US-APWR DCD Revision 2
RAI Tracking Report

March 2010

Revision History

Revision	Page	Description
0	All	Original issued Including RAI responses that were submitted through October 31, 2009
1	All	Including RAI responses that were submitted through December 31, 2009
2	All	Including RAI responses that were submitted through February 28, 2010 Including editorial changes to clarify the English language and to correct typographical errors

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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. Furthermore, the editorial changes to clarify the English language and to correct typographical errors are shown in the DCD Markups and Revision List.

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.

Chapter:1

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				
1	Introduction and Interfaces	531	01-7	2010/2/3	Y	N	N		-	DCD_01-7	2	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				
2.0	Site Charactoristics and Site	518	02-1	2010/2/15	Y	Y	N		-	DCD_02-1	2	3
	Parameters											
2.2.3	Evaluation of Potential Accidents											
2.3.1	Regional Climatology											
2.3.2	Local Meteorology											
2.3.3	Onsite Meteorological											
	Measurement Programs											
2.3.4	Short-term Dispersion Estimates											
	for Accident Releases											
2.3.5	Long-Term Atmospheric Dispersion											
	Estimates for Routine Releases											
2.4	Hydrology											
2.4.1	Hydrologic Description											
2.4.2	Floods											
2.4.3	Probable Maximum Flood (PMF)											
	on Streams and Rivers											
2.4.4	Potential Dam Failures											
2.4.5	Probable Maximum Surge											
	and Seiche Flooding											
2.4.6	Probable Maximum											
	Tsunami Hazards											
2.4.7	Ice Effects											

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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				
2.4.8	Cooling Water Canals											
	and Reservoirs											
2.4.9	Channel Diversions											
2.4.10	Flooding Protection Requirements											
2.4.11	Low Water Considerations											
2.4.12	Groundwater											
2.4.13	Accidental Releases											
	of Radioactive Liquid Effluents											
	in Ground and Surface Waters											
2.4.14	Technical Specifications											
	and Emergency											
	Operation Requirements											
2.5.1	Technical Specifications											
	and Emergency											
	Operation Requirements											
2.5.2	Vibratory Ground Motion											
2.5.3	Surface Faulting											
2.5.4	Stability of Subsurface Materials	OI	02.05.04-01A	2010/2/22	Y	N	N		-	DCD_02.05.04-01A	TBD	
	and Foundations											
2.5.5	Stability fo Slopes											

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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		03.04.02-1									
			03.04.02-2									
			03.04.02-3									
			03.04.02-4									
		489	03.04.02-5	12/26/2009	N	N	N		-	-	N/A	N/A
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	1	3
3.6.1	Plant Design for Protection											
	Against Postulated Piping Failures											

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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				
	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
	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	1	3
3.6.3	Leak-Before-Break	485	3.6.3-18	2010/1/18	N	N	N		-	-	N/A	N/A
	Evaluation Procedures	485	3.6.3-19	2010/1/18	Y	Y	N		-	DCD_3.6.3-19	2	3
		485	3.6.3-20	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-21	2010/1/18	Y	N	N		-	DCD_3.6.3-21	2	3
		485	3.6.3-22	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-23	2010/1/18	N	N	N		-	-	N/A	N/A
		485	3.6.3-24	2010/1/18	Y	N	N		-	DCD_3.6.3-24	2	3
		485	3.6.3-25	2010/1/18	N	N	N		-	-	N/A	N/A
3.7.1	Seismic Design Parameters	494	03.07.01-2	2010/1/29	N	N	N		-	-	N/A	N/A
		494	03.07.01-3	2010/1/29	N	N	N		-	-	N/A	N/A
		494	03.07.01-4	2010/1/29	Y	Y	N		-	DCD_03.07.01-4	2	3
3.7.2	Seismic System Analysis	495	03.07.02-2	2010/2/2	N	N	N		-	-	N/A	N/A
		212	3.7.2-3	2009/5/7	Y	N	N		-	DCD_3.7.2-3	TBD	
		495	03.07.02-3A	2010/2/2	N	N	N		-	-	N/A	N/A
		495	03.07.02-4	2010/2/2	Y	N	N		-	DCD_03.07.02-4	TBD	
		495	03.07.02-5	2010/2/2	Y	N	N		-	DCD_03.07.02-5	2	3
		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	493	03.07.03-2	2010/1/28	Y	N	N		-	DCD_03.07.03-2	2	3
		493	03.07.03-3	2010/1/28	N	N	N		-	-	N/A	N/A
		493	03.07.03-4	2010/1/28	N	N	N		-	-	N/A	N/A
		493	03.07.03-5	2010/1/28	Y	N	N		-	DCD_03.07.03-5	2	3
3.7.4	Seismic Instrumentation											
3.8.1	Concrete Containment											
		-	-	-	-	-	-	-	COL3.8(2) deleted	MAP-03-004	0	2, 3
		490	03.08.01-2	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-3	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-4	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-5	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-6	2010/2/4	N	N	N		-	-	N/A	N/A

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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				
		490	03.08.01-7	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-8	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-9	2010/2/4	N	N	N		-	-	N/A	N/A
		490	03.08.01-10	2010/2/4	N	N	N		-	-	N/A	N/A
3.8.3	Concrete and Steel	491	03.08.03-16	2010/3/3	N	N	N		-	-	N/A	N/A
	Internal Structures	491	03.08.03-17	2010/3/3	N	N	N		-	-	N/A	N/A
	of Steel or Concrete Containments	491	03.08.03-18	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-19	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-20	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-21	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-22	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-23	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-24	2010/3/3	N	N	N		-	-	N/A	N/A
		491	03.08.03-25	2010/3/3	N	N	N		-	-	N/A	N/A
3.8.4	Other Seismic Category I Structures	497	03.08.04-32	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-33	2010/2/19	Y	N	N		-	DCD_03.08.04-33	TBD	
		497	03.08.04-34	2010/2/19	Y	N	N		-	DCD_03.08.04-34	2	3
		497	03.08.04-35	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-36	2010/2/19	Y	N	N		-	DCD_03.08.04-36	2	3
		497	03.08.04-37	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-38	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-39	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-40	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-41	2010/2/19	Y	N	N		-	DCD_03.08.04-41	2	3
		497	03.08.04-42	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-43	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-44	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-45	2010/2/19	N	N	N		-	-	N/A	N/A
		497	03.08.04-46	2010/2/19	Y	N	N		-	DCD_03.08.04-46	2	3
		497	03.08.04-47	2010/2/19	N	N	N		-	-	N/A	N/A
3.8.5	Foundations	496	03.08.05-23	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-24	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-25	2010/2/4	Y	N	N		-	DCD_03.08.05-25	2	3
		496	03.08.05-26	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-27	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-28	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-29	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-30	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-31	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-32	2010/2/4	Y	N	N		-	DCD_03.08.05-32	2	3
		496	03.08.05-33	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-34	2010/2/4	N	N	N		-	-	N/A	N/A
		496	03.08.05-35	2010/2/4	Y	Y	N		-	DCD_03.08.05-35	2	3
3.9.1	Special Topics for											
	Mechanical Components											
3.9.2	Dynamic Testing and Analysis	498	03.09.02-59	2010/1/15	N	N	N		-	-	N/A	N/A
	of Systems, Structures,	498	03.09.02-60	2010/1/15	N	N	N		-	-	N/A	N/A
	and Components	498	03.09.02-61	2010/2/3	Y	N	N		-	DCD_03.09.02-61	TBD	
		498	03.09.02-62	2010/2/3	Y	N	N		-	DCD_03.09.02-62	TBD	
		498	03.09.02-63	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-64	2010/2/3	Y	N	N		-	DCD_03.09.02-64	2	3
		498	03.09.02-65	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-66	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-67	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-68	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-69	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-70	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-71	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-72	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-73	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-74	2010/1/15	N	N	N		-	-	N/A	N/A

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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				
		498	03.09.02-75	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-76	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-77	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-78	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-79	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-80	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-81	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-82	2010/2/3	N	N	N		-	-	N/A	N/A
		498	03.09.02-83	2010/1/15	N	N	N		-	-	N/A	N/A
		498	03.09.02-84	2010/2/3	N	N	N		-	-	N/A	N/A
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											
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	1	3
		486	03.10-12	2009/12/9	Y	N	N		-	DCD_03.10-12	1	3
		486	03.10-13	2009/12/25	Y	N	N		-	DCD_03.10-13	1	3
		486	03.10-14	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-15	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-16	2009/12/25	N	N	N		-	-	N/A	N/A
		486	03.10-17	2009/12/25	N	N	N		-	-	N/A	N/A
3.11	Environmental Qual	445	03.11-16	2009/9/29	Y	Y 1.8	N		-	DCD_03.11-16	0	3
	of Mech/Elec Eqmt	511	03.11-17	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-18	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-19	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-20	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-21	2010/2/2	Y	N	N		-	DCD_03.11-21	TBD	
		511	03.11-22	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-23	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-24	2010/2/2	Y	N	N		-	DCD_03.11-24	TBD	
		511	03.11-25	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-26	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-27	2010/2/2	N	N	N		-	-	N/A	N/A
		511	03.11-28	2010/2/2	N	N	N		-	-	N/A	N/A
		512	03.11-29	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-30	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-31	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-32	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-33	2010/1/28	N	N	N		-	-	N/A	N/A
		512	03.11-34	2010/1/28	Y	N	N		-	DCD_03.11-34	TBD	
		512	03.11-35	2010/1/28	N	N	N		-	-	N/A	N/A
3.12	ASME Code Class 1, 2, and 3	465	03.12-17	2009/12/2	Y	N	N		-	DCD_O3.12-17	1	3
	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

Chapter:3

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				
		465	03.12-23	2009/12/2	Y	N	N		-	DCD_03.12-23	1	3
		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											

Chapter:9

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	507	09.01.04-16	2010/2/15	Y	N	N		-	DCD_09.01.04-16	2	3
	(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	1	3
9.3.3	Equipment and Floor	426	09.03.03-15	2009/9/14	Y	N	N		-	DCD_09.03.03-15	-	2
	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	-	2
9.3.4	Chemical and Volume Control System											
	(PWR)											
	(Including Boron Recovery System)											
9.4.1	Control Room Area	327	09.04.01-9	2010/1/29	Y	N	N		-	DCD_09.04.01-9	2	3
	Ventilation System											
		475	09.04.01-12A	2009/11/20	Y	Y	N		-	DCD_09.04.01-12A	1	3
		475	09.04.01-13A	2009/11/20	Y	N	N		-	DCD_09.04.01-13A	1	3
		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	483	09.04.03-08	2010/2/5	Y	N	N		-	DCD_09.04.03-08	2	3
	Ventilation System	483	09.04.03-09	2010/2/5	Y	N	N		-	DCD_09.04.03-09	2	3
		483	09.04.03-10	2010/2/5	N	N	N		-	-	N/A	N/A

Chapter:9

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				
9.4.4	Turbine Area Ventilation System											
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	1	3
	Oil Storage and Transfer System	468	09.05.04-44	2009/12/10	Y	Y	N		-	DCD_09.05.04-44	1	3
		468	09.05.04-45	2009/12/10	Y	N	N		-	DCD_09.05.04-45	1	3
		468	09.05.04-46	2009/12/10	Y	N	N		-	DCD_09.05.04-46	1	3
		468	09.05.04-47	2009/12/10	Y	N	N		-	DCD_09.05.04-47	1	3
		468	09.05.04-48	2009/12/10	Y	N	N		-	DCD_09.05.04-48	1	3
		468	09.05.04-49	2009/12/10	N	N	N		-	-	N/A	N/A
9.5.5	Emergency Diesel Engine											
	Cooling Water System											
9.5.6	Emergency Diesel Engine											
	Starting System	504	09.05.06-24	12/23/09	Y	N	N		-	DCD_09.05.06-24	1	3
		504	09.05.06-25	12/23/09	Y	N	N		-	DCD_09.05.06-25	1	3
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
		506	09.05.07-20	2010/1/29	Y	N	N		-	DCD_09.05.07-20	2	3
		506	09.05.07-21	2010/1/29	N	N	N		-	-	N/A	N/A
		506	09.05.07-22	2010/1/29	Y	N	N		-	DCD_09.05.07-22	2	3
		506	09.05.07-23	2010/1/29	Y	N	N		-	DCD_09.05.07-23	2	3
9.5.8	Emergency Diesel Engine	470	09.05.08-18	2009/12/2	Y	N	N		-	DCD_09.05.08-18	1	3
	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	1	3
		470	09.05.08-21	2009/12/2	Y	N	N		-	DCD_09.05.08-21	1	3
		470	09.05.08-22	2009/12/2	Y	N	N		-	DCD_09.05.08-22	1	3
		505	09.05.08-23	2010/2/1	N	N	N		-	-	N/A	N/A
		505	09.05.08-24	2010/2/1	N	N	N		-	-	N/A	N/A
		505	09.05.08-25	2010/2/1	Y	N	N		-	DCD_09.05.08-25	2	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				
14.2	Initial Plant Test Program -	455	14.02-119	2009/10/1	Y	N	N		-	DCD_14.02-119	-	2
	Design Certification and	521	14.02-120	2010/2/5	Y	Y	N		-	DCD_14.02-120	2	3
	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	499	14.03.03-23	2009/12/16	Y	N	N		-	DCD_14.03.03-23	1	3
	and Acceptance Criteria											
14.3.4	Reactor Systems	503	14.03.04-42	2009/12/21	Y	N	N		-	DCD_14.03.04-42	1	3
14.3.5	Instrumentation and Controls -	515	14.03.05-32	2010/1/28	Y	N	N		-	DCD_14.03.05-32	2	3
14.3.6	Electrical Systems -											
	Inspections, Tests, Analyses,											
	and Acceptance Criteria											
14.3.7	Plant Systems -	456	14.03.07-48	2009/10/5	Y	N	N		-	DCD_14.03.07-48	-	2
	Inspections, Tests, Analyses,	456	14.03.07-49	2009/10/5	Y	N	N		-	DCD_14.03.07-49	-	2
	and Acceptance Criteria	508	14.03.07-50	2009/12/24	Y	N	N		-	DCD_14.03.07-50	1	3
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 -	488	14.03.11-40	12/25/09	Y	N	N					
	Inspections, Tests, Analyses,			2010/1/13	Y	N	N		-	DCD_14.3.4.11-40	1	3
	and Acceptance Criteria	488	14.03.11-41	12/25/09	N	N	N					
				2010/1/13	N	N	N		-	-	N/A	N/A
		488	14.03.11-42	12/25/09	Y	N	N					
				2010/1/13	Y	N	N		-	DCD_14.3.4.11-42	1	3
		396	14.03.12-20	2009/7/17	Y	N	N		-	DCD_14.03.12-20	2	3
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

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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				

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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	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:	520	16-300	2010/2/18	Y	Y	N		-	DCD_16-300	2	3
	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	1	3
		OI	16-1784/174	11/10/2009	Y	Y	N		-	DCD_16-1784/174	1	3
		OI	16-1784/186	11/10/2009	Y	Y	N		-	DCD_16-1784/186	-	2
		OI	16-1784/188	11/10/2009	Y	Y	N		-	DCD_16-1784/188	1	3
		OI	16-1784/192	11/10/2009	Y	Y	N		-	DCD_16-1784/192	-	2
		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	-	2
		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											

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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				
19	Probabilistic Risk Assessment	88	19-150	2008/11/27	Y	N	N	fin.	-	DCD_19-150	-	2
	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	-	2
		423	19-375	2009/9/7	Y	N	N		-	DCD_19-375	1	3
		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	1	3
		479	19-403	2009/11/25	Y	N	N		-	DCD_19-403	1	3
		479	19-404	2009/11/25	Y	N	N		-	DCD_19-404	1	3
		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
		480	19-*** (1)	2009/11/26	N	N	N		-	-	N/A	N/A
		480	19-*** (2)	2009/11/26	N	N	N		-	-	N/A	N/A
		480	19-*** (3)	2009/11/26	N	N	N		-	-	N/A	N/A
		480	19-*** (4)	2009/11/26	N	N	N		-	-	N/A	N/A
		480	19-*** (5)	2009/11/26	N	N	N		-	-	N/A	N/A
		480	19-*** (6)	2009/11/26	N	N	N		-	-	N/A	N/A
		528	19-407	2010/3/3	Y	N	N		-	DCD_19-407	2	3
		528	19-408	2010/3/3	Y	N	N		-	DCD_19-408	2	3
		528	19-409	2010/3/3	Y	N	N		-	DCD_19-409	2	3
		528	19-410	2010/3/3	Y	N	N		-	DCD_19-410	2	3
		528	19-411	2010/3/3	N	N	N		-	-	N/A	N/A
		528	19-412	2010/3/3	Y	N	N		-	DCD_19-412	TBD	
		528	19-413	2010/3/3	Y	N	N		-	DCD_19-413	TBD	
		528	19-414	2010/3/3	Y	N	N		-	DCD_19-414	2	3
		528	19-415	2010/3/3	Y	N	N		-	DCD_19-415	2	3
		528	19-416	2010/3/3	Y	N	N		-	DCD_19-416	2	3
		528	19-417	2010/3/3	Y	N	N		-	DCD_19-417	2	3
		528	19-418	2010/3/3	Y	N	N		-	DCD_19-418	2	3
		528	19-419	2010/3/3	Y	N	N		-	DCD_19-419	2	3
		528	19-420	2010/3/3	Y	N	N		-	DCD_19-420	2	3
		528	19-421	2010/3/3	Y	N	N		-	DCD_19-421	TBD	
		528	19-422	2010/3/3	Y	N	N		-	DCD_19-422	2	3
19.1	Determining the Technical Adequacy											
	of Probabilistic Risk Assessment											
	Results for Risk-Informed Activities											
		-	-	-	-	-	-	-	COL 19.3(5) deleted	MAP-19-001	-	2
19.2	Review of Risk Information											
	Used to Support Permanent Plant -											
	Specific Changes											
	to the Licensing Basis:											
	General Guidance											

Chapter 1

US-APWR DCD Chapter 1 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
1.5-4	Reference 1.5-8	Editorial: typographical error Changed the title of reference “HIS System description …” to “HSI System Description …”
1.8-5	Table 1.8-2 Sheet 1 COL 2.1(1)	Changed “The COL Applicant is to describe the site geography and demography including the site parameters.” to “The COL Applicant is to describe the site geography and demography including the site characteristics.” Reason: Corrected the use of the words “parameters” and “characteristics” [RAI 518-3967 Question 02-1]
1.8-10	Table 1.8-2 Sheet 6 COL 3.6(10)	Add new COL Item: “3.6(10) The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer.” Reason: New COL information item to provide operating and maintenance procedures to address water hammer for RCL branch piping. [RAI 485-3825 Question 03.06.03-19]
1.8-11	Table 1.8-2 Sheet 7 COL 3.7(7)	Changed “The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, and to evaluate the bearing load to this capacity.” to “The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the bottom of foundation elevations, and to evaluate the bearing load to this capacity.” Reason: Revised COL 3.7(7) to clarify that the properties of fill concrete used as a supporting medium are also discussed in DCD Subsection 3.7.1.3. [RAI 496-3735 Question 03.08.05-35]

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
1.8-11	Table 1.8-2 Sheet 7 COL 3.7(8)	Changed "The soil properties may be considered strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, the COL Applicant must institute dynamic testing to evaluate the strain-dependent variation of the material dynamic properties for site materials with initial shear wave velocities below 3,500 ft/s." to "The COL Applicant is to evaluate the strain-dependent variation of the material dynamic properties for site materials." Reason: Removed the ability to consider soil properties as strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher from the DCD. [RAI 494-3978 Question 03.07.01-4]
1.8-25	Table 1.8-2 Sheet 21 COL 6.6(1)	Editorial: typographical error Changed "… (pupms and valves), …" to "… (pumps and valves), …"
1.8-43	Table 1.8-2, COL 14.2(11)	RAI: No. 521, 14.02-120 Replaced the sentence as follows; "The COL holder for the first plant is to perform the first plant only test and prototype test." To "The COL holder for the first plant is to perform the first plant only tests and prototype test."
1.9-4	Table 1.9.1-1, Sheet 2 of 15, RG 1.21	RAI: No. 294, 09.03.02-6 Added the corresponding Subsection 9.3.2.
1.9-5	Table 1.9.1-1, Sheet 3 of 15, RG 1.35.1	RAI: No. 521, 14.02-120 Changed the corresponding Subsection 14.27 to 14.2.7.
1.9-6	Table 1.9.1-1, Sheet 4 of 15, RG 1.45	RAI: No. 522, 11.05-18 Added the corresponding Subsection 9.3.2.
1.9-11	Table 1.9.1-1, Sheet 9 of 15, RG 1.128	RAI: No. 388, 08.03.02-15 Changed the status to "Conformance with no exceptions identified."

1.5-4 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P (Proprietary) and MUAP-07011-NP (Non-Proprietary), Revision 0, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-5 Safety I&C System Description and Design Process, MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-6 Safety System Digital Platform -MELTAC-, MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., August 2008.

1.5-7 Defense-in-Depth and Diversity, MUAP-07006-P (Proprietary) and MUAP-07006-NP (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., June 2008.

1.5-8 H~~I~~SI System Description and HFE Process, MUAP-07007-P (Proprietary) and MUAP-07007-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., July 2007.

1.5-9 Qualification and Test Plan of Class 1E Gas Turbine Generator System, MUAP-07024, Revision 0, Mitsubishi Heavy Industries, Ltd., December 2007.

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

COL ITEM NO.	COL ITEM
COL 1.1(1)	*The COL Applicant is to provide scheduled completion date and estimated commercial operation date of nuclear power plants referencing the US-APWR standard design.*
COL 1.1(2)	*The Combined License (COL) Applicant is to identify the actual plant location.*
COL 1.2(1)	*The COL Applicant is to develop a complete and detailed site plan in the site-specific licensing process.*
COL 1.4(1)	*The COL Applicant is to identify major agents, contractors, and participants for the COL application development, construction, and operation.*
COL 1.8(1)	*The COL Applicant is to demonstrate that the interface requirements established for the design have been met.*
COL 1.8(2)	*The COL Applicant is to provide the cross-reference identifying specific FSAR sections that address each COL information item from the DCD*
COL 1.8(3)	*The COL Applicant is to provide a summary of plant specific departures from the DCD, and conformance with site parameters.*
COL 1.9(1)	*The COL Applicant is to address an evaluation of the applicable RG, SRP, Generic Issues including Three Mile Island (TMI) requirements, and operational experience for the site-specific portion and operational aspect of the facility.*
COL 2.1(1)	*The COL Applicant is to describe the site geography and demography including the specified site ~~parameters~~characteristics.*
COL 2.2(1)	*The COL Applicant is to describe nearby industrial, transportation, and military facilities within 5 miles of the site, or at greater distances as appropriate based on their significance. The COL Applicant is to establish the presence of potential hazards, determine whether these accidents are to be considered as DBEs, and the design parameters related to the accidents determined as DBEs.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 6 of 44)

COL ITEM NO.	COL ITEM
COL 3.6(6)	*Deleted*
COL 3.6(7)	*Deleted*
COL 3.6(8)	*Deleted*
COL 3.6(9)	*Deleted*
COL 3.6(10)	*The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer.*
COL 3.7(1)	*The COL Applicant is to confirm that the site-specific PGA at the basemat level control point of the CSDRS is less than or equal to 0.3 g.*
COL 3.7(2)	*The COL Applicant is to perform an analysis of the US-APWR standard plant seismic category I design to verify that the site-specific FIRS at the basemat level control point of the CSDRS are enveloped by the site-independent CSDRS.*
COL 3.7(3)	*It is the responsibility of the COL Applicant to develop analytical models appropriate for the seismic analysis of buildings and structures that are designed on a site-specific basis including, but not limited to, the following:* • *PSFSVs (seismic category I)* • *ESWPT (seismic category I)* • *UHSRS (seismic category I)*
COL 3.7(4)	*To prevent non-conservative results, the COL Applicant is to review the resulting level of seismic response and determine appropriate damping values for the site-specific calculations of ISRS that serve as input for the seismic analysis of seismic category I and seismic category II subsystems.*
COL 3.7(5)	*The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S, and the site-specific response spectra obtained from the response analysis.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 7 of 44)

COL ITEM NO.	COL ITEM
COL 3.7(6)	*The COL Applicant is to develop site-specific GMRS and FIRS by an analysis methodology, which accounts for the upward propagation of the GMRS. The FIRS are compared to the CSDRS to assure that the US-APWR standard plant seismic design is valid for a particular site. If the FIRS are not enveloped by the CSDRS, the US-APWR standard plant seismic design is modified as part of the COLA in order to validate the US-APWR for installation at that site.*
COL 3.7(7)	*The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the bottom of foundation elevations, and to evaluate the bearing load to this capacity.*
COL 3.7(8)	*~~The soil properties nay be considered strain-independent for subgrade materials with initial shear wave velocities of 3.500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, t~~The COL Applicant ~~must institute dynamic testing~~is to evaluate the strain-dependent variation of the material dynamic properties for site materials~~ with initial shear wave velocities below 3,500 ft/s~~.*
COL 3.7(9)	*The COL Applicant is to assure that the design or location of any site-specific seismic category I SSCs, for example pipe or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of non-seismic category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, non safety-related storage tanks, etc.*
COL 3.7(10)	*It is the responsibility of the COL Applicant to further address structure-to-structure interaction if the specific site conditions can be important for the seismic response of particular US-APWR seismic category I structures, or may result in exceedance of assumed pressure distributions used for the US-APWR standard plant design.*
COL 3.7(11)	*Deleted*
COL 3.7(12)	*It is the responsibility of the COL Applicant to design seismic category I below- or above-ground liquid-retaining metal tanks such that they are enclosed by a tornado missile protecting concrete vault or wall, in order to confine the emergency gas turbine fuel supply.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 21 of 44)

COL ITEM NO.	COL ITEM
COL 6.6(1)	*The COL Applicant is responsible for identifying the implementation milestone for ASME Section XI inservice inspection program for ASME Code Section III Class 2 and 3 systems, components (~~pupms~~pumps and valves), piping, and supports, consistent with the requirements of 10 CFR 50.55a (g).*
COL 6.6(2)	*The COL Applicant is responsible for identifying the implementation milestone for the augmented inservice inspection program.*
COL 7.3(1)	*Deleted*
COL 7.4(1)	*The COL applicant is to provide a description of component controls and indications required for safe shutdown related to the UHS.*
COL 7.5(1)	*The COL applicant is to provide a description of site-specific PAM variables.*
COL 7.5(2)	*The COL applicant is to provide a description of the site-specific EOF.*
COL 7.9(1)	*Deleted*
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 lighting 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.*

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 39 of 44)

COL ITEM NO.	COL ITEM
COL 14.2(8)	*Deleted*
COL 14.2(9)	*Deleted*
COL 14.2(10)	*The COL applicant is responsible for the testing outside scope of the certified design in accordance with the test criteria described in subsection 14.2.1. [14.2.12]*
COL 14.2(11)	*The COL holder for the first plant is to perform the first plant only tests and prototype test. For subsequent plants, either these tests are performed, or the COL applicant provides a justification that the results of the first-plant only tests are applicable to the subsequent plant and are not required to be repeated. [14.2.8]*
COL 14.2(12)	*The COL holder makes available approved test procedures for satisfying testing requirements described in Section 14.2 to the NRC approximately 60 days prior to their intended use. [14.2.3, 14.2.11, 14.2.12.1]*
COL 14.3(1)	*The COL applicant provides the ITAAC for the site specific portion of the plant systems specified in Subsection 14.3.5, Interface Requirements. [14.3.4.6,14.3.4.7]*
COL 14.3(2)	*The COL applicant provides proposed ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.10]*
COL 14.3(3)	*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. [14.3.4.12]*
COL 15.0(1)	*In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and 10 CFR 52.79 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criterion 19 are met for affected events using site-specific χ/Q values. Additionally, the Technical Support Center (TSC) dose should be evaluated against the habitability requirements in Paragraph IV.E. 8 to 10 CFR Part 50, Appendix E, and 10 CFR 50.47(b)(8) and (b)(11).*

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

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 3 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.29	Seismic Design Classification (Rev. 4, March 2007)	Conformance with no exceptions identified.	3.2.1, 5.2.5, 5.2.2.1, 5.4.11.1, 7.1.3.7, 8.1.5.3, 9.1.1, 9.1.2, 9.3.1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 1972)	Conformance with exceptions. Installation is not included in Design Certification phase.	14.2.7, 17.5
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	Conformance with no exceptions identified.	4.5.2, 5.2.3.4.4, 5.3.1.4, 6.1.1
1.32	Criteria for Power Systems for Nuclear Power Plants (Rev. 3, March 2004)	Conformance with no exceptions identified.	8.1.5.3, 16.3
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	Conformance with exceptions. Implementation of RG applies to a site-specific operational program for which COL Applicant will be responsible.	12.1.3 13.5
1.34	Control of Electroslag Weld Properties (Rev. 0, December 1972)	Not applicable. Electroslag welding is not employed in structural welds of low alloy steel. Electroslag welding is only applied for cladding.	5.2.3.3.2, 5.2.3.4.4, 5.3.1.4
1.35	In-Service Inspection (ISI) of Ungrouted Tendons in Prestressed Concrete Containments (Rev. 3, July 1990)	Conformance with no exceptions identified.	3.8.1.2, 3.8.1.7, 14.2.7
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	Conformance with no exceptions identified.	3.8.1.2, 3.8.1.7, 14.2.7
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 1973)	Conformance with no exceptions identified.	5.2.3.2, 6.1.1.2
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 1, March 2007)	Conformance with exception. Programmatic/operational aspect is not applicable to US-APWR design certification.	3.13.1, 4.5.1, 5.2.3, 5.3.1, 6.1.1, 14.2.7
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	Not applicable. RG applies to a site-specific operational program.	N/A
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	Not applicable. RG applies to a site-specific operational program.	N/A
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	Not applicable. US-APWR has no Class 1 continuous-duty motors in the containment.	N/A

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 4 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (Rev. 0, March 1973)	Conformance with no exceptions identified.	8.1.5.3, 14.2.7
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.2.3.3.2, 5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	3.6.3.3.4, 5.2.3.4.1, 5.2.3.4.2, 6.1.1
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage ~~Reactor Coolant Pressure Boundary Leakage Detection Systems~~ (Rev. 1, May 2008)	Conformance with no exceptions Identified.	5.2.5, 11.5
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	Conformance with no exceptions identified.	8.1.5.3, table 8.1-1, 18.7.3.2, table 18.7-1
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	This RG has been withdrawn by NRC.	N/A
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.3.1.2, 5.3.1.4, 5.2.3.3.2, 6.1.1
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	Conformance with no exceptions identified.	6.4.2, 6.4.6, Table 6.4-2, 6.5.1, Table 6.5-3, 9.4.1, 9.4.5, 12.3.3, 14.2.7
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	Conformance with no exceptions identified.	7.1.3. 2, 7.1.3.3, 8.1.5.3
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	Conformance with exceptions. Programmatic/operational and site-specific aspects are not applicable to US-APWR design certification. ASTM standard revision levels may differ from RG 1.54 as specifically referenced in the “Corresponding Chapter/Section/Subsection”	6.1.2 11.2 11.4
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	Not applicable. US-APWR has a concrete containment.	N/A
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	Conformance with exceptions. RG applies to a site-specific characterization for flooding.	2.4, 3.4.1.2
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, 2.5, 3.7
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified.	3.7, 3.9.2, 3.12.3, 3.12.5.4, 3.12.6.8

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 9 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, March 1978)	Not applicable. RG applies to a site-specific operational program.	N/A
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with no exceptions identified. ~~The hydrogen concentration limit required in RG 1.189 is appropriate for the fire protection scenario, over the RG 1.128.~~	8.1.5.3, 14.2.7
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Conformance with exceptions. Design certification applicability is to assure design features accommodate functions described in RG; full conformance in terms of program and activities will be the responsibility of the COL Applicant.	8.1.5.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 2, March 2007)	Conformance with no exceptions identified.	3.9.3.4, 3.12.6.1
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	Conformance with no exceptions identified.	8.1.5.3
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	Not applicable. RG applies to site-specific operational program.	N/A
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	Conformance with exceptions. C.3.a: Section 13.5 defines the responsibility for development of administrative and operating procedures. C.6: The COL applicant has the responsibility of this requirement.	4.4.6.3
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants (Rev. 3, March 1998))	Not applicable. RG applies to a site-specific operational program.	N/A
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	Conformance with exception. Site-specific aspect is not applicable to US-APWR design certification.	2.4
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments (Rev. 3, March 2007)	Conformance with no exceptions identified.	3.8.1.2, 14.2.7

Chapter 2

US-APWR DCD Chapter 2 Rev. 2, Trackinf Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
2.0-9	Table 2.0-1 Sheet 8 1st and 2nd Rows	Added Note 15. "(15)" Reason: Added notes to clarify allowable displacements. [RAI 496-3735 Question 03.08.05-32]
2.0-9	Table 2.0-1 Sheet 8 3rd and 4th Rows	Added Note 16. "(16)" Reason: Added notes to clarify allowable displacements. [RAI 496-3735 Question 03.08.05-32]
2.0-9	Table 2.0-1 Sheet 8 Notes	Added Note 15: "15. Settlements occurring during construction and operational life." Reason: Added notes to clarify allowable displacements. [RAI 496-3735 Question 03.08.05-32]
2.0-9	Table 2.0-1 Sheet 8 Notes	Added Note 16: "16. Settlements occurring during operational life only." Reason: Added notes to clarify allowable displacements. [RAI 496-3735 Question 03.08.05-32]
2.1-1	Section 2.1 1st Paragraph 1st Sentence	Change: "The Combined License (COL) Applicant is to describe the site geography and demography including the site parameters identified below." to "The Combined License (COL) Applicant is to describe the site geography and demography including the site characteristics identified below." Reason: Corrected the use of the words "parameters" and "characteristics" [RAI 518-3967 Question 02-1]
2.1-1	Subsection 2.1.4 *COL 2.1(1)*	Change: "*The COL Applicant is to describe the site geography and demography including the site parameters.*" to "*The COL Applicant is to describe the site geography and demography including the site characteristics*." Reason: Corrected the use of the words "parameters" and "characteristics" [RAI 518-3967 Question 02-1]
2.3-2	Subsection 2.3.4 5th Paragraph 1st Sentence	Change: "The 0-8 hrs χ/Q values of MCR and TSC are calculated by some formula based on both …" to "The 0-8 hrs χ/Q values of MCR and TSC are calculated by formulas based on both …" Editorial: clarify the language
2.3-3	Subsection 2.3.5	Change: "The *D/Q* values should be determined in a

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	5th Paragraph 1st Sentence	similar way as to how to determine the χ/Q values.” to “The *D/Q* values should be determined in a similar way as determining the χ/Q values.” Editorial: clarify the language

Table 2.0-1 Key Site Parameters
(Sheet 8 of 8)

Total settlement of R/B complex foundation[14][15]	6.0 in.
Differential settlement across R/B complex foundation[14][15]	2.0 in.
Maximum differential settlement between buildings[14][16]	0.5 in.
Maximum tilt of R/B complex foundation generated during operational life of the plant[14][16]	1/2000

NOTES:

1. The specified missiles are assumed to have a vertical speed component equal to 2/3 of the horizontal speed.
2. These dispersion factors are chosen as the maximum values at all intake points.
3. These dispersion factors are chosen as the maximum values at all inleak points.
4. These dispersion factors are used for a loss-of-coolant accident (LOCA) and a rod ejection accident.
5. These dispersion factors are used for a LOCA, a rod ejection accident, a failure of small lines carrying primary coolant outside containment and a fuel-handling accident inside the containment.
6. These dispersion factors are used for a steam generator tube rupture, a steam system piping failure, a reactor coolant pump rotor seizure and a rod ejection accident.
7. These dispersion factors are used for a fuel handling accident occurring in the fuel storage and handling area.
8. These dispersion factors are used for a steam system piping failure.
9. These dispersion factors are used for a LOCA.
10. These dispersion factors are used for a rod ejection accident, a failure of small lines carrying primary coolant outside containment and a fuel-handling accident inside the containment.
11. Normal winter precipitation roof load is determined by converting ground snow load p_g in accordance with ASCE 7-05. The ground snow load p_g is based on the highest ground-level weight of:
    - the 100-year return period snowpack,
    - the historical maximum snowpack,
    - the 100-year return period snowfall event, or
    - the historical maximum snowfall event in the site region.
12. The extreme winter precipitation roof load is based on the sum of the normal ground level winter precipitation plus the highest weight at ground level resulting from either the extreme frozen winter precipitation event or the extreme liquid winter precipitation event. The extreme frozen winter precipitation event is assumed to accumulate on the roof on top of the antecedent normal winter precipitation event. The extreme liquid winter precipitation event may not accumulate on the roof, depending on the geometry of the roof and the type of drainage provided. The extreme winter precipitation roof load is included as live load in extreme loading combinations using the applicable load factor indicated in DCD Section 3.8.
13. The 48-hour PMWP is based on interpolation of 24-hour PMP and 72-hour PMP data for the month of March in HMR-53 (Reference: Hydrometeorological Report No. 53, Seasonal Variation of 10-Square-Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, Figures 27 and 37)
14. Acceptable parameters for settlement without further evaluation.
15. Settlements occurring during construction and operational life.
16. Settlements occurring during operational life only.

