Chapter 11

US-APWR DCD Revision 2 Chapter 11 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
11.2-8	Subsection 11.2.2	RAI No.462, 11.02-27: Add statement referring newly added Table 11.2-20.
11.2-13	Subsection 11.2.2.2.6	RAI No.462, 11.02-25: Add information on the waste effluent strainers for clarification of the component data.
11.2-14	Subsection 11.2.2.2.7	RAI No.462, 11.02-25: Add information on the neutralizing agent measuring tank for clarification of the component data.
11.2-14	Subsection 11.2.2.2.8	RAI No.462, 11.02-25: Add information on the detergent drain strainer for clarification of the component data.
11.2-23	Table 11.2-3	RAI No.462, 11.02-25: Add information on the neutralizing agent measuring tank for clarification of the component data.
11.2-45	Table 11.2-20	RAI No.462, 11.02-27: Add new Table 11.2-20 for clarification of the LWMS component classification.

reference therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Reference 11.2-22) using the inspection plan guidance of ASTM D 5163 (Reference 11.2-23). Level-detecting instrumentation measuring the current tank inventories is provided. High- and low-level alarms are provided. These alarms are annunciated in the radwaste control room located in the A/B and also in the MCR.

The sump tanks are equipped with two pumps and level instruments. This redundancy serves to minimize the effect of pump failure.

11.2.2.2.3 Pumps

The LWMS pumps are constructed from material appropriate for their intended use and the material is listed in Table 11.2-4. Generally, these pumps are stainless steel, horizontal centrifugal type. A listing of codes applicable to pumps is presented in Table 11.2-1.

11.2.2.2.4 Cartridge Filter

Cartridge filters are housed in enclosures that facilitate simplified change out with minimal occupational dose in compliance with the ALARA principle. These filters are located inside a shielded environment commensurate with the design basis source term. The filters efficiently remove suspended solids and radioactive particulates. Differential pressure measured across the filter provides an indication of its performance and is indicated locally in the radwaste control room. Spent filters are transferred to the solid waste management system (SWMS) for packaging and disposal.

11.2.2.2.5 Activated Carbon Filter

The carbon filter is a column holding carbon media designed to remove organic contaminants. This serves to protect the downstream ion exchange media from fouling. Differential pressure measured across the filter is indicated locally in the radwaste control room and provides an indication of the performance of the filter. Spent filter media is transferred as slurry with primary make-up water to the SWMS for further processing and packaging. A listing of applicable codes is presented in Table 11.2-1.

11.2.2.2.6 Ion Exchange Columns (Demineralizers)

The ion exchange columns are designed to remove radionuclide impurities in the liquid stream. The ion exchange resins, which consist of anion and cation resins, are selected during the detailed design phase. Differential pressure measured across the columns is indicated in the radwaste control room and provides an indication of the performance of the columns. Spent resin is transferred as slurry with primary make-up water to the SWMS. A waste effluent strainer is installed downstream of the ion exchange columns for the purpose of removing any resin fines that may be carried over from the columns. These stainless steel strainers are basket-type with 25 micron to 550 micron mesh openings. Differential pressure measured across the strainer is indicated locally and provides an indication of the performance of the strainer. A listing of applicable codes is presented in Table 11.2-1.

11.2.2.2.7 Chemical Drain Subsystem

The chemical drainage subsystem collects laboratory wastes and some of the decontamination solutions. To the greatest extent practicable, all decontamination solutions and process liquids are inherently free of hazardous materials and toxic substances. The use of these decontamination solutions and process liquids must not generate mixed waste. Additionally, laboratory wastes are collected for treatment and disposed of in appropriate portable containers. Only small amounts of laboratory wastes, basically those associated with the cleaning of glassware and similar activities, are expected to be in the chemical drainage subsystem. Any such wastes, which do not contain significant quantities of chemical constituents, may be transferred to the floor drainage processing subsystem.

Dilute acids and bases, along with heavy metals, are captured by the chemical drainage subsystem, pH adjusted, sampled, and characterized. The waste is neutralized prior to being pumped to waste holdup tanks for further processing or transferred to a container for disposal. Figure 11.2-1 provides flexibility to process chemical effluent either way.

The neutralization agent measuring tank is provided to measure and add a basic solution to the chemical drain tank and the waste holdup tanks. The tank is typically part of a neutralization package with a metering pump and associated piping. The neutralization package is designed to provide pH adjustments before water from the waste water hold up tank and chemical drain tank is processed. The component data on the neutralizing agent measuring tank is presented in Table 11.2-3.

11.2.2.2.8 Detergent Drain Subsystem

Detergent waste is collected in the detergent drain tank. This waste stream consists primarily of material from sinks, showers, emergency showers, etc. This waste stream does not typically contain any significant levels of radioactive contaminants. A detergent drain strainer is installed upstream of the detergent drain tank for the purpose of removing any materials that may be carried over from the waste streams. These stainless steel strainers are basket-type with 25 micron to 550 micron mesh openings. This waste stream is filtered and released through the discharge header.

The detergent drain tank is based on ANSI/ANS 55.6 (Ref. 11.2-6). The requirements for maximum daily input is 2,000 gallons. The tank is sized but excluding the collection of laundry waste as contaminated laundry is sent off site for cleaning and/or disposal. This tank is sufficient for anticipated operations. The equipment for this subsystem consists of the following:

- Detergent drain tank and associated detergent drain tank pump
- Filtration system
- Detergent drain monitor tank and associated detergent drain monitor tank pump
- Sample points

Table 11.2-3 Component Data – Tanks (Sheets 2 of 2)

The tank effective volumes are tank volumes, the batch volumes are approximately 80% of these values.

Tank Type	Specifications
Containment Vessel Sump	
Number of items	1
Nominal volume (gal)	1,200
Design Pressure	Atmospheric
Design temperature (°F)	200
Material	Stainless Steel
Neutralizing Agent Measuring Tank	
Number of Items	1
Nominal Volume (gals)	100
Type	Vertical / Cylindrical
Design Pressure	0 (Atmosphere)
Design Temperature (°F)	50-104
Material	Stainless Steel

Table 11.2-20 LWMS Component Classification

Component	Safety Classification
Waste Holdup	RW IIc
Waste Monitor Tank	RW IIc
Detergent Drain Tank	RW IIc
Detergent Drain Monitor Tank	RW IIc
Chemical Drain Tank	RW IIc
LWMS Filter	RW IIc
LWMS IX Columns	RW IIc
Detergent Drain Filters	RW IIc

- Reactor coolant loop (RCL) drainage
- Leakage from valves inside the containment
- RCS vent drainage
- ACC drainage
- Pressurizer relief tank drainage

LWMS component data are identified in Table 11.2-3 through Table 11.2-6 and Table 11.2-20. Component American Society of Mechanical Engineers (ASME) Code, seismic design, and quality assurance requirements for the components in the LWMS are shown in Chapter 3, Table 3.2-2. The LWMS complies with the quality assurance requirements of ANSI/ANS-55.6 (Ref. 11.2-6).

The annual average release of nuclides from the plant is determined using the PWR-GALE Code. The code input parameters used are provided in Table 11.2-9. Associated projected annual releases from a single plant are provided in Table 11.2-10.

Components and structures of the mentioned systems are not under adverse vacuum conditions as there are no vacuum conditions existing due to component operations.

11.2.2.1 Liquid Waste Processing System Operation

Radioactive liquid wastes are collected in various collection tanks located within the A/B and reactor building (R/B). The wastes entering these tanks are transferred from a number of locations within the plant including the following:

- Equipment drainage
- Floor drainage and other waste sources with potentially high suspended solid content
- Detergent wastes, generally from plant sinks and showers, that contain soaps and detergent which are not compatible with ion exchange resins
- Chemical wastes (generated in very low volumes)
- SG blowdown (when radioactivity above a setpoint is detected)

The processing flow rate is selected based on the completion of sampling and processing of the volume of one tank in one shift of operation, assuming 40 hours work per week. Treated water is collected in one of two monitor tanks. When a tank is filled, the tank is isolated and the monitor tank pump is turned on to circulate the tank content for sampling and analysis to confirm that the quality of the treated water is suitable for reuse in radwaste systems (i.e., pipe flushing, sluicing, and SRST tank filling). In the event that there is a surplus of water in the plant, the water is discharged. Hence, the discharge is not a continuous process and the discharge valves are under supervisory

11.5.2.5 Effluent Liquid Monitor Component Description

11.5.2.5.1 Liquid Radwaste Discharge Radiation Monitor (RMS-RE-035)

The liquid radwaste discharge radiation monitor is a γ monitor; the detection range and other details are summarized in Table 11.5-4, item number 29. A process schematic for this monitor is shown in Figure 11.5-1d. The monitor for liquid radwaste discharge is located in the A/B as shown in Figure 11.5-2a.

This monitor is located downstream of the sample tanks and pumps in the LWMS (refer to Section 11.2). RMS-RE-035 measures the total gamma content in the discharge stream of the LWMS. The monitor is an inline monitor, measuring the liquid discharge stream before it reaches the discharge header. As discussed in Section 11.2, the treated liquid is normally recycled for plant use. If there is a surplus of liquid, then the water is discharged. The discharge valve is under supervisory control and requires approval to open for discharge. Detection of radioactivity levels in the stream exceeding the predetermined setpoint, the monitor pump is automatically shut off and automatically activates an alarm in the MCR and also automatically closes the discharge valve. These discharge valves are designed to open only when actuated by the radiation monitor for acceptable range for discharge. These valves normally stay close with all other conditions, including high radiation level and/or lack of signal (loss of power supply and/or radiation monitor failure).

The monitor is not safety-related and does not perform any safety function.

11.5.2.5.2 ESW Radiation Monitoring and Sampling System (RMS-RE-074A, RMS-RE-074B, RMS-RE-074C, RMS-RE-074D)

The ESW radiation monitors are γ monitors; their detection range, and other details as summarized in Table 11.5-4, item number 30. The process configuration of the monitor is schematically presented in Figure 11.5-1f. The monitors are located in the R/B as shown in Figure 11.5-2a.

These monitors measure the radiation level in the essential service water (ESW) due to potential contamination of radioactive material in the essential service water system (ESWS). The ESW is not normally expected to be radioactive. Detection of radioactive material in the ESW stream is an indication of leakage within the heat exchange equipment. Four monitors are provided: one for each of the four ESW trains and is located downstream of the ESW pumps. Detection of radiation exceeding the predetermined setpoints automatically activates an alarm in the MCR for operator actions.

Piping taps are provided for the purging and cleaning of the monitors. The monitors are not safety-related and do not perform any safety function.

Chapter 12

US-APWR DCD Revision 2 Chapter 12 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.2-7	12.2.1.1.10 1st paragraph 1st line	Replaced "… outside the buildings include the following" with "… outside the buildings <u>but inside the tank house</u> include the following". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 2nd column, 7th row	Replaced "17.7" with "16.7". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 2nd column, 8th row	Replaced "6.9" with "5.9". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 3rd column, 6th row	Replaced "140.2" with "132.4". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 5th column, 6th row	Replaced "35" with "26", and "59" with "68". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 5th column, 7th row	Replaced "11" with "13", "61" with "57", and "28" with "31". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 5th column, 8th row	Replaced "63" with "54", and "32" with "41". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 6th column, 5th row	Replaced "41.6" with "44.5". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 6th column, 6th row	Replaced "129.2" with "121.5". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 6th column, 7th row	Replaced "82.5" with "85.2". RAI No.427-2909 Revision 1
12.2-15	Table 12.2-1 (Sheet 1 of 6) 6th column, 8th row	Replaced "86.9" with "96.7". RAI No.427-2909 Revision 1

US-APWR DCD Revision 2 Chapter 12 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.2-16	Table 12.2-1 (Sheet 2 of 6) 5th column, 6th row	Replaced "49" with "48", and "41" with "42". RAI No.427-2909 Revision 1
12.2-16	Table 12.2-1 (Sheet 2 of 6) 6th column, 5th row	Replaced "91.8" with "92.2". RAI No.427-2909 Revision 1
12.2-16	Table 12.2-1 (Sheet 2 of 6) 6th column, 6th row	Replaced "97.6" with "98.2". RAI No.427-2909 Revision 1
12.2-82	Table 12.2-60 (Sheet 1 of 3) 2nd column, 4th row	Replaced "for noble gas" with "for noble gas & tritium", and "iodine&tritium" with "iodine". RAI No.427-2909 Revision 1
12.2-82	Table 12.2-60 (Sheet 1 of 3) 2nd column, 9th row	Replaced "3.5μCi/g" with "0.35μCi/g". RAI No.427-2909 Revision 1
12.2-83	Table 12.2-60 (Sheet 2 of 3) 2nd column, 3rd row	Replaced "for noble gas" with "for noble gas & tritium", and "iodine&tritium" with "iodine". RAI No.427-2909 Revision 1
12.2-83	Table 12.2-60 (Sheet 2 of 3) 2nd column, 6th row	Replaced "3.5μCi/g" with "0.35μCi/g". RAI No.427-2909 Revision 1
12.2-84	Table 12.2-60 (Sheet 3 of 3) 1st column, 2nd row	Added "normal operation/" after "… leak rate in". RAI No.427-2909 Revision 1
12.2-84	Table 12.2-60 (Sheet 3 of 3) 2nd column, 2nd row	Replaced "for Radiation Zone V to VI" with "for Radiation Zone V or higher". RAI No.427-2909 Revision 1
12.2-84	Table 12.2-60 (Sheet 3 of 3) 2nd column, 7th row	Replaced "for Radiation Zone V to VI" with "for Radiation Zone V or higher". RAI No.427-2909 Revision 1
12.2-88	Table 12.2-61 (Sheet 4 of 6) title	Replaced "Radiation Zone V to VI" with "Radiation Zone V or higher". RAI No.427-2909 Revision 1

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.3-3	12.3.1.1.1.2 E. Tanks 1st paragraph 2nd line	Replaced "Overflow lines are directed to the waste collection system to control any contamination within the plant structures. Tanks containing radioactive fluids are either equipped with open vents to the cubicle or the ventilation system." with "Overflow lines are directed to the waste collection system to minimize the potential for the spread of contamination within the plant structures. Tanks containing radioactive fluids have overboard line at least equal in size to the largest inlet line. The tank vent line is either open to the cubicle or connected directly to the ventilation system. The spent resin tank vents are equipped with a break-pot, which separates the air from the moisture and any entrained resin, which are subsequently sent to the A/B sump, and vents the air to the exhaust ductwork. These measures minimize the possible contamination of the area and the ductwork." RAI No.425-3264 Revision 1
12.3-13	12.3.1.2.2 3rd paragraph 1st line	Replaced "Projected dose rates and mission dose for the vital areas…" with "Projected dose rates and cumulative mission doses for tasks performed within the vital areas…". RAI No.429-3178 Revision 1
12.3-13	12.3.1.2.2 3rd paragraph 2nd line	Added "In this mission dose evaluation, it is assumed that workers use respiratory protection devices, thus only direct dose is considered. Alternatively, the dose calculation for the MCR personnel under accident conditions does not assume the use of respiratory protection devices, and exposure due to airborne activity is considered. The total mission doses for the post accident sampling activity and MCR personnel are given in Table 12.3-3 (Sheet 4 of 4)." after the 1st sentence. RAI No.429-3178 Revision 1

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.3-21	12.3.3.3 1st bulleted item	Added following paragraphs as the 7th and 8th paragraph: "Ventilation openings, in areas where flooding might occur, are located so that water entry in not possible. HEPA filters are specified not to fail for at least 20 inches wg differential pressure across them. Ventilation systems containing HEPA filters have fans with static pressure capacities well below 20 inches wg." Deleted the bulleted mark at the beginning of 6th paragraph. RAI No.425-3264 Revision 1
12.3-27	12.3.4.1.2 4th paragraph 1st line	Replaced "The area monitors are installed in the following locations" with "The <u>fixed</u> area monitors are installed in the following locations <u>to warn occupants of the area of a deteriorated radiological condition</u>". RAI No.429-3178 Revision 1
12.3-27	12.3.4.1.2 5th paragraph 1st line	Replaced "<u>Furthermore, during work activities,</u> a portable ARM<u>S</u> is installed in the following locations" with "<u>For areas with positive access control features, such as normally locked doors, or areas where a radiological hazard only exists during specific work activities, a fixed ARM is not required. Instead,</u> a portable ARM is installed <u>to warn occupants of a deteriorated radiological condition. Portable ARMs are utilized</u> in the following locations". RAI No.429-3178 Revision 1
12.3-30	12.3.4.1.9 2nd paragraph 2nd line	Added "And the methodology to determine the calibration interval and setpoints for the Area Radiation Monitors and Process and effluent Radiation Monitors are described in the Subsection 7.2.2.7. The calibration procedures are described in the Subsection 13.5.2.2." as the 2nd and 3rd sentence. RAI No.429-3178 Revision 1

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.3-34	12.3.6 COL 12.3(6) COL 12.3(7) COL 12.3(8)	Added three COL items as follows: "COL 12.3(6) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69. COL 12.3(7) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21. COL 12.3(8) IF the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.". RAI No.429-3178 Revision 1
12.3-43	Table 12.3-3 (Sheet 1 of 3) title	Replaced "sheet 1 of 3" with "sheet 1 of 4". RAI No.429-3178 Revision 1
12.3-44	Table 12.3-3 (Sheet 2 of 3) title	Replaced "sheet 2 of 3" with "sheet 2 of 4". RAI No.429-3178 Revision 1
12.3-45	Table 12.3-3 (Sheet 3 of 3) title	Replaced "sheet 3 of 3" with "sheet 3 of 4". RAI No.429-3178 Revision 1
12.3-46	Table 12.3-3 (Sheet 4 of 4)	Added the new table. RAI No.429-3178 Revision 1
12.3-61	Figure 12.3-1 (Sheet 11 of 34)	Moved the location of the ICIS Area Radiation Monitor. RAI No.429-3178 Revision 1

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.4-3	12.4.1 4th paragraph 8th line	Added "and wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air" at the end of this paragraph. RAI No.429-3178 Revision 1

12.2.1.1.10 Miscellaneous Sources

The principal sources of activity outside the buildings but inside the tank house include the following:

- The refueling water storage auxiliary tank
- The primary makeup water tank

The content of the water tanks is processed by the SFP purification system, or the boron recycle system until the activity in the fluids is sufficiently low to result in dose rates less than 0.25 mrem/h at 2 meters from the surface of the tank.

Radionuclide inventories of the refueling water storage auxiliary tank and primary makeup water tank are presented in Tables 12.2-50 and 12.2-51. There are no other significant amounts of radioactive fluids permanently stored outside the buildings.

Spent fuels are stored in the SFP. When the fuel is to be moved away from the SFP, it is placed in a spent fuel shipping cask for transport.

Storage space is allocated in the radwaste processing facility for storage of spent filter cartridges and packaged spent resins.

Radioactive wastes stored inside the plant structures are shielded so that areas outside the structures meet Radiation Zone I criteria. Additional storage space for radwaste is to be provided in the detailed design by the COL Applicant. If it becomes necessary to temporarily store radioactive wastes/materials outside the plant structures, radiation protection measures are to be taken by the radiation protection staff to ensure compliance with 10 CFR 20 (Reference 12.2-1), 40 CFR 190 (Reference 12.2-6) and to be consistent with the recommendations of RG 8.8 (Reference 12.2-2).

The SWMS facilities process and store dry active waste. If it becomes necessary to install additional radwaste facilities for dry active waste, it is to be provided by the COL Applicant. Radiation shielding is to be provided such that the dose rates comply with the requirements of 10 CFR 20 (Reference 12.2-1) and 40 CFR 190 (Reference 12.2-6). Interior concrete shielding is provided to limit exposure to personnel during waste processing. The ALARA methodology of RGs 8.8 (Reference 12.2-2) and 8.10 (Reference 12.2-3) has been used in the design of this facility.

Any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are to be provided by the COL Applicant.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent fuel, the residual heat removal system (RHRS), and the incore instrumentation system (ICIS). Individual components may

Table 12.2-1 Radiation Sources Parameters (Sheet 1 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Inside the containment vessel							
Steam generator Plenum Side Shell Side	 66.9 65.9	 63.0 434.2	 Homogeneous Homogeneous	 Source Water Source Water 22 wt%+ Secondary Water 9wt%+ Steel 69wt%	 ~~41.6~~44.5 69.2	 6.1 3.4	4
Regenerative heat exchanger * Plenum Side Shell Side	8.3	 23.2 ~~140.2~~132.4	 Homogenous Homogenous	 Water (Charging Line) Water (Letdown Line) ~~35~~26 wt%+ Water (Charging Line) 6 wt%+ Steel ~~59~~68 wt%	 62.4 ~~129.2~~121.5	2.0	3
Letdown heat exchanger Plenum Side Shell Side	~~17.7~~16.7	 24.4 189.8	 Homogenous Homogenous	 Source Water Source Water ~~11~~13 wt%+ Cooling water ~~61~~57 wt%+ Steel ~~28~~31 wt%	 62.4 ~~82.5~~85.2	ignored	1
Excess letdown heat exchanger Plenum Side Shell Side	~~6.9~~5.9	 21.7 130.2	 Homogenous Homogenous	 Source Water Source Water 5 wt%+ Cooling water ~~63~~54 wt%+ Steel ~~32~~41wt%	 62.4 ~~86.9~~96.7	 1.8 ignored	1

* The regenerative heat exchanger consists of three shells.

Table 12.2-1 Radiation Sources Parameters (Sheet 2 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Outside the containment vessel (Reactor Building)							
Containment spray/residual heat removal heat exchanger Plenum Side Shell Side	31.5	56.7 264.4	Homogenous Homogenous	Source Water Source Water 15 wt%+ Cooling water 48 wt%+ Steel 37 wt%	62.4 ~~91.8~~92.2	1.8 1.2	4
Seal water heat exchanger Plenum Side Shell Side	8.4	22.3 144.6	Homogenous Homogenous	Source Water Source Water 10 wt%+ Cooling water ~~49~~48 wt%+ Steel ~~41~~42 wt%	62.4 ~~97.6~~98.2	ignored	1
Volume control tank Liquid Phase Vapor Phase	47.2	179.2 107.5 71.7	Homogenous Homogenous	Air Water	7.6E-02 62.4	ignored	1

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Containment) (Sheet 1 of 3)

Parameter/ Assumption	Value
Reactor coolant leakage rate in normal operation	100 lb/d
Reactor coolant evaporation rate in refueling	1020 lb/h
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.45(others) (in refueling/shutdown) 1.0(for noble gas & tritium) 0.1(iodine~~&tritium~~) 0.001(others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor coolant tritium specific activity	3.5μCi/g
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	~~3.5~~0.35μCi/g
Low volume purge flow rate	(in normal operation) 2000 cfm
High volume purge flow rate	(in refueling) 30000 cfm
Purge flow duration	continuous

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Fuel Handling Area) (Sheet 2 of 3)

Parameter/ Assumption	Value
Reactor coolant evaporation rate in refueling	750 lb/h
Fraction of radioactive material to free volume	(in refueling/shutdown) 1.0(for noble gas & tritium) 0.1(iodine~~&tritium~~) 0.001(others)
Fuel defect	1%
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	~~3.5~~0.35μCi/g
Flow rate	24000 cfm
Flow duration	continuous

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building) (Sheet 3 of 3)

Parameter/ Assumption	Value
Reactor coolant leak rate in normal operation/refueling (Note)	100 lb/d (for Radiation Zone V ~~to VI~~or higher) 50 lb/d (for Radiation Zone IV) 2 lb/d (for Radiation Zone III)
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.1(iodine&tritium) 0.001(others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor coolant tritium specific activity	3.5μCi/g
Flow rate	1500 cfm(for Radiation Zone V ~~to VI~~or higher) 14000 cfm (for Radiation Zone IV) 76000 cfm (for Radiation Zone III)
Flow duration	continuous

(Note) Reactor coolant leak rates were derived from the leakage flow rates of the valves under consideration. Each Radiation Zone has a different number of valves handling radioactive fluids. Radiation Zones V and higher have many component cubicles and valve galleries. These zones have many radioactive valves. Zone IV has relatively high radiation level corridors, but has fewer radioactive valves than Zone V. Zone III has low radiation level corridors and access areas, and has fewer radioactive valves than Zone IV. As a result, the leak rate in Zone V or higher is high, while in Zones IV and III, the leak rates is low.

Table 12.2-61 Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone V ~~to VI~~or higher) (Sheet 4 of 6)

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)
Kr-83m	3.4E-07	1E-02	Ru-106	7.9E-14	5E-09
Kr-85m	1.3E-06	2E-05	Ag-110m	7.3E-16	4E-08
Kr-85	6.9E-05	1E-04	Te-125m	3.2E-13	2E-07
Kr-87	8.7E-07	5E-06	Te-127m	1.3E-12	1E-07
Kr-88	2.5E-06	2E-06	Te-127	6.8E-12	7E-06
Xe-131m	3.1E-06	4E-04	Sb-129	2.2E-14	4E-06
Xe-133m	3.1E-06	1E-04	Te-129m	4.4E-12	1E-07
Xe-133	2.3E-04	1E-04	Te-129	5.5E-12	3E-05
Xe-135m	5.7E-07	9E-06	Sb-131	8.9E-15	1E-05
Xe-135	7.7E-06	1E-05	Te-131m	1.2E-11	2E-07
Xe-138	5.0E-07	4E-06	Te-131	6.3E-12	2E-06
I-130	4.7E-09	3E-07	Te-132	1.3E-10	9E-08
I-131	1.2E-07	2E-08	Cs-132	6.2E-13	2E-06
I-132	6.4E-08	3E-06	Te-133m	1.2E-11	2E-06
I-133	2.1E-07	1E-07	Te-133	6.0E-12	9E-06
I-134	4.4E-08	2E-05	Cs-134	5.7E-10	4E-08
I-135	1.3E-07	7E-07	Te-134	2.2E-11	1E-05
Br-82	6.4E-12	2E-06	Cs-135m	6.7E-12	8E-05
Br-83	5.8E-11	3E-05	Cs-135	1.5E-15	5E-07
Br-84	3.1E-11	2E-05	Cs-136	1.5E-10	3E-07
Rb-86	5.6E-12	3E-07	Cs-137	3.2E-10	6E-08
Rb-87	-	6E-07	Cs-138	7.4E-10	2E-05
Rb-88	3.2E-09	3E-05	Ba-140	1.7E-12	6E-07
Rb-89	7.3E-11	6E-05	La-140	4.5E-13	5E-07
Sr-89	1.4E-12	6E-08	La-141	1.2E-13	4E-06
Sr-90	9.2E-14	2E-09	Ce-141	2.6E-13	2E-07
Y-90	2.1E-14	3E-07	Ce-143	2.2E-13	7E-07
Sr-91	9.5E-13	1E-06	Pr-143	2.4E-13	3E-07
Y-91m	4.9E-13	7E-05	Ce-144	2.0E-13	6E-09
Y-91	2.2E-13	5E-08	Pr-144	2.0E-13	5E-05
Sr-92	5.3E-13	3E-06	Pm-147	2.2E-14	5E-08
Y-92	4.1E-13	3E-06	Sm-147	-	2E-11
Y-93	1.8E-13	1E-06	Eu-154	2.1E-15	8E-09
Zr-93	-	3E-09	Na-24	2.9E-11	2E-06
Zr-95	2.7E-13	5E-08	Cr-51	2.8E-12	8E-06
Nb-95m	2.0E-15	9E-07	Mn-54	1.9E-12	3E-07
Nb-95	2.7E-13	5E-07	Mn-56	9.6E-11	6E-06
Mo-99	3.3E-10	6E-07	Fe-55	1.9E-12	8E-07
Tc-99m	1.3E-10	6E-05	Fe-59	3.3E-13	1E-07
Tc-99	-	3E-07	Co-58	4.5E-12	3E-07
Mo-101	1.5E-11	6E-05	Co-60	6.6E-13	1E-08
Tc-101	1.4E-11	1E-04	Zn-65	5.4E-13	1E-07
Ru-103	2.3E-13	3E-07	H-3	2.6E-07	2E-05
Rh-103m	2.2E-13	5E-04			

Back-flushable filters are designed so that the filter internals may be remotely removed and placed in a shielded cask for offsite shipping and disposal, in the unlikely event that a filter loses its back-flush capability.

Liquid systems containing radioactive cartridge filters are provided with a remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment for burial. The process is accomplished so that exposure to personnel and the possibility of an inadvertent radioactive release to the environment are minimized. Each filter is contained in a shielded compartment and provided with vent and drain valving, and individual compartments have drainage capabilities. The filter handling system has also been designed with a minimum of components susceptible to malfunction.

B. Demineralizers

Demineralizers for highly radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin storage tanks so that fresh resin can be loaded into the demineralizer remotely. The demineralizers and piping are designed with the ability to be flushed with demineralized water. Strainers are installed in the vent lines to prevent the entry of spent resin into the exhaust duct.

C. Evaporators

Adequate space and flanged connections for easy removal are provided for the maintenance of evaporator components. Additionally, the evaporator can be operated in an automatic operation mode that can reduce the exposure of the operator to radiation from the equipment.

D. Pumps

Wherever practicable, pumps are sealed with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed to allow easy removal, if necessary. All pumps in the radioactive waste systems are provided with flanged connections for ease of removal. Pump casings are provided with drain connections for draining pumps for maintenance.

E. Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to ~~control any~~ minimize the potential for the spread of contamination within plant structures. Tanks containing radioactive fluids ~~are~~ have overboard lines at least equal in size to the largest inlet line. The tank vent line is either ~~equipped with~~ open ~~vents~~ to the cubicle or connected directly to the ventilation system. The spent resin tank vents are equipped with a break-pot, which separates the air from the moisture and any entrained resin, which are subsequently sent to the A/B sump, and vents the air to the exhaust ductwork. These measures minimize the possible contamination of the area and the ductwork.

Projected dose rates and cumulative mission doses for tasks performed within the vital areas at various times after an accident are given in Table 12.3-3. In this mission dose evaluation, it is assumed that workers use respiratory protection devices, thus only direct dose is considered. Alternatively, the dose calculation for the MCR personnel under accident conditions does not assume the use of respiratory protection devices, and exposure due to airborne activity is considered. The total mission doses for the post accident sampling activity and MCR personnel are given in Table 12.3-3 (Sheet 4 of 4).

The US-APWR is designed to ensure the capability to achieve cold shutdown without subjecting personnel to excessive radiation exposure. This capability is further described in Chapter 7, Section 7.4. Radiation protection design features and access controls are described in Sections 12.3 and 12.5. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates defined on Figures 12.3-3 through 12.3-6 and Table 12.3-3, and appropriate time limits for presence in the area are imposed.

12.3.2 Shielding

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this section.

12.3.2.1 Design Objectives

The objective of the plant radiation shielding is to reduce personnel and population exposures, in conjunction with a program of controlled personnel access to, and occupancy of, radiation areas to levels that are within the requirements of 10 CFR 50 (Reference 12.3-7) and are ALARA within the dose standards of and requirements of 10 CFR 20 (Reference 12.3-2).

Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials, utilizing the design recommendations in accordance with the guidance in RG 8.8, Paragraph C.2 (Reference 12.3-1), where practicable.

Three plant conditions are considered in the nuclear radiation shielding design:

- Normal, full-power operation
- Shutdown conditions
- Emergency operations (for required access to safety-related equipment)

The shielding design objectives for the plant during normal operation (including anticipated operational occurrences), for shutdown operations, and for emergency operations are as follows:

- To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and individuals at and beyond the site boundary are in Low Population Area unrestricted areas beyond the site boundary is ALARA and within the requirements specified in 10 CFR 20.1301 (Reference 12.3-17) and 10 CFR 50, Appendix I (Reference 12.3-18).

- The requirements of 10 CFR 20, Appendix B (Reference 12.3-16) is satisfied in the control room following the DBAs described in Chapter 15, Subsection 15.6.5.5.
- The dose to control room personnel shall not exceed the limits specified in GDC 19 of Appendix A to 10 CFR 50 (Reference 12.3-8) following the DBAs described in Chapter 15, Subsection 15.6.5.5.

12.3.3.3 Design Features

To accomplish the design objectives and to conform to the design criteria, the following design guidelines are employed wherever practicable.

- Guidelines to minimize airborne radioactivity:

  Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

  Equipment vents and drains are piped directly to a collection device connected to the collection system. This is to prevent any contaminated fluid from flowing across the floor to a floor drain.