2.1 Geography and Demography

The Combined License (COL) Applicant is to describe the site geography and demography including the site ~~parameters~~ characteristics identified below.

2.1.1 Site Location and Description

- Site-specific information of the site location and description includes:
    - Plant and site property lines
    - Location and orientation of principal plant structures within the site area
    - Location of any industrial, military, or transportation facilities and commercial, institutional, recreational, or residential structures within the site area
    - Highways, railroads, and waterways that traverse or are adjacent to the site
    - Prominent natural and manmade features in the site area.

2.1.2 Exclusion Area Authority and Control

Site-specific information on the exclusion area includes the size of the area, and the exclusion area authority and control. If the EAB extends into a body of water, a discussion is provided with the bases upon which it has been determined that the applicant holds (or will hold) the authority required by 10 Code of Federal Regulations (CFR) 100.21(a), Non-Seismic Siting Criteria (Reference 2.1-1).

Non-related plant activities that occur, or could potentially occur, within the EAB, if any, are to be described, and their effects evaluated on plant operations and safety considered.

2.1.3 Population Distribution

Site-specific information regarding population distribution is based on the latest census data. The population is also projected through the anticipated life of the plant, and is to include the bases of the projections including methodology and sources used to obtain the data.

2.1.4 Combined License Information

COL 2.1(1) The COL Applicant is to describe the site geography and demography including the specified site ~~parameters~~ characteristics.

2.1.5 References

2.1-1 Non-seismic Siting Criteria, Reactor Site Criteria, Energy. Title 10, Code of Federal Regulations, Part 100.21(a), U.S. Nuclear Regulatory Commission, Washington, DC

2.3.4 Short-Term Atmospheric Dispersion Estimates for Accident Releases

For appropriate time periods up to 30 days after an accident, conservative estimates are provided of atmospheric dispersion factors (χ/Q values) at the site's EAB, at the outer boundary of the LPZ, and at the MCR for postulated accidental radioactive airborne releases.

The short-term χ/Q values are site-specific parameters. The χ/Q values listed in Table 2.0-1 are bounding factors for a typical US-APWR sited in most areas of the US and can be used to calculate radiological consequences of design basis accidents. There is no site-specific meteorological data in the stage of the DCD. The atmospheric dispersion factors (χ/Q values) are determined as follows.

The US-APWR χ/Q value of EAB should be determined as the representative of the US plants. Therefore, the US-APWR χ/Q value of EAB is selected to envelop most values at the corresponding EAB distance (0.5 miles) of the many existing plants. This value is reasonable in comparison with the existing plants values with different EAB distances.

The χ/Q values of LPZ are also determined by using the same method as EAB at every time interval. However, the LPZ distance of US-APWR can not be specified in the stage of the DCD. Therefore, the US-APWR χ/Q values of LPZ are determined to envelop most χ/Q values of many existing plants with LPZ distance of more than 1 mile.

The 0-8 hrs χ/Q values of MCR and TSC are calculated by ~~some~~ formulas based on both the diffusion equations used in ARCON96 (Reference 2.3-10) and the meteorological condition referred to RG 1.194 (Reference 2.3-9) (e.g. F stability and wind speeds of 1.0 m/s), not directly by ARCON96 itself. In this calculation formula, a multiplier is introduced to envelop the most χ/Q values of MCR and TSC of many existing plants.

By using the χ/Q values of MCR and TSC at various source-receptor distances of many existing plants, it is ensured that the above calculation formula envelops the most χ/Q values of the existing plants at any source-receptor distance, and then the US-APWR χ/Q value of MCR and TSC is determined by this calculation formula.

The other time interval χ/Q values (8-24 hrs, 24-96 hrs, 96-720 hrs) of MCR and TSC are calculated by using both the above formula of 0-8 hrs χ/Q values and the time interval factors described in RG 1.194 regulatory position 4.4. These calculated χ/Q values also envelop most existing plants values.

As a result, the US-APWR χ/Q values of EAB, LPZ, MCR and TSC in DCD Tier 2 Table 2.0-1 are representative of a reasonable number of the existing plants values. The COL Applicant is to provide conservative factors as described in SRP 2.3.4 (Reference 2.3-2). If a selected site will cause excess to the bounding χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 52.79(a)(1)(vi) (Reference 2.3-3) and the control room dose limits in 10 CFR 50, Appendix A, General Design Criteria 19 (Reference 2.3-4) are met using site-specific χ/Q values.

The necessary data to calculate χ/Q values of MCR and TSC by using ARCON96 are shown in Table 2.3-1 to 2.3-4.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

For annual average release, bounding limits of annual χ/Q values and deposition factors (*D/Q* values) are provided at the onsite (EAB) and offsite to evaluate individual dose.

The long-term χ/Q values at the US-APWR EAB are site-specific. There is no site-specific meteorological data and the food production area in the stage of the DCD.

The Depleted/Undepleted/Decayed χ/Q value of EAB should be determined to envelop the most existing plant values. The US-APWR χ/Q value of EAB (0.5 miles) is selected as representative of US-plants, to be around 70% of the highest value at the corresponding EAB distance of many existing plants. This US-APWR χ/Q value of EAB envelopes most values at the corresponding EAB distance of many existing plants.

The long-term offsite χ/Q value should be determined for the food production area. The offsite χ/Q value is defined almost to envelop the χ/Q values at locations more than the EAB distance of the US-APWR.

The *D/Q* values should be determined in a similar way as ~~to how to determine~~ determining the χ/Q values. The US-APWR *D/Q* value of the offsite boundary is conservatively assumed to be equal to the *D/Q* value of EAB. Therefore, the *D/Q* values of EAB are determined to envelop most values of some existing plants.

Therefore, it is ensured that the χ/Q values and the *D/Q* values of the US-APWR bound a reasonable number of existing plant values. The COL Applicant is to characterize the atmospheric transport and diffusion conditions necessary for estimating radiological consequences of the routine release of radioactive materials to the atmosphere, and provide realistic estimates of annual average χ/Q values and *D/Q* values as described in SRP 2.3.5 (Reference 2.3-5).

2.3.6 Combined License Information

COL 2.3(1) *The COL Applicant, whether the plant is to be sited inside or outside the continental US, is to provide site-specific pre-operational and operational programs for meteorological measurements, and is to verify the site-specific regional climatology and local meteorology are bounded by the site parameters for the standard US-APWR design or demonstrate by some other means that the proposed facility and associated site-specific characteristics are acceptable at the proposed site.*

COL 2.3(2) *The COL Applicant is to provide conservative factors as described in SRP 2.3.4 (Reference 2.3-2). If a selected site will cause excess to the bounding χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 52.79(a)(1)(vi) (Reference 2.3-3) and the control room dose limits in 10 CFR 50,Appendix A, General Design Criteria 19(Reference 2.3-4) are met using site-specific χ/Q values.*

Chapter 3

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	**Description of Change**
3.2-56	Table 3.2-2 Sheet 40 5th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Change the System and Components Description: “Main oil pumps” to “Lube oil main oil pumps” Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-56	Table 3.2-2 Sheet 40 6th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Change the System and Components Description: “Oil cooler” to “Lube oil cooler” Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-56	Table 3.2-2 Sheet 40 7th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Add new row for Item 27: “Lube oil reduction gear reservoir”; “3”; “PS/B”; “C”; “YES”; “5”; “I” Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-57	Table 3.2-2 Sheet 41 9th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Change the System and Components Description: “Combustion air intake equipment” to “Combustion air intake equipment and ductwork, turbine exhaust” Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-57	Table 3.2-2 Sheet 41 10th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Change the System and Components Description: “Exhaust equipment” to “GTG Room ventilation system supply side equipment and ductwork and exhaust side equipment and ductwork” Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-57	Table 3.2-2	Change the System and Components Description: “Piping and valves” to “Piping and valves (Safety related

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Sheet 41 11th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	portion: Off skid)" Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-57	Table 3.2-2 Sheet 41 Behind 11th Row under **Item 27. Emergency Gas Turbine Auxiliary System**	Add 2 new last rows for Item 27: "Piping and valves (Safety related portion: On skid)";"3"; "PS/B"; "C"; "YES"; "5"; "I" "PSFSV Ventilation system containing exhaust fan, back draft dampers, in-duct electric heater and ductwork";"5"; "PSFSV"; "N/A"; "N/A"; "5"; "II" Reason: Revised to identify classification of reduction gear reservoir and on and off skid piping and valves. [RAI 506-4029 Question 09.05.07-22]
3.2-58	Table 3.2-2 Sheet 42 **Item 28. Fuel Handling and Refueling System** Last Row for Item 28	Add new last row for Item 28: "Permanent Cavity Seal"; "4"; "PCCV"; "D"; "N/A"; "5"; "II" Reason: Provided description of the permanent cavity seal. [RAI 507-3993 Question 09.01.04-16]
3.2-66	Table 3.2-2 Sheet 50 6th and 7th Rows under **Item 41. Main Steam/Feedwater Piping Area Heating, Ventilation, and Air Conditioning System** Last Column	Change: "Dampers with areas containing safety-related equipment area…" to "Dampers within areas containing safety-related equipment are…" Reason: Correct typographical errors
3.2-67	Table 3.2-2 Sheet 51 8th and 9th Rows under **Item 42. Auxiliary Building**	Change: "Dampers with areas containing safety-related equipment area…" to "Dampers within areas containing safety-related equipment are…" Reason: Correct typographical errors

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	Heating, Ventilation, and Air Conditioning System Last Column	
3.3-7	Subsection 3.3.2.3 4th Paragraph 1st Sentence	Change: “The AC/B is not designed for a tornado and consequently it could potentially fail due to design basis tornado loading, including loss of its siding.” to “The AC/B is not designed for a tornado and consequently it could potentially fail due to design basis tornado loading.” Reason: Editorial correction. The AC/B is a concrete structure and has no siding.
3.6-23	Subsection 3.6.3 Last Paragraph 1st and 2nd Sentences	Change: “The COL Applicant is to identify the types of as-built materials and material specification used for base metal welds, weldments, and safe ends for piping evaluated for LBB. Additionally, the COL Applicant is to provide information related to as-built material…” to “The types of as-built materials and material specification is to be identified for base metal welds, weldments, and safe ends for piping evaluated for LBB. Additionally, information is to be provided related to as-built material…” Reason: Editorial Change
3.6-26	Subsection 3.6.3.3.1 4th Paragraph Last Sentence	Add at the end of paragraph: “Also, proper operating and maintenance procedures will be performed to prevent water hammer. The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer. The procedures should address the plant operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.” Reason: New COL information item to provide operating and maintenance procedures to address water hammer for RCL branch piping. [RAI 485-3825 Question 03.06.03-19]
3.6-34	Subsection 3.6.4 COL Item 3.6(10)	Add new COL Item: “*3.6(10) The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention*

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		of water hammer." Reason: New COL information item to provide operating and maintenance procedures to address water hammer for RCL branch piping. [RAI 485-3825 Question 03.06.03-19]
3.7-2	Subsection 3.7.1.1 6th Paragraph First Sentence	Change: "For the design of…" to "For the seismic design of…" Reason: Provide clarification [RAI 495-3980 Question 03.07.02-05]
3.7-2	Subsection 3.7.1.1 6th Paragraph Last Sentence	Change: "Examples of seismic category I buildings and structures which are not part of the standard plant include the essential service water pipe tunnel (ESWPT), the power source fuel storage vaults (PSFSVs), and the ultimate heat sink related structures (UHSRS)." to "Refer to Subsection 3.8.4 for discussion relating to the seismic design of seismic category I and seismic category II buildings and structures that are not part of the US-APWR standard plant." Reason: Clarify that PSFSVs and ESWPTs are functionally part of the standard design of the main power block but are seismically designed as a non-standard plant building. [RAI 495-3980 Question 03.07.02-05]
3.7-4	Subsection 3.7.1.1 **Site-Specific GMRS** 2nd Paragraph 5th Sentence	Change: "If materials are present at the site in which the initial (small strain) shear velocity is less than 3,500 ft/s, the site response analysis will address probable effects of non-linearity due to strain-dependence of the subgrade materials' response." to "The site response analysis will address probable effects of non-linearity due to strain-dependence of the subgrade materials' response." Reason: Removed the ability to consider soil properties as strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher from the DCD. [RAI 494-3978 Question 03.07.01-4]
3.7-10	Subsection 3.7.1.3 2nd Paragraph 3rd Sentence	Change: "The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, and to evaluate the bearing load to this capacity." to "The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		bottom of foundation elevations, and to evaluate the bearing load to this capacity." Reason: Revised COL 3.7(7) to clarify that the properties of fill concrete used as a supporting medium are also discussed in DCD Subsection 3.7.1.3. [RAI 496-3735 Question 03.08.05-35]
3.7-27	Subsection 3.7.2.3.11 1st Paragraph 2nd Sentence	Change: "…operation (ASCE 7, Subsection 12.7.2 [Reference 3.7-24]) or 75% of the roof snow load, whichever is…" to "…operation (ASCE 7, Subsection 12.7.2 [Reference 3.7-24]) and 75% of the roof snow load, whichever is…" Reason: Clarified by changing "or" to "and" for the use of live and snow loads on the seismic analyses. [RAI 497-3734 Question 03.08.04-36]
3.7-30	Subsection 3.7.2.4.1 7th Paragraph 3rd and 4th Sentences	Change: "The soil properties may be considered strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, the COL Applicant must institute dynamic testing to evaluate the strain-dependent variation of the material dynamic properties for site materials with initial shear wave velocities below 3,500 ft/s." to "The COL Applicant is to evaluate the strain-dependent variation of the material dynamic properties for site materials." Reason: Removed the ability to consider soil properties as strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher from the DCD. [RAI 494-3978 Question 03.07.01-04]
3.7-45	Subsection 3.7.3.1 4th Paragraph	Add as new 4th Paragraph: "The time history or response spectra generated at the support point of the subsystem are utilized as the input motion for performing the seismic dynamic analysis of the subsystem. However, where these data are not readily available, the data generated for a distance away from the structural support point may be used. To account for the structural linkage (i.e., intervening structural element) between these two locations, the additional amplification of the response due to the presence of the intervening structural element can be calculated and the remote input motion can be transformed. For cases where the intervening structure

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		is rigid (i.e., frequency > 50 Hz), the transformation can be achieved by adding the effect due to the rigid body motion of the intervening structure to the existing input motion at the remote location. The new translational time history at the interface location is generated by algebraic summation of the translational acceleration time history at the reference location and the time-history contribution arising from the rocking and torsional effects of the intervening structural element. The new translational response spectra are obtained by absolute sum of the translational response spectra at the reference location and the contributions arising from the rocking and torsional effects of the intervening structural element. For places where the intervening structural element is judged to be flexible, the new ISRS are generated by incorporating the flexibility of the intervening structural element. Or alternatively, the seismic dynamic analysis of the subsystem shall be expanded to include the flexibility of the intervening structural element." Reason: Provided information on ISRS for they analysis of SSCs. [RAI 493-3983 Question 03.07.03-5]
3.7-52	Subsection 3.7.3.9 Last Paragraph	Change: "Hydrodynamic loads including sloshing loads on these liquid-retaining vessels are determined using methods that conform to the provisions of Subsection II.14 of SRP 3.7.3 (Reference 3.7-35) and guidance of ASCE 4-98, Subsection 3.5.4 (Reference 3.7-9). The horizontal response analysis considers both the impulsive mode (in which a portion of the water moves in unison with the tank wall) and the horizontal sloshing convective mode. The seismic sloshing analysis of also considers potential slosh heights with respect to the potential of creating flooding, which is discussed in Section 3.4." to "Hydrodynamic loads on these liquid-retaining vessels are determined using methods that conform to the provisions of Subsection II.14 of SRP 3.7.3 (Reference 3.7-35) and guidance of ASCE 4-98, Subsection 3.5.4 (Reference 3.7-9). The horizontal response analysis considers both the impulsive mode (in which a portion of the water moves in unison with the tank wall) and the horizontal convective mode (water motion associated with wave oscillation). The seismic analysis of convective hydrodynamic effects also considers the maximum wave oscillation with respect to the potential of creating flooding, which is discussed in Section 3.4."

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Reason: Removed the term "sloshing" to provide clarification [RAI 497-3734 Question 03.08.04-34]
3.7-58	Subsection 3.7.5 COL 3.7(7)	Change: "*The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, and to evaluate the bearing load to this capacity*." to "*The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the bottom of foundation elevations, and to evaluate the bearing load to this capacity*." Reason: Revised COL 3.7(7) to clarify that the properties of fill concrete used as a supporting medium are also discussed in DCD Subsection 3.7.1.3. [RAI 496-3735 Question 03.08.05-35]
3.7-58	Subsection 3.7.5 COL 3.7(8)	Change: "The soil properties may be considered strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, the COL Applicant must institute dynamic testing to evaluate the strain-dependent variation of the material dynamic properties for site materials with initial shear wave velocities below 3,500 ft/s." to "The COL Applicant is to evaluate the strain-dependent variation of the material dynamic properties for site materials." Reason: Removed the ability to consider soil properties as strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher from the DCD. [RAI 494-3978 Question 03.07.01-4]
3.7-67	Table 3.7.1-3 2nd Row 2nd Column	Change: " 26'-8"/38'-10" " to " 38'-10" " Reason: Maintain dimensional consistency with plant drawings.
3.7-67	Table 3.7.1-3 4th Row 4th Column	Change: " 139'-6" " to " 175'-9" " Reason: Maintain dimensional consistency with plant drawings.
3.7-67	Table 3.7.1-3 Last Row	Change in the 2nd Column: " 37'-3" " to "38'-10" " Change in the 3rd Column: " 71' x 117' " to " (66'-0") x

Page	Location (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
	2nd through 4th Columns	(111'-6")(3) " Change in the 4th Column: " 51'-11" " to " 87'-4" " Reason: Maintain dimensional consistency with plant drawings.
3.7-67	Table 3.7.1-3 Note 3	Change: "Width and height are the distances between column lines of exterior walls." " to "Width and length are the distances between column lines of exterior walls." Reason: Correct typographical error.
3.8-49	Subsection 3.8.4.3.2 4th Sentence	Change: "Hydrodynamic loads due to seismic sloshing are determined as discussed in Subsection 3.7.3.9, and included in the earthquake load as described in Subsection 3.8.4.3.6." to "Impulsive and convective hydrodynamic loads due to seismic events are determined as discussed in Subsection 3.7.3.9, and included in the earthquake load as described in Subsection 3.8.4.3.6." Reason: Removed the term "sloshing" to provide clarification [RAI 497-3734 Question 03.08.04-34]
3.8-54	Subsection 3.8.4.3.6.2 2nd Paragraph 2nd Sentence	Change: "In addition to the dead load, 25% of the floor live load during normal operation or 75% of the roof snow load, whichever is applicable, is also considered as accelerated mass in the seismic models." to "In addition to the dead load, 25% of the floor live load during normal operation and 75% of the roof snow load, whichever is applicable, is also considered as accelerated mass in the seismic models." Reason: Clarified by changing "or" to "and" for the use of live and snow loads on the seismic analyses. [RAI 497-3734 Question 03.08.04-36]
3.8-63	Subsection 3.8.4.4.4 Last Paragraph Last Sentence	Change: "…the same load combinations and stress coefficients given in Table 3.8.4-4, except where noted therein." to "…the same load combinations and stress coefficients given in Table 3.8.4-4." Reason: Provided clarification that there are no exceptions to the acceptance criteria [RAI 493-3983 Question 03.07.03-2]
3.8-69	Subsection 3.8.4.6.3	Change: "There are no special construction techniques utilized in the construction of other seismic category I

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		structures." to "Standard provisions of ACI are to be applied where necessary to address issues related to the use of massive concrete pours. As stated in Subsection 3.8.4.6.1.1, volume changes in mass concrete are controlled where necessary by applying measures and provisions outlined in ACI 207.2R (Reference 3.8-52) and ACI 207.4R (Reference 3.8-53). The following summarizes the construction techniques commonly associated, either singularly or in combination, with massive concrete pours such as basemats: • Limit the size of concrete pour. • Use a checkerboard pattern of concrete placement in a single lift. To avoid a weak horizontal shear plane, a double lift placement of concrete, in general, is avoided. However, when it is absolutely needed to have two lifts, adequate design considerations and also, in general, shear stirrups are provided. • Schedule concrete pours for the most advantageous day and time to control temperature rise in the concrete. • Post-cooling can be performed by cooling the freshly placed concrete with running chilled water lines in the concrete." Reason: Revised to address special precautions and techniques that are required for massive concrete pours. [RAI 497-3734 Question 03.08.04-41]
3.8-72	Subsection 3.8.5.4.1 3rd Paragraph 2nd Sentence	Delete last Sentence: "The dissipation of energy in the subgrade media due to the soil material damping is conservatively neglected." Reason: [RAI 496-3735 Question 03.08.05-25]
3.8-100	Table 3.8.4-4 Sheet 2 Note 11	Change: "The stress limit coefficient where axial compression exceeds 20% of normal allowable, is 1.5 for load combinations 7, 8, 9, 9a, and 10, and 1.6 for load combination 11. For seismic category II members the stress limit coefficient applicable to axial and bending stresses for load combinations 7 through 11 is 1.7, however the allowable stress shall not exceed 1.0 F_y." to "The stress limit coefficient where axial compression exceeds 20% of normal allowable, is 1.5 for load combinations 7, 8, 9, 9a, and 10, and 1.6 for load combination 11. For load combinations 7 through 11 the allowable stress shall not exceed 1.0 F_y."

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		Reason: Provided clarification that there are no exceptions to the acceptance criteria [RAI 493-3983 Question 03.07.03-2]
3.8-163	Figure 3.8.1-13 Figure Title	Change: “Transient Conditions of Temperature of the Refueling Cavity Atmosphere and Sump Pool Water (Pipe Break in the Refueling Cavity)” to “Transient Conditions of Temperature of the Reactor Cavity Atmosphere and Sump Pool Water (Pipe Break in the Reactor Cavity)” Reason: Correct typographical error
3.9-22	Subsection 3.9.2.3 Last Paragraph	Add as last Paragraph: “The design of the US-APWR steam delivery system (including the safety relief valves and the steam separator) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the United States and around the world. The US-APWR steam delivery system is designed using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the USA for more than 20 years with sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the structural and vibration design bases are proven. This non-safety-related steam delivery system will not experience excessive vibration; therefore, the analysis of the flow excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes) is not expected.” Reason: Provided information on similarities to existing and currently operating steam delivery systems in the United States. [RAI 498-3782 Question 03.09.02-64]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3A-2	Subsection 3A.1.2 Last Sentence	Change: "Seismic category II ductwork and supports, including support anchorages, are therefore analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems, except where noted in Table 3.8.4-4." to "Seismic category II ductwork and supports, including support anchorages, are therefore analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems in Table 3.8.4-4." Reason: Provide clarification by removing phrase "except where noted". [RAI 493-3983 Question 03.07.03-2]
3B-iii	Figures Figure 3B-13	Changed: "US-APWR BAC for Surge Line (Heatup/Cooldown)" to "(deleted)" Reason: Deleted Figure 3B-13 [RAI 485-3825 Question 03.06.03-24]
3B-10	Subsection 3B.2.2.2 Next-to-Last Paragraph 2nd Sentence	Change: "Review of the curves against actual test data from the literature (e.g. as documented in NUREG-6004 – Reference 3B-4) has shown that the J-T curves should be achievable." to "Review of the curves against actual test data from the literature (e.g. Appendix B of NUREG/CR-6004 [Reference 3B-13] and Pipe Fracture Encyclopedia, Test Data – Volume 3 [Reference 3B-14]) has shown that the J-T curves should be achievable." Reason: Corrected reference number and directly references the Pipe Fracture Encyclopedia [RAI 485-3825 Question 03.06.03-21]
3B-17	Subsection 3B.5 Reference 3B-14	Add new reference: "3B-14 <u>Pipe Fracture Encyclopedia, Test Data – Volume 3</u>, U.S. Nuclear Regulatory Commission, December 1997." Reason: Added Reference used in Subsection 3B.2.2.2 [RAI 485-3825 Question 03.06.03-21]

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
3B-19	Table 3B-2 Item 8	Deleted information in columns 2-8 for Item 8. Changed Last Column: “Figure 3B-13” to “Figure 3B-13 (deleted)” Reason: Removed Figure 3B-13 since Figure 3B-12 is more conservative and will be used for all load combinations [RAI 485-3825 Question 03.06.03-24]
3B-32	Figure 3B-13	Deleted figure 3B-13, and changed title to “(deleted)” Reason: Removed Figure 3B-13 since Figure 3B-12 is more conservative and will be used for all load combinations [RAI 485-3825 Question 03.06.03-24]
3F-2	Subsection 3F.1.2 Last Sentence	Change: “Seismic category II conduit systems, including support anchorages, are therefore analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems, except where noted in Table 3.8.4-4.” to “Seismic category II conduit systems, including support anchorages, are therefore analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems in Table 3.8.4-4.” Reason: Provide clarification by removing phrase “except where noted”. [RAI 493-3983 Question 03.07.03-2]
3F-4	Subsection 3F.6.6 Last Sentence	Change: “The flexibility of base plates was considered in determining the anchor bolt loads.” to “The flexibility of base anchorage was considered in determining the anchor bolt loads.” Reason: Clarified statement [RAI 497-3734 Question 03.08.04-46]
3G-1	Subsection 3G.1.2 Last Sentence	Change: “Seismic category II cable tray systems including support anchorages, are therefore analyzed and designed for the applicable SSE, such as in-structure response spectra developed from the CSDRS within the standard plant Reactor Building and the East and West Power Source Buildings using the same methods and stress limits specified for seismic category I structures and subsystems, except where noted in Table 3.8.4-4.” to “Seismic category II cable tray systems including support anchorages, are therefore analyzed and designed for the applicable SSE, such as in-structure response spectra developed from the

Page	**Location** (e.g., subsection with paragraph/ sentence/ item, table with row/column, or figure)	Description of Change
		CSDRS within the standard plant Reactor Building and the East and West Power Source Buildings using the same methods and stress limits specified for seismic category I structures and subsystems in Table 3.8.4-4." Reason: Provide clarification by removing phrase "except where noted". [RAI 493-3983 Question 03.07.03-2]

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 40 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards[3]	Seismic Category[4]	Notes
Drain piping, valves in radiological controlled area	6	R/B A/B AC/B	N/A	N/A	6	Note 1	
Drain piping, valves, reactor building non-radioactive sump and sump pump in reactor building except for RCA	8	R/B	D	N/A	4	NS	
Drain piping, valves, turbine building sump and sump pump in turbine building	8	T/B	D	N/A	4	NS	
Drain piping, valves in auxiliary building and access control building, except for RCA	10	A/B,AC/B	N/A	N/A	5	NS	
Drain piping, valves in power source building	10	PS/B	N/A	N/A	5	NS	
Drain piping valves related to ESF rooms drain isolation FDS-VLV-001A,B,C,D	3	R/B	C	YES	3	I	
26. Potable and Sanitary Water System							
Potable and Sanitary Water System components, piping and valves	10	R/B,A/B,AC/B PS/B, T/B	N/A	N/A	5	NS	
27. Emergency Gas Turbine Auxiliary System							
Fuel oil storage tanks	3	PSFSV	C	YES	3	I	
Fuel oil transfer pumps	3	PSFSV	C	YES	3	I	
Fuel oil day tanks	3	PS/B	C	YES	3	I	
Air receivers	3	PS/B	C	YES	3	I	
Lube oil main ~~Main~~ oil pumps	3	PS/B	C	YES	5	I	
Lube oil ~~Oil~~ cooler	3	PS/B	C	YES	5	I	
Lube oil reduction gear reservoir	3	PS/B	C	YES	5	I	
Ventilation and cooling equipment	3	PS/B	C	YES	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 41 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Combustion air intake equipment and ductwork, turbine exhaust	3	PS/B	C	YES	5	I	
~~Exhaust~~ GTG Room ventilation system supply side equipment and ductwork and exhaust side equipment and ductwork	3	PS/B	C	YES	5	I	
Piping and valves (Safety related portion: Off skid)	3	PS/B, PSFSV	C	YES	3	I	
Piping and valves (Safety related portion: On skid)	3	PS/B	C	YES	5	I	
PSFSV Ventilation system containing exhaust fan, back draft dampers, in-duct electric heater and ductwork	5	PSFSV	N/A	N/A	5	II	
28. Fuel Handling and Refueling System							
Refueling machine	4	R/B	D	N/A	5	II	
Fuel handling machine	4	R/B	D	N/A	5	II	
Spent fuel assembly handling tool	5	R/B	N/A	N/A	5	NS	
New fuel storage rack	3	R/B	C	YES	5	I	
Spent fuel storage rack	3	R/B	C	YES	5	I	
Fuel transfer tube	2	R/B	B	YES	2	I	
Spent fuel Pit	3	R/B	C	YES	5	I	
New fuel pit	3	R/B	C	YES	5	I	
Fuel transfer canal	3	R/B	C	YES	5	I	
Cask pit	3	R/B	C	YES	5	I	
Cask washdown pit	3	R/B	C	YES	5	I	
Spent fuel pit gates	3	R/B	C	YES	5	I	
Fuel inspection pit	3	R/B	C	YES	5	I	
Fuel transfer system	4	R/B	D	N/A	5	II	
Suspension hoist and aux. hoist on spent fuel cask handling crane	4	R/B	D	N/A	5	II	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 42 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
New fuel elevator	4	R/B	D	N/A	5	II	
Containment rack	4	PCCV	D	N/A	5	II	
New fuel assembly handling tool	5	R/B	N/A	N/A	5	NS	
Rod control cluster handling tool	5	R/B	N/A	N/A	5	NS	
Thimble plug handling tool	5	R/B	N/A	N/A	5	NS	
Burnable poison rod assembly handling tool	5	R/B	N/A	N/A	5	NS	
Control rod drive shaft handling tool	5	R/B	N/A	N/A	5	NS	
Permanent Cavity Seal	4	PCCV	D	N/A	5	II	
29. Containment System							
Containment vessel	2	PCCV	B	YES	2	I	
Equipment hatch	2	PCCV	B	YES	2	I	
Personnel hatch	2	PCCV	B	YES	2	I	
30. Miscellaneous Plant Equipment							
PCCV polar crane	4	PCCV	D	N/A	5	II	These single-failure-proof cranes are designed in accordance with NUREG-0554 to maintain their position and hold their loads during an SSE.
Spent fuel cask handling crane	4	R/B	D	N/A	5	II	
(Deleted)							
Miscellaneous cranes and hoists in reactor building	4 or 10	R/B	D or N/A	N/A	5	II or NS	
Miscellaneous hoists in power source buildings	5	PS/B	N/A	N/A	5	II	
Crane for SWDS in auxiliary building	6	A/B	N/A	N/A	5	Note 1	
31. Containment Purge System							
Containment high volume purge air handling unit	10	R/B	N/A	N/A	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 50 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Ductwork and dampers	3	R/B, PS/B	C	YES	5	I	
41. Main Steam/Feedwater Piping Area Heating, Ventilation, and Air Conditioning System							
(Deleted)							
Main steam/feedwater piping area air handling units	9	R/B	N/A	N/A	5	NS	
Main steam/feedwater piping area air handling unit fans	9	R/B	N/A	N/A	5	NS	
Main steam/feedwater piping area air handling unit cooling coils	9	R/B	N/A	N/A	5	NS	
Main steam/feedwater piping area air handling unit electric heating coils	9	R/B	N/A	N/A	5	NS	
Dampers	5 or 9	R/B	N/A	N/A	5	II or NS	Dampers within areas containing safety-related equipment area supported as Seismic Category II.
Ductwork	5 or 9	R/B	N/A	N/A	5	II or NS	Ductwork within areas containing safety-related equipment area supported as Seismic Category II.
42. Auxiliary Building Heating, Ventilation, and Air Conditioning System							
Auxiliary building air handling units	9	A/B	N/A	N/A	5	NS	
Auxiliary building air handling unit fans	9	A/B	N/A	N/A	5	NS	
Auxiliary building air handling unit cooling coils	9	A/B	N/A	N/A	5	NS	
Auxiliary building air handling unit heating coils	8	A/B	D	N/A	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 51 of 57)

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category(4)	Notes
Auxiliary building exhaust fans	8	A/B	D	N/A	5	NS	
Penetration and Safeguard Component area isolation dampers and ductwork between Penetration and Safeguard Component area isolation damper	2	R/B	B	YES	5	I	
Exhaust line isolation dampers	2	R/B	B	YES	5	I	
Supply ductwork and dampers of the auxiliary building HVAC system	5 or 9	R/B, PS/B A/B, AC/B	N/A	N/A	5	II or NS	Dampers within areas containing safety-related equipment are~~a~~ supported as Seismic Category II.
Exhaust ductwork and dampers of the auxiliary building HVAC system	4 or 8	R/B, PS/B A/B, AC/B	D	N/A	5	II or NS	Ductwork within areas containing safety-related equipment are~~a~~ supported as Seismic Category II.
43. Non-Class 1E Electrical Room HVAC System							
Non-Class 1E electrical room air handling units	9	A/B	N/A	N/A	5	NS	
Non-Class 1E electrical room air handling unit fans	9	A/B	N/A	N/A	5	NS	
Non-Class 1E electrical room air handling unit cooling coils	9	A/B	N/A	N/A	5	NS	
Non-Class 1E electrical room air handling unit heating coils	8	A/B	D	N/A	5	NS	
Non-Class 1E electrical room Return air fans	9	A/B	N/A	N/A	5	NS	
Non-Class 1E battery room exhaust fans	5	A/B	N/A	N/A	5	NS	

the effective tornado wind pressure load on the building. This ensures that there is no overall failure of the T/B, due to tornado wind and/or atmospheric pressure change, which could affect the ability of adjacent buildings and structures to perform their intended safety functions. Localized failures of wind girts and other exposed SSCs are permitted. However, these items are designed to remain attached to the structure. Alternately, if such items could become dislodged, they are reviewed to ensure that no new missiles are generated that are not enveloped by the missiles addressed in Subsection 3.5.1.4.

The AC/B is not designed for a tornado and consequently it could potentially fail due to design basis tornado loading~~, including loss of its siding~~. However, since its location is sufficiently far away from seismic category I structures, and adjacent safety-related SSCs buried in the plant yard, the collapse of the AC/B would not impact any adjacent safety-related SSCs. The AC/B may also have localized failure due to tornado loading; however, the design precludes the generation of missiles that are not bounded by Subsection 3.5.1.4. The locations of any safety-related SSCs in the plant yard adjacent to the AC/B, including those which may be field routed, are reviewed prior to installation to ensure that their distances away from the AC/B and/or burial depths are sufficient to prevent potential failure effects that could jeopardize their function and integrity. Therefore, the ability of other SSCs to perform their intended safety functions is not affected by the potential collapse or localized failure of the AC/B due to tornado loading.

It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4. Where required by the results of investigations, structural reinforcement and/or missile barriers are implemented so as not to jeopardize safety-related SSCs.

3.3.3 Combined License Information

COL 3.3(1) *The COL Applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.*

COL 3.3(2) *These requirements also apply to seismic category I structures provided by the COL Applicant. Similarly, it is the responsibility of the COL Applicant to establish the methods for qualification of tornado effects to preclude damage to safety-related SSCs.*

COL 3.3(3) *It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.*

COL 3.3(4) *The COL Applicant is to provide the wind load design method and importance factor for site-specific category I and category II buildings and structures.*

expected range of impact energies demonstrate the capability to withstand the impact without rupture. Effects on environment and shutdown logics associated with the failure of the impacted pipe are considered.