  Welded piping systems are employed on systems containing highly radioactive fluids to the maximum extent practicable.

  Suitable coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.

  Diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

  ~~•~~The design of the equipment incorporates features that minimize the spread of radioactivity during maintenance operations.

  Ventilation openings, in areas where flooding might occur, are located so that water entry in not possible.

  HEPA filters are specified not to fail for at least 20 inches wg differential pressure across them. Ventilation systems containing HEPA filters have fans with static pressure capacities well below 20 inches wg.

- Guidelines to control airborne radioactivity:

  The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

  In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This minimizes the amount of uncontrolled exfiltration from the area.

The fixed area monitors are installed in the following locations to warn occupants of the area of a deteriorated radiological condition:

(a) MCR

(b) Inside of the containment

(c) Radio Chemical Lab

(d) SFP area

(e) Nuclear sampling room

(f) Inside of the containment (near the air lock)

(g) Inside of the containment (near the ICIS)

(h) Waste Management System (WMS) area

(i) TSC

~~Furthermore, during work activities,~~For areas with positive access control features, such as normally locked doors, or areas where a radiological hazard only exists during specific work activities, a fixed ARM is not required. Instead, a portable ARM~~S~~ is installed to warn occupants of a deteriorated radiological condition. Portable ARMs are utilized in the following locations:

(j) Refueling platform

(k) Residual heat removal pump and heat exchanger areas

(l) Hot machine shop

(m) HVAC filter area

(n) Cask handling area

(o) Equipment decontamination area

(p) Safe shutdown panel area

12.3.4.1.3 General System Description

The ARMSs are located at selected locations throughout the plant to detect, indicate, and store information through their associated data processing module on the radiation levels and, if necessary, annunciate abnormal radiation conditions.

Each monitor is composed of the requisite number of channels, with a channel consisting of a radiation detector and check source, except for monitors RMS-RE-091, RMS-RE-092, RMS-RE-093, RMS-RE-094.

The detectors for all area monitors are sensitive to gamma rays. If exposed to radiation in excess of full-scale indication, the area monitors indicate that the full-scale reading has been exceeded and remains at the full-scale value. If the radiation field causing the

To continuously indicate the radiation levels in the WMS. An alarm signal warns the occupants of the WMS of a deteriorated radiological condition.

12.3.4.1.9 Range and Alarm Setpoints

The range and control function of the ARMS is given in Table 12.3-4.

Alarm setpoints are controlled by plant procedures and the offsite dose calculation manual, where appropriate. And the methodology to determine the calibration interval and setpoints for the Area Radiation Monitors and Process and effluent Radiation Monitors are described in the Subsection 7.2.2.7. The calibration procedures are described in the Subsection 13.5.2.2.

Radiation zones for the normal operation of US-APWR are described in Table 12.3-2.

The following monitors are located in radiation Zones I or II:

- MCR Area Radiation Monitor
- Radio Chemical Lab. Area Radiation Monitor
- TSC Area Radiation Monitor

The MCR Area Radiation Monitor has a greater sensitivity than the other area radiation monitors since it is located in a Zone I radiation area and the reactor operators are present. The installed containment high radiation monitor has sufficient instrumentation range to measure radiation levels during an accident.

Each area radiation monitor has two alarm setpoints – intermediate and high. If a monitor has a control function (i.e., Containment High Range Area Radiation Monitor), the control function is triggered coincidentally with the high alarm setpoint. An intermediate alarm gives both a visual and audible indication in the control room (or alternate radwaste control room in the case of the ARMS) and near the detector where the radiation level has reached the intermediate setpoint. A high alarm gives both a visual and audible indication in the control room and near the detector where the high alarm setpoint has been reached.

12.3.4.2 Airborne Radioactivity Monitoring Systems

The Airborne Radioactivity Monitoring System is provided for monitoring in-plant airborne radioactivity levels.

12.3.4.2.1 Design Objectives

The design objectives of the Airborne Radioactivity Monitoring System during normal operating plant conditions and anticipated operational occurrences are as follows:

12.3.6 Combined License Information

COL 12.3(1) The COL Applicant is responsible for the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.

COL 12.3(2) Deleted.

COL 12.3(3) Deleted.

COL 12.3(4) The COL Applicant is to provide the site radiation zones that is shown on the site-specific plant arrangement plan.

COL 12.3(5) The COL Applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.

COL 12.3(6) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69.

COL 12.3(7) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.

COL 12.3(8) IF the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.

12.3.7 References

12.3-1 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Paragraph C.2, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.3-2 "Standards for Protection against Radiation," Energy. Title 10 Code of Federal Regulations Part 20, U.S. Nuclear Regulatory Commission, Washington, DC.

Table 12.3-3 Projected Dose Rates for the Vital Areas at Various times after an Accident (sheet 1 of ~~3~~4)

POST ACCIDENT Vital Areas	Various Times after an Accident			
	1 hour	1 day	1 week	1 month
MCR	≦1.0 mrem/h	≦1.0 mrem/h	≦1.0 mrem/h	≦1.0 mrem/h
TSC	≦1.0 mrem/h	≦1.0 mrem/h	≦1.0 mrem/h	≦1.0 mrem/h
Radio chemical Laboratory	≦2.5 mrem/h	≦2.5 mrem/h	≦2.5 mrem/h	≦2.5 mrem/h
Hot counting room	≦2.5 mrem/h	≦2.5 mrem/h	≦2.5 mrem/h	≦2.5 mrem/h
Post accident sampling system (Liquid sampling)	≦1 rem/h	≦15 mrem/h	≦15 mrem/h	≦15 mrem/h
Post accident sampling system (Gas sampling)	≦1 rem/h	≦15 mrem/h	≦15 mrem/h	≦15 mrem/h

Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 hour after) (sheet 2 of ~~3~~4)

Vital Area	Task description	Time when access required [h]	Max dose rate [rem/h]	Mission dose [rem]	Access route zone map No.
Main control room (MCR)	Access to MCR from AC/B for operation.	2.8E-02	1.0E-03	2.8E-05	Figure 12.3-3 Sheet 3,4,5
		7.7E-03	1.0	7.7E-03	
		Total		7.7E-03	
	Return to MCR from Radiochemical laboratory	5.1E-02	2.5E-03	1.3E-04	Figure 12.3-3 Sheet 3,4,5
		7.7E-03	1.0	7.7E-03	
		Total		7.8E-03	
Technical support center (TSC)	Access to TSC from AC/B for operation.	1.4E-02	1.0E-03	1.4E-05	Figure 12.3-3 Sheet 3,4,5
Postaccident sampling system (Liquid or gas sampling)	Access to PASS from MCR for sampling. (Sampling time is included)	3.7E-02	2.5E-03	9.2E-05	Figure 12.3-3 Sheet 3 to 8
		9.3E-02	1.0	9.3E-02	
		Total		9.3E-02	
Radiochemical laboratory & Hot counting room	Access to Radiochemical laboratory from PASS for sample analysis. Access to HOT counting room from PASS for sample counting. (Analysis and counting time are included)	8.5E-01	2.5E-03	2.1E-03	Figure 12.3-3 Sheet 3 to 8
		5.2E-02	1.0	5.2E-02	
		Total		5.4E-02	

(Note) Walk speed is usually about 13000 ft/h (4 km/h) and stairs are about 6500 ft/h (2 km/h). Sampling time is 2 minutes and analysis time is 50 minutes.

Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 day to 1 month after) (sheet 3 of ~~3~~4)

Vital Area	Task description	Time when access required [h]	Max dose rate [rem/h]	Mission dose [rem]	Access route zone map No.
Main control room (MCR)	Access to MCR from AC/B for operation.	2.8E-02	1.0E-03	2.8E-05	Figure 12.3-4 Sheet 3,4,5
		7.7E-03	1.5E-02	1.2E-04	
		Total		1.5E-04	
	Return to MCR from Radiochemical laboratory	5.1E-02	2.5E-03	1.3E-04	Figure 12.3-4 Sheet 3,4,5
		7.7E-03	1.5E-02	1.2E-04	
		Total		2.5E-04	
Technical support center (TSC)	Access to TSC from AC/B for operation.	1.4E-02	1.0E-03	1.4E-05	Figure 12.3-4 Sheet 3,4,5
Postaccident sampling system (Liquid or gas sampling)	Access to PASS from MCR for sampling. (Sampling time is included)	3.7E-02	2.5E-03	9.2E-05	Figure 12.3-4 Sheet 3 to 8
		9.3E-02	1.5E-02	1.4E-03	
		Total		1.5E-03	
Radiochemical laboratory & Hot counting room	Access to Radiochemical laboratory from PASS for sample analysis. Access to HOT counting room from PASS for sample counting. (Analysis and counting time are included)	8.5E-01	2.5E-03	2.1E-03	Figure 12.3-4 Sheet 3 to 8
		5.2E-02	1.5E-02	7.8E-04	
		Total		2.9E-03	

(Note) Walk speed is usually about 13000 ft/h (4 km/h) and stairs are about 6500 ft/h (2 km/h). Sampling time is 2 minutes and analysis time is 50 minutes.

Table 12.3-3 Activity Mission Dose (sheet 4 of 4)

Activity Description	Mission Dose [rem]	Note
Post accident sampling and analysis activity (1 hour after)	1.6E-01	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 2 of 4)
Post accident sampling and analysis activity (1 day to 1 month after)	4.6E-03	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 3 of 4)
Operation activity in the MCR (30 days after the LOCA)	4.5	Described in Table 15.6.5-16

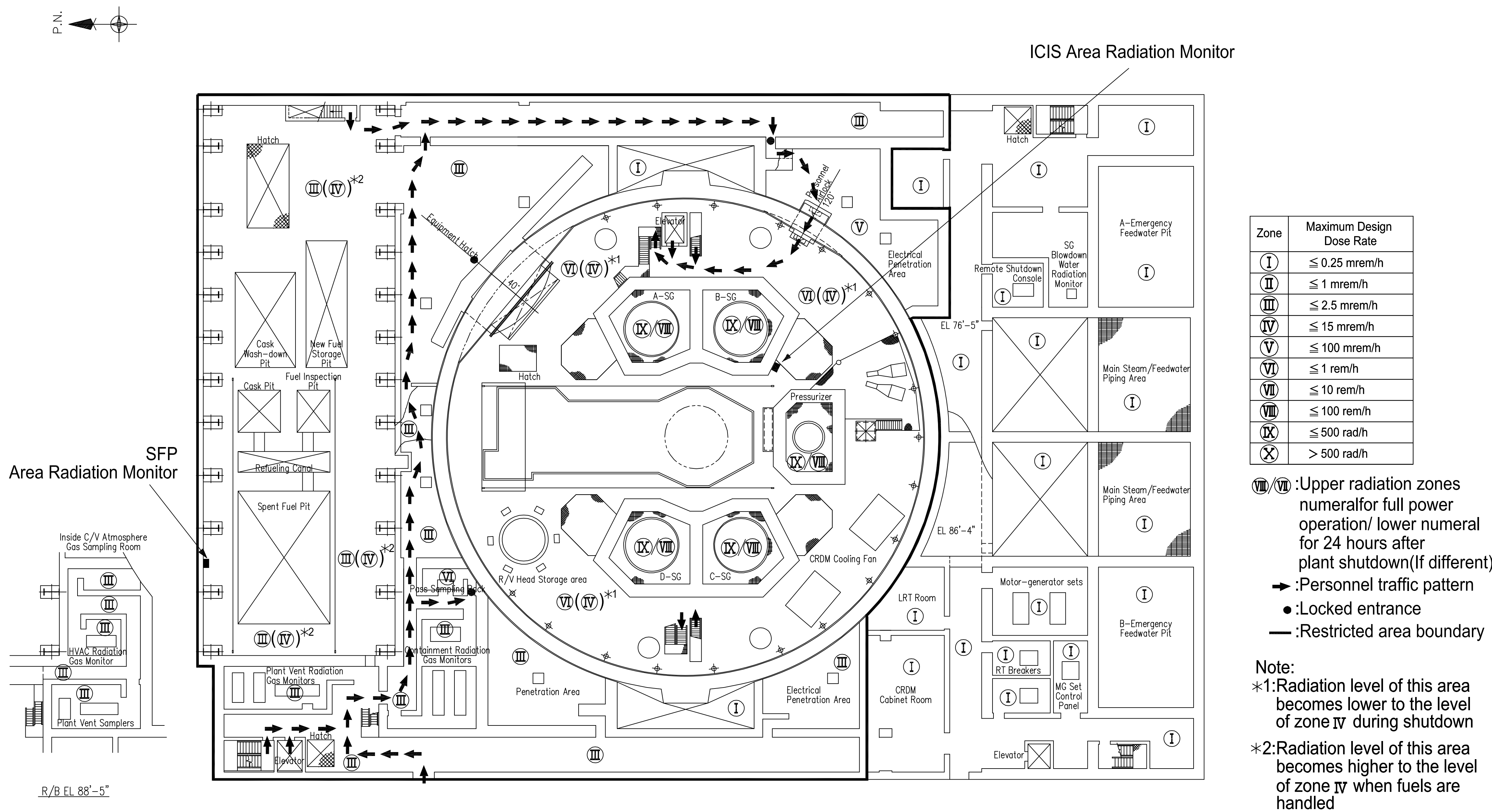

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 11 of 34)
Reactor Building at Elevation 76'-5"

(SRI)

expected that operation of the US-APWR results in radiation exposures of less than 100 person-rem per year.

12.4.1 Occupational Radiation Exposure

Radiation exposures to operating personnel are restricted to the limits of 10 CFR 20 (Reference 12.4-3). The Operational health physics program described in Section 12.5 and the radiation protection features described in Section 12.3 together maintain occupational radiation exposures ALARA. The airborne concentration is shown in Table 12.2-61. The airborne dose is less than the dose limit of 10 CFR 20 (Reference 12.4-3).

In the analysis of occupational radiation exposure data from operating plants of a design similar to the US-APWR, that is, domestic plants having Westinghouse-designed nuclear steam supply systems, the best operating plant performance is 0.1 rem per megawatt electrical per year of electricity produced. Major factors contributing to this level of occupational radiation exposure include low plant radiation fields, good layout and access provisions, and operational practices and procedures that minimize time spent in radiation fields. As discussed, the US-APWR design incorporates features to reduce occupational radiation exposure that goes beyond the designs provided for plants currently in operation.

The estimated annual occupational radiation exposures are developed within the following categories (Reference 12.4-1):

- Routine operations and surveillance
- Non-routine operations and surveillance
- Routine maintenance
- Waste processing
- Refueling operations
- ISI
- Special maintenance

Exposure data obtained from operating plants have been reviewed to obtain a breakdown of the doses incurred within each category. For several routinely performed operations, this information has been used to develop detailed dose predictive models. These models identify the various steps that are included in the operation, radiation zones, required number of workers, and the time to perform each step. This information has been used to develop dose estimates for each of the preceding categories. There is no separate determination of doses due to airborne activity. Experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses and wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air.