3.6.2.5 Implementation of Criteria Dealing with Special Features

Special features such as pipe whip restraints, barriers, and shields are discussed in Subsection 3.6.2.4.4.

3.6.3 LBB Evaluation Procedures

This subsection describes the design basis to eliminate the dynamic effects of pipe rupture (Subsection 3.6.2) for the selected high-energy piping systems of RCL piping, RCL branch piping, and main steam piping. GDC 4 of Appendix A to 10 CFR 50 (Reference 3.6-1) allows exclusion of dynamic effects associated with pipe rupture from the design basis, when analyses demonstrate that the probability of pipe rupture is extremely low for the applied loading resulting from normal conditions, anticipated transients and a postulated SSE. The LBB evaluation is performed in accordance with SRP 3.6.3 (Reference 3.6-4).

The LBB analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated pipe break. The critical crack size is compared to the size of a leakage crack for which detection is certain. If the leakage crack size is sufficiently smaller than the critical crack size, the LBB requirements are satisfied.

The piping systems, for which the LBB criterion is not applied, are evaluated for dynamic effects of postulated pipe rupture at locations defined in Subsection 3.6.2. For piping systems for which LBB is demonstrated, the evaluation of environmental effects including spray wetting, and flooding is still performed for breaks or leakage cracks in accordance with Subsection 3.6.2.

~~The COL Applicant is to identify the types of as-built materials and material specification used~~ The types of as-built materials and material specification is to be identified for base metal welds, weldments, and safe ends for piping evaluated for LBB. Additionally, ~~the COL Applicant~~ information is to be provided ~~information~~ related to as-built material and material specifications for piping including toughness (J-R curves) and tensile strength (stress-strain curves), yield and ultimate strength, welding process/methods used, provide confirmation that the actual plant-specific stress analysis based on final as-built plant piping layout and material properties and welds satisfy the bounding LBB analysis, and provide confirmation that the final bounding LBB analysis addresses all plant-specific and generic degradation mechanisms in the as-built piping systems. This issue is to be resolved in ITAAC described in Table 2.3-2 of Tier 1 Chapter 2.3.

3.6.3.1 Application of LBB Criteria

Piping systems to which LBB criteria are applied are high-energy systems with well defined loading combinations and conditions. LBB criteria are applied to the following high energy piping systems (see Appendix 3E).

- RCL Piping

assure integrity of piping and support design. However, LBB criteria are not applied to these piping.

As to other RCL branch piping, water hammer has been reported for ECCS piping in the past. In US-APWR, however, operational control is applied in a way that avoids water hammer.

Water hammer is not experienced in RCL branch piping other than in these areas and the piping is designed to preclude the voiding condition according to operation at a pressure greater than the saturation pressure of the coolant. Furthermore, no valve that requires immediate action, such as pressurizer safety valve or relief valve, is present in the piping. Also, proper operating and maintenance procedures will be performed to prevent water hammer. The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer. The procedures should address the plant operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.

From the above reasons, water hammer is not anticipated to occur regarding RCL branch piping that LBB criteria is applied.

Main Steam Piping

Steam hammer in the main stem line is prevented by the design features included in system design. These features include prevention of slug formation by use of drain pots and proper sloping of the line. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

Protection against the potential occurrence of steam hammer is provided through operations and maintenance procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), caution against fast closing of the main steam isolation valves except when necessary, and emphasize proper draining.

A turbine trip, which initiates a rapid closure of the stop valve, is a design condition analyzed for the safety-related portion of main steam piping and associated components. This stress analyses assure that rapid valve closure does not challenge the integrity of piping. Therefore, the main steam piping is adequately designed to sustain steam hammer or similar high frequency hydrodynamic events.

3.6.3.3.2 Creep Damage

Pipe materials are selected to satisfy operational temperature limits not to exceed 700°F for ferritic steel piping and not to exceed 800°F for austenitic stainless steel piping. Therefore, the piping is designed to operate at temperatures less than that for which creep and creep-fatigue is a concern.

COL 3.6(3) *Deleted*

COL 3.6(4) *The COL Applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to identify the postulated rupture orientation of each postulated break location for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.*

COL 3.6(5) *Deleted*

COL 3.6(6) *Deleted*

COL 3.6(7) *Deleted*

COL 3.6(8) *Deleted*

COL 3.6(9) *Deleted*

COL 3.6(10) *The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer.*

3.6.5 References

3.6-1 Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, U.S. Nuclear Regulatory Commission, Washington, DC.

3.6-2 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 3.6.1, Rev.3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.6-3 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 3.6.2, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.6-4 Leak-Before-Break Evaluation Procedures, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 3.6.3, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

Design Ground Motion Response Spectra

Horizontal and vertical response spectra define the design seismic ground motion used for the US-APWR standard plant seismic design. The SSE, OBE, and the spectra, which are used to characterize these earthquake motions, are discussed in the following paragraphs.

SSE

The SSE is the earthquake which produces the maximum vibratory ground motion for which certain SSCs are designed to remain functional and within applicable stress, strain, and deformation limits.

The SSCs that must remain functional are those necessary to assure the following:

(1) The integrity of the RCPB.

(2) The capability to shut down the reactor and maintain it in a safe-shutdown condition.

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100 (Reference 3.7-4).

The CSDRS define the site-independent SSE used for the site-independent design of the US-APWR standard plant seismic category I and seismic category II SSCs. The major seismic category I buildings and structures of the US-APWR standard plant include the R/B, PCCV and containment internal structure, and the east and west PS/Bs.

For the seismic design of seismic category I and seismic category II SSCs that are not part of the US-APWR standard plant, and for the detail design of the US-APWR standard plant structures that are modified for the site-specific condition which can affect their integrity, a site-dependent SSE that is derived from the site-specific GMRS can be used. ~~Examples of seismic category I buildings and structures which are not part of the standard plant include the essential service water pipe tunnel (ESWPT), the power source fuel storage vaults (PSFSVs), and the ultimate heat sink related structures (UHSRS).~~ Refer to Subsection 3.8.4 for discussion relating to the seismic design of seismic category I and seismic category II buildings and structures that are not part of the US-APWR standard plant.

CSDRS

The CSDRS are presented herein to be approved under 10 CFR 52, Subpart B (Reference 3.7-5) as the site-independent seismic design response spectra for an approved certified design of the US-APWR standard nuclear power plant. The CSDRS characterize the site-independent SSE design ground motion that is defined at a control point located at the bottom of each US-APWR standard plant building basemat.

The in-structure response spectra (ISRS), which are used to design the seismic category I and II SSCs contained within or mounted to the US-APWR standard plant seismic category I buildings and structures, are computed from the CSDRS using methodology and approaches discussed in Subsection 3.7.2.5.

Site-specific GMRS are developed at a sufficient number of frequencies (at least 25) that adequately represent the local and regional seismic hazards using the site-specific geological, seismological, and geophysical input data. A probabilistic seismic hazard analysis is performed that is based either on the reference-probabilistic approach as outlined in RG 1.165 (Reference 3.7-2) or on the performance-based approach outlined in RG 1.208 (Reference 3.7-3). Horizontal GMRS are developed using a site amplification function obtained from site response analyses performed on site-specific soil profiles that include the layers of soil and rock over the generic rock defined as the rock with shear wave velocity exceeding 9,200 ft/s. The site-specific soil profiles account for the uncertainties and variations of the site soil and rock properties. ~~If materials are present at the site in which the initial (small strain) shear velocity is less than 3,500 ft/s, t~~The site response analysis will address probable effects of non-linearity due to strain-dependence of the subgrade materials' response. Equivalent linear methodology can be utilized with soil stiffness and damping degradation curves that represent the stiffness and damping properties of the subgrade materials as a function of strain. However, the strain-compatible soil material damping shall not exceed 15% as stipulated in SRP 3.7.1 (Reference 3.7-10).

With respect to determining the site-specific GMRS, note that Section 2.5.4 requires site-specific characterization of subsurface materials and investigation of the associated engineering properties to assure consistency with Section 3.7.2. Further, vertical GMRS are developed by combining the horizontal GMRS and the most up-to-date vertical/horizontal response spectral ratios appropriate for the site obtained from the most up-to-date attenuation relationships.

FIRS

The site-specific GMRS serves as the basis for the development of FIRS that define the horizontal and vertical response spectra of the outcrop ground motion at the bottom elevation of the seismic category I and II basemats. Free-field outcrop spectra of site-specific horizontal ground motion are derived from the horizontal GMRS using site response analyses that consider only the wave propagation effects in materials that are below the control point elevation at the bottom of the basemat. The material present above the control point elevation can be excluded from the site response analysis.

Appendix S (IV)(a)(1)(i) of 10 CFR 50 (Reference 3.7-7) requires that the SSE ground motion in the free-field at the basemat level must be represented by an appropriate response spectra with a PGA of at least 0.1 g. This requirement is met on a site-specific basis by considering minimum horizontal response spectra that are tied to the shapes of the US-APWR CSDRS and anchored at 0.1g. Since the CSDRS are based on modified RG 1.60-spectra, this assures that there is sufficient energy content in the low-frequency range. The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S (Reference 3.7-7), and the site-specific response spectra obtained from the response analysis. The same requirements apply to the vertical FIRS, which are developed from the horizontal FIRS by using vertical/horizontal response spectral ratios appropriate for the site.

The COL Applicant is to perform an analysis of the US-APWR standard plant seismic category I design to verify that the site-specific FIRS at the basemat level control point of the CSDRS are enveloped by the site-independent CSDRS. If the verification analysis

The damping values for systems that include two or more substructures, such as a concrete and steel composite structure, can be obtained using the strain energy method. The strain energy dependent modal damping values are computed based on Reference 3.7-18, which is the same as the stiffness weighted composite modal damping method, and acceptable to SRP 3.7.2 (Reference 3.7-16).

The stiffness weighted modal damping ratio h_i of the j^{th} mode is obtained from the following equation:

$$h_j = \frac{\vec{\phi}_j^{\,T}\left[\overline{K}\right]\vec{\phi}_j}{\vec{\phi}_j^{\,T}\left[K\right]\vec{\phi}_j}$$

where

$[K]$ = the stiffness matrix of the combined soil-structure system

$\vec{\phi}_j$ = the j^{th} normalized mode shape vector

$[\overline{K}] = \sum [k_i] \cdot \xi_i$ = the modified stiffness matrix constructed from the products of the element stiffness matrices $[k_i]$ and the applicable damping ratio ξ_i

Formulation of damping values for the seismic analysis models which incorporate the combined soil-structure damping is discussed in Subsection 3.7.2.1. Damping values associated with site-specific SSI analyses are addressed in Subsection 3.7.2.4.1.

3.7.1.3 Supporting Media for Seismic Category I Structures

A range of soil parameters of the basemat supporting media are considered in the seismic design of seismic category I building structures for the US-APWR standard plant. The overall basemat dimensions, basemat embedment depths, and maximum height of the US-APWR R/B, PCCV, and containment internal structure on their common basemat are given in Table 3.7.1-3 and as updated by the COL Applicant to include site-specific seismic category I structures.

The required allowable static bearing capacity for seismic category I building structure basemats, including the R/B-PCCV-containment internal structure on their common basemat, is 15 ksf. The dynamic bearing loads for seismic category I structure basemats are dependent upon the magnitude of the seismic loads that can be obtained from a site-specific seismic analysis that considers FIRS. The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the bottom of foundation elevations, and to evaluate the bearing load to this capacity. A minimum factor of safety of 2 is suggested for the ultimate bearing capacity versus the allowable dynamic bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions.

The site-independent seismic design of seismic category I and seismic category II SSCs uses lumped parameter representation to model the interaction of seismic category I structures with the supporting media. The lumped parameter model

Comparisons of static deformations are made between the three-dimensional stick model and the FE model, as previously discussed.

iii) Comparison of ISRS

Comparisons of ISRS are made between the three-dimensional stick model and the FE model at various points in various elevations, as previously discussed.

3.7.2.3.10.3 Containment Internal Structure

i) Fixed-base FE model

Figure 3.7.2-10 shows the fixed-base FE model for the containment internal structure, which is compared with the three-dimensional stick model. To verify the three-dimensional stick model, the FE model is used to estimate its rigidity by both static and dynamic analyses.

ii) Rigidity estimation by static analysis

Comparisons of static deformations are made between the three-dimensional stick model and the FE model, as previously discussed.

iii) Comparison of ISRS

Comparisons of ISRS are made between the three-dimensional stick model and the FE model at various points in various elevations as previously discussed.

3.7.2.3.11 Equivalent Masses due to Dead and Live Loads

In the design of seismic category I and seismic category II buildings and structures, dead loads and various portions of live loads are treated as equivalent masses for consideration in the global seismic analysis models. For example, 25% of the design floor live loads during normal operation (ASCE 7, Subsection 12.7.2 [Reference 3.7-24]) ~~or~~ and 75% of the roof snow load, whichever is applicable depending on the specific location in the building or structure, have been considered in computing tributary mass at node points in the seismic models. This is consistent with SRP 3.7.2, Section II.3(d) (Reference 3.7-16). For the containment operating deck in the PCCV, the design floor live load for maintenance and refueling is 950 lb/ft^2 and the floor live load for normal operation is 200 lb/ft^2. Therefore, 50 lb/ft^2 (25% of 200 lb/ft^2) has been used as an equivalent live load (mass) for the seismic analysis models.

Equivalent dead loads used in the seismic analysis models also include the weight of SSCs not specifically identified or included as dead loads in the models such as the weight of minor piping systems, cables and cable trays, ducts, and all related supports. Similarly, equivalent live loads include fluid contained within the minor piping and equipment under operating conditions. The weight of permanently attached tanks (uniformly distributed over the room floor area) is included as equivalent dead load (mass) in the seismic models. For the seismic analysis models, an equivalent dead load of a minimum of 50 lb/ft^2 uniform load is applied to cover these conditions. This is consistent with SRP 3.7.2, Section II(3)(d) (Reference 3.7-16).

For floors with a significant number of small pieces of equipment (e.g., electrical cabinet

The site-specific seismic response analysis of R/B-PCCV building structure addresses factors that affect the response of the combined soil-structure dynamic system that include, but are not limited to, the following:

- Properties and layering of the soil, including fill concrete and backfill modeled depending on its horizontal extent
- Depth of the water table
- Basemat embedment
- Flexibility of the basemat
- Presence of nearby structures

Up-to-date modeling techniques capable of capturing the various site-specific SSI effects are used for the analysis. The computer program SASSI is used for the site-specific SSI analysis, because it is based on the use of the FE technique and sub-structuring method with frequency-dependent impedance functions to model the interaction of the embedded flexible basemat with the surrounding soil.

The input used for the site-specific analysis must be derived from geotechnical and seismological investigations of the site. The input control motion that is derived from the site-specific GMRS, is applied in the SASSI analysis as within motion at the bottom of the basemat. Site-specific SSI analyses account for the uncertainties and variations of the subgrade properties by using at least three sets of site profiles that represent the best estimate, lower bound, and upper bound (BE, LB, and UB for equations, respectively) soil and rock properties. If sufficient and adequate soil investigation data are available, the LB and UB values of the initial (small strain) soil properties are established to cover the mean plus or minus one standard deviation for every layer. In accordance with the specific guidelines for SSI analysis contained in Section II.4 of SRP 3.7.2 (Reference 3.7-16), the LB and UB values for initial soil shear modulii (G_s) are established as follows:

$$G_s^{(LB)} = \frac{G_s^{(BE)}}{(1+C_v)} \quad \text{and} \quad G_s^{(UB)} = G_s^{(BE)} \quad (1+C_v)$$

For well investigated sites, the C_v should be no less than 0.5. For sites that are not well investigated, the C_v for shear modulus shall be at least 1.0.

The SSI analysis must use stiffness and damping properties of the subgrade materials that are compatible with the strains generated by the site-specific design earthquake (SSE or/and OBE). However, soil material damping shall not exceed 15% as stipulated in SRP 3.7.1 (Reference 3.7-10). ~~The soil properties may be considered strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, t~~The COL Applicant ~~must institute dynamic testing~~ is to evaluate the strain-dependent variation of the material dynamic properties for site materials ~~with initial shear wave velocities below 3,500 ft/s~~. If the strains in the subgrade media are less than 2%, the strain compatible properties can be obtained from equivalent linear site-response analyses using soil degradation curves. Degradation curves that are published in literature can be used after demonstrating their

Time history analysis of seismic systems is discussed in Subsection 3.7.2. The time history seismic analysis of a subsystem can be performed by simultaneously applying the displacements and rotations at the interface point(s) between the subsystem and the system. These displacements and rotations are the results obtained from a model of a larger subsystem or a system that includes a simplified representation of the subsystem.

The choice of applied seismic analysis method depends on the desired level of precision and the level of complexity of the particular subsystem being designed. The equivalent static load method of analysis is predominantly used for civil structure-related seismic subsystems and is generally the preferred method because it is relatively simple and at least as conservative as the other more detailed methods. For example, the equivalent static load analysis method is generally used for miscellaneous steel platforms, stairs, and walkways, reinforced masonry block walls and enclosures, HVAC ducts and duct supports, electrical tray and tray supports, and conduits and conduit supports.

The time history or response spectra generated at the support point of the subsystem are utilized as the input motion for performing the seismic dynamic analysis of the subsystem. However, where these data are not readily available, the data generated for a distance away from the structural support point may be used. To account for the structural linkage (i.e., intervening structural element) between these two locations, the additional amplification of the response due to the presence of the intervening structural element can be calculated and the remote input motion can be transformed. For cases where the intervening structure is rigid (i.e., frequency > 50 Hz), the transformation can be achieved by adding the effect due to the rigid body motion of the intervening structure to the existing input motion at the remote location. The new translational time history at the interface location is generated by algebraic summation of the translational acceleration time history at the reference location and the time-history contribution arising from the rocking and torsional effects of the intervening structural element. The new translational response spectra are obtained by absolute sum of the translational response spectra at the reference location and the contributions arising from the rocking and torsional effects of the intervening structural element. For places where the intervening structural element is judged to be flexible, the new ISRS are generated by incorporating the flexibility of the intervening structural element. Or alternatively, the seismic dynamic analysis of the subsystem shall be expanded to include the flexibility of the intervening structural element.

Torsional effects due to the significant effect of eccentric masses connected to a subsystem are included in the subsystem analysis. For rigid components (i.e., those with natural frequencies greater than the ZPA cutoff frequency of 50 Hz), the lumped mass is modeled at the center of gravity of the component with a rigid link to the appropriate point in the subsystem. For flexible components having frequency less than the ZPA, the subsystem model is expanded to include an appropriate model of the component.

Regardless of the method chosen, to avoid resonance, the fundamental frequencies of components and equipment are preferably selected to be less than one half or more than twice the dominant frequencies of the support structure. If this is not practical, equipment and components with fundamental frequencies within this range are designed for any associated resonance effects in conjunction with all other applicable loads.

The equivalent static load method of analysis and the various modal response spectra analysis methods are described in the following subsections.

SASSI analysis with the exception that no stick model is required. Instead, plate elements are to be directly included to represent the tunnel in the SASSI model.

3.7.3.8 Methods for Seismic Analysis of Category I Concrete Dams

The US-APWR standard plant design does not include dams. It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.

3.7.3.9 Methods for Seismic Analysis of Aboveground Tanks

It is the responsibility of the COL Applicant to design seismic category I below- or above-ground liquid-retaining metal tanks such that they are enclosed by a tornado missile protecting concrete vault or wall, in order to confine the emergency gas turbine fuel supply.

The other seismic category I liquid-retaining vessels utilized in the design are reinforced concrete vessels whose walls and floors form part of the building structural framework, including the following:

- Spent fuel pit, located in the R/B with top of vessel at level 4F
- Refueling cavity, located in PCCV with top of vessel at level 4F
- Fuel transfer canal, which connects the spent fuel pit and refueling cavity
- Cask washdown pit located in the R/B with top of vessel at level 4F
- Cask loading pit and fuel inspection pit located in the R/B and connected to the spent fuel pit with a canal, with tops of vessels at level 4F
- New fuel storage pit located in the R/B with top of vessel at level 4F
- Refueling water storage pit, located in PCCV below level 2F

Hydrodynamic loads ~~including sloshing loads~~ on these liquid-retaining vessels are determined using methods that conform to the provisions of Subsection II.14 of SRP 3.7.3 (Reference 3.7-35) and guidance of ASCE 4-98, Subsection 3.5.4 (Reference 3.7-9). The horizontal response analysis considers both the impulsive mode (in which a portion of the water moves in unison with the tank wall) and the horizontal ~~sloshing~~ convective mode (water motion associated with wave oscillation). The seismic ~~sloshing~~ analysis of convective hydrodynamic effects also considers ~~potential slosh heights~~ the maximum wave oscillation with respect to the potential of creating flooding, which is discussed in Section 3.4.

3.7.4 Seismic Instrumentation

The proposed seismic instrumentation program for the US-APWR is in accordance with NUREG-0800, SRP 3.7.4 (Reference 3.7-39) and all aspects of 10 CFR 50, Appendix S (Reference 3.7-7), which requires that “suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be

COL3.7(5) The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S, and the site-specific response spectra obtained from the response analysis.

COL3.7(6) The COL Applicant is to develop site-specific GMRS and FIRS by an analysis methodology, which accounts for the upward propagation of the GMRS. The FIRS are compared to the CSDRS to assure that the US-APWR standard plant seismic design is valid for a particular site. If the FIRS are not enveloped by the CSDRS, the US-APWR standard plant seismic design is modified as part of the COLA in order to validate the US-APWR for installation at that site.

COL3.7(7) The COL Applicant is to determine the allowable dynamic bearing capacity based on site conditions, including the properties of fill concrete placed to provide a level surface that matches the bottom of foundation elevations, and to evaluate the bearing load to this capacity.

COL3.7(8) ~~The soil properties may be considered strain-independent for subgrade materials with initial shear wave velocities of 3,500 ft/s or higher, to be confirmed by the COL Applicant as part of the site-specific subsurface material investigations discussed in Section 2.5.4. However, t~~The COL Applicant ~~must institute dynamic testing~~ is to evaluate the strain-dependent variation of the material dynamic properties for site materials ~~with initial shear wave velocities below 3,500 ft/s~~.

COL3.7(9) The COL Applicant is to assure that the design or location of any site-specific seismic category I SSCs, for example pipe tunnels or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of non-seismic category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, non safety-related storage tanks, etc.

COL3.7(10) It is the responsibility of the COL Applicant to further address structure-to-structure interaction if the specific site conditions can be important for the seismic response of particular US-APWR seismic category I structures, or may result in exceedance of assumed pressure distributions used for the US-APWR standard plant design.

COL3.7(11) Deleted

Table 3.7.1-3 Major Dimensions of Seismic Category I Structures

Structure	Basemat Embedment Depth Below Grade (ft)	Basemat Width and Length (ft)	Max. Structure Height
R/B	~~26'-8"/~~38'-10"	210' x 309'(3)	190' - 9"
PCCV	See note 2.	See note 2.	268' - 3"
Containment Internal Structure	See note 2.	See note 2.	~~139' - 6"~~ 175'-9" (top of pressurizer compartment)
PS/B	~~37'-3"~~ 38'-10"	~~71' x 117'~~ (66'-0") x (111'-6")(3)	~~51'-11"~~ 87'-4"

Notes:

1. The dimensions shown are approximate and are based on the general arrangement drawings in Section 1.2.
2. The R/B, PCCV, and containment internal structure rest on a common basemat as shown on the general arrangement drawings in Section 1.2.
3. Width and ~~height~~ length are the distances between column lines of exterior walls.

utilized for design of individual members. Equivalent dead loads are used during global analyses as conservative uniform load allowances of minor equipment and distribution systems, including small bore piping.

3.8.4.3.1.1 Dead Loads (Uniform and/or Concentrated)

Dead loads include the weight of structures such as slabs, roofs, decking, framing (beams, columns, bracing, and walls), and the weight of permanently attached major equipment, tanks, machinery, cranes, elevators, etc. The deadweight of equipment is based on its bounding operating condition including the weight of fluids. In addition, permanently attached non-structural elements such as siding, partitions, and insulation are included. Dead loads of cranes and elevators do not include the rated capacity lift or impact.

3.8.4.3.1.2 Equivalent Dead Load (Uniform)

Equivalent dead load includes the weight of minor equipment not specifically included in the dead load defined in Subsection 3.8.4.3 and the weight of piping, cables and cable trays, ducts, and their supports. It also includes fluid contained within the piping and minor equipment under operating conditions. Floors are checked for the actual equipment loads. To account for permanently attached small equipment, piping, ductwork and cable trays, a minimum equivalent dead load of 50 lb/ft^2 is applied. Where piping, ductwork, or cable trays are supported from platforms or walkway beams, actual loads may be determined and used in lieu of a conservative loading.

For floors with a significant number of small pieces of equipment (e.g., electrical cabinet rooms), the equivalent dead load is determined by dividing the total equipment weight by the floor area that effectively supports the equipment within the room, plus an additional 50 lb/ft^2.

3.8.4.3.2 Liquid Loads (F)

The vertical and lateral pressures of liquids are treated as dead loads except for external pressures due to ground water which are treated as live loads. The effects of buoyancy and flooding on SSCs are considered, where applicable. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads. Impulsive and convective ~~H~~hydrodynamic loads due to seismic events ~~sloshing~~ are determined as discussed in Subsection 3.7.3.9, and included in the earthquake load as described in Subsection 3.8.4.3.6. For the purposes of evaluating flotation in Subsection 3.8.5.3, F_b is the buoyant force of the design-basis flood or high ground water table, whichever is greater.

3.8.4.3.3 Earth Pressure (H)

A static earth pressure acting on the structures during normal operation, considered as fully saturated to account for ground and flood water levels, is included in the analysis as *H*. The dynamic soil pressure, induced during an SSE event, is considered as an earthquake load E_{ss}.

3.8.4.3.5 Wind Load

3.8.4.3.5.1 Design Wind (W)

The design wind is determined as discussed in Subsection 3.3.1 for values specified in Chapter 2. Wind loads are not combined with seismic loads.

3.8.4.3.5.2 Tornado Load (Wt)

The design for tornado loads is in accordance with Subsection 3.3.2 for values specified in Chapter 2. In addition, extreme winds such as hurricanes and tornadoes have the potential to generate missiles. Missiles generated by tornadoes and extreme winds are listed in Subsection 3.5.1.4 and barrier design for missiles is discussed in Subsection 3.5.3. These subsections describe the determination of tornado loads applicable to the protection of safety-related equipment.

3.8.4.3.6 Seismic Loads

3.8.4.3.6.1 Operating Basis (E_{ob})

For seismic category I SSCs whose design is site-specific, that is, not included in the seismic design of the US-APWR standard plant, OBE loading has to be considered only if the value of site-specific OBE is set higher than 1/3 of the site-specific SSE. Therefore, the site-specific seismic design does not have to consider OBE loads if the OBE spectra are enveloped by 1/3 of the site-specific foundation input response spectra and ground motion response spectra.

3.8.4.3.6.2 Safe Shutdown (E_{ss})

E_{ss} is defined as the loads generated by the SSE specified for the plant, including the associated hydrodynamic loads and dynamic incremental soil pressure (based on three-dimensional SSI analysis results). Earthquake loads (E_{ss}), are derived for evaluation of seismic category I structures using ground motion accelerations in accordance with Section 3.7.

Seismic dynamic analyses of the buildings consider the dead load and the equivalent dead loads as the accelerated mass. In addition to the dead load, 25% of the floor live load during normal operation ~~or~~ and 75% of the roof snow load, whichever is applicable, is also considered as accelerated mass in the seismic models.

For the local design of members loaded individually, such as the floors and beams, seismic member forces include the vertical response due to masses equal to 50% of the specified floor live loads instead of 25% of floor live load, as follows:

$$a_v(0.5L)$$

where

a_v	=	Vertical seismic acceleration obtained from the seismic dynamic analysis results
L	=	Floor live load per Subsection 3.8.4.3.4

The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.

3.8.4.4.4 Seismic Category II Structures

Seismic category II structures need not remain functional during and after an SSE. However, such structures must not fall or displace to the point they could damage seismic category I SSCs.

Seismic Category II structures and subsystems are analyzed and designed using the same methods and stress limits specified for seismic Category I structures and subsystems, and the same load combinations and stress coefficients given in Table 3.8.4-4~~, except where noted therein~~.

3.8.4.5 Structural Acceptance Criteria

Structural acceptance criteria are listed in Table 3.8.4-3 for concrete structures and in Table 3.8.4-4 for steel structures, and are in accordance with ACI-349 (Reference 3.8-8) and AISC N690 (Reference 3.8-9), except as provided in the table notes.

The deflection of the structural members is limited to the maximum values as specified in ACI-349 (Reference 3.8-8) and AISC N690 (Reference 3.8-9), as applicable.

Subsection 3.8.5.5 identifies acceptance criteria applicable to additional basemat load combinations.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The following information pertains to the materials, quality control programs, and any special construction techniques utilized in the construction of the seismic category I structures for the US-APWR.

3.8.4.6.1 Materials

The major materials of construction in seismic category I structures are concrete, grout, steel reinforcement bars, splices of steel reinforcing bars, structural steel shapes, and anchors.

3.8.4.6.1.1 Concrete

Concrete utilized in standard plant seismic category I structures, other than PCCV and upper part of the tendon gallery in the basemat, has a compressive strength of f'_c = 4,000 psi. Concrete utilized in the PCCV and upper part of the tendon gallery in the basemat has a compressive strength of f'_c = 7,000 psi and is subject to the PCCV material requirements in Subsection 3.8.1.6, including the requirements of ASME III, Division 2 (Reference 3.8-2), as shown in Figure 3.8.5-4. The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures. A test age of 28 days is used for normal concrete. Batching and placement of concrete is performed in accordance with ACI 349 (Reference 3.8-8), ACI 304R (Reference 3.8-38), and ASTM C 94 (Reference 3.8-42). During construction, volume changes in mass concrete are controlled where necessary by applying measures and provisions outlined

3.8.4.6.2 Quality Control

Chapter 17 details the quality assurance program for the US-APWR.

3.8.4.6.3 Special Construction Techniques

Standard provisions of ACI are to be applied where necessary to address issues related to the use of massive concrete pours. As stated in Subsection 3.8.4.6.1.1, volume changes in mass concrete are controlled where necessary by applying measures and provisions outlined in ACI 207.2R (Reference 3.8-52) and ACI 207.4R (Reference 3.8-53). The following summarizes the construction techniques commonly associated, either singularly or in combination, with massive concrete pours such as basemats:

- Limit the size of concrete pour.
- Use a checkerboard pattern of concrete placement in a single lift. To avoid a weak horizontal shear plane, a double lift placement of concrete, in general, is avoided. However, when it is absolutely needed to have two lifts, adequate design considerations and also, in general, shear stirrups are provided.
- Schedule concrete pours for the most advantageous day and time to control temperature rise in the concrete.
- Post-cooling can be performed by cooling the freshly placed concrete with running chilled water lines in the concrete.

~~There are no special construction techniques utilized in the construction of other seismic category I structures.~~

3.8.4.7 Testing and Inservice Inspection Requirements

Seismic category I structures, except the PCCV, are monitored in accordance with paragraph (a)(2) of 10 CFR 50.65 (Reference 3.8-29), provided there is not significant degradation of the structure. Condition monitoring, is similar to that performed as part of the inservice inspection activities required by the ASME codes, is applied to these structures. The condition of all structures is assessed periodically. The appropriate frequency of the assessments is commensurate with the safety significance of the structure and its condition.

The COL Applicant is to establish a site-specific program for monitoring and maintenance of seismic category I structures in accordance with the requirements of NUMARC 93-01 (Reference 3.8-28) and 10 CFR 50.65 (Reference 3.8-29) as detailed in RG 1.160 (Reference 3.8-30). For seismic category I structures, monitoring is to include base settlements and differential displacements.

For water control structures, ISI programs are acceptable if in accordance with RG 1.127 (Reference 3.8-47). Water control structures covered by this program include concrete structures, embankment structures, spillway structures, outlet works, reservoirs, cooling water channels, canals and intake and discharge structures, and safety and performance instrumentation.

include acceptance criteria for overturning, sliding, and flotation as detailed in Table 3.8.5-1. The non-ASME portion of the basemat is designed in accordance with ACI-349 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19), where applicable. The reinforced concrete basemat for the PCCV and enveloped containment internal structure are designed in accordance with ASME Code Section III, Division 2, Subsection CC (Reference 3.8-2). Figure 3.8.5-4 delineates basemat regions applicable to each Code.

3.8.5.4 Design and Analysis Procedures

Based on the premise that seismic category I buildings basemats are not supported on bedrock, a computer analysis of the SSI is performed for static and dynamic loads. Subsection 3.7.2 provides further information. Two types of SSI analyses are required for the R/B and the PS/Bs: an overall seismic analysis of the building for the superstructure design, and a local analysis of the basemat for its design. For the basemat design, the basemat is modeled using solid finite elements with springs representing the subgrade.

The seismic category I structures are concrete, shear-wall structures consisting of vertical shear/bearing walls and horizontal floor slabs designed to SSE accelerations as discussed in Section 3.7. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the loads between them.

The reinforced concrete basemat for the PCCV and enveloped containment internal structure are designed in accordance with ASME Code Section III, Division 2, Subsection CC (Reference 3.8-2). Other seismic category I basemats of reinforced concrete are designed in accordance with ACI-349 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19) where applicable. Table 3.8.5-2 identifies the material properties of concrete and Figure 3.8.5-4 delineates the governing codes based on region of the R/B, PCCV and containment internal structure basemat.

3.8.5.4.1 Properties of Subgrade

For purposes of the US-APWR standard design, the SSI effects are captured by considering three generic subgrade types utilizing frequency independent springs. A fourth subgrade condition is also considered, that of a foundation resting on hard rock. For the fourth condition, it is not necessary to consider SSI effects because the foundation is considered to be resting on a fixed base that is rigid. Subsection 3.7.2.4 provides further discussion relating to SSI and the selection of subgrade types.

The four supporting media (subgrade) conditions for the US-APWR design are provided in Table 3.8.5-3.

The properties of conditions provided in Table 3.8.5-3 are considered to represent stiffness properties of the subgrade material that are compatible to the strains generated in the soil by the input design ground motion. ~~The dissipation of energy in the subgrade media due to the soil material damping is conservatively neglected.~~

Table 3.8.4-4 Load Combinations and Load Factors for Seismic Category I Steel Structures (Sheet 2 of 2)

Notes:

1. Coefficients are applicable to primary stress limits given in ANSI/AISC N690-1994 Sections Q1.5.1, Q1.5.2, Q1.5.3, Q1.5.4, Q1.5.5, Q1.6, Q1.10, and Q1.11. Calculated stresses shall not exceed allowable stresses for each of the load combinations shown in this table.
2. In no instance shall the allowable stress exceed $0.7F_u$ in axial tension nor $0.7F_u$ times the ratio Z/S for tension plus bending.
3. For primary plus secondary stress, the allowable limits are increased by a factor of 1.5.
4. The maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, is used in load combinations 9 through 11, unless an appropriate time history analysis is performed to justify otherwise.
5. In combining loads from a postulated high-energy pipe break accident and a seismic event, the SRSS may be used, provided that the responses are calculated on a linear basis.
6. All load combinations is checked for a no-live-load condition
7. In load combinations 7 through 11, the stress limit coefficient in shear shall not exceed 1.4 in members and bolts.
8. Secondary stresses which are used to limit primary stresses are treated as primary stresses.
9. Consideration is also given to snow and other loads as defined in ASCE 7.
10. This load combination is to be used when the global (non-transient) sustained effects of T_a are considered.
11. The stress limit coefficient where axial compression exceeds 20% of normal allowable, is 1.5 for load combinations 7, 8, 9, 9a, and 10, and 1.6 for load combination 11. For ~~seismic category II members the stress limit coefficient applicable to axial and bending stresses for~~ load combinations 7 through 11 ~~is 1.7, however~~ the allowable stress shall not exceed 1.0 F_y.
12. Load combinations and stress limit coefficients are applicable for AISI design of cold-formed steel structural members used in subsystem supports. Allowable strengths per AISI may be increased by the stress limit coefficients shown, subject to the limits noted in this table. The allowable strength shall equal or exceed the required strength calculated, in accordance with AISI, for each of the load combinations shown in this table.

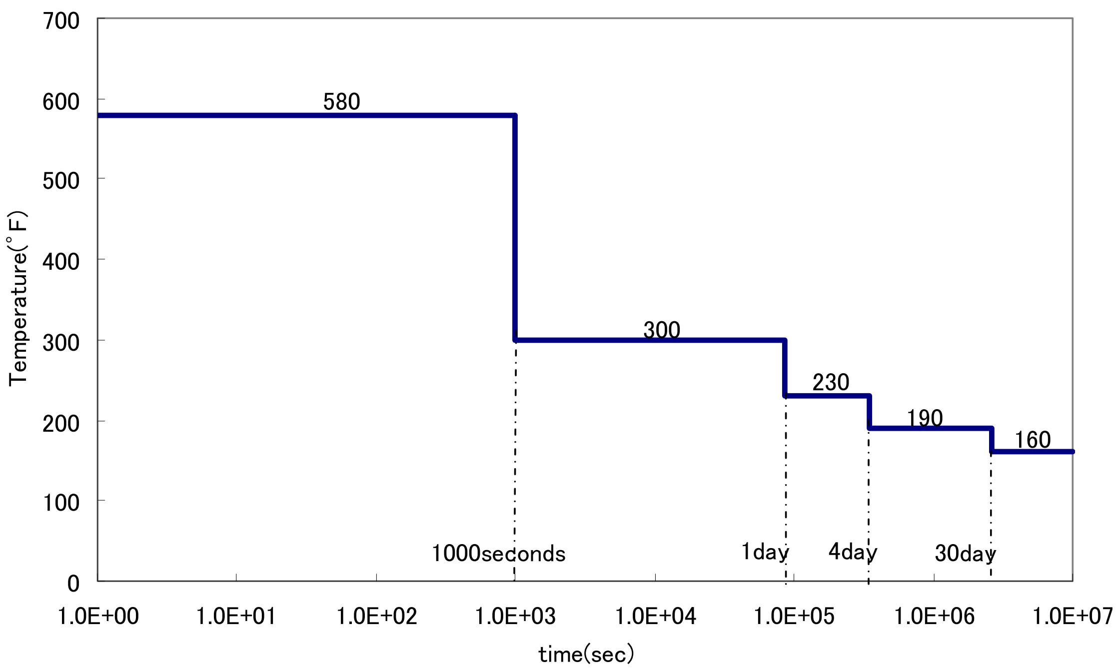

Figure 3.8.1-13 Transient Conditions of Temperature of the ~~Refueling~~ Reactor Cavity Atmosphere and Sump Pool Water (Pipe Break in the ~~Refueling~~ Reactor Cavity)

assessment program. Detail of the analysis is described in Reference 3.9-22. The evaluation of the SG is described in Subsection 5.4.2.1.

The design of the US-APWR steam delivery system (including the safety relief valves and the steam separator) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the United States and around the world. The US-APWR steam delivery system is designed using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the USA for more than 20 years with sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the structural and vibration design bases are proven. This non-safety-related steam delivery system will not experience excessive vibration; therefore, the analysis of the flow excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes) is not expected.

3.9.2.3.1 Classification of Reactor Internals in Accordance with the Comprehensive Vibration Assessment Program

The US-APWR reactor internals components are evolved from that of the well-proven current 4-loop plant design operating in United States and Japan. The differences are as follows:

- Design: the US-APWR uses neutron reflector instead of baffles
- Size: there are increases in the diameters of RV, core barrel and the secondary core support assembly
- Arrangement: RCCA guide tubes and upper support columns in the upper plenum
- Operating conditions: there is an increase in flow rate

The US-APWR reactor internals represent a unique, first of a kind design because of its design, size, arrangements and operating conditions. Therefore, the first US-APWR will be classified as a Prototype in accordance with Regulatory Guide 1.20 (Reference 3.9-21). Upon qualification of the first US-APWR as a valid prototype, subsequent plants will be classified as Non-Prototype Category I.

3.9.2.3.2 Comparative Analysis of the US-APWR and the Current Plant

In this section, flow-induced vibration characteristics of the US-APWR reactor internals are assessed in comparison to those of the current 4-loop plant. Subsection 3.9.5 provides general information on the reactor internals.

- **General**

  The basic design of the US-APWR reactor internals follows that of the current 4-loop plant but features a larger core barrel diameter and a neutron reflector instead of a baffle structure. However, the coolant flow velocities are carefully designed to remain the same as those in the current 4-loop plant so that any increase in the excitation force due to a larger surface area exposed to the coolant

Typically stress criteria for ductwork and supports results in selection of standard member sizes and maximum span lengths. Those HVAC systems that do not satisfy the parameters qualified for standard member sizes and maximum span lengths are designed to satisfy their specific load and operating conditions. Pressures due to flow velocity are based on the operability requirements of each HVAC system.

3A.1.2 Seismic Category II Ductwork

Seismic category II ductwork is not essential for the safe shutdown of the plant and need not remain functional during, and after, a SSE. However, such ductwork and supports must not fall or displace excessively where it could damage any seismic category I structures, systems, and components (SSCs). Seismic category II ductwork and supports, including support anchorages, are therefore analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems~~, except where noted~~ in Table 3.8.4-4.

3A.2 Applicable Codes, Standards and Specifications

The design and construction of seismic category I HVAC systems conform to AG-1-2003, Code on Nuclear Air and Gas Treatment, including Addendum AG-1a and AG-1b (Reference 3A-8). Sheet metal ducts are constructed in accordance with the American National Standards Institute (ANSI)/Sheet Metal and Air Conditioning Contractors National Association (SMACNA), HVAC Duct Construction Standards – Metal and Flexible (Reference 3A-1). The American Iron and Steel Institute (AISI), Specification for the Design of Cold-Formed Steel Members (Reference 3A-2), provides the methodology for evaluating the effects of shear lag and plate buckling appropriate for this type of duct construction. Structural steel duct supports are designed and constructed in accordance with the American Institute of Steel Construction (AISC) Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities (Reference 3A-3) or AISI as applicable.

Schedule round pipe used as ductwork is not discussed within this Appendix. Codes, standards, and specifications applicable to schedule pipe is in accordance with piping and pipe support criteria in Sections 3.9 and 3.12.

3A.3 Loads and Load Combinations

3A.3.1 Loads

Supports are designed for dead, seismic, thermal loads, and airflow forces at duct elbows, as applicable. Ducts are also designed for the operational and accident pressure loads. Construction live load is considered, however, it is not present during design seismic events. In addition, any accessory loads to the duct or supports are included in the qualification of the duct and duct supports.

The following loads are applicable for the ductwork load combinations:

ADL Additional dynamic loads resulting from system excitations due to structural motion, such as that caused by safety relief valve actuation and other hydrodynamic loads due to the design basis accident (DBA), small pipe break accident (SBA), and intermediate pipe break accident (IBA).

FIGURES

The required J-resistance curve relationship has been established based on the use of ASME Code modulus of elasticity and minimum strength properties at 550ºF in the fracture mechanics analysis. Review of the curves against actual test data from the literature (e.g. ~~as documented in~~ Appendix B of NUREG/CR-6004 ~~–~~ [Reference 3B-~~4~~13] and Pipe Fracture Encyclopedia, Test Data – Volume 3 [Reference 3B-14]) has shown that the J-T curves should be achievable. However, there is limited valid test data for material representative of the main steam line and its thickness.

It has been established that higher stress factors (and the associated lower J-T curves) will produce essentially equivalent results at the lower normal stress part of the BAC curves, and use of higher strength materials produce slightly higher BAC curves at higher stresses. Thus, use of Code minimum properties in establishing the BAC is conservative.

3B.3 LBB Evaluation for the US–APWR

The LBB evaluation method applied is briefly described below according to SRP 3.6.3 (Reference 3B-2).

In the LBB concept, it is necessary to detect a leak at normal operation to prevent the piping system from failure at the postulated maximum load. Therefore, both the stress under normal operation and the maximum load are required for evaluation.

(1) Applied load

a. Load under normal operation

The evaluation of crack opening area for the estimation of the leak rate is conducted using the stress under normal full power plant operation. The load is produced by internal pressure, dead weight, and thermal expansion.

$$F = F_{DW} + F_{Th} + F_P \quad (3B.3\text{-}1)$$

$$M = \sqrt{\left((M_X)^2 + (M_Y)^2 + (M_Z)^2\right)} \quad (3B.3\text{-}2)$$

$$M_X = (M_X)_{DW} + (M_X)_{Th}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{Th}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{Th}$$

where

F = Axial force

M = Bending moment

The subscripts indicate the following loads

DW = Dead weight

Th = Thermal expansion

P = Internal pressure

3B-5 Advances in Elastic-Plastic Fracture Mechanics. NP-3607, Final Report, Electric Power Research Institute, 1984.

3B-6 Robert E. Henry, The Two-Phase Critical Discharge of Initially Saturated or Subcooled Liquid. Nuclear Science and Engineering, Vol. 41, pp.336-342, 1970.

3B-7 Calculation of Leak Rates Through Cracks in Pipes and Tubes. NP-3395, Electric Power Research Institute, 1983.

3B-8 Part D Properties, ASME Boiler and Pressure Vessel Code, Section II, 2001 (Addenda 2003), American Society of Mechanical Engineers.

3B-9 Elastic-Plastic Fracture Mechanics Analysis of Through-Wall and Surface Flaws in Cylinders. NP-4496, Final Report, Electric Power Research Institute, 1988.

3B-10 Cofie, N.G., Miessi, G.A., and Deardorff, A.F., Stress-Strain Parameters in Elastic-Plastic Fracture Mechanics, Transactions of the 10th International Conference on Structural Mechanics in Reactor Technology, Volume L – Inelastic Behavior of Metals and Constitutive Laws of Materials, pp 91-96, 1989.

3B-11 State of the Art Report on Piping Fracture Mechanics, NUREG/CR-6540, U.S. Nuclear Regulatory Commission, November 1997.

3B-12 Assessment of Short Through-Wall Circumferential Cracks in Pipe, NUREG/CR-6235, U.S. Nuclear Regulatory Commission, April 1995.

3B-13 Probabilistic Pipe Fracture Evaluations for Leak-Rate Detection Applications. NUREG/CR-6004, U.S. Nuclear Regulatory Commission, April 1995.

3B-14 Pipe Fracture Encyclopedia, Test Data – Volume 3, U.S. Nuclear Regulatory Commission, December 1997.

Table 3B-2 List of BACs for LBB Evaluation

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)(1)	Pressure (psig)(1)	Inside Pipe		BAC Figure No.
										Water	Vapor	
1	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	615	2248	X		Figure 3B-6
2	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	615	2248	X		Figure 3B-7
3	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	551	2204	X		Figure 3B-8
4	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	551	2296	X		Figure 3B-9
5	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	551	2204	X		Figure 3B-10
6	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	551	2296	X		Figure 3B-11
7	RCS	Surge Line	16"-RCS-2501R B	16	16	1.594	SA-312 TP316	653	2248	X		Figure 3B-12
8	~~RCS~~	~~Surge Line~~	~~16"-RCS-2501R B~~	~~16~~	~~16~~	~~1.594~~	~~SA-312 TP316~~	~~449~~	~~400~~	~~X~~		Figure 3B-13 (deleted)
9	RCS	Residual Heat Removal System (RHRS) Hot Leg Branch Line off RCS	10"-RCS-2501R A,B,C,D, Hot Leg Side	10	10.75	1.125	SA-312 TP316	615	2248	X		Figure 3B-14
10	RCS	RHRS Cold Leg Branch Line off RCS	8"- RCS -2501R A,B,C,D (COLD LEG)	8	8.625	0.906	SA-312 TP316	551	2296	X		Figure 3B-15
11	SIS	Accumulator System	14"-RCS-2501R A,B,C,D	14	14	1.406	SA-312 TP316	551	2296	X		Figure 3B-16
12	RCS	Pressurizer Spray Line	6"-RCS-2501R B,C	6	6.625	0.719	SA-312 TP316	551	2296	X		Figure 3B-17
13	MSS	Main Steam Line	32"-MSS-1532 A,B,C,D	32	32	1.496	SA333 Gr.6	535	907		X	Figure 3B-18

Notes:
1. Conditions from Reactor Coolant System DCD, Table 5.1-2.
2. Use conservative lower 2243 psig for leakage which is the pressurizer end pressure.
3. Use conservative higher 2296 condition of cold leg for critical flaw sizing and 2235 for leakage based on upper portion connected to pressurizer steam space
4. No leakage case required since this condition is only for critical flaw sizing.

Figure 3B-13 (deleted)

analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems~~, except where noted~~ in Table 3.8.4-4.

3F.2 Applicable Codes, Standards, and Specifications

Conduits are manufactured to satisfy the American National Standard Institute (ANSI) C80.1 American Standard for Electrical Rigid Steel Conduit (ERSC), (Reference 3F-1) or ANSI C80.5, American Standard for Electrical Rigid Aluminum Conduit (ERAC), (Reference 3F-2), as applicable. Junction boxes are manufactured to satisfy the National Electrical Manufacturer Association (NEMA) Standards Publication 250 Enclosures for Electrical Equipment (1000 Volts Maximum) (Reference 3F-3). Installation of the conduit system conforms to the requirements of the National Fire Protection Associations (NFPA) 70, National Electric Code (NEC), (Reference 3F-4).

The American Iron and Steel Institute (AISI) Specification for the Design of Cold-Formed Steel Members (Reference 3F-5) provides the methodology for structurally evaluating cold formed steel shapes, as applicable. Structural steel shapes used for supports are designed and constructed in accordance with the American Institute of Steel Construction (AISC) Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities (Reference 3F-6). Welding is evaluated and performed in accordance with the American Welding Society (AWS) Standard D1.1 Structural Welding Code, (Reference 3F-7).

3F.3 Loads and Load Combinations

3F.3.1 Loads

Conduit systems are designed for dead, seismic, and thermal loads, as applicable. Design dead load includes the working load (weight) of cables permitted in the conduit. In addition, any accessory loads to the conduit and conduit supports are included in the qualification of the conduit and conduit supports.

3F.3.2 Load Combinations

Refer to Subsection 3.8.4.3 for various load combinations applicable to seismic category I SSCs.

Seismic category II conduit and conduit supports are qualified for the applicable SSE to assure they do not damage any seismic category I SSCs by falling or displacing excessively under any seismic loads. Seismic category II conduit supports are, therefore, qualified for maximum seismic load combinations and associated allowable stresses as discussed in Subsection 3.8.4.3.

3F.4 Design and Analysis Procedures

Refer to Section 3.7 for seismic system analysis and qualification requirements of seismic category I and seismic category II SSCs and their supports.

3F.6.2 Structural Steel Shapes

The design, fabrication and installation of structural steel supports, and structural shapes and plates used in support construction, comply with AISC-N690-1994 (Reference 3F-6).

3F.6.3 Conduit

ERSC conforms to ANSI C80.1 (Reference 3F-1).

ERAC conforms to ANSI C80.5 (Reference 3F-2).

3F.6.4 Electrical Boxes

Electrical Boxes conform to NEMA Standards Publication 250 (Reference 3F-3).

3F.6.5 Welding

Welding electrodes are E70 series for structural steel shapes greater than 3/16th inch thick or E60 series for structural steel shapes less than or equal to 3/16th inch thick, in accordance with AWS A5 series specifications (Reference 3F-8).

3F.6.6 Anchor Bolts

Anchor bolts used for conduit supports, seismic category I and II, are expansion anchors qualified in accordance with ACI 355.2 (Reference 3F-9). The flexibility of base ~~plates~~ anchorage was considered in determining the anchor bolt loads.

3F.6.7 Bolts

Bolts used in conduit support, seismic category I and II; conform to American Society for Testing and Materials (ASTM) A-307 (Reference 3F-10).

3F.7 References

3F-1 American Standard for Electrical Rigid Steel Conduit (ERSC). ANSI C80.1-2005, American National Standard Institute, 2005.

3F-2 American Standard for Electrical Rigid Aluminum Conduit (EARC). ANSI C80.5-2005, American National Standard Institute, 2005.

3F-3 NEMA Standards Publication 250-2003 Enclosures for Electrical Equipment (1000 Volts Maximum). National Electrical Manufacturer Association, 2003.

3F-4 National Electric Code (NEC). NFPA 70, National Fire Protection Association, 1999.

3F-5 Specification for the Design of Cold-Formed Steel Members, Part 1 and 2. 1996 Edition and 2000 Supplement, American Iron and Steel Institute.

3G Seismic Qualification of Cable Trays and Supports

3G.1 Description

This appendix provides the methodology used to qualify the structural integrity of seismic category I and seismic category II electrical cable trays and cable tray supports (hereafter referred to as “cable tray systems”). Cable tray systems containing non-Class 1E cable in non-seismic structures are not required to be qualified to the requirements of this appendix.

In general, the design of cable trays and cable tray supports is accomplished through the following steps:

- Determine applicable load combinations and corresponding allowable stresses for trays and supports
- Limit spacing of tray supports to maintain tray stresses within allowable stresses corresponding to the applicable load combination
- Assure that the maximum stresses of tray supports are within allowable stressescorresponding to the applicable load combination
- Provide system bracing to control seismic movement and interaction with other seismic category I structures, systems, or components (SSCs).

3G.1.1 Seismic Category I Cable Tray Systems

Seismic category I cable tray systems are designed for all applicable load combinations to maintain structural integrity within stress limits. This is achieved by analyzing the cable tray system (tray, fittings, connectors, fasteners, supports, etc.) and limiting the support spacing to maintain critical stresses to acceptably low levels. The seismic qualification of cable tray systems is to satisfy the safe-shutdown earthquake (SSE) requirements of the structure in which they are contained. Seismic category I cable tray systems, including support anchorages, in US-APWR standard plant seismic category I structures are analyzed and designed for a SSE which is equivalent to the in-structure response spectra developed from the certified seismic design response spectra (CSDRS). Site-specific seismic category I structures are analyzed and designed using as a minimum the site-specific SSE developed from the site-specific ground motion response spectra (GMRS) and foundation input response spectra (FIRS).

3G.1.2 Seismic Category II Cable Tray Systems

Seismic category II cable tray systems are designed to verify that the items will not fall or displace excessively where it could damage any seismic category I SSCs during, and after, a SSE. Seismic category II cable tray systems including support anchorages are, therefore, analyzed and designed for the applicable SSE, such as in-structure response spectra developed from the CSDRS within the standard plant Reactor Building and the East and West Power Source Buildings using the same methods and stress limits specified for seismic category I cable tray systems~~, except where noted~~ in Table 3.8.4-4.

Chapter 5

US-APWR DCD Chapter 5 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-40	Subsection 5.2.5.4.1.1	RAI: No. 521, 14.02-120 Changed “A leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators as stated in positions 5 and 7 of regulatory guide 1.45.” to “A leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators, consistent with regulatory positions 2.2 and 3.3 of regulatory guide 1.45.”
5.2-43	Subsection 5.2.5.7	RAI: No. 521, 14.02-120 Added to the beginning of the first paragraph as follows: “Consistent with Regulatory Position C.2.5 of RG 1.45, leakage monitoring systems, including those with location detection capability, have provisions to permit calibration and testing during plant operation, as appropriate.

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.2-44	Subsection 5.2.5.8	RAI: No. 521, 14.02-120 Changed “In accordance with the position 9 of regulatory guide 1.45 the limiting condition for identified and unidentified reactor coolant leakages are identified in the Chapter 16. Subsections 3.4.13 addresses RCS leak limits. Subsection 3.4.15 addresses RCS leak detection instrument requirements. The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the Technical Specification (TS) limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value.” to “In accordance with the position 4.1 of regulatory guide 1.45, the limiting conditions for identified, unidentified, RCPB and intersystem reactor coolant leakages are identified in the Chapter 16 Technical Specifications (TS). Subsections 3.4.13 and 3.4.14 address RCS operational leakage and pressure isolation valve (intersystem), leak limits, respectively. Subsection 3.4.15 addresses RCS leak detection instrument requirements. The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the TS limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value. In accordance with the guidance in RG 1.45 position C.2.1, the procedure includes the collection of leakage to the containment from unidentified sources so the total flow rate can be detected, monitored and quantified for flow rates greater than or equal to 0.05 gal/min.”

US-APWR DCD Chapter 5 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
5.4-2	Subsection 5.4.1.1.2, Third paragraph	RAI: No. 274, 05.04.01.01-3 Changed “The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel should be inspected for critical dimensions after the over speed test so that any dimensional changes can be detected. Qualified test procedure and the acceptance criteria should be decided with respect to this test procedure.” to “The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel will be inspected for critical dimensions after the over speed test so that any dimensional changes can be detected. With respect this test procedure, it should be decided qualified test procedure and the acceptance criteria.”
5.4-42	Subsection 5.4.7.2.3.1, Second paragraph, last sentence	Editorial: clarify the language Changed “Once the pressurizer steam bubble formation is complete, the RHRS would be isolated from the RCS.” to “Once the pressurizer steam bubble formation is complete, the RHRS is isolated from the RCS.”

Additionally, humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment. They do not quantify the reactor coolant leakage.

5.2.5.4.1 System Description of Unidentified Leakage detection

5.2.5.4.1.1 Containment Sump Level and Flow Monitoring System

Any leakage inside the containment from the RCPB and other components, not otherwise identified, condenses and flows by gravity through the floor drains and other drains to the containment sump, where the sump level meter measures the increase in the sump level indicating the leak rates. Indication of increasing sump level is transmitted from the sump to the MCR by means of a sump level transmitter and recorded.

A leak rate greater than or equal to 0.5 gpm is detectable within one hour, with an alarm actuating in the MCR to alert the operators, consistent with ~~as stated in~~ regulatory positions ~~5~~2.2 and ~~7~~3.3 of regulatory guide 1.45.

The sump level monitoring system is qualified for a safe shutdown earthquake.

5.2.5.4.1.2 Containment Airborne Particulate Radioactivity Monitor

In US-APWR, this monitor corresponds to the containment radiation monitor (RMS-RE-040). Refer to Chapter 11, Subsection 11.5.2. The containment airborne particulate radioactivity monitor performs continuous sampling of the containment air and measures the radiation level in the particulate. This monitor is qualified for a safe-shutdown earthquake (SSE). An air sample is drawn outside the containment and passed through a gamma monitor that monitors its gamma rays in radioactive particulate. After passing through the monitor, the sample is returned via the closed system to the containment atmosphere. The measuring range for the monitor is from $1\times10^{-10}\ \mu$Ci /cm^3. An indication of the monitor counting rate is provided to the MCR and electronically recorded.

The detection sensitivity of the airborne particulate radioactivity monitor for reactor coolant leak rate depends on conditions, such as radioactive concentration in the reactor coolant and a distribution coefficient of radioactive particles to the containment atmosphere.

In addition, provided that a radioactive concentration of airborne particulate in the containment is within the measuring range of the airborne particulate radioactivity monitor, an alarm is adjustable to actuate upon detection of a severalfold increase.

Assuming that corrosion and activation product concentration in the reactor coolant is $2\times10^{-1}\ \mu$Ci /g (Na-24,Cr-51,Zn-65,Mn-54,56, Co-58,60, Fe-55,59) and the distribution coefficient is 0.3, after leak occurrence, a leak rate of 0.5 gpm can be detected within one hour.

C. Containment air cooler condensate flow rate monitoring system - standpipe level

D. Containment sump level and flow monitoring system – sump level

E. Gross leakage detection methods - charging flow rate, letdown flow rate, pressurizer level, VCT level and reactor coolant temperatures are available as inputs for detection by RCS inventory balance. Containment sump levels and pump operation are also available. Total makeup water flow is available from the plant computer for liquid inventory.

F. Containment temperature, pressure, and humidity will only have readouts in the MCR and alarms to indicate occurrence of leakage within the containment. This method is used only to detect leaks and is not used to quantity leak rates.

5.2.5.7 Testing, Calibration and Inspection Requirements

Consistent with Regulatory Position C.2.5 of RG 1.45, leakage monitoring systems, including those with location detection capability, have provisions to permit calibration and testing during plant operation, as appropriate. Periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of detection equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks. A description of testing and calibration for the containment radioactivity monitoring system is presented in Subsection 11.5.2.

Periodic inspection of the floor drainage system to the containment sump is conducted to check for blockage and ensure unobstructed pathways.

The containment humidity monitoring systems and the containment air cooler condensate flow rate monitoring system are also periodically tested to ensure proper operation and verify sensitivity.

In service inspection criteria, equipment used, procedures, frequency of testing, inspection, surveillance, and examination of the structural and leak-tight integrity of RCPB components are described in Subsection 5.2.4.

5.2.5.8 Limits for Reactor Coolant Leakage Rates within the RCPB

In accordance with the position ~~9~~4.1 of regulatory guide 1.45, the limiting conditions for identified, ~~and~~ unidentified, RCPB and intersystem reactor coolant leakages are identified in the Chapter 16 Technical Specifications (TS). Subsections 3.4.13 and 3.4.14 address~~es~~ RCS operational leakage and pressure isolation valve (intersystem), leak limits, respectively. Subsection 3.4.15 addresses RCS leak detection instrument requirements.

The leakage management procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to identify leak source, monitor and trend leak rate, evaluate various corrective action plans in response to prolonged low leakage conditions that exceeds normal leakage rates and not exceed the ~~Technical Specification (~~TS~~)~~ limit in order to provide the operator sufficient time to take corrective actions before the leakage exceeds TS limit value. In accordance with the guidance in RG 1.45 position C.2.1, the procedure includes the collection of leakage to the containment from unidentified sources so the total flow rate can be detected, monitored and quantified for flow rates greater than or equal to 0.05 gal/min.

Section III of the ASME Code (Ref. 5.4-14).

The surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. The flywheel ~~should~~will be inspected for critical dimensions after the over~~-~~speed test so that any dimensional changes can be detected. ~~Qualified~~With respect this test procedure,~~ and the acceptance criteria~~ it should be decided ~~with respect to this~~qualified test procedure and the acceptance criteria.

Flywheels are inspected by a program based on the recommendations of RG 1.14, which references Section XI of the ASME Code (Ref. 5.4-9, 15). The inspection program is discussed in Technical Specification 5.5.7, Reactor Coolant Pump Flywheel Inspection Program and Technical Report "Justification for 20 Years Inspection Interval for Reactor Coolant Pump Flywheel" (Ref. 5.4-23).

5.4.1.1.3 Material Acceptance Criteria

RCP motor flywheels conform to the following material acceptance criteria:

- Nil ductility transition temperature (NDTT) of the flywheel material is obtained by two drop weight tests which exhibit no-break performance at 20°F in accordance with ASTM E-208. The tests prove that the NDTT of the flywheel material does not exceed 10°F.

- A minimum of three charpy v-notch (CVN) impact specimens from each plate are tested at ambient (70°F) temperature in accordance with ASME SA-370 specifications. The CVN energy in both the parallel and normal orientation with respect to the final rolling direction of the flywheel plate material is at least 50 ft-lb and 35-mil lateral expansion at 70°F, and therefore, the flywheel material has a reference nil ductility temperature (RT_{NDT}) of 10°F. An evaluation of the flywheel overspeed proves that integrity of the flywheel is maintained.

5.4.1.2 Reactor Coolant Pump Design Bases

The RCP is in the reactor containment and ensures adequate reactor cooling flow rate to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit that is evaluated in the safety analysis.

The RCP is designed, fabricated, and tested according to the requirements of 10CFR50, 50.55a, GDC 1 and ASME code, Section III (Ref. 5.4-7, 14). The pump is designed with the margin in integrity and exhibits safe operation under all postulated events.

In the event of loss of offsite power (LOOP), the pump is able to provide adequate flow rate during coastdown conditions because of the pump assembly rotational inertia which is provided by the flywheel (top of the motor), the motor rotor, and other rotating parts. This forced flow and the subsequent natural circulation effect in the reactor coolant system (RCS) adequately cools the core.

Figure 5.4.1-1 shows the RCP and Table 5.4.1-1 provides the design parameters of the RCP.

5.4.7.2.3 System Operation

5.4.7.2.3.1 Plant Startup

During the initial stage of the plant startup, the RCS is completely filled with water. Plant startup includes bringing the reactor from the cold shutdown condition to no-load operating temperature and pressure and subsequently to power operation. Generally, while in the cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load present at the time.

At the beginning of plant startup, at least one CS/RHR pump is operating, and the RHRS is aligned to the RCS to divert a portion of the RHR flow through a low pressure letdown path to the CVCS to control the RCS pressure. After the reactor coolant pumps are started, the RHRS is operated as necessary for heat removal. Once the pressurizer steam bubble formation is complete, the RHRS ~~would be~~is isolated from the RCS.

5.4.7.2.3.2 Normal Operation

CS/RHR pumps are not in-service during the normal operation. Normal operation includes the power generation and hot standby operation phases. During normal operation the RHRS is not used and the CS system is on standby. The CS/RHR pumps are normally aligned to take the suction from the RWSP. The tubes of the CS/RHR heat exchangers are filled with borated water and the shells of the heat exchangers are filled with CCW.

5.4.7.2.3.3 Plant Shutdown

Plant shutdown is the operation that brings the reactor plant from normal operating temperature and pressure to refueling condition. The initial phase of plant shutdown is accomplished by transferring heat from the RCS to the steam and power conversion system through SGs. Depressurization is accomplished by spraying reactor coolant into the pressurizer which cools and condenses the pressurizer steam bubble.

The second phase of cooldown starts with the RHRS being placed in operation when the reactor coolant temperature and pressure are reduced to approximately 350°F and 400 psig, respectively, approximately four hours after reactor shutdown. Startup of the RHRS includes a warm up period, during which time reactor coolant flow rate is slowly increased through the heat exchangers to protect the piping/components in the RHR system from thermal shock.

The rate of heat removal from the reactor coolant is manually controlled by the operator by regulating the coolant flow through the CS/RHR heat exchanger. This is accomplished by re-opening the CS/RHR heat exchanger outlet flow control valves in two subsystems. The CS/RHR heat exchanger outlet flow control valves are positioned by the operator who maintains the total flow rate constantly through the CS/RHR heat exchanger bypass-flow control valves.

Chapter 6

US-APWR DCD Chapter 6 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
6.1-3	6.1.1.2.1	Editorial: clarify the language Replaced “practicable” with “possible” in last sentence of 2nd paragraph.
6.1-4	6.1.1.2.2	Editorial: clarify the language Replaced “of” with “between the” and replaced “with” with “and the” in last sentence of 1st paragraph.
6.1-6	6.1.2	Editorial: clarify the language Replaced “to” with “from” and replaced “of” with “on” in last sentence of the last paragraph.
6.2-53	6.2.3	Editorial: clarify the language Replaced “describes” with “provides” and replaced “discussion” with “information” in last sentence of 1st paragraph.
6.2-53	6.2.4.	Editorial: clarify the language Replaced “that consist of flange” with “consisting of the flange” .
6.3-51	Table 6.3-4, GL2008-01	Editorial: clarify the language Replaced “is” with “are” and added “a” before “vortex” in 1st bullet in “US-APWR Design” column.

test data and concludes that no cracking is anticipated on any equipment (stressed, sensitized or non-sensitized) even in the presence of postulated levels of chlorides and fluorides, provided the emergency core cooling solution is maintained above pH of 7.0.

6.1.1.2 Composition and Compatibility of Core Cooling Coolants and Containment Sprays

Controls are instituted to maintain the chemistry of the borated reactor coolant and the borated water in the RWSP. Chlorides and fluorides, which promote intergranular stress-corrosion cracking corrosion, are managed such that their concentrations are below 0.15 ppm. During periods of high temperatures, dissolved oxygen concentrations remain below 0.10 ppm. The controls include the chemical and volume control system (CVCS) and the spent fuel pit cooling and purification system (SFPCS). Details on these control systems are provided in Chapter 9, Subsection 9.3.4, for the CVCS and in Subsection 9.1.3 for the SFPCS.

6.1.1.2.1 Compatibility of Construction Materials with Core Cooling Coolants and Containment Sprays

The provision of RG 1.44 (Ref. 6.1-4) are followed during the manufacture and construction of the ESF components and structures. The material used to fabricate the safety, significant portions of the ESF systems (including supports) is highly resistant to corrosion. The sources of corrosion may originate with the fluid (to include air in the ESF air clean-up applications) contained and delivered, as well as from external sources. Borated reactor coolant, borated emergency make-up water, and a wetting containment spray that combines these fluids with sodium tetraborate decahydrate (NaTB) are important potential sources of such internal and external corrosion.

The pH of the ESF fluids is controlled during a DBA using NaTB baskets as a buffering agent. NaTB baskets are placed in the containment to maintain the desired post-accident pH conditions in the recirculation water. Maintaining the pH in the RWSP avoids stress-corrosion cracking of the austenitic stainless steel components and avoids excessive generation of hydrogen attributable to corrosion of containment metals. The information regarding boric acid in the RWSP water and NaTB in the containment is described in Subsection 6.3.1.3, Subsection 6.3.2.2.5, and Table 6.3-5. Aluminum and zinc are materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions in the containment, and their use is limited as much as possible~~practicable~~.

The materials used in the fabrication of the ESF components are corrosion resistant in normal operation and the post-LOCA environment. General corrosion is negligible with the exception of low-alloy and carbon steels. Some materials within the containment would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions. Their use is limited as much as practicable (Ref. 6.1-7).

Borated water is used in the RCS and the RWSP. The water quality requirements for the RCS and RWSP are described in Chapter 9, Subsection 9.3.4 and Table 6.1-3, respectively. The pH of the RWSP during a LOCA is adjusted by the NaTB baskets. The concrete that forms the structure of the RWSP is clad in stainless steel which inhibits the leach-out of chlorides and other contaminants into the RWSP water. Therefore, the compatibility of the ESF components is preserved in the post-LOCA environment.

The use of particulate based insulation such as Min-K-based pipe insulation is prohibited in containment. Non-metallic (thermal) insulation is controlled in accordance with RG 1.36 (Ref. 6.1-8) to control the leachable concentrations of chlorides, fluorides, sodium compounds, and silicates. Chapter 5, Subsection 5.2.3.2.3, provides further details on the external insulation requirements which are also applicable to ESFs. Close attention to regulatory requirements and guidance ensures material compatibility between US-APWR construction materials and ESF fluids.

6.1.1.2.2 Controls for Austenitic Stainless Steel

Chapter 5, Subsection 5.2.3, describes the controls employed during material selection to preclude the severe sensitization of stainless steel materials to be used for fabrication. For example, cold worked austenitic stainless steel (300 series) typically is solution heat treated. Controls may be based on, but are not limited to, those imposed by Appendix B to 10CFR50, Appendix B part, 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", with particular emphasis on Criteria VII, “Control of Purchased Material, Equipment, and Services;” VIII, Identification and Control of Materials, Parts, and Components; and IX, Control of Special Processes (Ref. 6.1-9). When using fresh water to flush systems containing austenitic stainless steel components following construction, a chloride stress-corrosion cracking inhibitor is used in the flushing medium. The process of cleaning of materials and components, cleanliness control, and pre-operational flushing for systems that contain austenitic stainless steel components follows RG 1.37 (Ref. 6.1-11) and the quality assurance program complies with the provisions and recommendations provided by ASME NQA-1-1994, Part II (Ref. 6.1-10). This process includes documentation to verify the compatibility between the~~of~~ materials used in manufacturing ESF components and the~~with~~ ESF fluids.

Chapter 5, Subsection 5.2.3 describes control of welding, heat treatment, welder qualification, and contamination protection for ferritic and austenitic stainless steels material fabrication which are also applicable to ESFs. The ferrite content in stainless steel weld metal will be controlled in accordance with the recommendations of RG 1.31 (Ref. 6.1-13). The recommendations of RG 1.50, Control of Preheat Temperature for Welding of Low Alloy Steel, (Ref. 6.1-14) are applied during weld fabrication.

6.1.1.2.3 Composition, Compatibility and Stability of Containment and Core Coolants

607,500 gallons of borated water are available in the RWSP to meet LOCA and long-term post-LOCA coolant needs. The RWSP water is borated to approximately 4,000 ppm boric acid, at a pH of approximately 4.3. Crystalline NaTB spray additive is stored in containment and is used to raise the pH of the RWSP water from 4.3, to at least 7.0, post-LOCA. This pH is consistent with the guidance of NRC Branch Technical Position MTEB-6.1 for the protection of austenitic stainless steel from chloride-induced stress corrosion cracking. Subsection 6.3.2.2.5 describes the design of NaTB baskets.

plant's licensing standards. The standards apply to quality assurance and quality control for procurement and maintenance of coating systems, and training qualifications for protective coating inspectors and applicators. The procurement and application, or reapplication, of new and existing coating systems are monitored through the program according to the coating type, service level of qualification required for specific cases, the service level at which the coating was procured, and the significance and type of application (includes pertinent information such as coating repair, replacement, coating thickness, and overlapping areas). The COL applicant is responsible for identifying the implementation milestones for the coatings program.