Chapter 14

US-APWR DCD Revision 2Chapter 14 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
14.2-32	14.2.12.1.1	Added the following text as C.7: The leakage control program plant procedures which implement Technical Specifications program 5.5.2, Primary Coolant Sources Outside Containment, are performed while the plant is in hot standby. CP RAI86, 14.02-10 (MHI ref.: UAP-HF-09499)
14.2-107	14.2.12.1.78	Added the following text to the end of B.3: Type testing of the instrumentation used to detect primary-to-secondary leakage in the steam generators (see subsection 5.2.5.3) includes demonstrating that these instruments have the required sensitivity per NEI 97-06. CP RAI86, 14.02-11 (MHI ref.: UAP-HF-09510)
14.3-24, 25	14.3.4.12	Added the text as the second paragraph to specify the administrative control of the test program and test abstract for physical security hardware. RAI 481, 14.03-26, 14.03-27, 14.03-29, 14.03-30 (MHI ref.: UAP-HF-09516)
14.3-30	14.3.7	Added new references as “14.3-39” and “14.3-40”. RAI 481, 14.03-26, 14.03-27, 14.03-29, 14.03-30 (MHI ref.: UAP-HF-09516)

14.2.12.1.14 CVCS Preoperational Test - Letdown

14.2.12.1.15 RCS Lithium Addition and Distribution Test

14.2.12.1.19 Resistance Temperature Detectors (RTDs)/Thermocouple Cross-Calibration Preoperational Test

14.2.12.1.21 Main Steam Supply System Preoperational Test

14.2.12.1.22 Residual Heat Removal System (RHRS) Preoperational Test

14.2.12.1.23 Main Steam Isolation Valve (MSIV), Main Feedwater Isolation Valve (MFIV) and Main Steam Check Valve Preoperational Test

14.2.12.1.25 Turbine-Driven Emergency Feedwater System Preoperational Test

14.2.12.1.50 Dynamic State Vibration Monitoring of Safety Related and High-Energy Piping

14.2.12.1.51 Steady State Vibration Monitoring of Safety Related and High-Energy Piping

14.2.12.1.52 Thermal Expansion Test

14.2.12.1.54 Safety Injection System (SIS) Preoperational Test

14.2.12.1.56 Safety Injection Check Valve Preoperational Test

14.2.12.1.66 Reactor Cavity Cooling System Preoperational Test

14.2.12.1.69 Containment Fan Cooler System Preoperational Test

14.2.12.1.71 RCS Leak Rate Preoperational Test

14.2.12.1.72 Loose Parts Monitoring System Preoperational Test

14.2.12.1.76 Remote Shutdown Preoperational Test

14.2.12.1.83 Steam Generator Blowdown System Preoperational Test

14.2.12.1.84 Sampling System Preoperational Test

14.2.12.1.87 Component Cooling Water System Preoperational Test

14.2.12.1.107 Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test

7. The leakage control program plant procedures which implement Technical Specifications program 5.5.2, Primary Coolant Sources Outside Containment, are performed while the plant is in hot standby.

3. Test instrumentation is available and calibrated. Type testing of the instrumentation used to detect primary-to-secondary leakage in the steam generators (see subsection 5.2.5.3) includes demonstrating that these instruments have the required sensitivity per NEI 97-06.

4. Suitable check sources are available.

C. Test Method

1. The operation of each monitor is verified.

2. Setpoint, control logic, annunciation (e.g. high alarm of SFP area radiation monitor), and power failure alarms of each monitor is verified.

3. The uncertainty and determination of setpoint of each monitor is verified.

D. Acceptance Criterion

1. The process and effluent radiological monitoring system, area radiation monitoring system and airborne radioactivity monitoring system operate as described in Section 11.5 and Subsection 12.3.4.

14.2.12.1.79 High-Efficiency Particulate Air Filters and Charcoal Adsorbers Preoperational Test

A. Objective

1. To demonstrate operation of the high-efficiency particulate air (HEPA) filters and charcoal adsorbers. This includes the MCR HVAC system, technical support center (TSC) HVAC system, annulus emergency exhaust system and containment purge system.

B. Prerequisites

1. Required construction testing is completed.

2. Component testing and instrument calibration is completed.

3. Test instrumentation is available and calibrated.

4. Required support systems are available.

5. The ventilation systems containing HEPA filters and charcoal adsorbers are air balanced and are operational and available to support this test.

6. Replacement of HEPA filters and adsorber material used during system construction is completed.

C. Test Method

1. HEPA filters and charcoal adsorbers are tested in place.

provided in SRP 14.3 (Reference 14.3-2) and the applicable generic ITAAC in SRP 14.3.12 (Reference 14.3-16). They provide for verifying that:

- Vital equipment is located only within vital areas.
- The external walls, doors, ceiling and floors in the main control room and the central alarm station are bullet resistant.
- Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in the central alarm station.
- Security alarm annunciation and video assessment information are available in the central alarm station.
- The central alarm station is located inside a protected area and the interior of the alarm station is not visible from the perimeter of the protected area.
- The secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.
- Security alarm devices including transmission lines to annunciators are tamper indicating and self-checking (i.e., an automatic indication is provided when failure of the alarm system or a component occurs or when on standby power), and alarm annunciation indicates the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location.
- Intrusion detection and assessment systems are designed to provide visual display and audible annunciation of alarms in the central alarm station.
- Intrusion detection systems equipment exists to record onsite security alarm annunciation including the location of the alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, time and disposition of each alarm is recorded.
- Emergency exits through vital area boundaries are alarmed and secured by locking devices that allow prompt egress during an emergency.
- The central alarm station has conventional (land line) telephone service with local law enforcement authorities and a system for communication with the main control room and is capable of continuous communication with security personnel.

System tests of physical protection systems and related design features are performed as acceptance tests under the US-APWR Test Program Description, MUAP-08009 (Reference 14.3-39). Tests of installed physical security hardware to verify proper installation and functionality of security hardware components are performed as construction acceptance tests and installation tests as specified in MUAP-08009 (Reference 14.3-39). The organization, processes and controls for system acceptance tests, construction acceptance tests, and installation tests are as specified by MUAP-08009 (Reference 14.3-39). Descriptions of the specific inspections, tests and analysis for US-APWR physical protection systems are specified in the US-APWR Physical Protection System Test Abstracts, "LATER" (Reference 14.3-40)

The COL applicant provides proposed ITAAC for the facility's physical security hardware not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate.

14.3.4.13 ITAAC for the Design Reliability Assurance Program

Section 2.13 of Tier 1, which covers the design reliability assurance program, is prepared in accordance with the guidance in RG 1.206 (Reference 14.3-1), SRP 14.3 (Reference 14.3-2), and SRP 17.4 (Reference 14.3-36).

Section 17.4 describes the design reliability assurance program, which is developed in accordance with guidance in NUREG-0800, SRP 17.4 (Ref 14.3-36). The purposes of this program are to provide reasonable assurance that: (1) the US-APWR is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for the SSCs, (2) the risk-significant SSCs do not degrade to an unacceptable level during plant operations, (3) the frequency of transients that challenge risk-significant SSCs is minimized, and (4) the risk-significant SSCs function reliably when challenged. An additional goal is to facilitate communication among the PRA, the design, and the ultimate COL activity to assure that the design is consistent and integrated with the procurement process. To this end, Table 17.4-1 identifies risk-significant SSCs for the US-APWR design.

Section 2.13 of Tier 1 contains a brief summary of the design reliability assurance program based on details provided in Section 17.4. The risk significant SSCs will be identified by introducing site-specific information to the list shown in Table 17.4-1. A single ITAAC is provided to verify that that the design reliability assurance program provides reasonable assurance that the designs of these SSCs are consistent with the assumptions used in the associated risk analyses.

14.3.4.14 ITAAC for the Initial Test Program

Section 2.14 of Tier 1, which addresses the initial test program, is prepared in accordance with the guidance in RG 1.206 (Reference 14.3-1), SRP 14.3 (Reference 14.3-2), and SRP 14.2 (Reference 14.3-37).

Section 14.2 describes the initial test program for the US-APWR plant, which is developed in accordance with guidance in RG 1.68 (Reference 14.3-38), RG 1.206 (Reference 14.3-1) and SRP 14.2 (Reference 14.3-37). Some of the activities associated with the initial test program occur as a part of the initial plant startup.

Section 2.14, of Tier 1 provides a general description of the preoperational and startup test programs and the major program documents that define how the initial test program is to be conducted and controlled. This section also describes the key elements of the initial test program.

14.3-32 'Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50.49, U.S. Nuclear Regulatory Commission, Washington, DC.

14.3-33 "Environmental Radiation Protection Standards for Nuclear Power Operations," Protection of Environment. Title 40, Code of Federal Regulations, Part 190, U.S. Nuclear Regulatory Commission, Washington, DC.

14.3-34 Deleted.

14.3-35 Deleted.

14.3-36 'Reliability Assurance Program (RAP),' "Quality Assurance," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 17.4, Initial Issuance, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

14.3-37 'Initial Plant Test Program – Design Certification and New License Applicants,' "Initial Test Program and ITAAC – Design Certification," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 14.2, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

14.3-38 Initial Test Programs for Water-Cooled Nuclear Power Plants, Regulatory Guide 1.68, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

14.3-39 US-APWR Test Program Description, MUAP-08009, Rev. 1, October 2009.

14.3-40 US-APWR Physical Protection System Test Abstracts, LATER, 2009.

Chapter 16

US-APWR DCD Revision 2 Chapter 16 Tracking Report Revision 0 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
Technical Specifications		
3.5.2-2	SR 3.5.2.1	Open Item No. 16-135-1818/51 Added “SIS-MOV-024A,B,C and D”, “Safety Injection Pump Full-Flow Test Line Stop” and “ CLOSED”.
3.5.2-2	SR 3.5.2.4	Open Item No. 16-135-1818/53 Added “SR3.5.2.4 Verify each ECCS valve manually activated during a design basis accident event in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position.” and “In accordance with the Inservise Testing Program”. Replaced “SR 3.5.2.4” with “SR 3.5.2.5” Replaced “SR 3.5.2.5” with “SR 3.5.2.6”
3.8.1-1	REQUIRED ACTION A.2	Open Item No.16-134-1825/26 Added Required Action A.2.
3.8.1-2	REQUIRED ACTION A.3.1 and A.3.2	Open Item No.16-134-1825/26 Changed number of Required Action A.2.1 to A.3.1 and A.2.2 to A.3.2.
3.9.4-1	LCO	Open Item No.16-133-1827/15 Added brackets before and after “four”
3.9.5-2	ACTIONS A.4	Open Item No. 16-133-1827/15 Added brackets before and after “four”
3.9.6-2	ACTIONS B.3	Open Item No. 16-133-1827/15 Added brackets before and after “four”
Bases		

US-APWR DCD Revision 2 Chapter 16 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
B3.5.2-7, 8	SR 3.5.2.4 SR 3.5.2.5 SR 3.5.2.6	Open Item No. 16-135-1818/53 Added “SR 3.5.2.4 This Surveillance demonstrates that each ECCS valve manually activated during a design basis accident event actuates to the required position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in required position. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.” Replaced “SR 3.5.2.4” with “SR 3.5.2.5” Replaced “SR 3.5.2.5” with “SR 3.5.2.6”
B 3.8.1-5	REQUIRED ACTION A.2	Open Item No.16-134-1825/26 Added Required Action A.2.
B 3.8.1-6	REQUIRED ACTION A.3.1 and A.3.2	Open Item No.16-134-1825/26 Changed number of Required Action A.2.1 to A.3.1 and A.2.2 to A.3.2.
B 3.8.1-18	SURVEILLANCE REQUIREMNETS SR 3.8.1.8 Last Paragraph	Open Item No.16-72-853 Replaced “This power factor is representative of the actual inductive loading a Class 1E GTG would see under design basis accident conditions.” with “This power factor should be maintained as close as practicable to actual power factor which a Class 1E GTG would see under design basis accident conditions, such as 0.85.”
B 3.9.4-1	BACKGROUND	Open Item No.16-133-1827/15 Added brackets before and after “four”
B3.9.5-3	ACTIONS A.4, A.5, A.6, and A.6.2	Open Item No. 16-133-1827/15 Added brackets before and after “four”
B3.9.6-3	ACTIONS B.3, B.4, B.5.1, and B.5.2	Open Item No. 16-133-1827/15 Added brackets before and after “four”

SURVEILLANCE REQUIREMENTS

SURVEILLANCE				FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position (with power to the valve operator removed).			[12 hours OR In accordance with the Surveillance Frequency Control Program]
Number	Function		Position	
SIS-AOV -201B and C	Accumulator Makeup		CLOSED	
SIS-MOV -024A,B,C and D	Safety Injection Pump Full-Flow Test Line Stop		CLOSED	
SR 3.5.2.2	Verify each SIS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.			[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.5.2.3	Verify each SI pump's developed head at the test flow point is greater than or equal to the required developed head.			In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each ECCS valve manually activated during a design basis accident event in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position.			In accordance with the Inservice Testing Program
SR 3.5.2.~~4~~5	Verify each SI pump starts automatically on an actual or simulated actuation signal.			[24 months OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.~~5~~6	Verify by visual inspection, each SIS train ECC/CS STRAINER is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.	[24 months OR In accordance with the Surveillance Frequency Control Program]

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following ac electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E ac electrical power distribution system,

b. Three Class 1E Gas Turbine Generators (GTGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and

c. The associated automatic load sequencers for each required Class 1E GTG shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable to Class 1E GTGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit. AND A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	1 hour AND Once per 8 hours thereafter 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	<u>AND</u>	
	A.~~2~~3.1 Restore required offsite circuit to OPERABLE status.	72 hours
	[<u>OR</u>	
	A.~~2~~3.2 -------------NOTE-------------- This Required Action is not applicable in MODE 4. ----------------------------------- Apply the requirements of Specification 5.5.18.	72 hours]
B. One required Class 1E GTG inoperable.	B.1 Perform SR 3.8.1.1 for the required offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	B.2 Declare required feature(s) supported by the inoperable Class 1E GTGs inoperable when its required redundant feature in a train with an OPERABLE Class 1E GTG is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	B.3.1 Determine OPERABLE Class 1E GTGs are not inoperable due to common cause failure.	24 hours
	<u>OR</u>	

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3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is closed and held in place by [four] bolts, or if open, capable of being closed,

b. One door in the emergency air lock is closed and one door in the personnel airlock capable of being closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent or

2. Capable of being closed by an OPERABLE Containment Purge Isolation System.

--NOTE--
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend movement of irradiated fuel assemblies within containment.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.4 Close equipment hatch and secure with [four] bolts.	4 hours
	AND	
	A.5 Close one door in each air lock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify two RHR loops are in operation and circulating reactor coolant at a flow rate of ≥ 2645 gpm per pump.	[12 hours OR In accordance with the Surveillance Frequency Control Program]

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
	B.2	Initiate action to restore two RHR loops to operation.	Immediately
	AND		
	B.3	Close equipment hatch and secure with [four] bolts.	4 hours
	AND		
	B.4	Close one door in each air lock.	4 hours
	AND		

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

Periodic surveillance testing of SI pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4

This Surveillance demonstrates that each ECCS valve manually activated during a design basis accident event actuates to the required position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in required position. SRs are specified in the Inservice Testing Program of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.~~4~~5

This Surveillance demonstrates that each SI pump starts on receipt of an actual or simulated ECCS actuation signal. [The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length. This equipment is not at risk of imminent damage as it is designed to remain functional and in good condition while in operation, thus significant degradation due to a longer surveillance interval should not be of major concern. The design reliability is, therefore, maintained by taking these considerations based on sound engineering judgment. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.~~5~~6

Periodic inspections of the ECC/CS STRAINER ensure that it is unrestricted and stays in proper operating condition. [The 24 month Frequency is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length. This equipment is not at risk of imminent damage as it is designed to remain functional and in good condition while in operation, thus significant degradation due to a longer surveillance interval should not be of major concern. The design reliability is, therefore, maintained by taking these considerations based on sound engineering judgment. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. Subsection 6.2.1.
4. Subsection 15.6.5.
5. Chapter 19.

ACTIONS (continued)

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated Class 1E GTG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. This includes motor driven emergency feedwater pumps. Two train systems, such as turbine driven emergency feedwater pumps, may not be included.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. One required train has no offsite power supplying it loads and

b. A required feature on the other train (Train A, B, C or D) is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked. Discovering no offsite power to one required train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and Class 1E GTGs are adequate to supply electrical power to the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

ACTIONS (continued)

A.~~2~~3.1 [and A.~~2~~3.2]

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and Class 1E GTGs are adequate to supply electrical power to the onsite Class 1E distribution system.

[Required Action A.~~2~~3.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Requied Action is not applicable in MODE 4.]

The 72 hour Completion Time takes into account the capacity and capability of the remaining ac sources, a reasonable time for repairs, and the low probability of PA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable Class 1E GTG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that Class 1E GTGs in two trains are inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven emergency feedwater pumps. Two train systems, such as turbine driven emergency feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable Class 1E GTG.

SURVEILLANCE REQUIREMENTS (continued)

frequency values to which the system must recover following load rejection. [The 24 month Frequency of this SR is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. The reliability proved in Technical Report MUAP-07024 for Class 1E GTG is based on operating experience of non-nuclear gas turbine generator with reduced surveillance and its frequency from emergency generator of nuclear plant. The Class 1E GTG in US-APWR should be more reliable by performing the SRs. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. The 24 month Frequency is also consistent with the recommendation of Regulatory Guide 1.9 (Ref. 3). OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

Note 2 ensures that the Class 1E GTG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor ~~is representative of the actual inductive loading~~ should be maintained as close as practicable to actual power factor which a Class 1E GTG would see under design basis accident conditions, such as 0.85. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should

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B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 50.34. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least [four~~-~~] bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. Alternatively, the equipment hatch can be open provided it can be installed with a minimum of [four] bolts holding it in place.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of irradiated fuel assemblies within containment, the containment air locks must be capable of being closed.