The guidance provided in RG1.54 Rev. 1 is also applied for the evaluation of coatings on buried pipes and tanks. These coatings are evaluated to limit the expected damage from~~to~~ the soil and surrounding environments on~~of~~ the pipes and tanks.

6.1.3 Combined License Information

Any utility that references the US-APWR design for construction and Licensed operation is responsible for the following COL items:

COL 6.1(1) Deleted

COL 6.1(2) Deleted

COL 6.1(3) Deleted

COL 6.1(4) Deleted

COL 6.1(5) Deleted

COL 6.1(6) Deleted

COL 6.1(7) The COL Applicant is responsible for identifying the implementation milestones for the coatings program.

6.1.4 References

6.1-1 Codes and Standards, Title 10, Code of Federal Regulation, 10CFR50.55a January 2007 Edition.

6.1-2 ASME Boiler and Pressure Vessel Code Section III, Division 1, American Society of Mechanical Engineers, July 01 2002.

6.1-3 ASME Boiler and Pressure Vessel Code Section II, Division 1, American Society of Mechanical Engineers , July 01 2002.

6.1-4 U.S. Nuclear Regulatory Commission, Control of the Use of Sensitized Stainless Steel, Regulatory Guide 1.44, May 1973.

penetration areas at a negative pressure during accident conditions. Subsection 6.5.3.2 provides~~describes~~ additional information ~~discussion~~ on the function of the containment penetration areas.

6.2.4 Containment Isolation System

The containment prevents or limits the release of fission products to the environment. The containment isolation system allows the free flow of normal or emergency-related fluids through the containment boundary in support of reactor operations, but establishes and preserves the containment boundary integrity. The containment isolation system includes the system and components (piping, valves, and actuation logic) that establish and preserve the containment boundary integrity.

The criteria for isolation requirements and the associated system design are set forth in GDC 55 through 57 of Appendix A to 10CFR50. Unless acceptable on some other specific and defined basis (e.g., instrument lines), two isolation barriers are required; one inside and one outside of the containment. Isolation barriers are valves, unless the piping system inside the containment is neither part of the RCPB, nor communicates directly with the containment atmosphere, and is both suitably protected and robust. This section of the DCD describes the design and functional capabilities of the US-APWR containment isolation system in compliance with these GDC.

The containment penetration barriers consisting~~that consist~~ of the flange closure, personnel airlock and equipment hatch are under administrative control.

6.2.4.1 Design Bases

As described in Chapter 3, Subsection 3.1.5, the containment isolation system conforms to GDC 54, 55, 56, and 57, and is designed to seismic category I, quality group B.The containment isolation valves are identified as Equipment Class 1 or 2, as described in Chapter 3, Section 3.2. In addition to being protected from the effects of a postulated pipe rupture and containment missiles, closed systems inside the containment considered an isolation barrier under GDC 57 are designed to withstand the containment design temperature, pressure from the containment structural acceptance test, LOCA conditions, and to accommodate the internal fluid pressure associated with the containment temperature resulting from a design basis LOCA. Instrument lines closed both inside and outside containment are designed in accordance with the guidance provided by RG 1.11, RG 1.141 and satisfy NUREG-0800, SRP 6.2.4 (Ref. 6.2-27), acceptance criterion 1. The containment isolation system is designed in accordance with the Three Mile Island (TMI)-related requirements of 10CFR50.34(f)(2)(xiv)(A) through (E). The discharge side of the relief valves in the CS/RHR pump suction lines is designed to withstand and be tested at the containment design pressure.

Chapter 3, Sections 3.3 and 3.4 describe how the containment isolation system is designed to accommodate the wind and tornado loadings, and to withstand flood levels. The design requirements for protection from internally generated missiles (for isolation system components inside and outside of the containment) are described in Chapter 3, Section 3.5. The design for protection against the dynamic effects associated with the postulated rupture of piping is described in Chapter 3, Section 3.6, while the

Table 6.3-4 Response of US-APWR to Generic Letters and Bulletins (Sheet 14 of 15)

No.	Regulatory Position	US-APWR Design
GL 2008-01	**MANAGING GAS ACCUMULATION IN EMERGENCY CORE COOLING, DECAY HEAT REMOVAL, AND CONTAINMENT SPRAY SYSTEM** The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter (GL) to address the issue of gas accumulation in the emergency core cooling, decay heat removal (DHR), and containment spray systems for following purposes: (1) to request addressees to submit information to demonstrate that the subject systems are in compliance with the current licensing and design bases and applicable regulatory requirements, and that suitable design, operational, and testing control measures are in place for maintaining this compliance (2) to collect the requested information to determine if additional regulatory action is required	In the US-APWR, the following design provisions are provided in order to prevent void forming in the system: - To reduce gas intrusion into the safety-related pump system, fully submerged strainers are~~is~~ installed to function as a vortex suppressor. - To mitigate any possible gas buildup in the RCS, a temperature instrument is installed on the line from the Engineered Safety Feature to the RCS for detection in the MCR. - To prevent boric acid water containing dissolved nitrogen from flowing back from the accumulator tank to RHRS, RHRS return line and accumulator injection line are segregated. - Pump test line is provided in order to allow the dynamic venting of the system through the periodic pump full-flow testing.

Chapter 8

US-APWR DCD Chapter 8 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
8.3-39	Subsection 8.3.1.3.1 3rd sentence	Editorial: clarify the language Replaced “is satisfied” with “meets” .

This regulatory guide endorses revision 2 of NUMARC 93-01 (Reference 8.2-6) with some provisions and clarifications for complying with 10 CFR 50.65 (Reference 8.2-7). Conformance to this regulatory guide is generically addressed in Section 1.9.

- RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants"

  This regulatory guide endorses Section 11 of NUMARC 93-01 (Reference 8.2-6) dated February 11, 2000 with some provisions and clarifications for complying with 10 CFR 50.65(a)(4) (Reference 8.2-7). Conformance to this regulatory guide is generically addressed in Section 1.9.

- RG 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants"

  This RG endorses four IEEE Standards, IEEE Std 665 (Reference 8.2-8), IEEE Std 666 (Reference 8.2-9), IEEE Std 1050 (Reference 8.2-10) and IEEE Std C62.23 (Reference 8.2-11), in their entirety with one exception to IEEE Std 665 (Reference 8.2-8), Subsection 5.7.4, which misquotes Subsection 4.2.4 of IEEE Std 142 (Reference 8.2-12). The US-APWR onsite power supply design fully confirms to the requirements of the endorsed IEEE standards that pertain to the lightning protection of nuclear power plants.

8.3.1.3 Electrical Power System Calculations and Distribution System Studies for AC System

Load flow, voltage regulation and short circuit studies are performed using the computer software program titled Electrical Transient Analyzer Program (ETAP) published by Operation Technology, Inc. The ETAP computer software program conforms to the requirements of 10 CFR Part 21 (Reference 8.3.1-36); 10 CFR Part 50 Appendix B (Reference 8.3.2-11); and American Society of Mechanical Engineers (ASME) NQA-1 (Reference 8.3.1-37). Onsite ac power system calculations are presented in Technical Report MUAP-09023 (Reference 8.3.1-38).

8.3.1.3.1 Load Flow/Voltage Regulation Studies and Under-/Overvoltage Protection

Load flow studies are performed to evaluate acceptable voltage range is maintained at equipment terminal in worst case loading condition. Voltage drop at equipment terminal is also calculated in largest motor starting condition. As a result, terminal voltage of equipment ~~is satisfied~~meets the acceptable voltage range indicated in Table 8.3.1-2.

8.3.1.3.2 Short Circuit Studies

Short circuit studies are performed to determine the magnitude of the prospective currents flowing throughout the power system due to a fault occurrence. The studies are performed to calculate most severe fault condition. This condition is a three phase bolted short circuit at the output terminal of a circuit breaker in the onsite ac distribution system. The studies are performed with ETAP based on ANSI/IEEE C37 standards. The acceptance criteria are that the calculated maximum short circuit current conforms to applicable breaker capability. Table 8.3.1-1 shows the breakers nominal ratings for the

Chapter 9

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-13	Subsection 9.2.2.1.2.2	Editorial: clarify the language Replaced “CCWSis” with “CCWS is” .
9.2-14	Subsection 9.2.2.2	Editorial: clarify the language Replaced “is branch off” with “branches” .
9.2-14	Subsection 9.2.2.2	Editorial: clarify the language Replaced “system. The CCW water the filters to protect the plate type CCW heat exchangers are not deemed necessary and not provided.” with “system, therefore, the CCW filter is not necessary.” .
9.2-15	Subsection 9.2.2.2.1.2	Editorial: clarify the language Replaced “Since the difference of installation elevation between the surge tanks and the pumps is large enough, as NPSH available, there is sufficient margin.” with “The surge tanks are located at a higher elevation than the pumps to ensure sufficient NPSH margin is available.” .
9.2-16	Subsection 9.2.2.1.2.3	Editorial: clarify the language Replaced “in the surge tank as a countermeasure of the negative pressure in a tank at the time of a sudden fall of tank” with “on the surge tank to prevent damaging the tank in the event of a sudden decrease in” .
9.2-16	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “can be attained even if assumming single failure, since there are two header tie line isolation valves. Since a header tie line isolation valve will be closed in about 10 seconds or less, it is satisfactory to isolate by” with “meets the single failure criteria by incorporating two header tie line isolation valves. The header isolation valves are designed to close within 30 seconds upon a” .
9.2-16	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “S+UV signal, P signal, and surge tank water low-low level.” with “S+UV signal, P signal, or surge tank water low-low level.” .

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-16	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “the valve close signal currently sent is made to bypass and the valve is made to open.” with “isolation signal can be bypassed and the isolation valves reopened.” .
9.2-16	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “CCW pumps are designed such that one CCW pump can supply water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) during normal operation. Therefore, the header isolation valves are maintained to be open.” with ” In addition, the header isolation valves are opened in order to supply cooling water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) by one CCW pump during normal operation.” .
9.2-17	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “Two air-operated isolation valves are provided in series on each CCW supply line isolation valves are provided on each CCW supply line (A2 and C2) to the components located in the non-seismic category I buildings (turbine building (T/B) and auxiliary building (A/B). These valves close to protect against CCW seismic category I out-leakage through the non-seismic category I portions by automatic closure upon the demand signals.” with “The CCW system supplies cooling water to components located in the non-seismic Category I buildings (turbine building and auxiliary building). Each CCW supply line (A2 and C2) has two in-series air operated isolation valves. These valves close automatically to isolate the non-seismic Category I portion of the CCW system upon receipt of a S+UV signal, P signal or surge tank low-low level signal.” .
9.2-17	Subsection 9.2.2.1.2.5	Editorial: clarify the language Replaced “CCW out-leakage through the non-seismic CCW return lines (A2 and C2) is prevented by check valves series located on the return line for components located in the non-seismic Category I buildings (i.e. the turbine (T/B) and auxiliary building (A/B))” with “In-series check valves are provided on the CCW return lines from the non-seismic Category I portion of the CCW system.” .
9.2-18	Subsection 9.2.2.1.2.5	Editorial: clarify the language Deleted eighth bullet.

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-18	Subsection 9.2.2.1.2.5	Editorial: clarify the language Add the following to the last of this item “The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).”
9.2-19	Subsection 9.2.2.2.2.4	Editorial: clarify the language Replaced “The signal to the pump is setting up delay time.” with “The start signal to the pumps is delayed.” .
9.2-20	Subsection 9.2.2.2.2.6	Editorial: clarify the language Replaced “The CCWS is designed in consideration of the water hammer prevention and mitigation of its in accordance with the following as discussed in NUREG-0927.” with “The CCWS is designed in consideration of water hammer prevention and mitigation in accordance with the following as discussed in NUREG-0927.” .
9.2-35	Subsection 9.2.7.1.1	Editorial: clarify the language Replaced at first sentence “require for” with “to” .
9.2-35	Subsection 9.2.7.1.1	Editorial: clarify the language Replaced at first sentence “GDC45” with “GDC 45” .
9.2-37	Subsection 9.2.7.2.1	Editorial: clarify the language Replaced “The motor operated three-way control valves provide the retune line of safety-related air handling unit cooling coils. These valves control a heat removal capacity of coil by modulating the flow rate of chilled water through the cooling coil in response to temperature control signal during AHU in operation. The valve failure position at the loss of a control signal and electrical power is “as is”.” with “The motor operated three-way control valves are located on the retune lines from each safety-related air handling unit cooling coils. These valves control the heat removal capacity by modulating the flow rate of chilled water through the AHU cooling coils in response to a temperature control signal. The motor operated three-way control valves fail “as is” upon a loss of control signal or electrical power.” .

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-37	Subsection 9.2.7.2.1	RAI No.338, Question 06.04-6 Add the following to sixth paragraph. “The chillers are protected by a pressure-relief device to safely relieve pressure and are piped to outside of the building in accordance with ANSI/ASHRAE Standard 15. And the chiller mechanical equipment rooms meet ANSI/ASHRAE Standard 15, so that are equipped with refrigerant leak detectors and actuate a dedicated ventilation system.”
9.2-37	Subsection 9.2.7.2.1	Editorial: clarify the language Replaced “. The ECWS water the filters to protect the chillers and cooling coils are not deemed necessary and not provided.” with “, therefore, the ECWS filter is not necessary.” .
9.2-37	Subsection 9.2.7.2.1	Editorial: clarify the language Replaced at first sentence of ninth paragraph “the chemical feed tank” with “the chemical feed tanks” .
9.2-37	Subsection 9.2.7.2.1	Editorial: clarify the language Replaced “The chemical feed tank is a constructed of carbon steel.” with “The chemical feed tanks are constructed of carbon steel.” .
9.2-37	Subsection 9.2.7.2.1	Editorial: clarify the language Replaced “The isolation valves that are installed in piping between chemical addition feed and ECWS piping chemical feed line.” with “Manual isolation valves are installed in the piping between the chemical feed tank and the ECWS piping.” .
9.2-38	Subsection 9.2.7.2.1.1	Editorial: clarify the language Replaced at second sentence of forth paragraph “accommodated” with “accommodates” .
9.2-38	Subsection 9.2.7.2.1.1	Editorial: clarify the language Replaced “The chemical feed tank is a constructed of carbon steel.” with “The chemical feed tank is constructed of carbon steel.”.

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.2-38	Subsection 9.2.7.2.1.1	Editorial: clarify the language Replaced “The isolation valves that are installed in piping between chemical addition feed and ECWS piping chemical feed line.” with “Manual isolation valves are installed in the piping between the chemical feed tank and the ECWS piping.” .
9.2-39	Subsection 9.2.7.2.1.1	Editorial: clarify the language Replaced “The check valves provide the nitrogen supply and makeup water line to maintain the system pressure due to failure of the non seismic support system.” with “The nitrogen supply line and makeup water supply line check valves are designed to maintain ECW system pressure in the event of failure of the non-seismic support system.” .
9.2-39	Subsection 9.2.7.2.1.1	Editorial: clarify the language Replaced “The motor operated three-way control valves provide the retune line of safety-related air handling unit cooling coils. These valves control a heat removal capacity of coil by modulating the flow rate of chilled water through the cooling coil in response to temperature control signal during AHU in operation. The valve failure position at the loss of a control signal and electrical power is “as is”.” with “The motor operated three-way control valves are located on the retune lines from each safety-related air handling unit cooling coils. These valves control the heat removal capacity by modulating the flow rate of chilled water through the AHU cooling coils in response to a temperature control signal. The motor operated three-way control valves fail "as-is" upon a loss of control signal or electrical power.” .
9.2-40	Subsection 9.2.7.2.2	RAI No.338, Question 06.04-6 Add the following to second paragraph. “The chillers are protected by a pressure-relief device to safely relieve pressure and are piped to outside of the building in accordance with ANSI/ASHRAE Standard 15. And the chiller mechanical equipment rooms meet ANSI/ASHRAE Standard 15, so that are equipped with refrigerant leak detectors and actuate a dedicated ventilation system.”

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.4.8	Subsection 9.4.1.3	RAI No.327, Question 09.04.01-9 Addition of the description of the closest potential source of fresh air contamination for MCR HVAC system intake.
9.4-15	Subsection 9.4.3.2	RAI No.483, Question 09.04.03-9 Addition of the description of the design features which assure that the US-APWR design meets the GDC 60.
9.4-30	Subsection 9.4.5.2.2	Editorial: clarify the language Replaced at second paragraph “ACC” with “AAC” .
9.4-32	Subsection 9.4.5.2.4	Editorial: clarify the language Replaced “blow” with “below”
9.4-33	Subsection 9.4.5.2.5	Editorial: clarify the language Replaced “blow” with “below”
9.4-67	Table 9.4.5-1 Annulus Emergency Filtration Unit Area Air Handling Unit Cooling Coil Quantity	Editorial: Typographical error Replaced “Cooling coil Quantity” with “Cooling Coil Quantity”. Replaced “A Unit is provided the chilled water of a B train” with “A Unit is provided the chilled water of A and B train”.
9.5-3	Subsection 9.5.1.1	CP34 COLA RAI 010, Question 09.05.01-7 Addition of a description of the deviation from the National Fire Protection Association (NFPA) codes and standards.
9.5-40	Subsection 9.5.6.2 3rd paragraph 2nd sentence	Editorial Replaced: “The GTG starts properly by turning effort of four air start motors.” with “The GTG starts by four air start motors.”
9.5-43	Subsection 9.5.7.2 1st paragraph 4th sentence	Editorial Replaced: “Keep-warm system is not installed basically, since gas turbine lubrication oil has cold-adapted feature.” with “Keep-warm system is not installed, since gas turbine lubrication oil performs under cold condition.”

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.5-44	Subsection 9.5.7.2 3rd paragraph Last sentence	RAI No.506, Question No.09.05.07-23 Replaced: "Requirement specification of fuel oil consumption is 0.053 gal/h or less." with "Requirement specification of lube oil consumption is 0.053 gal/h or less."
9.5-44	Subsection 9.5.7.2 4th paragraph 4th sentence	Editorial Replaced: "During starting of the gas turbine, GTG does not need pre-circulation of lube oil, because ball bearings are adopted." with "During starting of the gas turbine, GTG does not need pre-circulation of lube oil, because ball bearings are used."
9.5-44	Subsection 9.5.7.2 5th paragraph 2nd sentence	Editorial Replaced: "The fail to open of temperature regulating valves would also cause a high lube oil temperature condition." with "The failure of the temperature regulating valves to open would also cause a high lube oil temperature condition."
9.5-44	Subsection 9.5.7.3 Item B 1st sentence	RAI No.506, Question No.09.05.07-22 Replaced: "The components of the systems are designed to ASME Boiler & Pressure Vessel Code, Section III, Class 3. When a component is commercially unavailable as ASME Boiler & Pressure Vessel Code, Section III, Class 3 design, the component is proven of equivalent quality." with "All components of the systems are provided in GTG enclosure as a GTG package and proven of equivalent quality to ASME Boiler & Pressure Vessel Code, Section III, Class 3."
9.5-45	Subsection 9.5.7.3 Item F	RAI No.506, Question No.09.05.07-20 Replaced: "Power section of gas turbine is designed, so that the absorbed energy of casing is beyond the kinetic energy of rotational parts. The missiles generated by GTG are not postulated as Section 3.5." with "The power section of the gas turbine is designed so that the capacity of the casing to absorb energy is greater than the kinetic energy of rotational parts of the turbine. Missiles are not postulated to be generated by the GTG as described in Section 3.5."

US-APWR DCD Chapter 9 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
9.5-45	Subsection 9.5.7.5 1st paragraph 2nd sentence	Editorial Replaced: “Low lube oil pressure, high lube oil temperatures, are alarmed in the MCR and in the GTG room.” with “Low lube oil pressure and high lube oil temperatures, are alarmed in the MCR and in the GTG room.”
9.5-45	Subsection 9.5.7.5 2nd paragraph 1st sentence	Editorial Replaced: “Lube oil tank level instrumentation is installed and low level is alerted in the MCR and in the GTG room.” with “Lube oil tank level instrumentation is installed and low level is alarmed in the MCR and in the GTG room.”
9.5-45	Subsection 9.5.7.5 2nd paragraph 2nd sentence	Editorial Replaced: “Differential pressure instrumentation for filter and strainer are installed and high pressure is alerted the MCR and in the GTG room.” with “Differential pressure instrumentation for filter and strainer are installed and high pressure is alarmed the MCR and in the GTG room.”
9.5-158	Table 9.5.7-1 Reduction Gear Reservoir	RAI No.506, Question No.09.05.07-22 Replaced: “Code: ASME Section III, Class 3” with “Codes and Standards: Manufacturer's standards”

- Provide sufficient cooling capacity for the components required during normal operating conditions such as normal power operation, normal shutdown and refueling as described below.

- Detect leakage of radioactive material into the system and control leakage of radioactive material out of the system.

- Prevent long term corrosion that may degrade system performance.

9.2.2.1.2.1 Normal Operation

The CCWS is designed to transfer heat from the plant components required to support normal power operation with one train (pump and heat exchanger) unavailable due to on line maintenance and a single active component failure. The CCWS is sized such that the component cooling water supply temperature to plant components is not more than 100°F. Normal operating heat loads are reactor coolant pump, charging pump, letdown heat exchanger, instrument air, spent fuel pool cooling heat exchanger, sample heat exchanger, seal water heat exchanger, blowdown sample cooler, B.A. evaporator, waste gas compressor, and so on. The CCWS provides sufficient surge tank capacity below the low level alarm to allow for operators to take action.

9.2.2.1.2.2 Normal Plant Cooldown

The CCWS is designed to remove both decay and sensible heat from the core and the reactor coolant system in addition to some normal operating heat loads during the latter stages of plant cooldown. The component cooling water system is sized to reduce the temperature of the reactor coolant system from 350°F at approximately 4 hours after reactor shutdown to 140°F using 4 trains while maintaining the component cooling water supply below 110°F. Failure of one train of CCW with another train unavailable due to maintenance will not prevent achieving cold shutdown conditions. The CCWS continues to provide cooling water to the residual heat removal system throughout the shutdown after cooldown is complete.

9.2.2.1.2.3 Refueling

During refueling, cooling water flow is provided to spent fuel pool heat exchangers to cool the spent fuel pool. For a full core off-load cooling water is also supplied to a normal residual heat removal heat exchanger as part of spent fuel pool cooling. The CCWS maintains the spent fuel pit water temperature below 120°F. System operation is with both CCWS divisions available.

9.2.2.2 System Description

The system flow diagram is shown in Figure 9.2.2-1.

The CCWS is the closed loop system that functions as an intermediate system between the various components cooled by CCWS and the ESWS, (Subsection 9.2.1). The CCWS transfers heat and prevents direct leakage of the radioactive fluid from the components to the ESWS.

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. Each train has one CCWP and one CCW HX and provides 50% of the cooling capacity required for safety function.

Electrical power to the CCWS is supplied from Class 1E buses that are backed up by Class 1E power supply so that the system is capable to operate during a loss of off site power.

There is the header tie line between trains A and B, and between trains C and D. The header tie line in each subsystem branches~~is branch off~~ into two loops. See Table 9.2.2-1 for the components supplied by each loop.

Each subsystem is served by one CCW surge tank. The CCW surge tank is installed at the highest point of the system to facilitate system air venting to ensure a water solid closed loop and to provide the net positive suction head at the CCWP suction. In addition, the surge tank accommodates the thermal expansion and contraction of the cooling water and potential leakage into or out of the CCWS.

Deminerlized quality water with corrosion inhibitors is circulated in the CCWS. No outside impurities are expected to be infiltrated in the system, therefore, the CCW filter is not necessary~~. The CCW water the filters to protect the plate type CCW heat exchangers are not deemed necessary and not provided~~. The impacts of non-safety related SSC failures in the CCW system will not adversely affect safety-related SSCs to perform their safety related function since the direct impact of a pipe break in the non-safety portion of the system can be accommodated. The CCW system's safety function will be maintained as a result of the nonsafety-related piping failure, and the indirect impact of the pipe break will not impact any SSC safety function.

9.2.2.2.1 Component Descriptions

The CCWS components are described below. Design parameters for major components of CCWS are provided in Table 9.2.2-2.

9.2.2.2.1.1 CCW HX

The CCW HXs transfer heat from the CCWS to the ESWS. The CCW HXs are plate type. The CCW HXs are designated quality group C as defined in Regulatory Guide 1.26 (Ref. 9.2.11-3), seismic category I, and are designed in accordance with the requirements of the ASME Section III, class 3.

9.2.2.2.1.2 CCWP

The CCWP circulates cooling water through the CCW HX and the components cooled by CCWS.

The pumps are horizontal centrifugal pumps and driven by an ac powered induction motor.

The pumps are designated quality group C as defined in Regulatory Guide 1.26, seismic category I, and are designed in accordance with the requirements of the ASME Section III, class 3.

The pumps are designed in consideration of head losses in the cooling water inlet piping based on full power flow conditions, increased pipe roughness, maximum pressure drop through the system heat exchangers, and the actual amount of excess margin etc.

~~Since the difference of installation elevation between the surge tanks and the pumps is large enough, as NPSH available, there is sufficient margin~~The surge tanks are located at a higher elevation than the pumps to ensure sufficint NPSH margin is available.

9.2.2.2.1.3 CCW Surge Tank

The CCW surge tanks are connected to the suction side of the CCWP. The surge tank accommodates the thermal expansion and contraction of the cooling water and potential leakage into or from the CCWS. Makeup water is supplied to the respective surge line.

The CCW surge tank is designated quality group C as defined in Regulatory Guide 1.26, seismic category I, and is designed to the requirements of the ASME Section III, class 3.

In case of a small leak out of the system, makeup water is supplied as necessary until the leak is isolated.

The makeup water can be supplied from the following systems:

- Demineralized water system (DWS) which supplies the demineralized water
- Primary makeup water system (PMWS) which supplies the deaerated water and primary makeup water
- Refueling water storage system (RWS) which supplies the refueling water

Deaerated water is used for initial filling of this system and demineralized water is used for automatic makeup when the tank water level reaches a low level setpoint.

If necessary, primary makeup water and refueling water may be used during an emergency. Refueling water storage pit is water source of seismic category I.

Water chemistry control of CCWS is performed by adding chemicals to the CCW surge tank to prevent long term corrosion that may degrade system performance. The CCW in the surge tank is covered with nitrogen gas to maintain water chemistry.

In order to provide redundancy for a passive failure (a loss of system integrity resulting in abnormal leakage), an internal partition plate is provided in the tank so that two separate surge tank volumes are maintained.

The CCW surge tank capacity of 50% is able to receive the amount of inleak from RCP thermal barrier Hx in consideration of isolation time. Regarding the makeup water source of the RWSP to be seismic category I, this makeup water source provides capacity to accommodate system leakage for seven days. Makeup water supply is performed by an operator by locally operating the manual valves. A vacuum breaker is installed ~~in the surge tank as a countermeasure of the negative pressure in a tank at the time of a sudden fall of tank~~ on the surge tank to prevent damaging the tank in the event of a sudden decrease in water level.

9.2.2.2.1.4 Piping

Carbon steel is used for the piping of the CCWS. Piping joints and connections are welded, except where flanged connections are required.

9.2.2.2.1.5 Valves

・Header tie line isolation valve

The function of this motor operated valve is to separate each subsystem into two independent trains during abnormal and accident conditions. This ensures each safety train is isolated from any potential passive failure in the non-safety portion or another safety train of the CCWS. This valve automatically closes at once upon the following signals:

- Low- low water level signal of a CCW surge tank
- ECCS actuation signal and under voltage signal
- Containment Spray signal

Header isolation meets the single failure criteria by incorporating two header tie line isolation valves. The header isolation valves are designed to close within 30 seconds upon a ~~can be attained even if assumming single failure, since there are two header tie line isolation valves. Since a header tie line isolation valve will be closed in about 10 seconds or less, it is satisfactory to isolate by~~ S+UV signal, P signal, ~~and~~or surge tank water low-low level. Then, in order to resume supply of the cooling water to the RCP thermal barrier heat exchanger and the spent fuel pit heat exchanger, the isolation signal can be bypassed and the isolation valves responed.~~valve close signal currently sent is made to bypass and the valve is made to open. CCW pumps are designed such that one CCW pump can supply water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) during normal operation. Therefore, the header isolation valves are maintained to be open.~~ In addition, the header isolation valves are opened in order to supply cooling water to A, B, A1 and A2 trains (or C, D, C1 and C2 trains) by one CCW pump during normal operation.

・Containment Spray/Residual Heat Removal Heat Exchanger (CS/RHRS HX) CCW Outlet Valve

The CCW which is supplied to the CS/RHR heat exchanger is shutoff by the CCW outlet isolation valve during standby. However, this normal closed motor operated valve automatically opens at once upon ECCS actuation signal plus the respective train CCW pump start signal to establish cooling water flow to the CS/RHR heat exchanger.

- **RCP Thermal Barrier HX CCW Return Line Isolation valve**

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.

- **CCW Surge Tank Vent Valve and Relief Valve**

The surge tank vent valve opens upon CCW surge tank high pressure and this valve closes when the radiation monitor level exceeds its set point. The surge tank relief valve provides surge tank overpressure protection.

- **Other Relief Valve**

Other relief valves are provided to relieve the pressure buildup caused by potential thermal expansion when equipment is isolated.

- **Containment Isolation Valve**

Containment isolation valves are installed on CCW lines penetrating containment as described in Subsection 6.2.4.

- **Isolation valve between seismic category I portion and non-seismic category I portion**

~~Two air-operated isolation valves are provided in series on each CCW supply line isolation valves are provided on each CCW supply line (A2 and C2) to the components located in the non-seismic category I buildings (turbine building (T/B) and auxiliary building (A/B). These valves close to protect against CCW seismic category I out-leakage through the non-seismic category I portions by automatic closure upon the demand signals~~The CCW system supplies cooling water to components located in the non-seismic Category I buildings (turbine building and auxiliary building). Each CCW supply line (A2 and C2) has two in-series air operated isolation valves. These valves close automatically to isolate the non-seismic Category I portion of the CCW system upon receipt of a S+UV signal, P signal or surge tank low-lwo level signal.

~~CCW out-leakage through the non-seismic CCW return lines (A2 and C2) is prevented by check valves series located on the return line for components located in the non-seismic Category I buildings (i.e. the turbine (T/B) and auxiliary building (A/B))~~ In-series check valves are provided on the CCW return lines from the non-seismic Category I portion of the CCW system (See Figure 9.2.2-1, Sheet 9 of 9).

The CCW supply header (A2 and C2) isolation valves close automatically when one of the following occurs (See Figure 9.2.2-1, Sheet 9 of 9).

The isolation valves on auxiliary building supply line

- Low- low water level signal of the component cooling water surge tank
- ECCS actuation signal
- Containment spray signal

b) The isolation valves on turbine building supply line

- Low- low water level signal of the component cooling water surge tank
- ECCS actuation signal and under voltage signal
- Containment spray signal

~~· RCP Thermal Barrier HX CCW Return Line Isolation valve~~

~~These Valves function to supply cooling water to the RCPs of header A-1 (or C-1) in the event cooling is lost due to a single failure during on-line maintenance of a CCW pump. The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).~~

· RCP CCW tie line isolation valve

This normally closed motor operated valve opens when it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance, and ensures the thermal barrier cooling water.

· RCP motor CCW supply line isolation valve

This normally open motor operated valve closes when it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance, and ensures the thermal barrier cooling water.

· RCP CCW supply line isolation valve

This normally open motor operated valve closes automatically upon P signal to shutoff the component cooling water flow to the containment vessel.

· RCP CCW return line isolation valve

This normally open motor operated valve closes to establish the return line of the thermal barrier cooling water in the case it becomes impossible to supply cooling water to the RCP of A1 (or C1) header due to the single failure of the CCW pump and on-line maintenance. The cooling water for the thermal barrier is ensured by opening NCS-MOV-232A/B and NCS-MOV-233A/B, and closing NCS-MOV-234A (or 234B).

9.2.2.2.2 System Operations

Table 9.2.2-4 and 9.2.2-5, respectively, provide heat loads and water flow balance for various operating modes.

9.2.2.2.2.1 Normal Power Operation

During normal operation, at least one train from each subsystem is placed in service. A total of two CCWP and two CCW HXs are in operation. A combination of trains in service is trains A or B and trains C or D.

During this operating condition, an operating CCWP in each subsystem supplies CCW to all loops in the particular subsystem with cooling water temperature not exceeding 100 ^{0}F maximum.

CCWPs which are not in service are placed in standby and automatically start upon a low pressure signal of CCW header pressure.

9.2.2.2.2.2 Normal Plant Shutdown

After approximately four hours of normal plant cool down, when the reactor coolant temperature and pressure are reduced to approximately 350 ^{o}F and 400 psig, the standby CCW HXs and pumps are placed in service resulting in four trains (i.e. four CCWPs and four CCW HXs) in operation. The CCWS isolation valve for each of the CS/RHR HXs is opened to supply cooling water to these HXs.

The failure of one cooling train (i.e. failure in one pump or one HX) increases the time for plant cool down, however, it does not affect the safe operation of the plant. The plant can be safely brought to the cold shutdown condition with a minimum of two trains.

During plant cool down by the residual heat removal system, the CCW supply temperature to the various components is permitted to increase to 110 ^{o}F.

9.2.2.2.2.3 Refueling

During refueling, the required number of CCW HXs and pumps is determined by the heat load. Normally, three trains operate in this mode. The remaining train may be taken out of service for maintenance. An operating CCWP in each subsystem supplies CCW to all loops in service in the particular subsystem with a maximum CCW supply water temperature not exceeding 100 ^{o}F.

9.2.2.2.2.4 Loss of Coolant Accident

All CCWP are automatically actuated by ECCS actuation signal. ~~The signal~~The start signal to the ~~pump~~pumps is ~~-setting up delay time~~delayed. (Refer to Figure 8.3.1-2 Logic diagrams (Sheet 18 of 24)) The isolation valves for the CS/RHR HXs are automatically opened by the ECCS actuation signal and the same train CCWP start signal. ~~-~~The header tie line isolation valves are closed by an ECCS actuation signal in coincidence with an undervoltage signal, and the CCWS is separated into four individual trains (A, B, C and D). The header tie line isolation valves can be manually reopened from the MCR to restore RCP seal and SFP HX cooling, if required.

As a minimum, two trains are required to operate during a LOCA.

9.2.2.2.2.5 Loss of Offsite Power (LOOP)

In the case of a LOOP, all CCWPs are automatically loaded onto their respective Class 1E power sources. The CCWS continues to provide cooling of the required components.

As a minimum, two trains are required to operate during a LOOP.

9.2.2.2.2.6 Water Hammer Prevention

The CCWS is designed in consideration of ~~the~~ water hammer prevention and mitigation ~~of its~~ in accordance with the following as discussed in NUREG-0927.

• An elevated surge tank to keep the system filled.
• Vents for venting components and piping at all high points in the system.
• After any system drainage, venting is assured by personnel training and procedures.
• System valves are slow acting.

The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer prevention. The procedures should address the operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.

9.2.2.3 Safety Evaluation

The CCWS is designed to perform its safety function with only two out of four trains operating. As shown in Table 9.2.2-3, the CCWS is completely redundant and a single failure does not compromise the system's safety function even if one train is out of service for maintenance.

The safety-related portions of the CCWS is protected against natural phenomena and internal missiles. The following sections addresses natural phenomena and missiles protection.

- Section 3.3, Wind and tornado loadings;
- Section 3.4, Water Level (Flood) Protection;
- Section 3.5, Missile Protection;
- Section 3.7, Seismic Design;

Pipe rupture protection is addressed in Section 3.6, Protection against Dynamic Effects Associated with Postulated Rupture of Piping.

The function of the non-essential chilled water system is to provide, during plant normal operation and LOOP, chilled water for the plant air cooling and ventilation systems serving the non-safety related areas.

9.2.7.1 Design Bases

9.2.7.1.1 Essential Chilled Water System

The essential chilled water system provides cooling water ~~require for~~to various HVAC components during all plant operating conditions, including normal plant operation, abnormal and accident conditions. The essential chilled water system is designed to meet the relevant requirements of GDC 45, and GDC 46 (Ref.9.2.11-1).

9.2.7.1.1.1 Safety Design bases

The essential chilled water system is designed to satisfy the following safety design bases.

- The essential chilled water system equipment and component pressure boundary are designed in compliance with ASME Section III.
- A single failure of any active component, or LOOP, cannot result in a loss of chilled water service to the plant safety-related cooling and ventilation systems.
- The essential chilled water system and its distribution piping loop are designed to equipment class 3 and seismic category I to remain functional during and following a SSE.
- The safety-related portions of the ECWS are protected against natural phenomena and internal missiles.
- The essential chilled water system withstands the effects of adverse environmental, operating and accidental conditions.
- The essential chilled water system withstands the effects of tornadoes and tornado missiles.
- The essential chilled water system withstands the design loadings.
- The essential chilled water system meets GDC 2, by compliance, meeting the guidance of Regulatory Guide (RG) 1.29. The applicable sections of RG 1.29 include Position C.1 for safety related portions and Position C.2 for non-safety related portions.

9.2.7.1.1.2 Power Generation Design Bases

The essential chilled water system is designed to satisfy the following power generation design bases.

The essential chilled water system flow diagram is shown in Figure 9.2.7-1, equipment and component data is presented in Table 9.2.7-1.

The essential chilled water system consists of four independent trains and each train consists of one 50% capacity system. Each system includes, a water-cooled chiller, a chilled water pump, a compression tank with a make-up water line, a chilled water distribution loop, and instrumentation and control system. The condenser (heat rejection) section of each chiller is supplied with cooling water from the respective essential service water system during both normal and emergency operating conditions. The ECWS heat transfer and flow requirements for normal plant operation and abnormal conditions are shown in Table 9.2.7-2.

The motor operated three-way control valves are located on ~~provide~~ the retune lines from each ~~of~~ safety-related air handling unit cooling coils. These valves control ~~a~~the heat removal capacity ~~of coil~~ by modulating the flow rate of chilled water through the AHU cooling coils in response to a temperature control signal. ~~during AHU in operation.~~ The motor operated three-way ~~valve failure position at the loss of a~~ control valves fail~~signal and electrical power is~~ "as is" upon a loss of control signal or electrical power.

During LOOP, each of the essential chilled water system is powered from the respective safety emergency power source.

The chiller of each essential chilled water system is equipped with an integral chilled water temperature control system.

The chillers are protected by a pressure-relief device to safely relieve pressure and are piped to outside of the building in accordance with ANSI/ASHRAE Standard 15. And the chiller mechanical equipment rooms meet ANSI/ASHRAE Standard 15, so that are equipped with refrigerant leak detectors and actuate a dedicated ventilation system.

The essential chilled water system control maintains the chilled water supply temperature. The compression tank maintains the system pressure within the design operating range.

Upon receipt of an ECCS actuation signal, the operating essential chillers and pumps continue to run and the standby essential chillers and pumps start.

Demineralized quality water with corrosion inhibitors is circulated in the ECWS. No outside impurities are expected to be infilterated in the system. ~~The ECWS water the filters to protect the chillers and cooling coils are not deemed necessary and not provided~~, therefore, the ECWS filter is not necessary.

Water chemistry control of ECWS is performed by adding chemicals to the chemical feed tanks to prevent long-term corrosion that may degrade system performance. The chemical feed tanks ~~is a~~ are constructed of carbon steel. The chemical feed tanks are designed as non safety-related but seismic category II and are designed in accordance with ASME Section VIII. ~~The isolation valves that are installed in piping between chemical addition feed and ECWS piping chemical feed line. These valves are normally locked closed.~~Manual isolation valves are installed in the piping between the chemical feed tank and the ECWS piping. These valves are normally locked closed.

The essential chilled water system is designed in consideration of the water hammer prevention and mitigation of its in accordance with the following as discussed in NUREG-0927.

- A compression tank to keep the system filled
- Vents for venting components and piping at all high points in the system.
- After any system drainage, venting is assured by personnel training and procedures.
- System valves are slow acting.

The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer prevention. The procedures should address the plant operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.

9.2.7.2.1.1 Component Descriptions

The ECWS components are described below.

Essential Chiller Unit

The essential chiller unit is water-cooled type. Each essential chiller unit is designed to remove heat load from all the cooling coil of safety-related HVAC system of respective train it serves during all plant condition. Each essential chiller unit is designed to provide a sufficient quantity of chilled water to associated HVAC system chilled water cooling coils at a minimum 40^{o}F of water temperature. Environmental safe refrigerants are being utilized in the chilled water systems chillers.

Essential Chilled Water Pump

Each essential chilled water pump is designed to supply chilled water to all the cooling coils of safety-related HVAC system for the respective train it serves during all plant condition. The pump is designed in consideration of fluctuation in the supplied electrical frequency, increased pipe roughness, and maximum pressure drop through the system components. The pumps are horizontal centrifugal pumps and driven by an ac induction motor. The pumps are designed quality group C as defined in Regulatory Guide 1.26, seismic category I, and are designed in accordance with the requirements of the ASME Section III, Class 3. The essential chilled water pumps have sufficient NPSH available due to system pressure pressurized by compression tank.

Essential Chilled Water Compression Tank

The essential chilled water compression tanks are connected to the suction side of the respective essential chilled water pump. The compression tank accommodateds the thermal expansion and contraction of the cooling water and potential leakage from the ECWS. The compression tank provides the net positive suction head (NPSH) at the essential chilled water pump suction. The compression tanks are compressed by nitrogen gas (compressed gas supply system (GGS)).

Makeup water is supplied to the respective surge line. The makeup water is supplied from the following systems.

- Demineralized water system (DWS) which supplies the demineralized water
- Primary makeup water system (PMWS) which supplies the deaerated water

Deareated water is used for initial filing of this system and demineralized water is used for makeup when the tank water level reaches a low level setpoint.

The compression tank contains sufficient water volume to assure reliable system operation without makeup for at least seven days.

Chemical Feed Tank

Water chemistry control of ECWS is performed by adding chemicals to the chemical feed tank to prevent long-term corrosion that may degrade system performance. The chemical feed tank is ~~a~~ constructed of carbon steel. The chemical feed tanks are designed as non safety-related but seismic category II and are designed in accordance with ASME Section VIII. ~~The isolation valves that are installed in piping between chemical addition feed and ECWS piping chemical feed line~~ Manual isolation valves are installed in the piping between the chemical feed tank and the ECWS piping. These valves are normally locked closed.

Piping

Carbon steel piping designed, fabricated, installed and tested in accordance with ASME Section III, class 3 requirements, is used for the safety-related portion of the ECWS. Piping is arranged to permit access for inspection.

Valves

- **ECW Compression Tank relief Valve**

The ECW compression tank relief valve provides compression tank and system overpressure protection. The valves discharge to the non-radioactive drain sump.

- **Check Valves**

The nitrogen supply line and makeup water supply line check valves are designed to maintain ECW system pressure in the event of ~~provide the nitrogen supply and makeup water line to maintain the system pressure due to~~ failure of the non-~~-~~seismic support system.

- **Chilled Water Control Valves**

The motor operated three-way control valves are located on ~~provide the~~ retune lines ~~of~~ from each safety-related air handling unit cooling coils. These valves control ~~a~~the heat removal capacity ~~of coil~~ by modulating the flow rate of chilled water through the AHU cooling coils in response to a temperature control signal. ~~during AHU in operation.~~ The motor operated three-way control valves fail~~ure position at the loss of a control signal and electrical power is~~ "as is" upon a loss of control signal or electrical power.

9.2.7.2.2 Non-Essential Chilled Water System

The non-essential chilled water system flow diagram is shown in Figure 9.2.7-2. The non-essential chilled water system consists of four water-cooled chillers, four chilled water pumps, a compression tank with a make-up water line, a chilled water distribution loop, and an instrumentation and control system. The condenser (heat rejection) section of each chiller is supplied with cooling water from a dedicated cooling tower. Each chiller is sized for one-third of the total non-essential chilled water load.

The chillers are protected by a pressure-relief device to safely relieve pressure and are piped to outside of the building in accordance with ANSI/ASHRAE Standard 15. And the chiller mechanical equipment rooms meet ANSI/ASHRAE Standard 15, so that are equipped with refrigerant leak detectors and actuate a dedicated ventilation system.

When the non-essential chilled water system is energized, the chilled water pump, the condenser water pump, and the cooling tower fans will start. When both the chilled and condenser water flows are established, the chillers will start to satisfy the plant non-safety cooling load. The non-essential chilled water system control maintains the chilled water supply temperature at the design setpoint. The compression tank maintains the system pressure within the design operating range.

During the LOOP condition, the non-essential chilled water system is powered from the alternate ac power source.

9.2.7.3 Safety Evaluation

9.2.7.3.1 Essential Chilled Water System

The essential chilled water system is designed to perform its safety function with only two out of four trains operating. The essential chilled water system is completely separate and a single failure does not compromise the system's safety function even if one train is out of service for maintenance.

The physical separation of the redundant system and the associated components assures the continuous operation of the essential chilled water system.
The system is classified as equipment class 3, seismic category I. The system pressure boundary is designed in accordance with ASME Section III to assure the continuous integrity of the system pressure boundary under all modes of operation.

Redundant systems are powered by separate safety related buses and their heat rejection sections (condenser) are provided with cooling from separate safety related essential service water system.

Casings of the chiller refrigerant compressor and the chilled water pumps are designed to withstand penetration by internally generated missiles.

The MCR HVAC system is protected against piping failure due to high energy line breaks and is not affected by any of the effect of postulated break of the piping. The basis for protection against postulated piping failure is discussed in Section 3.6.2.

The closest potential source of fresh air contamination is the exhaust from the Emergency Gas Turbine Generators (GTGs). For each GTGs, there are two exhaust sources which are the GTG room ventilation exhaust and the exhaust from the GTG. The minimum horizontal distance from the GTG exhaust to the MCR HVAC system's outside air intakes is approximately 72 feet. And the minimum horizontal distance from the GTG room ventilation fan exhaust vents to the outside air intakes is approximately 65 feet. These are well above the minimum of 10ft. required according to the International Mechanical Code (Ref. 9.4.8-26).

9.4.1.4 Testing and Inspection Requirements

The MCR HVAC system is provided with adequate instrumentation, temperature, flows, and differential pressure indicating devices to facilitate testing and verification of equipment function, heat transfer capability and flow blockage.

The MCR HVAC system is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters and ducts to verify their integrity, provided with proper access for initial and periodic inspection and maintenance activities.

Preoperational testing of the MCR HVAC system is performed as described in Chapter 14, Verification Programs, to verify that system is installed in accordance with applicable programs and specifications.

Routine testing of the MCR HVAC system is conducted in accordance with normal power plant requirements, facilitated by testing programs and written procedures. This testing demonstrates system and component operability and integrity.

Periodic surveillance testing of the MCR HVAC system is carried out in accordance with IEEE-338. This standard invokes periodic testing consisting of functional tests and checks, calibration verification and time response measurements as required, to ensure system function and availability.

During normal operation, equipment rotation is performed to minimize and equalize wear on redundant equipment.

Air handling units are factory tested in accordance with Air Movement and Control Association (AMCA) standards. Air filters are tested in accordance with the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) standards. Cooling coils are hydrostatically tested in accordance with ASME AG-1 (Ref. 9.4.8-2) and their performance is rated in accordance with the Air Conditioning and Refrigeration Institute (ARI) standards.

Ductwork is leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage

R/B controlled area, A/B controlled area, and sampling/laboratory area. The airflow from the containment low volume purge exhaust filtration unit exhausts through the vent stack, which also contains radiation monitors. These radiation monitors are used during all modes of operation. This design complies with GDC 64, Monitoring Radioactivity Releases, and GDC 63, Monitoring fuel and waste storage, as indicated in Section 11.5.

This redirects normal exhaust from radiological controlled area to HEPA and charcoal absorber filters in the containment low volume purge system. Thereby, this system arrangement meets the requirements of GDC 61 for normal plant conditions.

The auxiliary building HVAC system and containment low volume purge system arrangement for the fuel handling area meets the GDC 60 requirements for normal plant operation based on compliance with RG 1.140. However, based on the fuel handling accident analysis (Section 15.7.4) no credit is given for any filtration of released radionuclide's and the calculated offsite dose is well within the guideline dose limit values of 10 CFR 50.34. Therefore, compliance with GDC 60 and 61 is not required for the postulated fuel handling accident condition.

Airborne radioactivity is monitored inside the charging pumps areas. As shown in Figure 9.4.3-1 the merging in one duct of the A, B charging pump areas and the A, B annulus emergency exhaust filtration unit areas within the controlled area of the reactor building, the airflow in this duct is monitored by radiation monitor to determine if high levels of radioactivity are present. Under normal operating conditions, when high levels of radioactive material are not present, the airflow is routed through the normally open, air operated damper to the auxiliary building exhaust fans and then to the vent stack for release. Upon detection of high levels of radioactivity in this duct existing the controlled area of the reactor building, the normally closed, air operated damper is opened and the normally open damper is closed. The airflow in the duct is then routed to connect with the duct to the containment low volume purge exhaust filtration units, as shown on Figure 9.4.3-1, which will pass the radioactive exhaust air through a HEPA filter as well as through charcoal absorber filters. This filter arrangement will effectively remove the majority of radioactive materials from the exhaust air stream before it is sent to the vent stack for release. The vent stack also contains radiation monitors which are used during all modes of operation to provide assurance that the release of radioactive materials contained in gaseous effluents will not exceed the limits specified in 10 CFR 20. The arrangement shown in Figure 9.4.3-1, which allows the radiological controlled areas of the auxiliary building and reactor building to be filtered by the containment low volume purge exhaust filtration units, meets the GDC 60 requirements for normal plant operation based on compliance with RG 1.140.

To minimize the buildup of radioactive contamination within the ducts, the exhaust ducts are design/sized for the transport velocities needed to convey the radioactive contaminants without settling. Ducts for most nuclear exhaust and post-accident air cleanup systems should be sized for a minimum duct velocity of approximately 2,500 feet per minute (fpm).

The exhaust from the auxiliary building HVAC system is combined with the treated gaseous waste flow from the GWMS before being routed to the plant vent stack. This

Train pair A&B and train pair C&D, each is connected to a single air distribution system. The air distribution system is qualified in accordance with seismic category I requirements. Conditioned air is distributed to the following areas:

- Class 1E instrumentation and control (I&C) rooms
- Class 1E electrical rooms
- Class 1E uninterruptible power supply (UPS) rooms
- Class 1E Battery and battery charger rooms
- MCR/Class 1E electrical HVAC equipment rooms
- Remote shutdown console room
- Control rod drive mechanism (CRDM) cabinet room (non-safety)
- M-G set and M-G set panel rooms (non-safety)
- Leakage rate testing (LRT) room (non-safety)
- Reactor trip breaker room
- AA~~C~~C selector circuit panel room

The return air from these areas is drawn by the corresponding HVAC train through the seismic category I ductwork.

The volume of the air exhausted from battery rooms by the corresponding battery exhaust fans is sufficient to maintain the hydrogen concentration well below 1% by volume of battery room.

Rooms with high heat loss during the cold season are provided with non safety-related unit heaters or in-duct electric heaters in their supply air branches. These electric heaters are classified as equipment class 5 and seismic category II.

Upon the Class 1E electrical room high temperature, the chilled water control valve for the served air handling units is automatically positioned for full chilled water flow to prevent the temperature rise.

Upon the electric heating coil outlet high temperature, the electric heating coil is automatically tripped to prevent the abnormal heating.

9.4.5.2.2.1 Normal Operation Mode

During normal plant operation, safeguard component areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the safeguard component areas are cooled by individual safeguard component area air handling units. The safeguard component area includes the CS/RHR pump rooms, SI pump rooms, CS/RHR heart exchanger rooms, AHU rooms, R/B sump tank rooms.

A rise of the safeguard component area temperature reaching the setpoint of the switch is to a cause the associated fan to start. Reverse operation occurs upon a temperature decrease below the setpoint of the switch.

Each air handling unit consists of, in the direction of airflow, an electric heating coil, a cooling coil, a supply fan and associated controls. The safeguard component area HVAC system is shown in Figure 9.4.5-3 and the equipment design data is presented in Table 9.4.5-1. The COL Applicant is to determine the capacity of heating coils that are affected by site specific conditions. The cooling coils are supplied with chilled water from the essential chilled water system (Section 9.2.7).

Upon safeguard component area high temperature, the chilled water cooling coil control valve for the corresponding air handling units is automatically positioned for full chilled water flow to prevent the temperature rise.

Upon electric heating coil outlet high temperature, the electric heating coil is automatically tripped to prevent the abnormal heating.

The air handling unit trains A, B, C and D provide 100% of the heating and cooling requirements of their associated equipment room.

The function of the backdraft dampers at the common duct section that interfaces between the annulus emergency exhaust system and the auxiliary building HVAC system is described in Subsection 9.4.5.2.1.

9.4.5.2.4 Emergency Feedwater Pump Area HVAC System

During normal plant operation, emergency feedwater pump (motor-driven) areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the auxiliary building HVAC system is unavailable. The emergency feedwater pump (motor-driven) areas are cooled by individual air handling units. The emergency feedwater pump (turbine-driven) areas are cooled during normal plant operation and design basis accident or LOOP by an independent air handling unit. A rise of the emergency feedwater pump area temperature reaching the setpoint of the switch is to cause the associated fan to start. Reverse operation occurs upon a temperature decrease below the setpoint of the switch.

The emergency feedwater pump (motor-driven) area air handling unit consists of, in the direction of airflow, an electric heating coil, a cooling coil, a supply fan, and associated controls. The emergency feedwater pump (turbine-driven) area air handling unit consists of, in the direction of airflow, a low efficiency filter, an electric heating coil, a cooling coil, a supply fan, and associated controls. The emergency feedwater pump area HVAC system is shown in Figure 9.4.5-4 and the equipment design data is presented in Table

9.4.5-1. The COL Applicant is to determine the capacity of heating coils that are affected by site specific conditions. The cooling coils of the emergency feedwater pump area air handling units are supplied with chilled water from the essential chilled water system (Section 9.2.7).

Each of the air handling units provides 100% of the heating and cooling requirements of the associated equipment room.

Upon the emergency feedwater pump area high temperature, the chilled water control valve for the corresponding air handling units is automatically positioned for full chilled water flow to prevent the temperature rise.

Upon the electric heating coil outlet high temperature, the electric heating coil is automatically tripped to prevent the abnormal heating.

9.4.5.2.5 Safety Related Component Area HVAC System

During normal plant operation ESF equipment areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the auxiliary building HVAC system is unavailable. The safety related component areas are cooled by individual air handling units. A rise of the safety related component area temperature reaching the setpoint of the switch is to cause the associated fan to start. Reverse operation occurs upon a temperature decrease below the setpoint of the switch.

Each of the air handling units in the Penetration Areas, CCW pump Areas, Essential Chiller Unit Areas and Charging Pump Areas, each consists of, in the direction of airflow, an electric heating coil, a cooling coil, a supply fan, and associated controls. Each of the air handling units provides 100% of the heating and cooling requirements of the associated equipment room.

Each of the annulus emergency filtration unit air handling units, consists of, in direction of airflow, an electric heating coil, two cooling coils, a supply fan, and associated controls. Each of the air handling units provides 100% of the heating requirements of the associated equipment room. Each of the air handling units contains two 100% capacity cooling coils. Each cooling coil is served by a dedicated train of the essential chilled water system. Hence, the loss of one train will not affect the cooling capacity of the annulus emergency filtration unit area air handling units.

The safety-related component area HVAC system is shown in Figure 9.4.5-1 and 9.4.5-5 and the equipment design data is presented in Table 9.4.5-1. The COL Applicant is to determine the capacity of heating coils that are affected by site specific conditions. The cooling coils are supplied with chilled water from the essential chilled water system (Section 9.2.7).

Upon safety-related component area high temperature, the chilled water control valve for the corresponding air handling units is automatically positioned for full chilled water flow to prevent the temperature rise.

Table 9.4.5-1 Equipment Design Data (Sheet 3 of 4)

Annulus Emergency Filtration Unit Area Air Handling Unit	
Number of Units	2
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	1,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling ~~c~~Coil Quantity	2 per Unit Note; A Unit is provided the chilled water of ~~a~~A and B train. B Unit is provided the chilled water of C and D train.
Cooling Coil Capacity, btu/hr	10,000 / Coil
Heating Coil Type	Electric
Charging Pump Area Air Handling Unit	
Number of Units	2
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	1,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	10,000
Heating Coil Type	Electric
CCW Pump Area Handling Unit	
Number of Units	4
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	1,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	30,000
Heating Coil Type	Electric
Essential Chiller Unit Area Air Handling Unit	
Number of Units	4
Equipment Class	3
Seismic Category	I
Unit Airflow Capacity, cfm	1,000
Unit Fan Type	Centrifugal
Cooling Coil Type	Chilled Water
Cooling Coil Capacity, btu/hr	30,000
Heating Coil Type	Electric

- Assure floor drains (Subsection 9.3.3) are provided in safety-related equipment areas to remove expected fire fighting water flow.

- Provide fire-fighting personnel access and escape routes for each fire area or fire zone/compartment.

- Provide communications (Subsection 9.5.2) and emergency lighting (Subsection 9.5.3) that facilitate safe-shutdown following a fire.

- Minimize exposure to personnel and releases to the environment of radioactivity or hazardous chemicals as a result of a fire.

The fire protection system is classified as a non-safety related, non-seismic system. The fire protection system is not required to remain functional following a plant accident or the most severe natural phenomena. Seismic design requirements are applied to portions of the system located in areas containing equipment required for safe-shutdown following a safe-shutdown earthquake (SSE). In addition, the containment isolation valves and associated piping for the fire protection system are safety-related (Equipment Class 2) and seismic category I.

The fire protection system is designed to perform the following functions:

- Detect and locate fires and provide operator indication of the location.

- Provide the capability to extinguish fires in any plant area, to protect site personnel, limit fire damage, and enhance safe-shutdown capabilities.

- Supply fire suppression water at a flow rate and pressure sufficient to satisfy the demand of any automatic sprinkler system plus 500 gpm for fire hoses, for a minimum of 2-hours, but not less than 300,000gallons.

- Maintain 100% design capacity of fire pump, assuming failure of the largest fire pump or the loss of offsite power (LOOP).

- Following a SSE, provide water to hose stations for manual fire fighting in areas containing safe-shutdown equipment.

In order to accomplish the goals of the fire protection program, appropriate industry codes and standards are consulted in the design, construction, and operation of the US-APWR. Fire protection SCCs designed to NFPA codes and standards will use, as the code of record, those NFPA codes and standards which are in effect 180 days prior to the submittal of the application under 10 CFR 52. Deviations to any NFPA codes and standards are identified and justified in the fire hazards analysis. These deviations are not to degrade the performance of the fire protection systems or features.

The US-APWR design has four separate and redundant safety trains. Two safety trains can achieve safe-shutdown from the MCR, which eliminates the need for any operator manual actions that would require operators to enter any fire-involved areas. A remote

9.5.6 Gas Turbine Generator Starting System

The GTG starting system provides for a reliable GTG start following a LOOP. Each GTG consists of two gas turbines that drive one generator.

9.5.6.1 Design Bases

A. The GTG starting system initiates a start of the GTG such that within 100 seconds after receipt of the start signal, the GTG is operating at rated speed and is ready to begin load sequencing. This time frame is consistent with that assumed in the accident analyses presented in Chapter 15.

B. The GTG starting system is designed so that no single active failure, assuming a LOOP, can result in a complete loss of the standby power source function.

C. The GTG starting system is required to start the GTG upon receipt of a Class 1E bus undervoltage or an ECCS actuation signal.

D. The GTG starting system is designed to remain functional after a SSE.

E. Active components of the system can be tested during plant operation.

F. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with postulated rupture of piping is discussed in Section 3.6. Environmental design is discussed in Section 3.11.

G. Codes and standards applicable to the GTG starting system are listed in Section 3.2.

9.5.6.2 System Description

The GTG starting system is an air-powered system designed to start the GTG. Control for starting the GTG system are discussed in Chapter 7 and Section 8.3. The standby emergency power supply from the GTG (electrical side) is discussed in detail in Section 8.3.

The GTG air starting system is shown schematically in Figure 9.5.6-1. Each GTG starting air system is equipped with six (6) air compressors with an air cooler in each, three (3) drain chambers, two (2) air receivers, compressor air intake filters, two (2) air starting units that include solenoid valves, piping, valves, and associated instrumentation. Design parameters for the major system components are summarized in Table 9.5.6-1.

A GTG is composed of two gas turbine engines which have two air start motors respectively. The GTG starts ~~properly~~ by ~~turning effort of~~ four air start motors. An air receiver supplies air to two air start motors through two starting valves mounted on a starting unit.

The pressure switches support the automatic control modes of compressor and receiver operation.

9.5.7 Gas Turbine Lubrication System

A GTG lubrication system for each of the four GTGs provides essential lubrication to the GTG components. Each GTG consists of two gas turbines that drive one generator though one gear box.

9.5.7.1 Design Bases

The GTG lubrication system is designed to provide adequate lubrication under all operating conditions, including full load operation after starting, as required by the design basis.

Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against the dynamic effects associated with postulated rupture of piping is discussed in Section 3.6. Environmental qualification is discussed in Section 3.11.

A. The GTG lubrication system provides lubricating oil to all gas turbine bearings during GTG operation and shutdown.

B. The GTG lubrication system is designed to remain functional during and after a safe shutdown earthquake.

C. The GTG lubrication system is designed so that a single failure of any active component, assuming a LOOP, cannot result in complete loss of the power source function.

D. Active components of the system can be tested during plant operation.

E. Codes and standards applicable to the GTG lubrication system are listed in Section 3.2.

9.5.7.2 System Description

The lubrication system is shown schematically in Figure 9.5.7-1. Major components of the system include two gas turbine shaft driven pumps, a reduction gear box (including its oil reservoir), suction strainer at each oil pump's suction line, a full flow filter, a lube oil cooler for each pump, oil cooler fan, and associated valves, piping, and instrumentation. All components of the system are contained in GTG enclosure. Keep-warm system is not installed~~ basically~~, ~~ ~~since gas turbine lubrication oil ~~has cold-adapted feature~~performs under cold condition. Design parameters for major system components are provided in Table 9.5.7-1.

The GTG lubrication system circulates oil through a lube oil filter, a strainer, and then through the entire gas turbine.

When the GTG is operating, circulation is accomplished by the gas turbine shaft driven pumps, which draw oil from the reduction gear oil reservoir through a suction strainer, and passes it through a full-flow filter, a strainer, and air cooled lube oil cooler before distribution to the bearings. Requirement specification of lube~~fuel~~ oil consumption is 0.053 gal/h or less.

During operation of the gas turbine, failure of the gas turbine shaft driven pumps and spurious open of pressure regulating valves results in unsatisfactorily low lube oil pressure. Receipt of a low lube oil pressure signal from the trip logic will shut down the GTG during routine operation. The low lube oil temperature shutdown signal is bypassed or defeated during accident conditions. During starting of the gas turbine, GTG does not need pre-circulation of lube oil, because ball bearings are ~~adopted~~used. GTG can start without circulation of lube oil until shaft driven pumps start.

Loss of cooling to the lube oil cooler would cause a high lube oil temperature condition and alarm. The ~~fail to open of temperature regulating valves~~ failure of the temperature regulating valves to open would also cause a high lube oil temperature condition. Receipt of a high lube oil temperature signal from the trip logic will shut down the GTG during routine operation. The high lube oil temperature shutdown signal is bypassed or defeated during accident conditions.

9.5.7.3 Safety Evaluation

A. The gas turbine shaft driven pumps provide oil to the gas turbine bearings during GTG operation. Oil is kept at a constant pressure and temperature by use of regulating valves and a lube oil cooler.

B. ~~The~~All components of the systems are provided in GTG enclosure as a GTG package ~~designed to ASME Boiler & Pressure Vessel Code, Section III, Class 3. When a component is commercially unavailable as ASME Boiler & Pressure Vessel Code, Section III, Class 3 design, the component is~~and proven of equivalent quality to ASME Boiler & Pressure Vessel Code, Section III, Class 3. Equivalent quality of a component is interpreted to mean an item designed for commercial use is upgraded to ASME Boiler & Pressure Vessel Code Section III, Class 3 requirements through seismic design, testing, qualification and documentation (Ref. 9.5.4-7).

C. The lubrication system is designed in accordance with seismic category I requirements as specified in Section 3.2. System, equipment, and components which are not normally required to be seismic category I based on their safety function, but whose failure could impair the functioning of the air starting system are upgraded in design to seismic category I.

D. The lubricating oil supply to each gas turbine is sized to provide gas turbine lubrication. The lubrication system for each generator is capable of supplying lube oil for an extended period without augmentation from other sources. The lube oil pump is driven by the gas turbine with which it is associated. Because of these arrangements and the redundancy of emergency GTG design and installation, a failure of any single active component of the GTG lubrication system cannot result in a complete loss of the power source. A single failure is assessed as a failure of the gas turbine with which it is associated; in such a circumstance, safe shutdown is attained and maintained by the remaining GTGs.

E. All active components are capable of being tested during power generation operation to ensure proper functioning of the system (Subsection 9.5.7.4).

F. The ~~P~~power section of the gas turbine is designed~~,~~ so that the ~~absorbed energy~~ capacity of the casing to absorb energy is ~~beyond~~ greater than the kinetic energy of rotational parts of the turbine. ~~The m~~Missiles are not postulated to be generated by the GTG ~~are not postulated~~ as described in Section 3.5.

9.5.7.4 Inspection and Testing Requirements

The lubrication system is tested prior to initial startup. Preoperational testing is described in Section 14.2. System performance is verified during periodic GTG testing.

Inservice inspection of piping is performed in accordance with the requirements of ASME Section XI, as discussed in Section 6.6 (Ref. 9.5.4-11).

Technical Specification surveillance testing and inspection of the GTG lubricating oil system is performed to assure operational readiness, as described in Chapter 16.

The lubrication system is operationally tested during the startup and checkout of the gas turbine. Lube oil pressure and temperature are monitored to ensure operability of the gas turbine shaft driven pump. Inspection and testing of the system can be performed without disturbing normal plant operations. The lube oil in the gas turbine will be analyzed periodically for wear and failure parameters. The lube oil will have the following tests performed: kinematic viscosity, water content, wear metal content and all acid value. These tests will be performed and accepted in accordance with manufacturer's recommendation. Strainers may be removed and inspected for the buildup of impurities on a periodic basis.

9.5.7.5 Instrumentation Requirements

Instrumentation provided for the lubrication system includes pressure and temperature switches and indicators. Low lube oil pressure and~~,~~ high lube oil temperatures, are alarmed in the MCR and in the GTG room. In addition, local indications associated with the lubrication system that are provided, including oil temperature and pressure.

Lube oil tank level instrumentation is installed and low level is ~~alerted~~alarmed in the MCR and in the GTG room. Differential pressure instrumentation for filter and strainer are installed and high pressure is ~~alerted~~alarmed the MCR and in the GTG room. Low lube oil pressure and high lube oil temperature during operation of the GTG initiates a GTG trip, without postulated accident condition. GTG oil pressure and oil temperature trip logic initiates a GTG trip and alarms at the GTG control panel and the MCR. Both of these sensors are connected to common supply piping for No.1 bearing and No.2 bearing.

Table 9.5.7-1 Lubrication System Component Data

Main oil pump	
Quantity (per GTG)	2
Capacity (gpm)	56
Lube Oil Consumption (gpm)	0.00088
Relief valve set pressure (psig)	49
Design code	Manufacturer's standards
Driver	Gas turbine shaft driven
Seismic Category	I
Oil cooler	
Quantity (per GTG)	2
Type	Air Cooled
Air Cooling Fan	2
Codes and standards	Manufacturer's standards
Seismic Category	I
Fluid	Lubricating oil
Temperature in/out (ºF)	180/151.5
Flowrate (gpm)	56
Design pressure (psig)	125
Design temperature (ºF)	200
Material	Carbon steel
Reduction gear reservoir	
Quantity (per GTG)	1
Type	Horizontal, cylindrical
Capacity, each (gal)	95.1
Operating pressure/temperature (psig/ºF)	atm/170-180
Material	Carbon steel
Codes and standards	~~ASME Section III, Class 3~~Manufacture's standards
Seismic Category	I
Main oil filter	
Quantity (per GT)	2
Type	Full-flow, duplex, cartridge
Flowrate (gpm)	56
Particle retention capability (μm)	10
Design pressure/temperature (psig/μF)	150/200
Housing	Carbon steel
Code (pressure boundary)	Manufacture's Standard
Seismic Category	I
Main lube oil strainer	
Quantity (per GTG)	2
Flowrate (gpm)	56
Design pressure/temperature (psig/ºF)	150/200
Filtering capacity (μm)	150 mesh
Housing	Carbon steel
Screen	Stainless steel
Code (pressure boundary)	Manufacture's Standard
Seismic Category	I
Piping, fittings, and valves	
Material	Carbon steel
Design code, safety-related portion	Manufacture's Standard
Seismic Category	I

Chapter 10

US-APWR DCD Chapter 10 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.2-19	Section 10.2.3.5 The first sentence of the first paragraph	Editorial: clarify the language. Replaced “is to provide” with “provides”.
10.2-19	Section 10.2.3.5 The second sentence of the first paragraph	Editorial: clarify the language. Replaced “2008a” with “IWA-2430 of the 2007 Edition with 2008 Addenda of”.
10.2-19	Section 10.2.3.5 The second sentence of the first paragraph	Editorial: clarify the language. Deleted “IWA-2430 of the”.
10.2-19	Section 10.2.3.5 The third sentence of the first paragraph	Editorial: clarify the language. Deleted the third sentence.
10.3-10	Section 10.3.2.4.3 The first sentence of the first paragraph	Editorial: clarify the language. Replaced “consider” with “takes into consideration”.
10.3-10	Section 10.3.2.4.3 The second sentence of the first paragraph	Editorial: clarify the language. Replaced “steam” with “water (steam)”.
10.3-18	Section 10.3.6.3 The second sentence of the second paragraph	Editorial: clarify the language. Inserted “piping”.
10.3-18	Section 10.3.6.3 The third sentence of the second paragraph	Editorial: clarify the language. Inserted “piping”.
10.3-19	Section 10.3.6.3 The second paragraph Single-Phase Line Main feedwater line	Editorial: clarify the language. Deleted “in the”.
10.3-19	Section 10.3.6.3 The second paragraph Two-Phase Line Main steam line	Editorial: clarify the language. Replaced “is” with “of”.

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.3-19	Section 10.3.6.3 The second paragraph Two-Phase Line Feedwater heater drain piping	Editorial: clarify the language. Deleted “of” and “portion”.
10.3-19	Section 10.3.6.3 The second paragraph Two-Phase Line Feedwater heater drain piping	Editorial: clarify the language. Replaced “are” with “is”.
10.3-20	Section 10.3.6.3 The eighth sentence of the third paragraph	Editorial: clarify the language. Replaced “inspection” with “inspections”.
10.4-3	Section 10.4.1.3 The seventh sentence of the second paragraph	Editorial: clarify the language. Replaced ““During” with “during”.
10.4-3	Section 10.4.1.3 The eighth sentence of the second paragraph	Editorial: clarify the language. Deleted the double apostrophe.
10.4-3	Section 10.4.1.3 The last sentence of the second paragraph	Editorial: clarify the language. Separated the last sentence into two sentences.
10.4-7	Section 10.4.2.2.1 The last sentence of the fifth paragraph	Editorial: clarify the language. Deleted “Furthermore,”.
10.4-12	Section 10.4.3.2.2 The sixth sentence of the last paragraph	Editorial: clarify the language. Deleted “Furthermore,”.
10.4-12	Section 10.4.3.2.2 The last sentence of the last paragraph	Editorial: clarify the language. Deleted “Furthermore,”.