BASES

ACTIONS

RHR loop requirements are met by having two RHR loops OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation or insufficient forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If no RHR is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with [four] bolts,

b. One door in each air lock must be closed, and

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore two RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of three OPERABLE RHR loops and at least two operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

If no RHR is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with [four] bolts,

b. One door in each air lock must be closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

Chapter 19

US-APWR DCD Revision 2 Chapter 19 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
19.1-42 to 19.1-43	19.1.4.2 Sensitivity analysis case (CASE 14)	RAI#423_19-371 and 19-373 Inserted new sensitivity analysis case in which test interval of valve result in CDF impact.
19.1-108	19.1.6.1 Bullet of “Indications of temperature”	RAI#443_19-397 Changed “As for inaccurate hot leg temperature measurement after loss of decay heat removal, reactor coolant hot leg temperature instruments are located in the flow path during RHR operation, so this parameter can be accurately indicated.” to ”As for inaccurate hot leg temperature measurement while RHR flow is maintained whereas RHR heat exchanger function has degraded, reactor coolant hot leg temperature and core exit temperature instruments are located in the flow path during RHR operation. Accordingly the malfunction of RHR system can be identified.”
19.1-115	19.1.6.1 Third bullet of heading description in LOOP event tree	Editorial Changed the stated allowable time for the LOOP recovery to six hours to reflect the allowable time results for POS 8-1 (Table 20.7-1 of Ref. 19.1-47) Revised the probability that LOOP duration does not exceed six hours.
19.1-380	Table 19.1-38 Sheet 2	RAI# 423 19-362 Added discussion regarding uncertainties associated with the partial loss of CCW initiating event frequency.
19.1-380	Table 19.1-38 Sheet 2	RAI#423 19-376 Enhanced discussion regarding uncertainties associated with system unavailability.
19.1-382	Table 19.1-38 Sheet 4	RAI#423 19-371, 19-373 Added discussion regarding uncertainties associated with failure probabilities of valves with long test intervals.
19.1-383	Table 19.1-38 Sheet 5	RAI# 423 19-363 Added discussion regarding uncertainties associated with the CCF parameters for normally running pumps.

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Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
19.1-948 to 19.1-971	Table 19.1-119	Editorial Changed total sheet number to 24 to insert new sheet (sheet 9) of Table 19.1-119 to reflect RAI #423 19-368.
19.1-956	Table 19.1-119 Sheet 9	RAI#423 19-368 Inserted PRA key assumption for CCF of check valve in sheet 9 of Table 19.1-119.
19.1-956	Table 19.1-119 Sheet 9	RAI#423 19-387 Inserted PRA assumption regarding maintenance and configuration risk management programs
19.1-1049	Figure 19.1-20	RAI#39 19-68 Changed the loss of offsite power event tree figure to the one that has been revised associated with change in loss of offsite power initiating event frequency.
19.2-34 to 19.2-35	19.2.5 (3) After the last sentence of first paragraph in (During LPSD Operations)	RAI#443_19-396 Inserted new sentence “However, the ability … for LPSD risk reduction.”
19.2-35	19.2.5 (4) After the last sentence of first paragraph in (During LPSD Operations)	RAI#443_19-396 Inserted new sentence “However, the ability … for LPSD risk reduction.”

In the base case, common cause failure of sump screens are evaluated from with generic failure data and generic common cause failure parameters. Although sump screens of US-APWR are design to minimize failure due to clogging, common cause failure CCF probability of sump screen may increase at for large LOCA. In this sensitivity analysis, the probability of all four sump screens to clog at large LOCA has been assumed to be 0.0625 (=0.5^4) per demand. The resulting CDF is 1.1E-06/RY. This CDF is 7% higher than the base case CDF.

Valve Reliability

Sensitivity analysis of valve reliability that has high FV importance and long test interval is performed to study the impact of its uncertainty on plant CDF for internal initiating events at power.

- CASE 14: Test Interval of Valves

  Failure probabilities of valves reported in NUREG/CR-6928 and used in the US-APWR PRA are independent from test intervals. The failure probability of valves reported in NUREG/CR-6928 is based on failure data of valves that have average test intervals less than 12 months. Sensitivity analyses are performed applying higher failure probabilities to valves that have FV importance higher than 2.0E-03, considering longer test intervals. Valves that have high FV importance and have test intervals sufficiently longer than the NUREG data are the followings.

  - Main steam isolation valves (NMS-SMV-515A, B, C, D)
  - EFW pit outlet check valves (EFS-VLV-008A, B)
  - EFW pump outlet check valves (EFS-VLV-012A, B, C, D)
  - EFW line check valves (EFS-VLV-018A, B, C, D)
  - Safety depressurization valves (RCS-MOV-117A, B)
  - Pressurizer safety valves (RCS-VLV-120, 121, 122, 123)

  All of these valves are under control of the in-service test program and are required to be tested every 24 months except the pressure safety valves, which is tested every 60 months. Demand failure probabilities of NUREG/CR-6928 are adjusted for long test intervals, based on the mean testing interval of the NUREG data and the actual test interval of the valves. Failure probabilities were adjusted in two ways, either by assuming that 100% (Type A) or 50% (Type B) failures in the NUREG are standby failures. The resulting CDF and increment ratio are shown below.

Component Description	CDF [/RY]	
	Type A	Type B
Base Case	1.0E-06	
Main steam isolation valves	1.2E-06 (+19.4%)	1.1E-06 (+8.7%)
EFW pit outlet check valves	1.1E-06 (+4.9%)	1.1E-06 (+1.9%)
EFW pump outlet valves	1.1E-06 (+2.9%)	1.0E-06 (+1.0%)
EFW line check valves	1.1E-06 (+2.9%)	1.0E-06 (+1.0%)
Safety depressurization valves	1.0E-06 (+1.8%)	1.0E-06 (+1.0%)
Pressurizer safety valves	1.1E-06 (+5.8%)	1.1E-06 (+2.9%)

For the first case, if higher failure probability of valves considering long test interval are used, the increase of CDF is approximately 20%. For latter case, the resulting increase in CDF is approximately 9%.

The major conclusions of the importance and sensitivity analyses are:

- Basic events that are related to failure to prevent RCP seal LOCA are important.
- The CCF basic events are important individually, as well as a group with respect to plant CDF. This is expected for a plant with highly redundant safety systems.
- The CDF is 4.5E-06/RY if one safety train is out of service all year. This compares well with existing plants, even where periodic online maintenance is performed. Even if one accumulator and one safety train is out of service, the CDF is still below 1.0E-05/RY.
- If one safety train and another safety injection pump are simultaneously taken out of service, the CDF is 4.3E-05/RY. The four train safety system of the US-APWR enables to maintain CDF below a considerable value under conditions where two trains of a safety system are out of service.
- If no credit is taken for operator actions, the CDF is 1.6E-03/RY. If operator actions are assumed to succeed, the CDF is 3.8E-07/RY. CDF of US-APWR is sensitive to the reliability of operator actions.
- Reliability data of gas turbine generators does not have significant impact on CDF. If the reliability of generic gas turbine generators is applied the CDF increases 29%. However, the reliability of gas turbine generators that will be installed in

Loss of SFP cooling is also progress the phenomena and has sufficient time to recovery because of large coolant inventory in the pool. Furthermore, both events have not been risk significant in previous PRA studies. Therefore, both events are excluded as an initiating event for LPSD PRA.

Indications of temperature and water level are provided to detect unfavorable events that occur during shutdown. Indications are listed below.

- Indications of temperature

As for inaccurate hot leg temperature measurement ~~after loss of decay heat removal~~while RHR flow is maintained whereas RHR heat exchanger function has degraded, reactor coolant hot leg temperature and core exit temperature instruments are located in the flow path during RHR operation~~, so this parameter can be accurately indicated.~~. Accordingly the malfunction of RHR system can be identified.

- Indications of water

Three types of instruments are provided in US-APWR design to measure RCS water level for shutdown. The first one is narrow range water level instrument, the second one is mid range water level and the third one is wide range water level. Narrow range and mid range water level instruments that refer pressure at the bottom of cross over leg and pressurizer gas phase are provided to measure RCS water level during midloop operation.

Freeze plug may not be used for US-APWR because the isolation valves are installed considering maintenance and CCWS has been separated individual trains. Therefore, the freeze plug failure is excluded from the potential initiator.

The methods for data analysis and common cause analysis are the same as for Level 1 internal events PRA at power. The details of data analysis and CCF analysis are given in Subsection 19.1.4.1.1.

Mitigating functions during LPSD can be categorized into two groups: decay heat removal function and RCS inventory make up function. Systems that provide these functions are listed below. It is postulated that if these systems fail following an initiating event, bulk boiling and core damage will occur.

- Decay heat removal functions
  - RHR system

    If RHR pumps are available, the RCS is cooled by the RHR system through RHR suction line.

The ET for the LOOP is shown in Figure 19.1-20. The ET top events are described as follows:

- GT: Power supply by the gas turbine generators

  The automatic start up of the gas turbine generators is initiated with blackout sequence after the LOOP, and the gas-turbine generators supply electricity to components important for RHR operation.

- SP: Power supply by the gas turbines or AAC power

  If operation of the gas turbine generators fails, alternate power supply can supply the emergency power. The operation time of the alternate power supply is longer than 24 hours. If this function succeeds, it is assumed that sufficient time has elapsed for offsite power to be recovered.

- AC: Offsite power recovery

  The recovery of the LOOP within an allowable time is considered. The allowable time is assumed to be ~~1~~six hours. The probability that the LOOP duration does not exceed~~s~~ six hours is taken as 0.91 from Reference 19.1-41.

- PR: CCW pumps / essential service water pumps restart

  Following blackout sequence, CCW pumps and essential service water pumps automatically start (or re-start) up after power supply to the safety bus is re-established. If this function fails, the mitigation systems to require CCWS are unavailable.

The other top events are the same as described previously for a LOCA or LOCS.

The process of FT analysis is same as for the Level 1 internal events PRA at power (see Subsection 19.1.4.1.1).

In general, the success criteria for the LPSD PRA are the same as for the Level 1 internal events PRA at power (see Subsection 19.1.4.1.1).

The assumptions of success criteria specific to the LPSD PRA are as follows:

- For manual operation, one hour is conservatively assumed to be the allowable time until the exposure of reactor core from previous PRA studies and experience which mid-loop operation.

- When the RCS is under atmospheric pressure, it is assumed that the gravitational injection from SFP is effective. The gravitational injection from SFP is established by opening the injection flow path from SFP to RCS cold legs, and the water supply path from the RWSP to SFP. The validity of this function is determined by engineering judgment based on the previous PRA studies.

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 1 of ~~5~~6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Unique Equipments and their Duty to the US-APWR Design	SDVs	M	Motor-operated valves will be more reliable than air-operated valves for feed and bleed operation.	NA
	Motor-Operated Main Steam Relief Valves (MSRVs)	M	Hardware failure probabilities of MSRVs are not significant contributors to CDF.	NA
	Advanced Accumulators	M	The failure modes of the advanced accumulators are assumed similar to existing accumulators in the current PWR plants. Advanced accumulators are not significant contributors to CDF.	NA
	CSS/RHRS system	M	Appropriate conservative and simplified assumptions are made in the event tree / fault tree models.	NA
	Gas turbine generators	M	Sensitivity analysis of failure probability and failure rates was performed.	Sensitivity Analysis (Case 9)
	Digital I&C	M	Applied requirement or reliability for digital I&C.	Sensitivity Analysis (Case 10)
Initiating Event Analysis	Completeness of initiating events to the US-APWR design	C	Rare initiating events to the US-APWR design are assessed.	NA
	Statistical uncertainty of initiating event frequency	P	(Statistical uncertainty is considered)	Uncertainty Analysis

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 2 of ~~5~~6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Initiating Event Analysis	Event frequency of partial loss of CCW/ESW given on-line maintenance of CCW and ESW pumps ~~Identification of accident sequences~~	P~~M~~	If the unavailability of standby pumps due to maintenance is increased compared to operating plants, the assumed partial loss of CCW/ESW frequency can be underestimated. The contribution of partial loss of CCW to the at-power internal events CDF is approximately 1% and is relatively low compared to other initiating events. Hence, increase in the partial loss of CCW event frequency due to on-line maintenance has only small impact on the CDF. ~~Considered realistic accident sequences.~~	NA~~NA~~
Event Tree Analysis	Identification of accident sequences	M	Considered realistic accident sequences.	NA
Success Criteria Analysis	Boundary conditions Plant parameters	M	Appropriate simplified evaluations for the US-APWR have been performed.	NA
System Analysis	Plugging before events occurred is not modeled.	M	It would be hard to plug during normal operation in RCS and safety related systems.	NA
	System unavailability due to maintenance	M	US generic data is considered appropriate at design stage. ~~However, Sensitivity analyses were performed.~~ The following components have risk achievement worth higher than 2.0 and adequate control for on-line maintenance. - Turbine driven EFW pump - Essential service water pump Component unavailability will be adequately controlled by	Sensitivity Analysis (Case 01, Case 02, Case 03, Case 04)

			the maintenance rule.	

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 3 of ~~5~~6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
System Analysis	Class 1E electrical room HVAC are reliable and do not impact risk	M	Even if losses of HVAC occur, actuation signals of all trains will actually complete within a short time after the occurrence of an initiating event, and therefore, losses of HVAC may not affect the signal actuation. Even if HVAC function were to have impact on signals they will be limited to those that are required to operate hours after the initiating event. It is unlikely for losses of HVAC to actuate spurious signal and lead to functional failure of system so HVAC failure are likely to cause plant trip or malfunction of operating mitigation systems. To relax room heat up after losses of Class 1E electrical room HVAC, the operator will be open the room door and utilize available portable fans.	If Class 1E electrical room heat up were to occur and impact components in the most undesirable way, conditional core damage frequency will be 1.0 and the consequences will be severe.
Data Analysis	Applicability of failure modes to the US-APWR equipment design	M	Potentially valuable generic data sources were collected. All the failure modes of the US-APWR component types were considered.	NA

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 4 of ~~5~~6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Data Analysis	Failure probability and failure rates for diesel generators are applied to gas turbine generators.	M	Sensitivity analysis of failure probability and failure rates was performed.	Sensitivity Analysis (Case 09)
	Statistical uncertainty of failure rate	P	(Statistical uncertainty is considerable)	Uncertainty Analysis
	Failure probability of digital I&C software	M	Sensitivity analysis of failure probability was performed.	Sensitivity Analysis (Case 10)
	Failure probabilities of valves with long test intervals	M	Sensitivity analyses of failure probabilities were performed.	Sensitivity Analysis (Case 14)
Common Cause Failure Analysis	CCF parameters of emergency diesel generators are applied to gas turbine generators.	M	Sensitivity analysis of gas turbine generator CCF parameters was performed.	Sensitivity Analysis (Case 08)
	CCF of inter-systems is not included in the CCF model.	M	The environment, operation or service, design, and maintenance are different between inter-systems.	NA
	Statistical uncertainty of CCF probabilities.	P	(Statistical uncertainty is involved in data base)	Uncertainty Analysis

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 5 of ~~5~~6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Common Cause Failure Analysis ~~HRA~~	CCF in normally running components ~~Human error probability~~	M~~M~~	There are published data for the CCF of normally running pumps to continue to run. Based on expert judgment, the PRA applies a CCF parameter, lower than those reported in the NUREGs, for the normally running CCW and ESW pumps. Uncertainty associated with the CCF parameters for normally running pumps impact the initiating event frequency of total loss of CCW, which has large contribution to the CDF. The PRA considers CCF between normally running pumps and standby pumps, and applies a value of 0.1. This value is expected as a conservative estimation since the running pumps and the standby pumps are initially in an asymmetric configuration. When taking into consideration of the conservative CCF parameter set used for the CCF of pumps with asymmetric configuration, the evaluated initiating event frequency of total loss of CCW for the US-APWR design is not expected be higher than the current value even if uncertainty of CCF for normally running pumps may result in higher CCF parameters. ~~Sensitivity analyses of post initiating event operator action failure probabilities were performed to study the impact of human errors to CDF. Set all the HEPs to 1.0 or 0.0, and~~	NA~~Sensitivity~~

Table 19.1-38 Key Sources of Uncertainty and Key Assumptions (Level 1 PRA for Internal Events at Power)

(Sheet 6 of 6)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
HRA	Human error probability	M	Sensitivity analyses of post initiating event operator action failure probabilities were performed to study the impact of human errors to CDF. Set all the HEPs to 1.0 or 0.0, and change lower bound HEPs to mean value.	Sensitivity Analysis (Case 05, Case 06, Case 07)
	Statistical uncertainty of human error probability	P	(Statistical uncertainty is considered)	Uncertainty Analysis

Note - Uncertainty sources are categorized into three types, Parametric (P), Modeling (M) or Completeness(C).