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-17	Section 10.4.4.4 The last sentence of the last paragraph	Editorial: clarify the language. Replaced “at the valve” with “at a valve”.
10.4-17	Section 10.4.4.4 The last sentence of the last paragraph	Editorial: clarify the language. Replaced “pressure 774 psig” with “pressure of 774 psig”.
10.4-21	Section 10.4.5.2.2.4 The first sentence	Editorial: clarify the language. Replaced “wakeup” with “makeup”.
10.4-82	Section 10.4.9.2.1 A. Emergency feedwater pumps The fourth sentence of the fourth paragraph	Editorial: clarify the language. Replaced “shares” with “share”.
10.4-82	Section 10.4.9.2.1 A. Emergency feedwater pumps The fifth sentence of the fourth paragraph	Editorial: clarify the language. Replaced “is given” with “has”.
10.4-82	Section 10.4.9.2.1 A. Emergency feedwater pumps The fifth sentence of the fourth paragraph	Editorial: clarify the language. Deleted “become”.
10.4-83	Section 10.4.9.2.1 D. Emergency feedwater pits The last sentence of the first paragraph	Editorial: clarify the language. Replaced “for the pit” with “from the pit”.
10.4-83	Section 10.4.9.2.1 D. Emergency feedwater pits The third sentence of the second paragraph	Editorial: clarify the language. Replaced “of keeping” with “by keeping”.

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Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-84	Section 10.4.9.2.1 D. Emergency feedwater pits The last sentence of the second paragraph	Editorial: clarify the language. Inserted “will”.
10.4-84	Section 10.4.9.2.1 D. Emergency feedwater pits The fourth paragraph	Editorial: clarify the language. Replaced “the stress” with “stress”.
10.4-84	Section 10.4.9.2.1 D. Emergency feedwater pits The second sentence of the last paragraph	Editorial: clarify the language. Replaced “bleed and feed” with “feed and bleed”.
10.4-87	Section 10.4.9.2.2 B. Operation during Plant Transients and Accidents (f) Station Blackout (SBO) The last sentence	Editorial: clarify the language. Replaced “are” with “is”.
10.4-88	Section 10.4.9.2.2 C. Water Hammer Prevention The second bullet of the first paragraph	Editorial: clarify the language. Combined two sentences into one sentence.
10.4-89	Section 10.4.9.2.3 The last sentence of the first paragraph	Editorial: clarify the language. Replaced “buse” with “buses”.
10.4-91	Section 10.4.9.3 The second sentence of the eighth paragraph	Editorial: clarify the language. Replaced “a back leakage” with “back leakage”.
10.4-91	Section 10.4.9.3 The second sentence of the eighth paragraph	Editorial: clarify the language. Replaced “retain” with “remain”.

Page	Location (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-91	Section 10.4.9.3 The second sentence of the eighth paragraph	Editorial: clarify the language. Replaced “, and then steam voids may be formed due to the back leakage, which may become the cause of” with “resulting in the formation of steam voids which could lead to”.
10.4-91	Section 10.4.9.3 The third sentence of the eighth paragraph	Editorial: clarify the language. Replaced “When the leakage continues the voids reaches into EFW-pump casing and into suction line and therefore, steam binding may occur which would make the EFW pump inoperable” with “As the leakage continues, the voids may reach the EFW pump casing and suction line creating the possibility for steam binding which would render the EFW pump inoperable”.
10.4-91	Section 10.4.9.3 The fourth sentence of the eighth paragraph	Editorial: clarify the language. Replaced “binding to the EFW pump” with “binding of the EFW pump”.
10.4-91	Section 10.4.9.3 The fourth sentence of the eighth paragraph	Editorial: clarify the language. Replaced “monitoring of the” with “monitoring the”.
10.4-91	Section 10.4.9.3 The fourth sentence of the eighth paragraph	Editorial: clarify the language. Replaced “check valves provides detection” with “check valve will provide early detection”.
10.4-91	Section 10.4.9.3 The fifth sentence of the eighth paragraph	Editorial: clarify the language. Replaced the sentence as following. “This is especially important during on-line maintenance that requires the pump discharge tie line to be open increasing the possibility for all EFW pumps to become inoperable.”
10.4-91	Section 10.4.9.3 The last sentence of the eighth paragraph	Editorial: clarify the language. Replaced the sentence as following. “Should leakage be detected when the tie line is open, prompt restoration will be performed by the following procedure.”
10.4-91	Section 10.4.9.3 The second bullet of the eighth paragraph	Editorial: clarify the language. Replaced “area” with “piping”.

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Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
10.4-91	Section 10.4.9.3 The second bullet of the eighth paragraph	Editorial: clarify the language. Replaced "the maintenance" with "maintenance".
10.4-91	Section 10.4.9.3 The third bullet of the eighth paragraph	Editorial: clarify the language. Replaced "performing the water filling of the isolated area, complete the restoration verifying" with "check valve maintenance refill the piping and verify".
10.4-91	Section 10.4.9.3 The third bullet of the eighth paragraph	Editorial: clarify the language. Deleted "in the".

- Tangential stresses will not cause a flaw, which is assumed to be twice the corrected ultrasonic examination reportable size, to grow to critical size in the design life of the rotor (refer to Subsection 10.2.3.2).

The low-pressure turbine has fully integral rotors forged from a single ingot of low alloy steel. This design is inherently less likely to have a failure resulting in a turbine missile than designs with shrunk-on discs. A major advantage of the fully integral rotor is the elimination of disc bores and keyways, which can be potential locations for stress risers and corrosive contaminant concentration. This difference results in a substantial reduction of rotor peak stresses, which in turn reduces the potential for crack initiation. The reduction in peak stress also permits selection of a material with improved ductility, toughness, and resistance to stress corrosion cracking.

The non-bored design of the high-pressure and low-pressure turbine rotor provides the necessary design margin by virtue of its inherently lower centerline stress. Metallurgical processes permit fabrication of the rotors without a center borehole. The use of solid rotor forgings was verified by an evaluation of the material removed from center-bored rotors for fossil power plants. This evaluation demonstrated that the material at the center of the rotors satisfied the rotor material specification requirements. Forgings for no-bore rotors are provided by suppliers who have been qualified based on bore material performance.

All the low-pressure turbine rotating blades are attached to the rotor using christmas tree, side entry type root.

10.2.3.5 Inservice Inspection

The inservice inspection program for the LP turbine ~~is to~~ provides assurance that rotor flaws that might lead to brittle failure of a rotor at speeds up to design speed will be detected. This inspection includes disassembly of the turbine at equal or less than 10-year intervals during plant shutdowns coincident with the inservice inspection schedule required by IWA-2430 of the 2007 Edition with 2008 Addenda of ~~2008a~~ Section XI, Division 1 ~~IWA-2430 of the~~ ASME Boiler & Pressure Vessel Code. ~~Disassembly of the turbine is conducted during plant shutdown.~~ Inspection of parts that are normally inaccessible when the turbine is assembled for operation (couplings, coupling bolts, turbine rotors, and low pressure turbine blades) is conducted.

The maintenance and inspection program plan for the turbine assembly and valves is based on turbine missile probability calculations, operating experience of similar equipment and inspection results. The turbine missile generation probability due to rotor material failure below design overspeed was submitted in Reference 10.2-9. The analysis of missile generation probability due to failure of the overspeed protection system is used to determine turbine valve test frequency and is described in Reference 10.2-10. The maintenance and inspection program includes the activities outlined below:

- This inspection consists of visual, surface, and volumetric examinations as indicated below:

10.3.2.3.6 Main Steam to Emergency Feedwater Pump Turbine

See Subsection 10.4.9, Emergency Feedwater System.

10.3.2.4 System Operation

10.3.2.4.1 Normal Operation

During startup, the main steam piping is heated by opening the MSBIV and thus controlling the steam flow. Main steam is not admitted to the main turbine until warmup of the main steam piping is accomplished. After warmup mode, secondary side no-load temperature and pressure are maintained automatically by the turbine bypass system which is maintained in the pressure control mode. When the reactor coolant temperature reaches 557°F (which is the no load temperature), the MSIVs are opened in a controlled manner. As the piping downstream of MSIVs is heated up, MSIVs are fully open and the MSBIVs are closed.

The MS/R 2nd reheat supply steam shutoff valve, control valve, bypass valve and warmup valve remain closed below 10% turbine load. With turbine load greater than 10%, heating steam is admitted by opening the warmup valve to the tube bundle.

During hot standby condition, the SG pressure is controlled by modulating TBVs and dumping steam to the condenser.

During plant cool down, decay and sensible heats are removed by dumping steam into the condenser via the TBVs. When the steam pressure falls below 125 psia, the steam dump is then stopped and cooldown is switched to the residual heat removal operation.

10.3.2.4.2 Emergency Operation

In the event that the plant must be shutdown due to accident or transient, the MSIVs with associated MSBIVs are closed. The MSDVs are used to remove the reactor decay heat and primary system sensible heat in order to cooldown the primary system to the conditions at which the residual heat removal system can perform the remaining cooldown function. If one of the MSDVs is unavailable, the respective safety valves associated with that main steam line provide overpressure protection. The remaining MSDVs are sufficient to cooldown the plant.

In the event of a design-basis accident, such as a main steam line break, the MSIVs with associated MSBIVs are automatically closed. In case the line break is downstream of the MSIV, even if a single failure of this valve is assumed, the MSIVs on the main steam piping of the intact SGs would prevent the steam blowdown through more than one SG.

10.3.2.4.3 Water (Steam) Hammer Prevention

The MSS design ~~consider~~ takes into consideration water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed. Refer to DCD subsection 3.12.5.3.5 Fluid Transient Loads for a description of water (steam) hammer caused by rapid valve closure and relief valve discharge loads in the piping analysis.

The material selection and fabrication methods used for Class 2 and 3 components conform to the following:

- In designing US-APWR, the material used for the piping and components of the CFS and the MSS conform with Appendix I to Section III (Reference 10.3-12), Parts A (Reference 10.3-13), Parts B (Reference 10.3-14), and Parts C (Reference 10.3-15) of Section II of the ASME Code Regulatory Guide 1.84 (Reference 10.3-16).

- Cleaning and handling of Class 2 and Class 3 components of the MSS and CFS are conducted in accordance with the acceptable procedures described in RG 1.37.

- The welding of low-alloy materials conform to the guidance provided in Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel" (Reference 10.3-19) for the MSS and the CFS. The minimum preheat temperatures for carbon steel and low alloy materials conform to the recommendations in ASME Section III, Appendix D, Article D-1000 (Reference 10.3-6).

- As for welds in areas of limited accessibility, the qualification procedure is specified in conformances with the guidance of Regulatory Guide 1.71 (Reference 10.3-20) (i.e., assurance of the integrity of welds in locations of restricted direct physical and visual accessibility) and as described with respect to all applicable components.

- The nondestructive examination procedures and acceptance criteria used for the examination of tubular products conform to the provisions of the ASME Code, Section III, Paragraphs NC/ND-2550 through 2570 (Reference 10.3-6). Refer to Section 6.6 for details on equipment class 2 and 3 components.

10.3.6.3 Flow-Accelerated Corrosion (FAC)

As noted in Subsection 10.3.6.2, MSS and CFS piping materials selected are corrosion resistant. CFS chemistry is controlled to have an environment that minimizes corrosion. This is further described in Subsection 10.3.5.

~~The following portions have the potential for FAC from past experiences in operating power plants and are included in FAC monitoring program. These~~ The following piping portions with potential for FAC are basically based on NUREG-1344 attached to GL 89-08. Generally, most of these piping portions are entirely made of carbon steel, however, materials for the piping portions extremely susceptible to FAC are FAC-resistant alloy (Cr-Mo steel, austenite stainless steel) taking into consideration past experiences. For other safety/non-safety carbon pipelines with relatively-mild FAC degradation identified in NUREG-1344 attached to GL 89-08, the initial thinning rate is prepared based on the actual measurement records from Japanese PWR nuclear power plants. Setting the initial thinning rate due to FAC is 0.2×10^{-4} mm/hr and also estimating operational rates, the additional thickness 70 mils is applied for the 10-year of design life.

Single-Phase Line

- Main feedwater line
  The piping from steam generator up to and excluding main feedwater equalization piping ~~in the~~ upstream of feedwater flow meter is made of high content of chrome-moly materials as shown in table 10.3.2-3. This portion is resistant to FAC. Other feedwater lines upstream of feedwater equalization piping are generally made of carbon steel with 10 year corrosion margin, however, material of extremely susceptible to FAC portion such as feedwater heater and elbows are made of FAC resistant alloy.
- Steam generator blowdown line (upstream of angle valves)
  This portion is made of carbon steel with 10 year corrosion margin.
- Main feedwater recirculation to condenser
  ~~This portion is made of carbon steel.~~ Most of this entire portion is made of carbon steel with 10 year corrosion margin, however, material of extremely susceptible to FAC portion such as condenser and elbows are changed to FAC-resistant alloy.
- Feedwater pump suction line
  ~~This portion is made of carbon steel.~~ Most of this entire portion is made of carbon steel with 10 year corrosion margin, however, material of extremely susceptible to FAC portion such as elbows are made of FAC-resistant alloy.
- Feedwater pump discharge line
  ~~This portion is made of carbon steel.~~ Most of this entire portion is made of carbon steel with 10 year corrosion margin, however, material of extremely susceptible to FAC portion such as feedwater pump discharge, feedwater pump minimum flow line and elbows are changed to FAC-resistant alloy.
- Condensate pump recirculation to condenser line
  ~~This portion is made of carbon steel.~~ Most of this entire portion is made of carbon steel with 10 year corrosion margin, however, material of extremely susceptible to FAC portion such as elbows are made of FAC-resistant alloy.

Two-Phase Line

- Main steam line
  This portion is made of carbon steel. There is no portion which is susceptible to FAC because of the low moisture ~~is~~of approximately 0.1 %.
- Turbine cross-over piping
  Most of this entire portion is made of carbon steel with 10 year corrosion margin.
- Turbine ~~C~~cross-under piping
  This portion is made of FAC-resistant alloy as shown in table 10.3.2-4. This portion is immune to FAC.
- Extraction steam line
  This portion is made of FAC-resistant alloy. This portion is immune to FAC.
- Feedwater heater drain piping
  Most of this entire portion is made of carbon steel with 10 year corrosion margin, however, material ~~of~~ extremely susceptible to FAC ~~portion~~ such as downstream of control valves ~~are~~is made of FAC-resistant alloy.

- Steam generator blowdown line (downstream of angle valves)
  ~~Most of this portion is entirely made of carbon steel, however, material for the portion extremely susceptible to FAC portion such as downstream of angle valves are~~ This portion is made of stainless steal or chrome-moly materials.

~~Corrosion allowance is the difference between the actual minimum wall thicknesses after any wall thinning that occurs during fabrication, and the required design wall thickness.~~ As for the safety/non-safety carbon pipelines with relatively-mild FAC degradation, ~~T~~the required design wall thickness is determined based on piping design pressure/ temperature and allowable stress in accordance with ASME Sec.III NX-3641 or ASME B31.1 paragraph 104. The specified wall thickness (prior to fabrication) is a standardized wall thickness stipulated in ASME B36.10M and ASME B36.19M. It is specified to exceed the required design wall thickness with consideration of minus tolerances of the thicknesses by a large and appropriate amount to account for the expected wall thinning during fabrication and FAC aging degradation of 70 mils. ~~The fabrication thinning is controlled by establishing fabrication tolerances.~~ As for the safety/non-safety carbon pipelines with relatively-mild FAC degradation, the FAC monitoring program based on EPRI "Recommendations for an effective Flow-Accelerated Corrosion Program (NSAC-202L-R2)" shall be prepared and implemented by using knowledge acquired from experiences of pipe wall thinning managements via Electric Power Companies in USA and Japan. The FAC monitoring program ~~provided by COL applicant~~ will include preservice thickness measurements of as-built piping considered susceptible to FAC. By performing this preservice measurement, the piping thickness margin that will be used as a wall thinning margin will be known, and then by combining the measurement with regular inspections, the frequency of the pipe replacement will be predicted. Integrity and safety of a plant is assured ~~by the COL applicant~~ by conducting inspection and maintenance during ~~over 60 years of~~ the service life of the plant and replacing piping if necessary.

The US-APWR design and piping layout has considered several features for the various piping systems to minimize incidence of FAC in piping. These features include:

- elimination of high turbulence points wherever possible (example: adequate straight pipe length downstream of flow orifice or control valve, etc)
- use of long radius elbows
- smooth transition at shop or field welds
- selection of pipe diameter to have velocities within industry recommended values
- use of corrosion resistant materials
- use of austenite stainless steel and P11 and P22 chrome-moly materials

The type of fluid, flow rates, fluid temperatures and pressure of ASME Code Class 2 and 3 piping for steam and feedwater system are shown in Table 10.3.2-6.

The Combined License Applicant will provide a description of the FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam and are susceptible to erosion-corrosion damage. The

Condensate polishing system is taken into service, when the circulating water in-leakage is detected. The permissible cooling water in-leakage and the length of time the condenser may operate with in-leakage without affecting the condensate/feedwater quality for safe reactor operation is described in Subsection 10.4.6.

The condenser tube cleaning system performs mechanical cleaning of the circulating water side of the titanium tubes. This cleaning, along with chemical treatment of the circulating water, reduces fouling and helps to maintain the thermal performance of the condenser.

10.4.1.3 Safety Evaluation

The main condenser has no safety-related function and therefore requires no nuclear safety evaluation.

During normal operation and shutdown, the main condenser has no significant inventory of radioactive contaminants. Radioactive contaminants may enter through a steam generator tube leak. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated operating concentrations of radioactive contaminants, is included in Chapter 11. Concerning secondary side chemical injection under normal operating conditions, pH controller and oxygen scavenger are injected as described in Subsection 10.4.10. Regarding source of hydrogen, thermal decomposition of hydrazine described in Subsection 10.4.10.2.2.2 can be considered. But “Air, nitrogen, and ammonia are mainly included in these noncondensable gasses.” as described in the third paragraph of Subsection 10.4.2.2.1. Therefore, the potential for hydrogen buildup within the condenser shells does not exist. Furthermore, since ~~“D~~during normal plant operation, noncondensable gases are removed from the main condenser by the operation of one or two vacuum pumps. If one pump trips, the condition is alarmed in the main control room, and the standby pump is started~~.”~~ as described in the second paragraph of Subsection10.4.2.2.3, the potential for hydrogen buildup within the condenser shells does not exist due to pump failure. Therefore, no hydrogen buildup in the main condenser is anticipated. The failure of the main condenser and any resultant flooding will not preclude operation of any essential system since no safety-related equipment is located in the turbine building and the water cannot reach safety-related equipment located in Category I plant structures, since in the yard area, the flood volume is directed away from the plant structures by virtue of the site grading and yard drainage system~~, i~~In addition, the water tight doors are installed in the doorways at ground level between T/B and R/B as described in Subsection 3.4.1.3.

10.4.1.4 Tests and Inspections

The condenser water boxes are hydrostatically tested after erection. Condenser shells are tested by completely filling them with water. Tube joints are leak tested during construction.

10.4.1.5 Instrumentation Applications

The main condenser hotwell is equipped with level control devices for control of automatic makeup and rejection of condensate. The condensate level in the condenser this oxygen will be released in the condenser, and the amounts are considered negligible compared to the large amounts of air being evacuated by the MCES. Therefore, the potential for explosive mixtures within the condenser shells does not exist.

The turbine component cooling water system provides the cooling for the vacuum pump seal water cooler. The vacuum pump seal water cooler uses turbine component cooling water so that the seal water is kept cooler than the saturation temperature of the condenser at its operating pressure to maintain the required pump performance.

The noncondensable gases removed from the main condenser and exhausted by the vacuum pumps are directed to the vent of the MCES. The exhaust flow is monitored for radioactivity prior to exhaust to environment. The noncondensable gases that are exhausted to the environment from the MCES are not normally radioactive. However, it is possible for the noncondensable gases to become contaminated in the event of primary-to-secondary system leakage. When an unacceptable radioactivity level is detected in the exhaust flow, adequate operating procedures are implemented. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated release from the system, is included in Chapter 11. The statement regarding the key elements, unacceptable levels of radiation and alarm set points to preclude significant releases radiation is addressed in Subsection 11.5.2.4.2. ~~Furthermore, t~~The statement regarding the location of the detectors is shown in Figure 11.5-1i and Figure 11.5-2c.

As long as the MCES is operable, the reactor coolant system operation is not affected. When the MCES becomes inoperable, a gradual decrease in condenser vacuum would result from the buildup of noncondensable gases. This decrease in condenser vacuum would cause a decrease in the turbine cycle efficiency. If the MCES remains inoperable, the condenser vacuum decreases to the turbine trip setpoint and a turbine trip is initiated.

A loss of condenser vacuum incident is described in Subsection 15.2.3.

10.4.2.2.2 Component Description

The MCES consists of three vacuum pumps. Each vacuum pump is supplied as packaged units and includes a liquid ring type vacuum pump, seal water cooler, seal water pump, separator tank.

The seal water pump supplies seal water from the separator tank to the vacuum pump. The seal water is used to seal clearances in the pump and also to condense vapor at the inlet to the pump.

The seal water cooler is installed between the vacuum pump and the seal water pump and cools the seal water by the turbine component cooling water. The seal water flows through the shell side of the seal water cooler and the turbine component cooling water flows through the tube side.

The separator tank separates mist water from noncondensable gases and store up the separated water. Seal water make up is provided to the separator tank by the condensate system and demineralized water system.

returned to the No. 1 feedwater heaters via the spillover control valve which automatically opens to bypass excess steam from the GSS.

During the initial startup phase of turbine-generator operation, steam is supplied to the gland seal system from the auxiliary steam header which is supplied from the auxiliary boiler. At times other than the initial startup, turbine-generator sealing steam is supplied either from the auxiliary steam system, or from the main steam system.

At the outer ends of the glands, collection piping routes the mixture of air and excess seal steam to the gland steam condenser. The gland steam condenser is a shell and tube type heat exchanger where the steam-air mixture from the turbine seals is discharged into the shell side and condensate flows through the tube side as a cooling medium. The gland seal condenser internal pressure is maintained at a slight vacuum by a motor-operated exhaust fan. There are two-100-percent exhaust fans mounted in parallel. Condensate from the steam-air mixture drains to the main condenser via the condensate recovery tank while non-condensable gases are exhausted to the atmosphere.

The mixture of non-condensable gases discharged from the gland steam condenser exhaust fan is not normally radioactive; however, in the event of significant primary-to-secondary system leakage due to a steam generator tube leak, it is possible to discharge radioactively contaminated gases. The GSS effluents are monitored by a radiation monitor installed on the gland steam condenser exhaust fan discharge line. Upon detection of unacceptable levels of radiation, operating procedures are implemented. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated release from the system, is addressed in Chapter 11. The statement regarding the key elements and system is addressed in 2nd paragraph of Subsection 11.5.2.4.3. ~~Furthermore, t~~The statement regarding unacceptable levels of radiation and provision of alarms and corresponding set points to preclude significant release of radiation is addressed in 1st paragraph of Subsection 11.5.2.4.3. ~~Furthermore, t~~The detail on how the effluents are discharged to the environment is shown on Figure 11.5-1j and Figure 11.5-2g.

10.4.3.3 Safety Evaluation

The gland seal system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.3.4 Tests and Inspections

The testing and the inspection will be performed in accordance with written procedures during the initial testing and operation program in accordance with the requirements of Chapter 14.

10.4.3.5 Instrumentation Applications

A pressure controller is provided to maintain the steam-seal supply header pressure by providing signals to the steam-seal control valve. Pneumatic control valves are used to provide appropriate pressure to both the low- and high-pressure turbine glands.

The load rejection controller prevents a large increase in the reactor coolant temperature following a large, sudden load decrease. The error signal is a difference between the lead-lag compensated selected T_{avg} and the selected Tref based on turbine inlet pressure and a difference between the nuclear power signal and the turbine inlet pressure with a rate-lag compensation.

Following a turbine trip, the load rejection controller is defeated and the turbine trip controller becomes active. The error signal is a difference between the lead-lag compensated T_{avg} and the no-load reference T_{avg}.

The pressure control mode is used at no-load operational mode. Pressure mode control is used to remove decay heat during plant startup and cooldown. The difference between the steam equalization piping pressure and a pressure set point is used to control the turbine bypass flow. The pressure set point is manually adjustable and is based on the desired reactor system coolant temperature.

10.4.4.4 Safety Evaluation

The TBS serves no safety function and has no safety design basis. There are no safety-related equipment/components in the vicinity of the TBS components. All high-energy lines of the TBS are located in the turbine building.

The failure of a TBS high-energy line will not disable the turbine speed control system.

The bypass valves fail closed upon loss of motive air power or electric signal. This is to prevent the possibility of the primary side of the plant from over cooling. In this case, MSRVs provide the controlled cooldown. In the unlikely event that one of the TBVs sticks wide open, the maximum steam flow through one valve at full load main steam pressure is less than the maximum permissible flow to limit a reactor transient.

The TBS is designed to bypass steam to the main condenser during normal plant shutdown. The system removes the residual heat and cools the reactor coolant system to a point where the RHR system is placed in service for further cooldown. Three TBVs with 13.4 % of rated main steam flow of 20,200,000 lb/h at ~~the~~a valve inlet pressure of 774 psig perform adequate decay heat removal to keep the cooldown rate of reactor coolant system at 50 deg.F/h during normal plant shutdown and thereby reduce the demands on systems important to safety in meeting GDC 34.

10.4.4.5 Inspection and Tests

Before the system is placed in service, all TBVs are tested for operability. The pipelines are hydrostatically tested to verify leak tightness. All piping and valves are accessible for inspection.

Additional description of inspection and tests is provided in Section 14.2.

10.4.4.6 Instrumentation Application

Instrumentation for the TBS is described in Section 7.7. Controls are provided in the main control room for the system operating mode selection. Pressure indication and the valve position indication are provided in the main control room.

Mechanical draft cooling towers have been selected for the CWS.

There are two CTWs each with 30 cells. Each cooling tower is arranged in two rows of 15 cells in each row, with the rows arranged back to back.

The cooling towers are located outdoors, a sufficient distance from any equipment or structure important to reactor safety.

The cooling towers and foundation are designed for wind load and earthquake loads.

10.4.5.2.2.3 Condenser tube cleaning

A condenser tube cleaning system is provided.

10.4.5.2.2.4 Cooling Tower Makeup Water Pumps

Two 100% capacity makeup water pumps provide ~~wakeup~~makeup water. The makeup water pump provides the makeup water to the cooling tower basins. The makeup water pumps are vertical, driven by electric motors and are located in the raw water intake structure.

10.4.5.2.2.5 Blowdown Pumps

Two 100% capacity CTW blowdown pumps are located in each cooling tower basin. These pumps take suction from the CTW basin and discharge into the raw water source.

10.4.5.2.2.6 Piping and Valves

All above ground CWS piping is carbon steel piping designed, fabricated, installed and tested in accordance with ASME B31.1 Power Piping Code (Reference 10.4-3), with an internal coating of corrosion preventive compound. The underground portions of the circulating water system piping are constructed of pre-stressed concrete pressure piping with lining. The piping is arranged to allow easy access for inspection (i.e., access man ways for the large CWS underground pre-stressed concrete headers).

Motor-operated butterfly valves are provided in each of the circulating water lines at the inlet and exit from the condenser shell to allow isolation of portions of the condenser. Motor-operated butterfly valves are also provided at the discharge of each circulating water pump. Control valves are provided for the regulation of cooling tower blowdown and makeup.

10.4.5.2.2.7 Main Condenser

Refer to Subsection 10.4.1.

10.4.5.2.2.8 Chemical Injection

Biocide, algaecide, pH adjuster, corrosion inhibitor, and silt dispersant are injected into the CWS by the chemical injection system to maintain a non-scale forming condition and main feedwater line; isolation of the faulted SG and the termination of flow to the faulted SG limits the RCS cooldown and mass/energy release to the containment.

C. Steam generator tube rupture

Upon detection of a water level increase of the SG, the EFW isolation valves and EFW control valves are automatically closed.

The failure modes and effects analysis given in Table 10.4.9-4 demonstrates that required EFW flow is ensured to the SGs during postulated accident conditions with a single failure in the EFWS.

10.4.9.2.1 Description of Major Components

A description of the major components and features in the EFWS is as follows:

A. Emergency feedwater pumps

Each EFW pump is normally aligned to feed one SG. Each EFW pump takes suction from one of two EFW pits and the discharge flow is directed to one of the four SGs.

The EFW pump is designed to develop adequate head to supply the design flow of at least 400 gpm to each SG, when the SG pressure is equivalent to the set pressure of the first stage of the main steam safety valve (safety valve with lowest set pressure) plus 3% of accumulation.

The maximum EFW pump flow is limited by the motor-operated EFW control valves which have a preset open position.

A mini flow line from the EFW pump discharge line to the EFW pit with a normally open valve and an orifice is provided to maintain minimum recirculation flow required for pump protection. The minimum flow line ensures a minimum recirculation flow for pump cooling whenever the pumps are running. A and B EFW pump shares their minimum flow line. C and D EFW pump also share~~s~~ their minimum flow line. Following the requirements in NRC IE Bulletin IEB 88-04, the minimum flow line ~~is given~~has sufficient capacity so that either of the pumps which share a minimum flow line does not ~~become~~ dead-head. A separate full flow line with a normally closed valve and an orifice allows pump testing during normal plant operation at the pump design flow rate without injection to the SGs. Both the mini flow line and full flow line are routed to the EFW pit by a common header.

Two motor-driven and two turbine-driven EFW pumps, with different power supplies are provided. Two motor-driven EFW pumps connect to each different safety ac bus to achieve the specific safety functions in case of off-site power loss; each bus is backed by a redundant emergency power source. Table 10.4.9-6 presents the power sources for EFWS components.

The EFW pumps automatically start on receipt of LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low steam generator water level signal in any one of SGs.

B. Motor-driven (M/D) emergency feedwater pumps

Two of the four EFW pumps are horizontal, centrifugal pumps driven by electric motors which are supplied with power from independent, Class 1E Safety ac bus. Each motor-driven pump has a capacity of 450 gpm. The capacity of each motor-driven pump is based on the required flow of 400 gpm to SG and 50 gpm through miniflow line. The design parameters of the pump and the motor are provided in Table 10.4.9-1.

C. Turbine-driven (T/D) emergency feedwater pumps

Two of the four EFW pumps are turbine-driven providing diversity of motive pumping power. The pump is a horizontal, centrifugal unit with a capacity of 550 gpm. The capacity of each turbine-driven pump is based on the required flow of 400 gpm to SG and 150 gpm through miniflow line.

The steam supply line to each T/D EFW pump turbine is connected to main steam lines from two SGs. Steam supply piping to the turbine driver for the A-EFW pump is taken from the two main steam lines (A-main steam Line and B-main steam Line) and the steam supply piping to the turbine driver for the D-EFW pump is taken from the two main steam lines (C-main steam Line and D-main steam Line). The steam supply connection is made upstream of the MSIVs. The motor-operated isolation valve and a check valve are provided in each of these steam lines to the EFW pump turbine. The check valves prevent blowdown from an intact SG into a faulted SG. The MOV provides isolation of these lines in case of a SGTR. The steam line to each T/D-EFW pump is also provided with a normally closed motor-operated EFW pump actuation valve. Opening of this valve starts the T/D EFW pumps. The steam discharge from the T/D-EFW pumps is routed to the atmosphere. The design parameters of the pump and the motor are provided in Table 10.4.9-1.

D. Emergency feedwater pits

Two 50% EFW pits are provided. The EFW pits are completely enclosed stainless steel lined structures that do not contain any operating equipment. All components inside the pit are also constructed of stainless steel. No foreign materials intrusion is anticipated. An access hatch located above the 100 % water level is available for inspections of pit interior areas. The EFW pits are filled with clean demineralized water. Filtration is not required. Both EFW pits together contain the minimum water volume required for maintaining the plant at hot standby condition for 8 hours and performing plant cooldown for 6 hours until the RHRS can start to operate. The inside dimensions of each pit is approximately 28 feet long, approximately 42 feet wide and approximately 35 feet depth. With the minimum pit level at approximately 26 feet during normal plant condition, the volume of water in each pit available for the EFW is 186,200 gallon. With two pits, each pit with a capacity of 204,850 gallons, is sufficient to perform hot standby and plant cooldown until the RHRS starts to perform heat removal. And also each pit has adequate capacity ~~for~~from the pit low level alarm setpoint to allow at least 20 minutes for operator action in accordance with the additional short-term recommendation “Primary EFW Water Source Low Level Alarm,” of generic recommendations of NUREG-0611 and NUREG-0635.

The makeup line routed from the demineralized water storage tank to the EFW pit is used for initial water fill of the EFW pits and to provide makeup water to maintain the water level in the EFW pits during normal plant operation. The demineralized water storage tank provides a backup source for EFWS. Due to a sufficient volume of water in the EFW pits for safe shutdown ~~of~~by keeping the plant at hot standby for 8 hours and performing plant cooldown to RHR entry condition for 6 hours after accident or transient, this backup supply is not required to be safety-related. The manual valves from the demineralized water storage tank to the EFW pumps are normally closed. If the water level of both EFW pits reaches low-low water level after an accident or transient without stabilizing at MODE 4 condition, the manual isolation valve will be opened by an operator. Before opening the isolation valve, the operator will verify that the storage tank has adequate water level to keep sufficient NPSH of the EFW pumps.

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. In this case, to prevent depletion of the water source from one pit, the tie line valves at the EFW pit outlet are required to be opened within about 8 hours after starting EFW pumps to perform continuous feedwater supply to the intact SGs. The design parameters of the EFW pit are provided in Table 10.4.9-1.

Because the EFW pits have the water supplied directly from condensate storage tank without deaerating and the inventory water of the pit has direct contact with atmosphere, the dissolved oxygen level of the pit inventory is not zero, however, because the design temperature of the EFW pit is 105 Deg F, which is determined to exceed assumed maximum operating temperature of the EFWS, ~~the~~ stress corrosion cracking would not occur in such low temperature condition even if the level of dissolved oxygen is high, therefore, the EFW pits have adequate integrity.

Sampling of the EFW pits is performed monthly, and turbidity is ensured to be not over 1 ppm. Any deviation is corrected by utilizing ~~bleed~~feed and ~~feed~~bleed method. Demineralized water from the demineralized water storage tank (make-up water source) is used for feeding the water inventory. Complete inspections with the pits drained will be performed periodically per the ISI program.

E. Emergency feedwater control valves

The normally open motor-operated globe control valves are provided in the EFW pump discharge lines to each SG for controlling the EFW flow. The control valve pre-set open position is established during pre-operational testing to limit the maximum flow during steam line break accidents. These flow control valves also provide isolation function of the EFW to the faulty SG.

The motor-operated valves are normally-open and verified whether they are in pre-set open position at startup of the EFW pump on receipt of open check signal such as LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low steam generator water level signal in any one of the SGs. The design parameters of these valves are provided in Table 10.4.9-1.

F. Emergency feedwater isolation valves

The motor-operated gate isolation valves are provided in the EFW lines routed from the EFW pump to each SG for isolation of the EFW to the faulty SG.

The motor-operated valves are normally-open and verified whether they are in fully open position at startup of the EFW pump on receipt of a open check signal such as LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low SG water level signal in any one of SGs. The motor-operated valves are also closed on

EFW spills out of the break, resulting in reduction of heat removal in the secondary side and leading to temperature increase of the RCS. Hence, it is necessary to isolate the faulted SG and supply EFW to the intact SGs.

The EFW pump automatically starts following FLB. Upon detection of a main steam pressure decrease in the faulted loop, the faulted loop is automatically isolated and continuous EFW is supplied to the intact SGs.

(e) Main Steam Line Break (MSLB)

The most limiting condition resulting from a spectrum of MSLB is a double-ended rupture of a main steam line, occurring at zero power. The accident results in a severe cooldown transient. The EFWS is expected to provide the maximum SG feedwater flow rate because that makes the cooldown more severe until the affected SG is isolated. The EFWS is required to limit its feed flow to the SGs, especially to the faulted SG. The flow from the EFW line to the faulted SG is isolated automatically as described in the FLB accident analysis. The EFW function is not needed during the mitigation of the MSLB accident, but is needed only for cooldown up to the RHR system initiation.