Table 19.1-119 Key Insights and Assumptions (Sheet 1 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Design features and insights	
1. High Head Safety Injection System	
- The high head safety injection system consists of four independent and dedicated SI pump trains.	6.3.2.1.1
- The SI pump trains are automatically initiated by a SI signal, and supply borated water from the RWSP to the reactor vessel via direct vessel injection line.	6.3.2.1.1
2. Accumulator System	
- There are four accumulators, one supplying each reactor coolant cold leg.	6.3.2.1.2
- The accumulators incorporate internal passive flow dampers, which function to inject a large flow to refill the reactor vessel in the first stage of injection, and then reduce the flow as the accumulator water level drops. Thus the accumulators provide integrated function of low head injection system in the event of LOCA.	6.3.2.1.2
3. Chemical and Volume Control System	
- The charging pumps are arranged in parallel with common suction and discharge headers. Each pump provides full capability for normal makeup.	9.3.4.2.6
- Charging injection is provided by the CVCS. One CVCS charging pump is capable of maintaining normal RCS inventory with small system leak if the leakage rate is less than that from a break of a pipe 3/8 inch in inside diameter.	9.3.4.2.7.4
- Normally, one charging pump is operating and takes suction from the VCT, supplies charging flow to the RCS and seal water to the reactor coolant pumps.	9.3.4.2.6
- The pump can take suction from the VCT, the reactor makeup control system, the refueling water storage auxiliary tank and the spent fuel pit.	9.3.4.2.6
- During normal operation, the VCT water level is controlled by automatic makeup. In case the automatic makeup fails to actuate and the water level in the VCT decreases, low VCT water level is detected and actuates a low-low level signal that opens the stop valves in the refueling water storage auxiliary tank supply line, and closes No. 1 and No. 2 stop valves in the VCT outlet to provide emergency makeup.	9.3.4.5.4.1

Table 19.1-119 Key Insights and Assumptions (Sheet 2 of ~~23~~24)

Key Insights and Assumptions	Dispositions
4. Containment Spray System / Residual Heat Removal System	
- The containment spray system (CSS) and the residual heat removal system (RHRS) share major components which are containment spray/residual heat removal (CS/RHR) pumps and heat exchangers.	6.2.2
- The CSS/RHRS consists of four independent subsystems, each of which receives electrical power from one of four safety buses. Each subsystem includes one CS/RHR pump and one CS/RHR heat exchanger, which have functions in both the CS system and the RHRS.	6.2.2 5.4.7.2.1
- CS/RHRS provides multiple functions such as, (1) containment spray to decrease pressure and temperature in the CV, (2) alternate core cooling in case all safety injection systems fails during LOCA in conjunction with a fast depressurization of the RCS by using the EFW pumps to remove heat through the SGs and by manually opening the MSRVs especially in high RCS pressure sequences, (3) RHR operation for long term core cooling, (4) heat removal function for long term C/V cooling, (5) providing water to flood the reactor cavity and (6) fission product removal. During plant shutdown, RHRS provides function to remove decay heat from the RCS.	3.2.2 6.2.5 5.4.7.1
- The RHRS is designed and equipped with pressure relief valves to prevent RHRS over-pressurization and low temperature over-pressurization.	5.4.7.1
- Two motor operated valves in series on the RHR suction line with power lockout capability during normal power operation minimize the probability of RCS pressure entering the RHR system. Even if both these valves are opened during normal power operation, the RHR system is designed to discharge the RCS inventory to the in-containment RWSP. The RHRS is designed to prevent an interfacing system LOCA by having a design rating of 900 lb. The RHR 900 lb. design rated system can withstand the full RCS pressure.	6.3.1.4

Table 19.1-119 Key Insights and Assumptions (Sheet 3 of ~~23~~24)

Key Insights and Assumptions	Dispositions
- The RHR system is used to provide core cooling when the RCS must be partially drained to allow maintenance or inspection of the reactor head, SGs, or reactor coolant pump seals.	5.4.7.2.3.6
- During mid-loop operation, if the water level of RCS drops below the mid-loop level, low pressure letdown lines are isolated automatically. This interlock is useful to prevent loss of reactor coolant inventory.	5.4.7.2.3.6
5. Refueling Water Storage Pit	
- The RWSP is located on the lowest floor inside the containment. The coolant and associated debris from a pipe or component rupture (LOCA), and the containment spray drain into the RWSP through transfer pipes.	6.3.2.2.5 6.3.2.2.6
- Four independent sets of ECC/CS strainers located in the RWSP. The strainer design includes redundancy, a large surface area to account for potential debris blockage and maintain safety performance, corrosion resistance, and a strainer hole size to minimize downstream effects.	
6. Reactor Trip System	
- Reactor trip signal is provided by the reactor protection system (RPS), which consists of four redundant and independent trains. Four redundant measurements using sensors from the four separate trains are made for each variable used for reactor trip.	7.2.1
- One channel of sensor is allowed to be unlimitedly bypassed. One train of reactor trip breaker is allowed to be unlimitedly bypassed.	16.3.3
- Each train of the RPS consists of two separate digital controllers to achieve defense-in-depth through functional diversity. Each functionally diverse digital controller within a train can initiate a partial reactor trip signal.	7.2.1.9

Table 19.1-119 Key Insights and Assumptions (Sheet 4 of ~~23~~24)

Key Insights and Assumptions	Dispositions
7. Engineered Safety Function System	
- There are four redundant engineered safety function (ESF) trains. Within each train, ESF actuation system (ESFAS) and signal logic system (SLS) controllers are redundant.	7.3.1.8
- All ESF systems are automatically initiated from signals that originate in the RPS. Manual actuation of ESF systems is carried out through a diverse signal path that bypasses the RPS.	7.3.1.9
8. Diverse Actuation System	
- The diverse actuation system (DAS) provides monitoring, control and actuation of safety and non-safety systems required to cope with abnormal plant conditions concurrent with a CCF that disables all functions of the PSMS and PCMS.	7.8
- DAS design consists of conventional equipment that is totally diverse and independent from the MELTAC platform of the PSMS and PCMS. Therefore, a software CCF in the digital safety and non-safety systems, would not affect the DAS.	7.8.2.2
- DAS hardware for anticipated transient without scram (ATWS) mitigation functions – Reactor trip, turbine trip, and EFW actuation, is diverse from the reactor trip hardware used in the PSMS. The reactor trip is actuated by tripping the non-safety CRDM motor-generator set.	7.8.1.2.1 7.8.2.2
- The DAS is electrically and physically isolated from the PSMS.	7.8.2.3
9. Emergency Feed Water System	
- EFWS consists of two motor-driven pumps and two steam turbine-driven pumps with two emergency feedwater pits.	10.4.9.2
- Each EFW pump discharge line connects with a tie line with a motor-operated isolation valve. During normal plant operation (at non-OLM), the discharge tie line isolation valves of each EFW pump discharge tie line are in the closed position to provide separation of four trains. During OLM, the tie line isolation valves of each EFW pump discharge tie line are kept in the open position.	10.4.9.2
- Upon detection of a water level increase of the SG, the EFW isolation valves and EFW control valves are automatically closed.	10.4.9.2

Table 19.1-119 Key Insights and Assumptions (Sheet 5 of ~~23~~24)

Key Insights and Assumptions	Dispositions
- The motor-operated EFW isolation valves and EFW control valves are provided in each EFW pump discharge line to close automatically to terminate the flow to the affected SG.	10.4.9.2
- The common suction line from each EFW pit is connected by a tie line with two normally closed manual valves. When the two EFW pumps taking suction from the same pit are not available (OLM of one EFW pump and the single failure of other EFW pump), the tie line connections to EFW pits need to be established.	10.4.9.2
- The demineralized water storage tank provides a backup source for EFWS. The manual valves from the demineralized water storage tank to the EFW pumps are normally closed.	
10. Reactor Coolant System High Point Vents	
- Safety depressurization valves (SDVs) are provided at top head of the pressurizer in order to cool the reactor core by feed and bleed operation when loss of heat removal from steam generator occurs.	5.4.12.2
- RCS depressurization system dedicated for severe accident is provided to prevent high pressure melt ejection. The location of release point from the valve is in containment dome area.	5.4.12.2
11. Main Steam Supply System MSIVs are installed in each of the main steam lines to (1) limit uncontrolled steam release from one steam generator in the event of a steam line break, and to (2) isolate the faulted SG in the event of SGTR.	10.3
12. Component Cooling Water System	
- The CCWS consists of two independent subsystems. One subsystem consists of trains A & B, and the other subsystem consists of trains C & D, for a total of four trains.	9.2.2.2
- The CCWS is designed to withstand leakage in one train without loss of the system's safety function.	9.2.2.1.1
- Two motor operated valves are located at the CCW outlet of the RCP thermal barrier Hx and close automatically upon a high flow rate signal at the outlet of this line in the event of in-leakage from the RCS through the thermal barrier Hx, and prevents this in-leakage from further contaminating the CCWS.	9.2.2.2.1.5

Table 19.1-119 Key Insights and Assumptions (Sheet 6 of ~~23~~24)

Key Insights and Assumptions	Dispositions
- CCWS supplies cooling water to containment fan cooler unites to when performing alternate CV cooling during severe accident conditions. The cooling water system is switched from the non-essential chilled water system to CCW system to supply the cooling water to the containment fan cooler units. - In the case of loss of CCW, a non-essential chilled water system or a fire suppression system is able to connect to the CCWS in order to cool the charging pump and maintain RCP seal water injection.	9.4.6.2.1
13. Essential Service water system The ESWS is arranged into four independent trains (A, B, C, and D). Each train consists of one ESWP, two 100% strainers in the pump discharge line, one 100% strainer upstream of the CCW HX, one CCW HX, one essential chiller unit, and associated piping, valves, instrumentation and controls.	9.2.1.2.1 COL19.2(3) COL19.2(4)
14. Onsite Electric Power System - The onsite Class 1E electric power systems comprise four independent and redundant trains, each with its own power supply, buses, transformers, and associated controls.	8.3.1.1.2.1
- One independent Class 1E GTG is provided for each Class 1E train.	8.3.1.1.2.1
- Non-Class 1E 6.9kV permanent buses P1 and P2 are also connected to the non-Class 1E A-AAC GTG and B-AAC GTG, respectively. The loads which are not safety-related but require operation during LOOP are connected to these buses.	8.3.1.1.1
- In the event of SBO, power to one Class 1E 6.9kV bus can be restored manually from the AAC GTG.	8.3.1.1.2.4
- Common cause failure between class 1E GTG and non-class 1E GTG supply is minimized by design characteristics. Different rating GTGs with diverse starting system, independent and separate auxiliary and support systems are provided to minimize common cause failure.	8.4.1.3
- The non-safety GTG can be started manually when connecting to the class 1E bus in the event of SBO.	8.4.1.3
- Power to the shutdown buses can be restored from the AAC sources within 60 minutes	8.4.1.3

Table 19.1-119 Key Insights and Assumptions (Sheet 7 of ~~23~~24)

Key Insights and Assumptions	Dispositions
- Power to the shutdown buses can be restored from the AAC sources within 60 minutes	8.4.1.3
- The GTG does not need cooling water system. Cooling of GTG is achieved by air ventilation system	9.5.5
- GTG combustion air intake and exhaust system for each of the four GTGs supply combustion air of reliable quality to the gas turbine and exhausts combustion products from the gas turbine to the atmosphere. The air intake also provides ventilation/cooling air to the GTG assembly.	9.5.8
15. RCP seal	
- RCP seal can keep its integrity for at least one hour without water cooling.	8.4.2.1.2
- If loss of seal injection should occur, CCW continues to provide flow to the thermal barrier heat exchanger; which cools the reactor coolant. The pump is able to maintain safe operating temperatures and operate safely long enough for safe shutdown of the pump.	5.4.1.3.3
- If loss of CCW should occur, seal injection flow continues to be provided to the RCP. The pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling.	5.4.1.3.4
16. Containment System	
- The containment prevents or limits the release of fission products to the environment.	3.1.2.7 3.8.1
- Hydrogen control system that consists of igniters is provided to limit the combustible gas concentration. The igniters start with the ECCS actuation signal and are powered by two non-class 1E buses with non-class 1E GTGs.	6.2.5.2
- Alternate containment cooling system using the containment fan cooler units is provided to prevent containment over pressure even in case of containment spray system failure.	9.4.6.2.1
- Reactor cavity flooding system by firewater injection is provided to enhance heat removal from molten core ejected into the reactor cavity. This system is available as a countermeasure against severe accidents even in case of fire.	9.5.1.2.2

Table 19.1-119 Key Insights and Assumptions (Sheet 8 of ~~23~~24)

Key Insights and Assumptions	Dispositions
- The FSS is also utilized to promote condensation of steam. The FSS is lined up to the containment spray header when the CSS is not functional, and provides water droplet from top of containment. This will temporarily depressurize containment.	9.5.1.2.2
- A set of drain lines from SG compartment to the reactor cavity is provided in order to achieve reactor cavity flooding. Spray water which flows into the SG compartment drains to the cavity and cools down the molten core after reactor vessel breach.	3.4.1.5.1
- Reactor cavity has a core debris trap area to prevent entrainment of the molten core to the upper part of the containment.	3.8.1 19.2.3.3.4
- Reactor cavity is designed to ensure thinly spreading debris by providing sufficient floor area and appropriate depth.	3.8.1 19.2.3.3.3
- Reactor cavity floor concrete is provided to protect against challenge to liner plate melt through.	3.8.1 19.2.3.3.3
- Main penetrations through containment vessel are isolated automatically with the containment penetration signal even in case of SBO.	6.2.4
17. Main equipments and instrumentations used for severe accident mitigation are designed to perform their function in the environmental conditions such as containment overpressure and temperature rise following hydrogen combustion.	19.2.3.3.7
18. Instrumentations for detecting core damage with high reliability are provided.	5.3.3.1
19. Risk significant SSCs are identified for the RAP.	17.4
20. Instrumentation piping are installed at upside of the RV. No penetrations through the RV are located below the top of the reactor core. This minimizes the potential for a loss of coolant accident by leakage from the reactor vessel, allowing the reactor core to be uncovered.	5.3.3.1

Table 19.1-119 Key Insights and Assumptions (Sheet 9 of 24)

Key Insights and Assumptions	Dispositions
21. Check valves in accumulator, high head injection system, and other systems are in diverse configuration because: - The accumulator does not have any pumps to drive upon a failed closed check valve but other systems have pumps so the forces acting on the valves to open them (even if the valves are similar) are different - The duty cycles in the systems are different. They are cycled at different dimes when the systems are tested. - Maintenance practices including testing may also be different. Common cause failure between the check valves in accumulator and high head injection system is therefore unlikely to occur.	PRA Assumption
22. The availability and reliability of all trains of safety related systems will be controlled by the maintenance and configuration risk management programs. Availability goals will be set for each train of all safety related systems and their availability will be tracked and compared to these goals.	COL 17.6(1)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~9~~ 10 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Operator actions (At Power)	
1. Operator actions modeled in the PRA are based on symptom oriented procedures. Risk significant operator actions identified in the PRA will be addressed in plant operating procedures including AOP, EOP, etc.	19.2.5 COL 19.3(6) COL 13.5(5) COL 13.5(6) COL 13.5(7) COL 19.3(6)
2. Maintenance procedures indicate to check valve positions from the main control room after outages or testing. Valves that have been aligned in the wrong position will be detected and fixed to the correct position within a short period of time.	19.2.5 COL 19.3(6) COL 13.5(7)
3. In the operational VDU of US-APWR, the layout of controllers & monitoring alignment in each window are different and this feature would make the operator perceive them as different locations.	18.4 19.2.5 COL 19.3(6) COL 13.5(5)
4. In the case of loss of CCW, operators connect a non-essential chilled water system or a fire suppression system to the CCWS in order to cool the charging pump and maintain RCP seal water injection. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.1.4 COL 13.5(5)
5. When station blackout occurs, operators connect the alternative ac power to class 1E bus in order to recovery emergency ac power. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.6 19.2.5 COL 19.3(6) COL 13.5(6)
6. If emergency feed water pumps cannot feed water to two intact SGs, operators will attempt to open the cross tie-line of EFW pump discharge line in order to feed water to two more than SGs by one pump.	19.2.5 COL 19.3(6) COL 13.5(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~10~~ 11 of ~~23~~24)

Key Insights and Assumptions	Dispositions
7. The CS/RHR System has the function to inject the water from RWSP into the cold leg piping by switching over the CS/RHR pump lines to the cold leg piping if all safety injection systems failed (Alternate core cooling operation). In high RCS pressure sequences, a fast depressurization of the RCS by using the EFW pumps to remove heat through the SGs and by manually opening the MSRVs allows alternate core cooling injection using the CS/RHR pumps. Alternate core cooling operation may be required under conditions where containment protection signal is valid. In such cases, alternate core cooling operation is prioritized over containment spray, because prevention of core damage would have higher priority than prevention of containment vessel rupture.	19.2.5 COL 19.3(6) COL 13.5(6)
8. When any two EFW pumps that commonly utilize at EFW pit have failed, operators supply water to operating EFW pumps from alternate EFW pit or demineralized water storage pit in order to ensure the water source.	19.2.5 COL 19.3(6) COL 13.5(6)
9. In the case of failure to isolate failed SG, but success to sufficiently depressurize RCS by secondary side cooling and Safety depressurization valve in SGTR event, operators do RCS pressure control in order to prepare to early RHR cooling in order to ensure long term heat removal. (RCS pressure control means stopping SI safety injection and starting charging pump. RCS pressure under SI injection remains higher for connecting RHR system. Charging pump is back up for failure of RHR cooling after stopping SI injection.)	19.2.5 COL 19.3(6) COL 13.5(6)
10. In the case of above, if operators fail to move RHR cooling after SI injection control, operators start to bleed and feed operation. Operators open safety depressurization valve and start the safety injection pump in order to ensure long term heat removal.	19.2.5 COL 19.3(6) COL 13.5(6)
11. When the main steam isolation valve fail to close in SGTR event, with status signal of this valve, operators try to close this valve in order to stop leakage of RCS coolant from the failed SG.	19.2.5 COL 19.3(6) COL 13.5(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~11~~ 12 of ~~23~~24)

Key Insights and Assumptions	Dispositions
12. When the main steam isolation valve fail to close in SGTR event, with SG pressure indication after above operation, operators close turbine bypass stop valves in order to stop leakage of RCS coolant from the failed SG.	19.2.5 COL 19.3(6) COL 13.5(6)
13. In the case of loss of failed SG isolation function in SGTR event, with SG pressure indication after above operation, operators open main steam depressurization valve of intact SG loop in order to promote SG heat removal and to depressurize RCS and move to cool down and recirculation operation.	19.2.5 COL 19.3(6) COL 13.5(6)
14. In the case of loss of secondary side cooling function by emergency feedwater system in transient events including turbine trip, load loss event etc., with emergency feedwater pump flow rate, operators start to recover main feedwater system in order to maintain secondary side cooling.	19.2.5 COL 19.3(6) COL 13.5(6)
15. In the case of loss of SI injection function entirely in LOCA event, with SI flow rate and RCS temperature indication, operators provide secondary side cooling to reduce RCS pressure and temperature by opening the main steam depressurization valves manually and supplying water from the emergency feedwater system in order to enable low pressure injection with containment spray system / residual heat removal system.	19.2.5 COL 19.3(6) COL 13.5(6)
16. In the case of loss of containment spray system function, alternate containment cooling operation is implemented utilizing CV natural recirculation in order to remove heat from CV. This preparation contains CCW pressurization with N2 gas, disconnection heat load of non-safety chiller and CRDM etc. and connection to containment fan cooler units. This operation is implemented when the containment pressure reaches the design pressure.	19.2.5 COL 19.3(6) COL 13.5(6)
17. In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	19.2.5 COL 19.3(6) COL 13.5(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~12~~ 13 of ~~23~~24)

Key Insights and Assumptions	Dispositions
18. When the CV isolation signal fail to automatically actuate, with CV pressure abnormally high signal, operators manually actuate the CV isolation signal in order to remove heat from the containment vessel.	19.2.5 COL 19.3(6) COL 13.5(6)
19. When the CCW header tie-line isolation valves fail to automatically close with specific signals which contain SI signal plus UV signal, P signal, and surge tank level low signal, operators manually close these valves in order to separate CCW header.	19.2.5 COL 19.3(6) COL 13.5(5)
20. RCS is depressurized through operating the depressurization valve after onset of core damage and before reactor vessel breach. This operation prevents events due to high pressure melt ejection.	19.2.5 COL 19.3(6) COL 13.5(5)
21. Operation of firewater injection to reactor cavity is implemented to flood reactor cavity in case of containment spray system failure, after onset of core damage and before reactor vessel breach.	19.2.5 COL 19.3(6) COL 19.3(6)
22. When the CCW header tie-line isolation valves fail to automatically close with specific signals which contain SI signal plus UV signal, P signal, and surge tank level low signal, operators manually close these valves in order to separate CCW header.	19.2.5 COL 19.3(6) COL 19.3(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~13~~ 14 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Operator actions (LPSD)	19.1.6
1. When the RCS is under atmospheric pressure, gravity injection from SFP is effective. Operator will perform the gravity injection by opening the injection flow path from SFP to RCS cold legs, and supplying water from RWSP to SFP.	19.2.5 COL 19.3(6) COL 13.5(7) 5.4.7.2.3.6
2. When station blackout occurs, operators connect the alternative ac power with alternate gas turbines to class 1E bus in order to recover emergency ac power. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.8 18.9 19.2.5 COL 19.3(6) COL 13.5(7)
3. In the case of loss of CCW/ESW, operators connect the fire suppression system to the CCWS and start the fire suppression pump in order to cool the charging pump and maintain injection to RCS. . This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.8 18.9 19.2.5 COL 19.3(6) COL 13.5(7)
4. In the case of loss of decay heat removal functions by RHRS and SGs operators start the charging pump in order to recover water level in the RCS. If water level in the RWSAT, which is the water source of charging pumps, indicates low level the operator will supply RWSP water to the RWSAT by the refueling water recirculation pump. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.8 18.9 19.2.5 COL 19.3(6) COL 13.5(7)
5. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves and stop leakage of RCS coolant from RHRS where LOCA occurs.	19.2.5 COL 19.3(6) COL 13.5(7)
6. In case the RCS water level decreases during mid-loop operation and the failure of automatic isolation valve occurs, operator will perform the manual isolation of low-pressure letdown line.	19.2.5 COL 19.3(6) COL 13.5(7)
7. When over-draining occurs and the automatic isolation valve fails, with RCS water level – low, operators close the valve on the letdown line in order to stop draining.	19.2.5 COL 19.3(6) COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~14~~ 15 of ~~23~~24)

Key Insights and Assumptions	Dispositions
8. In the case of loss of decay heat removal functions by RHRS and SGs, operators start the safety injection pump in order to maintain RCS water level. This operator action is risk important. Activities to minimizes the likelihood of human error in the human factors engineering is important in developing procedures, training and other human reliability related programs.	18.8 18.9 19.2.5 COL 19.3(6) COL 13.5(7)
9. In the case of failure of running RHRS, with RHR flow rate – low, operators open the valves on the standby RHR suction line and discharge line and start the standby RHR pump in order to maintain RHR operating.	19.2.5 COL 19.3(6) COL 13.5(7)
10. In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	19.2.5 COL 19.3(6) COL 13.5(7)
11. In the case of failure of running CCWS, with CCW flow rate – low, operators start the standby CCW pump in order to maintain CCWS operating.	19.2.5 COL 19.3(6) COL 13.5(7)
12. In the case of failure of running ESWS, with CCW flow rate – low, operators start the standby ESW pump in order to maintain ESWS operating.	19.2.5 COL 19.3(6) COL 13.5(5)
13. When ESW strainer plugs up, with ESW pump pressure – normal, ESW flow rate – low and differential pressure – significant, operators switch from plugged strainer to standby strainer in order to maintain ESWS operating.	19.2.5 COL 19.3(6) COL 13.5(5)
14. In the case of loss of decay heat removal functions from RHR, with RCS temperature – high or RCS water level – low, operators feed water to SGs by motor-driven EFW pump and open safety depressurization valve in order to remove decay heat from RCS.	19.2.5 COL 19.3(6) COL 13.5(7)
15. In the case of failure of feed or steam line associated with available motor-driven EFW pump during secondary side cooling, operators open the EFW tie-line valves in order to feed water to multiple SGs.	19.2.5 COL 19.3(6) COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~15~~ 16 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Operator actions (Severe Accidents)	
1. Operators manually initiate severe accident mitigation systems in accordance with the instructions from the technical support centre staff.	COL 19.3(6)
2. In the loss of support system sequences, operators will attempt to recover CCW/ESW or ac power while suppressing containment overpressure with firewater injection into spray header.	COL 19.3(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~16~~ 17 of ~~23~~24)

Key Insights and Assumptions	Dispositions
LPSD assumptions	
1. Freeze plug may not be used for US-APWR because the isolation valves are installed considering maintenance and CCWS has been separated individual trains. Therefore, the freeze plug failure is excluded from the potential initiator.	
2. Hydrogen peroxide addition is adopted instead of aeration because it decreases the duration of the mid-loop operation. As a result, the mid-loop operation is needed only to drain the SG primary side water while being able to maintain a high RCS water level for most of the oxidation operation.	5.4.7.2.3.6
3. Installation of a redundant water narrow level instrument enhances reliability of the mid-loop operation.	5.4.7.2.3.6
4. For manual operation, one hour is conservatively assumed to be the allowable time until the exposure of reactor core. This allowable time is determined from previous PRA studies and experience which mid-loop operation.	19.2.5 COL 19.3(6) COL 13.5(6)
5. When the RCS is mid-loop operation, it is assumed that the reflux cooling with the SGs is effective.	19.2.5 COL 19.3(6) COL 13.5(6)
6. Various equipments will be possible temporary in the containment during LPSD operation for maintenance. However, there are few possibilities that these materials fall into the sump because the debris interceptor is installed on the sump of US-APWR. Therefore, potential plugging of the suction strainers due to debris is excluded from the PRA modeling.	6.2.2
7. For the US-APWR, low-pressure letdown line isolation valves are installed. One normally closed air-operated valve is installed in each of two low-pressure letdown lines that are connected to two of four RHR trains. During normal plant cooldown operation, these valves are opened to divert part of the normal RCS flow to the CVCS for purification and the RCS inventory control. These valves are automatically closed and the CVCS is isolated from the RHRS by the RCS loop low-level signal to prevent loss of RCS inventory at mid-loop operation during plant shutdown. There are no features that automate the response to loss of RHR.	19.2.5 COL 19.3(6)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~17~~ 18 of ~~23~~24)

Key Insights and Assumptions	Dispositions
8. The time when loss of RHR occur were set to be 12 hours after plant trip, which is the time POS 4 (mid-loop operation) is entered after plant trip, since this condition gives the most severe condition for mid-loop operation from a decay heat perspective. The pressurizer spray-line vent line with 3/4 inch diameter is assumed to be open at the initial condition. One hour after loss of RHR function, the operator is assumed to perform the following actions: - Close pressurizer spray line vent, - Start emergency feed water (EFW) pump, and - Open main steam depressurization valve.	19.2.5 COL 19.3(6) COL 13.5(7)
9. Nitrogen will not be injected in the SG tubes to speed draining in the US-APWR design. The SG tubes will be filled with air during midloop operation.	19.2.5 COL 19.3(6) COL 13.5(7)
10. Operator actions assumed in the PRA will be considered in the shutdown response guideline, which will be developed satisfying NUMRAC 91-06 and following other recent guidelines such as INPO 06-008.	19.2.5 COL 19.3(6) COL 13.5(7)
11. Cleanliness, housekeeping and foreign material exclusion areas are administrative controls and programs to be developed by any applicant referencing the certified US-APWR design for construction and operation	6.2 Table 6.2.2-2 19.2.5 COL 19.3(6) COL 13.5(7)
12. The reactivity insertion event due to boron dilution has been judged to be insignificant to risk because of the following factors: - Strict administrative controls are in place to prevent boron dilution - Boron dilution events are highly recoverable - The consequences of re-criticality are minor unless they continue for very long.	15.4.6.2 19.2.5 COL 19.3(6) COL 13.5(7)
13. Administrative controls ensure the RCS water level, temperature and pressure indication are available during shutdown.	19.2.5 COL 19.3(6) COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~18~~ 19 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Seismic insights	
1. Table 19.1-54 provides the list of HCLPFs for US-APWR SSCs. This table demonstrates that the SSC HCLPF values are greater than 1.67 times the design basis SSE although the assessment performed by conservative generic data from EPRI URD. This insight will be certified by the following assessment.	19.1.5.1 Table 19.1-54
- Perform seismic margin assessment using US-APWR plant specific in-structure response and stress analyses. - Conduct plant walkdown to certify the SSCs retain seismic margin under as-built conditions prior to fuel loading.	3.7

Table 19.1-119 Key Insights and Assumptions (Sheet ~~19~~ 20 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Seismic assumptions	
1. Failure of the RHRS isolation valves is not included in the analysis, because the pipe sections are assumed to fail before the valves fail and these valves are normally closed. Also, the US-APWR design has provided further protection against interfacing system LOCA by upgrading design pressure. Therefore, interfacing system LOCA is not modeled.	5.4.7.1
2. Failure of buildings that are not seismic Category I (i.e., turbine building, auxiliary building and access building) does not impact SSCs designed to be seismic Category I. Seismic spatial interactions between SSCs design to be seismic Category I and any other buildings will be avoided by proper equipment layout and design. The following seismic Category I buildings and structures are identified as buildings and structures that involve safety-related SSCs to prevent core damage. - Reactor building - Safety power source buildings - Essential service water intake structure - Essential service water pipe tunnel	3.2.1
3. Relay chatter does not occur or does not affect safety functions during and after seismic event.	3.10 Table 19.1-51

Table 19.1-119 Key Insights and Assumptions (Sheet ~~20~~ 21 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Internal fire insights	
1. Fire protection seals are provided for walls, floors, and ceilings, which compose the fire area boundaries divided by four train areas.	9.5.1
2. Turbine building electric rooms are segregated into two groups by qualified fire barriers. This feature is possible to prevent loss of offsite power by a turbine building fire.	9.5.1
3. In case of LOCA or loss of RHR caused by over drain or failure of water level maintain by a fire during LPSD, the flow pathway could be isolated by automatic closing of the low pressure letdown line isolation valve.	5.4.7.2.2.3

Table 19.1-119 Key Insights and Assumptions (Sheet ~~21~~ 22 of ~~23~~24)

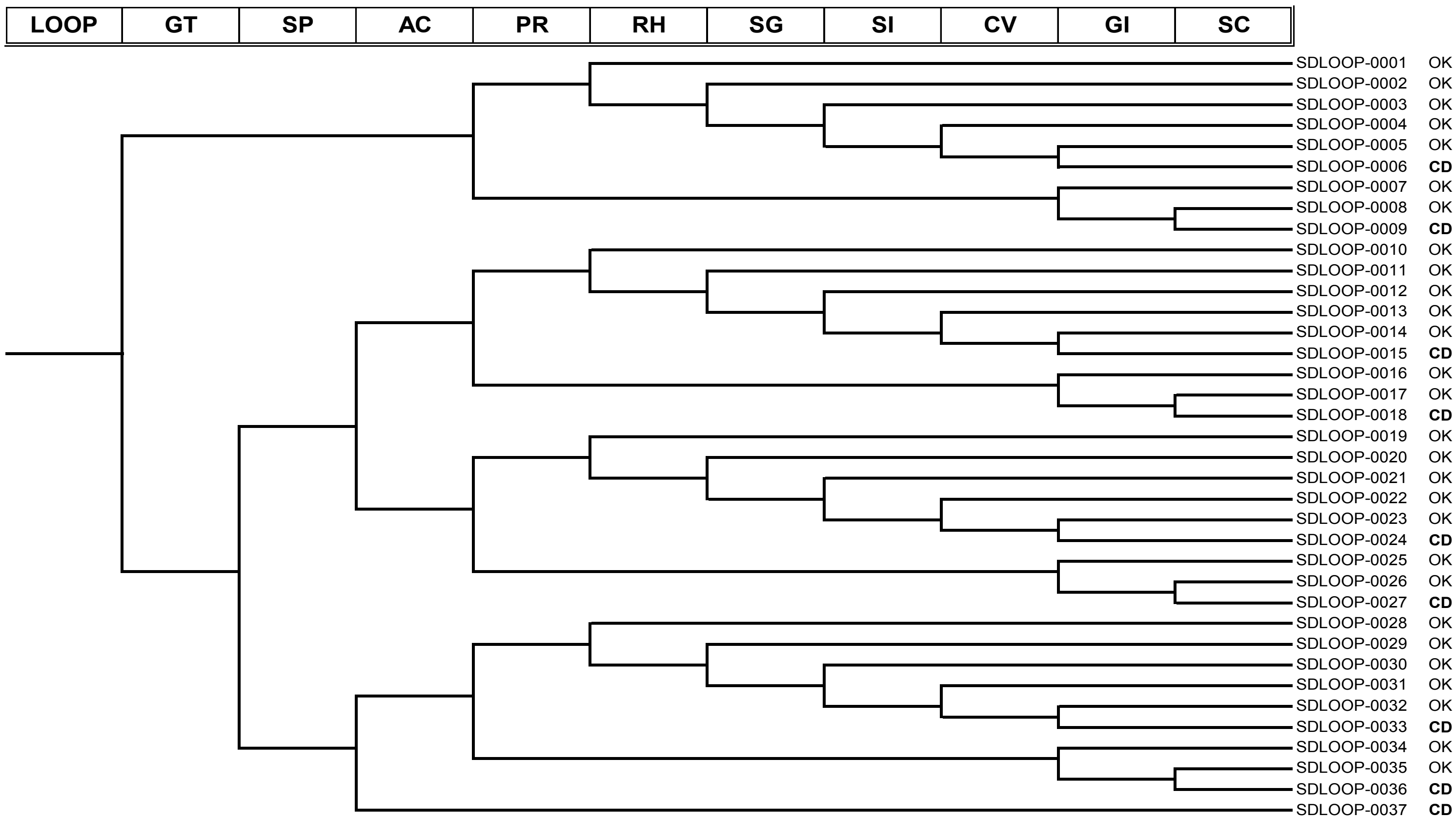
Key Insights and Assumptions	Dispositions
Internal fire assumption	
1. All fire doors serving as fire barriers between redundant safety train fire compartments are normally closed.	9.5.1.2.1 COL 9.5(1)
2. For transient combustibles, “three Airline trash bags” has been assumed in each fire compartment.	9.5.1.2.1 COL 9.5(1)
3. Transient combustibles with total heat release capacity of 93,000 Btu (obtained from NUREG/CR-6850, “AppendixG-table-7LBL-Von Volkinburg, Rubbish Bag” Test results) is assumed for Fire ignition source within Containment Vessel.	9.5.1.2.1 COL 9.5(1)
4. The Heat Release Rate of various items as specified in Chapter-11 of NUREG/CR-6850 is used.	9.5.1.2.1 COL 9.5(1)
5. Damage temperature of thermoplastic cables as shown in Appendix-H of NUREG/CR-6850 is used as the target damage temperature.	9.5.1.2.1 COL 9.5(1)
6. Operators are well trained in responding to fire event.	9.5.1.2.1 COL 9.5(1)
7. One of RCS letdown isolation valves and one of RCS vent line isolation valves are locked close by administrative controls	COL 13.5(1) COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~21~~ 23 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Internal flood insights	
1. East side and west side of reactor building are physically separated by flood propagation preventive equipment and the connections are kept closed and locked.	3.4.1.3
2. Areas between the reactor building and the turbine building are physically separated by flood propagation prevention equipment.	3.4.1.3
3. The flood barriers that separate the reactor building between east side and west side are important to safety for the operation of the facility. These doors should be monitored and controlled during plant operation and maintenance.	3.4.1.3 19.2.5 COL 19.3(6) COL 19.5(1) COL 13.5(7) (RAI 19-207)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~23~~ 24 of ~~23~~24)

Key Insights and Assumptions	Dispositions
Internal flood assumption	
1. Drain systems are designed to compensate with flood having flow rate below 100 gpm. Flood with flow rate below 100 gpm will not propagate to other areas due to the drain systems.	3.4.1.3
2. R/B is separated in two divisions (i.e. east area and west area). This design is prevents loss of all safety systems though postulated major floods that leak water over the capacities of flood mitigation systems. East side and west side of reactor building (R/B) are physically separated by flood propagation preventive equipment such as water tight doors. Therefore, flood propagation between east side and west side in the reactor building is not considered.	3.4.1.3 19.2.5 COL 19.3(6) COL 13.5(1) COL 13.5(7)
3. Watertight doors are provided for the boundaries between R/B and A/B in the bottom floor and between R/B and T/B in flood area 1F. This measure prevents flood propagation from non-safety building to R/B.	3.4.1.3
4. Flooding of ESW system can to be isolated within 15 minutes.	
5. Four trains of ESW system have physical separation and flooding in one train does not propagate to other trains.	9.2.1.2.1 COL 9.2(3) COL 9.2(4)
6. The components that are environmentally qualified are considered impregnable to spraying or submerge effects. Also component failure by flooding will not result in the loss of an electrical bus.	
7. Penetrations within the boundaries between the restricted area and non-restricted area are sealed and doors or dikes are provided for openings. Therefore, flood propagation, except for major flood events is not considered.	3.4.1.3
8. The administrative controlled flood barriers that separated the reactor building between the east side and the west side are effective. The other water tight doors may be opened during maintenance.	19.2.5 COL 19.3(6) (RAI 19-50) COL 13.5(1) COL 13.5(7)
9. The outage states of mitigation systems are important for LPSD risk. From the insight of flooding risk, one train of mitigation system on each side in R/B should be available. So that assumed the available safety injection pumps trains A and C are available during POS 8-1. B and D pumps are assumed out of service.	19.2.5 COL 19.3(6) COL 13.5(7)



Event	Description
LOOP	: Loss of offsite power
GT	: Power supply by class 1E GTG
SP	: Power supply by AAC
AC	: Offsite power recovery
PR	: CCWS restart
RH	: Heat removal by standby CS/RHR pumps
SG	: Heat removal via SGs
SI	: Injection by HHIS
CV	: Injection by CVCS
GI	: Gravitational injection
SC	: Injection by CVCS using alternate component cooling

Figure 19.1- 20 Loss of Offsite Power Event Tree

spray header and provides water as spray droplet. This operation temporarily depressurizes containment however the fire protection water supply system does not contain a heat exchanger, and thus has no ability to remove heat from containment to terminate the containment pressurization.

- Accident management of prevention of early containment failure is through prevention of containment bypass, HPME and hydrogen detonation. RCS depressurization is in order for prevention of HPME and temperature-induced SGTR. When core damage is detected, severe accident dedicated depressurization valve is opened and if necessary safety depressurization valve is opened. In case water supply to SG is available, main steam depressurization valve is opened to enhance primary system cooling and depressurization if needed. Water supply to SG is recovered or controlled to avoid FP release due to temperature induced SGTR through secondary system, also to depressurize RCS. Main feedwater system or emergency feedwater system are employed for this function and operation is required when SG water level decreases below a criterion if available. Combustible gas control is in order to prevent containment failure especially due to hydrogen detonation. Although the combustible gas control is automatically achieved by hydrogen ignition system, in case CSS fails and containment vessel atmosphere is kept inerted for certain duration, CSS recovery or operation of alternate containment cooling may lead containment vessel atmosphere to combustible condition under high hydrogen concentration. In such case containment depressurization is suspended at a relatively high containment pressure. It is widely known that the low inert limit of steam concentration is approximately 55% and the low flammability limit of hydrogen concentration is approximately 4%. Hydrogen impact when depressurizing containment is evaluated and a material, such as a map of hydrogen concentration vs. containment pressure to show if hydrogen burn is safe or potential danger, is prepared to support the containment depressurization operation. MCR alarm for hydrogen concentration is also provided through the containment hydrogen monitoring system when the hydrogen concentration reaches 4% and 8%. The control room operators are required to carefully monitor the condition of containment.

(During LPSD operations)

It is likely that containment is not isolated during LPSD operations in order for various maintenance activities. The accident management functions to maintain containment integrity during LPSD include firstly recovery of containment isolation from the environment, and secondary heat removal from the isolated containment. However, the ability to close the containment and to recover heat removal without ac power is minimal and may not be possible. It is evaluated for the LPSD PRA that the losses of offsite power contribute approximately 30% of shutdown risk in total. As a result any period in which the RCS level is low should be planned to be undertaken with maximum confidence in offsite and onsite power reliability. Maintenance activities in the switchyard are minimal or precluded by risk management during mid-loop for example. It may also be preferable to limit undertaking the maintenance activities which require opening the equipment hatch during the inventory is low in the reactor. This limitation will fundamentally eliminate the necessary operator actions for containment closure during mid-loop, and will significantly contribute for LPSD risk reduction.

- According to the identification of some symptoms, such as loss of decay heat removal capability and onset of boiling in core, operators are required to take actions of containment isolation.
- For decay heat removal, accident management functions are fundamentally same with the ones for operations at power, i.e. reactor cavity flooding, activation of CSS or alternate containment cooling by natural circulation, or otherwise firewater injection to spray header.

(4) To minimize offsite release

(During operations at power)

Key function of accident management to minimize offsite release during operations at power is fission products removal from containment vessel atmosphere. CSS and fire protection water supply system are utilized to reduce the amount of airborne FP in the containment atmosphere. Countermeasures and operator actions for each function are described below.

- Operator recovers CSS even after containment vessel failure if available.
- If CSS is not available, operator recovers fire protection water supply system connected to the spray header if available.

(During LPSD operations)

It is likely that containment is not isolated during LPSD operations in order for various maintenance activities. The accident management functions to minimize offsite release during LPSD include firstly recovery of containment isolation from the environment, and secondary deposition of fission products within the containment. However, the ability to close the containment without ac power is minimal and may not be possible. It is evaluated for the LPSD PRA that the losses of offsite power contribute approximately 30% of shutdown risk in total. As a result any period in which the RCS level is low should be planned to be undertaken with maximum confidence in offsite and onsite power reliability. Maintenance activities in the switchyard are minimal or precluded by risk management during mid-loop for example. It may also be preferable to limit undertaking the maintenance activities which require opening the equipment hatch during the inventory is low in the reactor. This limitation will fundamentally eliminate the necessary operator actions for containment closure during mid-loop, and will significantly contribute for LPSD risk reduction.

- According to the identification of some symptoms, such as loss of decay heat removal capability and onset of boiling in core, operators are required to take actions of containment isolation.

Tier 1

US-APWR DCD Revision 2 Tier 1 Tracking Report Revision 0 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.7-202	Table 2.7.5.2-3, ITAAC #4.b	Added “and relative humidity limits in the Remote Shutdown Console Room during normal plant operations” at the end of DC and AC column text. RAI 474, 09.04.05-10
2.7-202	Table 2.7.5.2-3, ITAAC #4.c	Added “during all plant operating conditions, including normal plant operations, abnormal and accident conditions” at the end of DC and AC column text. Replaced “below 2%” with “below 1%” in the AC column text. RAI 474, 09.04.05-10
2.7-203	Table 2.7.5.2-3, ITAAC #4.d	Added “during a design basis accident or LOOP” at the end of DC and AC column text. RAI 474, 09.04.05-10
2.7-203	Table 2.7.5.2-3, ITAAC #4.e	Added “during a design basis accident or LOOP” at the end of DC and AC column text. RAI 474, 09.04.05-10
2.7-203	Table 2.7.5.2-3, ITAAC #4.f	Added “during a design basis accident or LOOP” at the end of DC and AC column text. RAI 474, 09.04.05-10

Table 2.7.5.2-3 Engineered Safety Features Ventilation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a The annulus emergency exhaust system is capable of meeting the selected numerical performance values used in the safety analysis listed in Subsection 2.7.5.2.1.1.	4.a.i Type tests, tests and analyses of filter efficiencies for the annulus emergency exhaust system will be performed on both divisions.	4.a.i The annulus emergency exhaust system is capable of meeting the filter efficiencies identified in Subsection 2.7.5.2.1.1 on both divisions.
	4.a.ii A Test of negative pressure arrival time for the as-built annulus emergency exhaust system will be performed on both divisions.	4.a.ii The as-built annulus emergency exhaust system is capable of drawing down all four penetration areas and all four safeguard component areas to less than or equal to -0.25 inches w.g. relative to adjacent areas within the arrival time identified in Subsection 2.7.5.2.1.1 on both divisions.
4.b The Class 1E electrical room HVAC system provides conditioning air to maintain area design temperature limits in the Class 1E electrical rooms during all plant operating conditions, including normal plant operations, abnormal and accident conditions and relative humidity limits in the remote shutdown console room during normal plant operations.	4.b Tests and analyses of the as-built Class 1E electrical room HVAC system will be performed for all four divisions.	4.b The as-built Class 1E electrical room HVAC system is capable of providing conditioning air to maintain area design temperature limits within the Class 1E electrical rooms during all plant operating conditions, including normal plant operations, abnormal and accident conditions and relative humidity limits in the remote shutdown console room during normal plant operations.
4.c The Class 1E electrical room HVAC system provides battery room ventilation to maintain hydrogen concentration within the design limit during all plant operating conditions, including normal plant operations, abnormal and accident conditions.	4.c Tests and analyses of the as-built Class 1E electrical room HVAC system will be performed for all four divisions.	4.c The as-built Class 1E electrical room HVAC system is capable of providing battery room ventilation to maintain hydrogen concentration below ~~2~~1% by battery room volume during all plant operating conditions, including normal plant operations, abnormal and accident conditions.

Table 2.7.5.2-3 Engineered Safety Features Ventilation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.d The safeguard component area HVAC system provides conditioning air to maintain area design temperature limits within the safeguard component areas when the respective equipment is operating during a design basis accident or LOOP.	4.d Tests and analyses of the as-built safeguard component area HVAC system will be performed for all four divisions.	4.d The as-built safeguard component area HVAC system is capable of providing conditioning air to maintain area design temperature limits within the safeguard component areas when the respective equipment is operating during a design basis accident or LOOP.
4.e The emergency feedwater pump area HVAC system provides conditioning air to maintain area design temperature limits within the emergency feedwater pump areas when the respective equipment is operating during a design basis accident or LOOP.	4.e Tests and analyses of the as-built emergency feedwater pump area HVAC system will be performed for all four divisions.	4.e The as-built emergency feedwater pump area HVAC system is capable of providing conditioning air to maintain area design temperature limits within the emergency feedwater pump areas when the respective equipment is operating during a design basis accident or LOOP.
4.f The safety-related component area HVAC system provides conditioning air to maintain area design temperature limits in each individual safety-related component area, when the respective equipment is operating during a design basis accident or LOOP.	4.f Tests and analyses of the as-built safety-related component area HVAC system will be performed for each safety-related component area.	4.f The as-built safety-related component area HVAC system is capable of providing conditioning air to maintain area design temperature limits in each individual safety-related component area, when the respective equipment is operating during a design basis accident or LOOP.
5.a The dampers identified in Table 2.7.5.2-1 perform an active safety function to change position as indicated in the table.	5.a.i Tests of the as-built dampers identified in Table 2.7.5.2-1 will be performed using a simulated signal.	5.a.i Each as-built damper identified in Table 2.7.5.2-1 perform the active safety function identified in the table after receiving an ECCS actuation signal or a high temperature signal.
	5.a.ii Tests of the as-built tornado dampers identified in Table 2.7.5.2-1 will be performed under preoperational test pressure, and fluid flow conditions.	5.a.ii Each as-built tornado damper changes position as identified in Table 2.7.5.2-1.