(f) Station Blackout (SBO)

A SBO results in the loss of normal offsite and emergency onsite ac power sources. The M/D-EFW pumps are inoperable because there is no ac power. Both T/D EFW pumps are available because of the dc power supplied by class 1E batteries with two hours capacities. EFW flow control is also available because the EFW flow control valves are powered by dc power which is available from class 1E batteries. In addition, at least within one hour after the SBO occurrence, one unit of the AAC-GTG is started, and by the operation of one unit of emergency feedwater pump (turbine-driven) area air handling units, the integrity of one unit of T/D EFW pump is ensured. The AAC-GTGs minimize the potential for common cause failures with the Class 1E GTG as discussed in Section 8.4.1.3. From the above, because the AAC GTGs are available during SBO event, in accordance with the generic recommendations of NUREG-0611 and NUREG-0635 Generic Short Term Recommendation No. 5 (GS-5), the EFWS is capable of providing required EFW flow for at least two hours from one T/D-EFW pump. After starting the operation of the AAC-GTG, charging to the Class 1E batteries ~~are~~is resumed, therefore, the turbine-driven EFW pump is able to continue to operate after two hours of the SBO and is independent of any ac power source.

(g) Anticipated Transient Without Scram (ATWS)

The acceptance criteria for an ATWS is to provide adequate heat removal such that the maximum RCS pressure is limited to less than the emergency stress limit. For this event, the EFWS is actuated by the DAS (diverse actuation system).

(h) Steam Generator Tube Rupture (SGTR)

The SGTR is a postulated accident that assumes that, a SGTR and the reactor coolant flows to the secondary side of the SG. The EFW pump automatically starts on receipt of an ECCS actuation signal. Upon detection of a water level increase in the faulted SG, the EFW isolation valve to the all SG is automatically closed. When all pumps start and operate without failure, the SG water level is verified in all

SGs. If there is no potential for decrease in SG level, the pump is stopped depending on the condition. The emergency operating procedures provide additional details for operator actions during the accident conditions.

A summary of system performance for various accident conditions is provided in Table 10.4.9-3. The table includes flows to both the faulted and intact SGs. Comparing these data with those in Table 10.4.9-2, it is seen that minimum flow requirements for the intact SGs are satisfied under all failure modes.

C. Water Hammer Prevention

The following items are identified as water hammer prevention and mitigation measures in EFWS.

- Automatic initiation of EFW flow following a loss of main feedwater flow to prevent draining of the SG feedring in accordance with NUREG-0927
- Implementation of EFW pipe refill flow limits to minimize steam-water entrainment and subsequent formation of water slug~~. I~~in accordance with BTP 10-2
- Detection of a high temperature main feedwater back leakage from an EFW check valve which becomes the cause of water hammer

The Combined License Applicant is to provide operating and maintenance procedures in accordance with NUREG-0927 and a milestone schedule for implementation of the procedure. The procedures should address:

- Prevention of rapid valve motion
- Introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Introduction of steam or heated water that can flash into water-filled lines and components
- Introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

10.4.9.2.3 Testing and Inspection Requirements

The EFW pumps are hydrostatically tested by the pump vendor in accordance with American Society of Mechanical Engineers (ASME) Section III (Reference 10.4-8), Class 3. Prior to initial plant start-up, the entire EFWS is hydrostatically tested after the installation is complete in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (Reference 10.4-8), Class 3. Chapter 14, Initial Test Program, describes testing to verify component installation and initial operation including a pump endurance test in accordance with the additional short-term recommendation “EFW Pump Endurance Test” in the generic recommendations of “NUREG-0611 and

NUREG-0635” and the testing of transfer between normal and emergency buses, as well as integrated system testing.

Periodic testing in accordance with Technical Specifications is performed during normal plant operation. The EFWS is designed with provisions for full design flow testing of EFW pumps during normal plant operation. Each pump has a higher capacity orifice line in parallel with the miniflow orifice line to allow the pump to be operated at its design flow rate without injecting water into the SGs during periodic inservice testing. See Section 3.9 for inservice testing and inspection requirements. The EFWS, its initiating signals, and its circuits are capable of being tested periodically while the plant is at power, in accordance with the frequency specified in the Technical Specifications.

During periodic testing of the EFW pumps, manual valve alignment is required. Only one EFW pump is tested at a time. Because each EFW pump is capable of providing 50% of the total required flow, full system flow requirements is available at all times. Additionally, when these valves are changed from their normal position, an alarm is annunciated in the control room to alert the operators. After finishing the periodic testing of EFW pumps, an operator determines that the EFWS valves are properly aligned and a second operator independently verifies that the valves are properly aligned.

10.4.9.2.4 Instrumentation Requirements

The EFWS includes appropriate instrumentation inputs to the safety-related instrumentation and control systems to perform the following functions:

- Automatic actuation of safeguards systems and components following an accident or transient.
- Monitoring of the EFWS process parameters to confirm proper EFWS operation.

The automatic initiation signals and circuits are designed so that their failure does not result in the loss of manual initiation from the control room in accordance with Regulatory Guide 1.62 (Reference 10.4-16). The engineered safety features system details are provided in Section 7.3.

The EFWS also includes appropriate controls to allow for manual actuation and/or control of EFWS components if necessary, such as backup manual actuation of components that did not automatically actuate.

The EFW flow element/transmitter is provided in each EFW line to the SG, to transmit the flow rate signal to the indication in the MCR. The pressure transmitter is provided at the discharge line of the EFW pump to transmit the pressure signal to the indication in the MCR. Two channels of the level transmitters are provided at each pit to indicate the water level of the EFW pits during normal plant condition, monitor water level following an accident and annunciate abnormal water level. The EFW discharge line temperature upstream of the EFW flow control valves is monitored. A high temperature alarm in the MCR is an indication of the back leakage of the check valve, requiring operator action.

Safety-related display instrumentation related to the EFWS is discussed in Section 7.5. Information indicative of the readiness of the EFWS prior to operation and the status of active components during system operation is displayed for the operator in the MCR and at the remote shutdown console. See Section 7.4 for details. The indication and controls provided for the EFWS are summarized in Table 10.4.9-5.

The EFW Pump capacities and start times (maximum of 140 seconds for M/D pump and 60 seconds for T/D pump) are established such that the above objectives are met and the EFW Pumps can deliver the required flow for all conditions as given in Tables 10.4.9-2 and 10.4.9-3. Pump head is sufficient to establish the minimum necessary flow rate against the SG pressure corresponding to the first stage main steam safety valve set pressure plus 3% accumulation pressure. The maximum time to start the electric motors and the steam turbines which drive the EFW pumps are chosen so that sufficient flow can be supplied to SGs during the feedwater line break event which can result in reactor core damage. See Section 15.2 for details.

With the low-low water level in the EFW pits the available net positive suction head (NPSH) to the M/D EFW pumps is 97 feet, while the maximum required NPSH is 73 feet providing adequate margin. The available net positive suction head (NPSH) to the T/D EFW pumps is 100 feet, while the maximum required NPSH is 76 feet providing sufficient margin.

The EFWS is designed to reduce the probability of steam binding. When ~~a~~ back leakage from an EFW check valve occurs, high temperature water from the main feedwater line will ~~retain~~remain around the check valve~~, and then steam voids may be formed due to the back leakage, which may become the cause of~~ resulting in the formation of steam voids which could lead to water hammer. ~~When~~As the leakage continues, the voids may reach~~es~~ ~~into~~the EFW~~-~~ pump casing and ~~into~~ suction line ~~and therefore, steam binding may occur which would make~~ creating the possibility for steam binding which would render the EFW pump inoperable. ~~-~~To avoid water hammer and steam binding ~~to~~of the EFW pump, monitoring ~~of~~ the EFW discharge line temperature upstream of the EFW check valve~~s~~ will provide~~s~~ early detection of back leakage~~, which requires~~ and allow for prompt corrective action. ~~These are~~This is especially important during ~~OLM because the pump discharge tie line is opened and the possibilities of all EFW becoming inoperable increases~~on-line maintenance that requires the pump discharge tie line to be open increasing the possibility for all EFW pumps to become inoperable. ~~In the case leakage from the EFW check valve is detected, restoration is~~ Should leakage be detected when the tie line is open, prompt restoration will be performed by the following procedure.

1. Isolate the relevant line using the EFW isolation valve (EFS-MOV-019), EFW pump outlet manual isolation valve (EFS-VLV-013) and EFW pump discharge cross-connect line isolation valve (EFS-MOV-014).
2. After draining the isolated ~~area~~piping, perform ~~the~~ maintenance of the check valve.
3. After ~~performing the water filling of the isolated area, complete the restoration~~ check valve maintenance refill the piping and verify~~ing~~ that there is no temperature rise at the temperature gauge ~~in the~~ upstream of the EFW check valve.

Also, in the case 1 train is isolated for the restoration, the condition of the EFWS should be shifted to T-spec 3.7.5 CONDITION B. In this case, it is necessary to complete the restoration within the completion time of 72 hours. In the case restoration cannot be performed within 72 hours, the condition of the EFWS must be shifted to CONDITION C and plant operation condition shall be shifted to MODE 3 within 6 hours, then, it must be

Chapter 11

US-APWR DCD Chapter 11 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
11.4-21	11.4.9 References 11.4-32	Editorial: clarify the language. Replaced "American Society for Testing and Materials, "Standard Guide for Establishing Procedures to Quality and Certify Inspection Personnel for Coating Work in Nuclear Facilities, "ASTM D 4537-04a" with "American Society for Testing and Materials, <u>Standard Guide for Establishing Procedures to Quality and Certify Personnel for Coating Work Inspection in Nuclear Facilities</u>, ASTM D 4537-04a.".
11.4-21	11.4.9 References 11.4-33	Editorial: clarify the language. Replaced "American Society for Testing and Materials, "Standard Guide for Condtion Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants."11.4-32 ASTM D 5163-08." with "American Society for Testing and Materials, <u>Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants</u>, ASTM D 5163-08.".

11.4-27 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Regulatory Guide 1.111, Rev. 1, July 1977.

11.4-28 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Regulatory Guide 1.113, Rev. 1, April 1977.

11.4-29 Deleted.

11.4-30 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800.

11.4-31 Packaging and transportation of radioactive material, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 71.

11.4-32 American Society for Testing and Materials, ~~"~~Standard Guide for Establishing Procedures to Quality and Certify ~~Inspection~~ Personnel for Coating Work Inspection in Nuclear Facilities, ~~"~~ASTM D 4537-04a.

11.4-33 American Society for Testing and Materials, ~~"~~Standard Guide for Establishing a Program for ~~Condtion~~ Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants, ~~."11.4-32~~ ASTM D 5163-08.

Chapter 12

US-APWR DCD Chapter 12 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.2-10	12.2.1.3 2nd paragraph From 3rd line to 6th line	Replaced "... The airborne radioactivity in containment is calculated based on the assumptions that all the radioactive material released into containment is airborne, decreases due to deposit and attachment and spray, dissolving into recirculation water are not taken into consideration. ..." with "... The airborne radioactivity in containment is calculated based on the assumption that all the radioactive material released into containment is airborne. Decreases due to deposition, leakage, spray, or dissolving into the recirculation water are not taken into consideration. ..." Editorial: clarify the language
12.2-10	12.2.1.3 2nd paragraph From 7th line to 8th line	Replaced " can be calculated using MicroShield code, and also using the airborne radioactive concentration" with " can be calculated with the MicroShield code using the airborne radioactivity concentration " Editorial: clarify the language
12.2-10	12.2.1.3 2nd paragraph From 8th line to10th line	Replaced "... The beta ray source strengths can be calculated by multiplying the airborne radioactive concentration in containment by the effective energy of beta ray. ..." with "... The beta source strengths are calculated by multiplying the airborne radioactivity concentration in containment by the effective energy of beta. ..." Editorial: clarify the language
12.2-10	12.2.1.3 3rd paragraph 2nd line	Replaced "The sources are based on the assumptions in RG 1.183…" with "The sources are based on the assumptions given in RG 1.183…" Editorial: clarify the language

US-APWR DCD Chapter 12 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.2-10	12.2.1.3 3rd paragraph From 3rd line to 6th line	Replaced "... The radioactivity in recirculation water is calculated based on the assumptions that all the radioactive material released into containment except for noble gas is dissolved, decreases due to deposit and attachment, being airborne in containment are not taken into consideration. ..." with "... The radioactivity in the recirculation water is calculated based on the assumption that all the radioactive material released into containment, except for noble gases, is dissolved in the recirculation water. Decreases due to deposition, leakage, or radioactivity being airborne in containment are not taken into consideration. ..." Editorial: clarify the language
12.2-10	12.2.1.3 3rd paragraph From 7th line to 8th line	Replaced "... The gamma ray source strengths can be calculated using MicroShield code, and also using the radioactive concentration in recirculation water." with " … The gamma ray source strengths can be calculated with the MicroShield code using the radioactivity concentration in recirculation water." Editorial: clarify the language
12.2-10	12.2.1.3 3rd paragraph From 9th line to 11th line	Replaced "The beta ray source strengths can be calculated by multiplying the radioactive concentration in recirculation water by the effective energy of beta ray. ..." with "The beta source strengths are calculated by multiplying the radioactivity concentration in recirculation water by the effective energy of beta. ..." Editorial: clarify the language
12.3-2	12.3.1.1.1.1 D. SGs 2nd paragraph 4th line	Replaced "… for the US-APWR steam generator tube" with "… for the US-APWR steam generator tubing". Editorial: clarify the language
12.3-31	12.3.4.2.2 4th paragraph 2nd line	Replaced "The detailed flow diagram of HVAC system installed …" with "The detailed flow diagram of HVAC system includes … Editorial: clarify the language

US-APWR DCD Chapter 12 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
12.4-6	12.4.1.8 1st paragraph 11th line	Replaced "accident are not exceeded reflect the time-dependency of the area dose rates and the" with " accident are not exceeded reflect the time-dependence of the area dose rates and the " Editorial: clarify the language
12.4-6	12.4.1.8 1st paragraph From 12th line to 17th line	Replaced "... The doses on the radiation zone maps are determined by adding the upper limit dose on the radiation zone maps under normal conditions to the gamma dose from airborne radioactive materials in containment after LOCA, which are calculated by modeling outer shield and containment as cylinder with the containment free volume. ..." with "... The post-accident doses on the radiation zone maps are determined by adding the upper limit dose on the radiation zone maps under normal conditions (from Figure 12.3-1) to the gamma dose from the airborne radioactive materials in containment after LOCA. The doses are calculated by modeling the outer shield and containment as a cylinder with the containment free volume. ..." Editorial: clarify the language
12.4-6	12.4.1.8 1st paragraph From 17th line to 19th line	Replaced "... Then the outer shield is ignored in dose calculation of the penetration areas and the other shields having sufficient shielding effect are considered." with "... Then the outer shield is ignored in the penetration areas dose calculation although other shields having sufficient shielding effect are considered." Editorial: clarify the language

All these activities are calculated using the following equation:

$$A = \frac{1}{3.7 \times 10^4} N \cdot \sigma \cdot \varphi[1 - \exp(-\lambda t_1)] \cdot \exp(-\lambda t_2) \qquad \text{Eq. 12.2-3}$$

where:

A	=	activity (µCi/cm^3)
N	=	isotope number density (1/cm^3)
σ	=	activation cross section (cm^2)
φ	=	neutron flux (n/cm^2/s)
λ	=	decay constant (1/s)
t_1	=	irradiation period (s)
t_2	=	time after shutdown (s)

Other calculation parameters are tabulated in Table 12.2-71.

12.2.1.3 Sources for the Design-Basis Accident

The radiation sources of importance for the DBA are the containment source and the RHRS and Containment Spray System sources.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in RG 1.183 (Reference 12.2-4). The airborne radioactivity in containment is calculated based on the assumption~~s~~ that all the radioactive material released into containment is airborne~~,~~. ~~decreases~~ Decreases due to deposition, ~~and attachment~~leakage~~ and~~, spray, or dissolving into the recirculation water are not taken into consideration. The gamma ray source strengths can be calculated ~~using~~ with the MicroShield code~~, and also~~ using the airborne ~~radioactive~~ radioactivity concentration in containment. The beta ~~ray~~ source strengths ~~can be~~are calculated by multiplying the airborne ~~radioactive~~ radioactivity concentration in containment by the effective energy of beta~~ ray~~. The integrated gamma ray and beta particle source strengths for various time-periods following the postulated accident are tabulated in Table 12.2-58.

The RHRS and shielding are designed to allow limited access to the RHR pumps following a DBA. The sources are based on the assumptions given in RG 1.183 (Reference 12.2-4). The radioactivity in the recirculation water is calculated based on the assumption~~s~~ that all the radioactive material released into containment, except for noble gases, is dissolved in the recirculation water~~,~~. ~~decreases~~ Decreases due to deposition, ~~and attachment,~~leakage, or radioactivity being airborne in containment are not taken into consideration. The gamma ray source strengths can be calculated ~~using~~ with the MicroShield code~~, and also~~ using the ~~radioactive~~ radioactivity concentration in recirculation water. The beta ~~ray~~ source strengths ~~can be~~are calculated by multiplying the ~~radioactive~~ radioactivity concentration in recirculation water by the effective energy of beta ~~ray~~. Noble gases formed by the decay of halogens in the sump water are assumed to be retained in the water. Credit has been taken for dilution by the RCS volume plus the contents of the refueling water storage. Gamma ray source strengths for radiation sources circulating in the RHR loop and associated equipment are tabulated in Table 12.2-59.

12.2.2 Airborne Radioactive Material Sources

This section deals with the models, parameters, and sources required to evaluate the airborne concentration of radionuclides during the plant operations in the various plant radiation areas where personnel occupancy is expected.

Radioactive material that becomes airborne may come from the RCS, spent fuel pit, and refueling water storage pit. The calculation of potential airborne radioactivity in equipment cubicles, corridors, or operating areas normally occupied by operating personnel is based on reactor coolant activities given in Chapter 11, Section 11.1.

The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various applicable regions are tabulated in Table 12.2-60 and table 12.2-72.

The CVCS and the RHRS are designed to provide the capability to purify the reactor coolant through the purification demineralizer after the reactor shutdown and cooldown. This mode of operation will ensure that the effect of activity spikes does not significantly contribute to the containment airborne activity during refueling operations.

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance) and operating procedures followed during and after shutdown. The permissible coolant activity levels following de-pressurization are based on the noble gases evolved from the RCS water upon the removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification is established based on the maximum permissible concentration considerations and containment ventilation system capabilities of the plant.

The exposure rates at the surface of the refueling cavity and spent fuel pit water are dependent on the purification capabilities of the refueling cavity and spent fuel pit cleanup systems. A water total activity level of less than 0.005 μCi/g for the dominant gamma-emitting isotopes at the time of refueling leads to a dose rate at the water surface less than 2.5 mrem/h.

The detailed listing of the expected airborne isotopic concentrations in all the various plant regions is presented in Table 12.2-61. The final design of the plant ensures that all the expected airborne isotopic concentrations in all normally occupied areas are well below the derived air concentration (10 CFR 20 Appendix B [Reference 12.2-5]). If entry is needed in areas where airborne concentrations exceed the limit (such as containment

C. Reactor Vessel Insulation

Insulation, in the area of the reactor vessel nozzle welds, is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate the removal of the insulation for the inspection of the welds.

D. SGs

The SGs incorporate several design features to facilitate maintenance and inspection in reduced radiation fields. The SGs have the following design aspects:

1. Manways of the channel head are sized to facilitate access for tube bundle inspections and maintenance.
2. The channel head has a cylindrical region just below the tube sheet primary side to enhance the access of tooling to all tubes, including those on the periphery of the tube bundle.
3. Rapid entry/exit nozzle dam systems are provided in both primary nozzles to minimize occupational radiation exposure and to enhance personnel safety.

The specification of low cobalt tubing material for the US-APWR steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content is controlled to be not more than 0.016 mass percent for the US-APWR steam generator ~~tube~~tubing.

E. Materials

Equipment specifications for components exposed to high temperature reactor coolant contain limitations on the cobalt content of the base metal as given in Table 12.3-7. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations.

Nickel-based alloys in the reactor coolant system (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the inconel steam generator tube.

12.3.1.1.1.2 Balance of Plant Equipment

A. Filters

Filters that accumulate radioactivity are supplied with the means either to back-flush the filter remotely or to perform cartridge replacement with semi-remote tools.

For cartridge filters, adequate space is provided to allow removal, cask loading, and transportation of the cartridge to the solid radwaste area.

- To measure the airborne radioactivity in the HVAC exhaust ducts of the air exhausted from cubicles
- To warn of an abnormal release of radioactive material from cubicles

12.3.4.2.2 Criteria for Location of Airborne Radioactivity Monitors

Considerations for airborne monitor sampling points are HVAC exhaust ducts that are installed in the Radioactive Controlled Area.

The Airborne Radioactivity Monitors are sampled at locations where airborne radioactivity may normally exist. If the gas is detected, the existence of Iodine or other radioactive materials are to be determined. The Airborne Radioactivity Monitors are installed in the following areas:

- Fuel Handling Area
- Annulus and Safeguard Area
- R/B
- A/B
- Sample and Lab Area

All of these areas are RCA.

The sampling points of the airborne radioactivity monitors are shown in the Figure 12.3-10. The detailed flow diagram of HVAC system ~~installed~~ includes the airborne radioactivity monitors are shown in Figure 9.4.3-1.

12.3.4.2.3 General System Description

The system description of the airborne radioactivity monitors is the same as process gas monitors (see Chapter 11, Subsection 11.5.2.1).

12.3.4.2.4 Data Processing Module and Display Console

A description of these components is given in Chapter 11, Section 11.5.

12.3.4.2.5 Local Annunciation

All airborne radiation monitors have local annunciation consisting of an audible alarm and a warning light at the local readout.

any individual from exceeding 5 rem to the whole body or 50 rem to the extremities. Figure 12.3-2 shows the general plant arrangement with the vital areas that must be accessed in the post-accident environment identified. Figures 12.3-3 through 12.3-6 contain radiation zone maps for plant areas including those areas requiring post-accident access. This figure shows projected radiation zones in areas requiring access and access routes or ingress, egress, and performance of actions at these locations. The radiation zone maps reflect maximum radiation fields over the course of an accident. The analyses that confirm that the individual personnel exposure limits following an accident are not exceeded reflect the time-~~dependency~~ dependence of the area dose rates and the required post-accident access times. The post-accident doses on the radiation zone maps are determined by adding the upper limit dose on the radiation zone maps under normal conditions (from Figure 12.3-1) to the gamma dose from the airborne radioactive materials in containment after LOCA~~,~~. ~~which~~ The doses are calculated by modeling the outer shield and containment as a cylinder with the containment free volume. Then the outer shield is ignored in the penetration areas dose calculation ~~of the penetration areas and the~~although other shields having sufficient shielding effect are considered. The areas that require post-accident accessibility are as follows:

- Main Control Room (MCR)
- Technical Support Center (TSC)
- Postaccident sampling system (PASS)
- Radiochemistry laboratory (sample analysis)
- Hot counting room

Accident parameters and sources are discussed and evaluated in Chapter 15, Subsection 15.6.5.5.

12.4.1.9 Dose to Construction Workers

For multiunit plants, the COL Applicant is to provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).

12.4.2 Radiation Exposure at the Site Boundary

12.4.2.1 Direct Radiation

The direct radiation from onsite contained sources is described in this subsection. The direct radiation from the containment and other plant buildings is negligible.

Chapter 13

US-APWR DCD Chapter 13 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
13-ii	Subsection 13.6.5	Editorial: typographical error Changed the page 13.6-5 to 13.6-6.

Chapter 14

US-APWR DCD Chapter 14 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
14.2-47	14.2.12.1.15	Replaced "the its lithium concentration" with "its lithium concentration" in C.3. RAI 521, 14.02-120
14.2-139	14.2.12.1.115	Separated the merged paragraph A.2 and A.3. RAI 521, 14.02-120
14.2-181	14.2.13, COL 14.2(11)	Changed "test" to "tests" in the first sentence. RAI 521, 14.02-120
14.3-2	14.3.1.2	Reworded the definition of "As-built" to reflect the discussion with the NRC and NEI. RAI 531, 01-7
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. Amended RAI 396, 14.03.12-20 (MHI ref.: UAP-HF-10067) RAI 481, 14.03.12-26, 14.03.12-27, 14.03.12-29, 14.03.12-30 (MHI ref.: UAP-HF-10067)
14.3-26	14.3.6	Revised COL 14.3(3) for COL applicant to provide the abstracts for the site-specific PS-ITAAC. Amended RAI 396, 14.03.12-20 (MHI ref.: UAP-HF-10067) RAI 481, 14.03.12-26 (MHI ref.: UAP-HF-10067)
14.3-30	14.3.7	Added new references as "14.3-39" and "14.3-40". Amended RAI 396, 14.03.12-20 (MHI ref.: UAP-HF-10067) RAI 481, 14.03.12-26, 14.03.12-27, 14.03.12-29, 14.03.12-30 (MHI ref.: UAP-HF-10067)
14.3-61	Table 14.3-2 (sheet 3 of 4), Item 5.b.ii	Replaced "inspection for the existence of the report" to "inspections and analyses for the piping" in the ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
14.3-62	Table 14.3-2 (sheet 3 of 4 and 4 of 4), Item 6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21

US-APWR DCD Chapter 14 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	**Description of Change**
14.3-69	Table 14.3-8 (sheet 1 of 4)	Broke out the combined entry for IEEE 603 Sections 4.4 and 4.6 into two separate items. RAI 515, 14.03.05-32

2. At several sampling points (e.g., RCS loops, pressurizer liquid, and demineralizer inlet), each lithium concentration is measured until the lithium concentration is almost equal at each sampling point.

3. The RCS volume without fuel assembles is calculated from the amount of lithium to be added and ~~the~~ its lithium concentration. The quantity and concentration of injected lithium divided by increased concentration of lithium is used to estimate the RCS volume. This volume is reference data.

4. Following completion of hot functional testing, lithium is removed until it is approximately equal to the initial concentration (example: almost 0.5 ppm).

D. Acceptance Criteria

1. The lithium concentrations from all sample points are within +/-0.05 ppm following analytical measurement with an accuracy of +/- 0.05 ppm or better.

2. Lithium mixing, charging to the RCS, and removal performs as described in Subsections 9.3.4.

14.2.12.1.16 Primary Makeup Water System (PMWS) Preoperational Test

A. Objective

1. To demonstrate the operation of the PMWS.

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, including the Demineralized Water Storage Tank (DWST) and the Demineralized Water Transfer Pumps.

C. Test Method

1. Verify manual and automatic system controls.

2. Verify system flowrates.

3. Verify indications and alarms.

D. Acceptance Criteria

1. The PMWS operates as described in Subsection 9.2.6.

2. Indications and alarms operate as described in Subsections 9.2.6.5 and 9.3.4.5.5.6.

4. Operate exhaust fans in battery room and verify operating condition.

D. Acceptance Criteria

1. Turbine building area ventilation system (electric equipment area) operates as described in Subsection 9.4.4.

2. Indications and alarms operate as described in Subsection 9.4.4.

3. Battery room exhaust fan operation maintains the hydrogen concentration below 1% by volume in the battery room per Subsection 9.4.4.1.2.

14.2.12.1.112 Reserved

14.2.12.1.113 Reserved

14.2.12.1.114 Reserved

14.2.12.1.115 RCPB Leak Detection Systems Preoperational Test

A. Objective

1. To verify operability of RCPB leak detection systems and adjust the alarm setpoints.

2. To demonstrate the function described in Subsection 5.2.5 with reference to RG 1.45.

3. To determine quantitative conversion data from measured quantities that correspond to RCS leak rate.

Note: This test may be performed in conjunction with subsection 14.2.12.1.80, ”Liquid Waste Management System Preoperational Test.”

B. Prerequisites

1. Component testing and instrument calibration is completed.

2. Test instrumentation is available and calibrated.

C. Test Method

1. Verify the calibration, alarm setpoints and alarm functions to each channel of RCPB leak detection systems and associated systems used to determine RCS leakage identified below.

Note: Instrument channel verification should be performed in conjunction with the associated tests identified below.

a. Intersystem leakage, SG tube leakage and unidentified leakage detection design features:

14.2.13 Combined License Information

COL 14.2(1) *Deleted*

COL 14.2(2) *The COL Applicant reconciles the site-specific organization, organizational titles, organizational responsibilities, and reporting relationships to be consistent with US-APWR Test Program Description Technical Report, MUAP-08009 (Reference 14.2-29) [14.2.2].*

COL 14.2(3) *Deleted*

COL 14.2(4) *Deleted*

COL 14.2(5) *Deleted*

COL 14.2(6) *Deleted*

COL 14.2(7) *The COL applicant provides an event-based schedule, relative to fuel loading, for conducting each major phase of the test program, and a schedule for the development of plant procedures that assures required procedures are available for use during the preparation, review and performance of preoperational and startup testing. For multiunit sites, the COL applicant discusses the effects of overlapping initial test program schedules on organizations and personnel participating in each ITP. The COL applicant identifies and cross-references each test or portion of a test required to be completed prior to fuel load which satisfies ITAAC requirements. [14.2.9] [14.2.11]*

COL 14.2(8) *Deleted*

COL 14.2(9) *Deleted*

COL 14.2(10) *The COL applicant is responsible for the testing outside scope of the certified design in accordance with the test criteria described in subsection 14.2.1. [14.2.12]*

COL 14.2(11) *The COL holder for the first plant is to perform the first plant only tests and prototype test. For subsequent plants, either these tests are performed, or the COL applicant provides a justification that the results of the first-plant only tests are applicable to the subsequent plant and are not required to be repeated. [14.2.8]*

COL 14.2(12) *The COL holder makes available approved test procedures for satisfying testing requirements described in Section 14.2 to the NRC approximately 60 days prior to their intended use. [14.2.3, 14.2.11, 14.2.12.1]*

The type of information and the level of detail in Tier 1 are based on a graded approach commensurate with the safety significance of the SSCs for the design. Top-level design features of safety-related SSCs are addressed in Tier 1 with consideration of performance capabilities required to perform their safety functions. Non-safety related SSCs are evaluated on a case-by-case basis to ascertain the level of detail considered appropriate for Tier 1 based on their safety significance. Design-specific and unique features of the facility are included in Tier 1, as appropriate. The Tier 1 material is derived from the Tier 2 material.

The top-level information selected for Tier 1 includes the principal performance characteristics and safety functions of the SSCs, which are to be verified appropriately by ITAAC. ITAAC for non-safety related SSCs are developed based on their importance to safety, considering such factors as internal and external hazards analysis, fire protection, PRA insights and severe accident prevention and mitigation. The successful completion of all ITAAC is a prerequisite for fuel load and a condition of the license.

The ITAAC included in the Tier 1 material support the requirement in 10 CFR 52.97(b) (Reference 14.3-3) that ITAAC be used to verify the complete facility. To this end, the Tier 1 portion of the DCD provides ITAAC for all structures and systems within the scope of the certified design.

The primary basis for ITAAC appears in 10 CFR 52.80(a) (Reference 14.3-4). These requirements specify that a Combined License Application (COLA) must include the proposed inspections, tests, and analyses (including those that apply to emergency planning) that the licensee shall perform and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and applicable NRC regulations.

14.3.1.2 Definitions

The following definitions are used in the design descriptions and the related ITAAC to assure precision and consistency. Most are based on RG 1.206 (Reference 14.3-1) and SRP 14.3 (Reference 14.3-2).

- **Acceptance criteria** refer to the performance, physical condition, or analysis result for an SSC, to demonstrate that the design requirement/commitment is met.

- **Analysis** means a calculation, mathematical computation, or engineering/technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar SSCs.

- **As-built** means the physical properties of a structure, system, or component~~the SSC~~ following ~~the~~ completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, ~~D~~determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur 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 analyses~~is~~ for US-APWR physical protection systems provided in Table 2.12-1 of Tier 1 of the DCD are specified in ~~the~~ "US-APWR Physical ~~Protection System~~Security Hardware ITAAC ~~Test~~ Abstracts, "~~LATER~~MUAP-10003" (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, and provides abstracts describing the specific inspections, tests and analyses for the facility's physical security hardware ITAAC not addressed in the DCD.

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.

No ITAAC are necessary for the initial test program because all ITAAC are to be completed prior to fuel load.

14.3.5 Chapter 3 of Tier 1, Interface Requirements

Chapter 3 of Tier 1 focuses on the interface requirements of the safety-significant design attributes. The interface requirements in Chapter 3 of Tier 1 define the safety-significant design attributes and performance characteristics that assure that the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design that are dependent on characteristics of the site.

Chapter 3 of Tier 1 also identifies the scope of the design to be certified by specifying the systems that are completely or partially out of scope of the certified design. Thus, interface requirements are defined for: (a) systems that are entirely outside the scope of the design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the standard design based on the above methodology.

14.3.6 Combined License Information

COL 14.3(1) *The COL applicant provides the ITAAC for the site specific portion of the plant systems specified in Subsection 14.3.5, Interface Requirements. [14.3.4.6, 14.3.4.7]*

COL 14.3(2) *The COL applicant provides proposed ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.10]*

COL 14.3(3) *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, and provides abstracts describing the specific inspections, tests and analysis for the facility's physical security hardware ITAAC not addressed in the DCD. [14.3.4.12]*

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~~Security Hardware ITAAC Abstracts, Rev. 0, ~~LATER,~~ March 20~~09~~10.

Table 14.3-2 Example of ITAAC Table

(Sheet 3of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b Each of the seismic category piping, including supports, identified in Table ___is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table ___ are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table____ is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table ____ can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table____ can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table ____ as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table ____as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.

Table 14.3-2 Example of ITAAC Table

(Sheet 4of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	6.a.ii Inspection will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table ____as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table ______, is powered from their respective Class 1E division.	6.b Tests will be performed on the ____ system by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the Class 1E equipment identified in Table _____under test.
6.c Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 14.3-8 IEEE 603-1991 Compliance Matrix by DCD Tier 1 Section

(Sheet 1 of 4)

IEEE Std. 603-1991		Tier 1 DCD Subsection Number					
Section Number	**Section Title or Topic**	**2.5.1 (PSMS)**	**2.5.2 (SSD)**	**2.5.3 (DAS)**	**2.5.4 (PAM/BISI/ SPDS et. al.)**	**2.5.5 (PCMS)**	**2.5.6 (DCS)**
4.4	Analytical limits, ranges and rates of change	X Table 2.5.1-6 Item # 19, 22	(1) (5)	N/A	(1) (5)	N/A	N/A
~~4.4 and~~ 4.6	Number and location of sensors; spatial dependence	X Table 2.5.1-6 Item # 28	(1) (5)	N/A	(1) (5)	N/A	N/A
4.8	Potential for functional degradation	X Table 2.5.1-6 Item # 7, 8	(1) (5)	N/A	(1) (5)	N/A	(1) (5) Table 2.5.6-1 Item # 4
5.1	Single Failure	X Table 2.5.1-6 Item # 10, 21	(1) (5)	N/A	(1) (5)	N/A	N/A
5.2 and 7.3	Completion of Protective Action	X Table 2.5.1-6 Item # 14	(1) (5) Table 2.5.2-3 Item # 7	N/A	(1) (5)	N/A	N/A
5.3	Quality	(2)	(2)	(2)	(2)	(2)	(2)
5.4	Equipment Qualification	X Table 2.5.1-6 Item # 5, 6, 7	(1) (5)	N/A	(1) (5)	N/A	(1) (5)
5.5	System Integrity	X Table 2.5.1-6 Item # 5, 6, 7, 8, 22	(1) (5)	N/A	(1) (5)	N/A	(1) (5) Table 2.5.6-1 Item # 2

Chapter 16

US-APWR DCD Chapter 16 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
Technical Specifications		
1.1-2	3rd paragraph	RAI No.520-4183 Question No.16-300, Editorial Replaced “setpoint” with “setpoints”.
3.3.1-8	Action T	RAI No.520-4183 Question No.16-300, Editorial Added “OR” connector.
3.3.1-14	Table 3.3.1-1 Item 3.b	RAI No.520-4183 Question No.16-300, Editorial Added a space. It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.1-1 ware deleted according to “UAP-HF-09493”.
3.3.1-17	Table 3.3.1-1 Item 13.b	RAI No.520-4183 Question No.16-300, Technical Added footnote (k). It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.1-1 ware deleted according to “UAP-HF-09493”.
3.3.2-16	Table 3.3.2-1 Item 5A	RAI No.520-4183 Question No.16-300, Editorial “valve” in Item 5A in Table 3.3.2-1 was capitalized. It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.2-1 ware deleted according to “UAP-HF-09493”.
3.3.2-21	Table 3.3.2-1 Item 13. b	RAI No.520-4183 Question No.16-300, Editorial Added a space between SR and 3.3.2.4. It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.2-1 ware deleted according to “UAP-HF-09493”.
3.3.2-22	Table 3.3.2-1 Item 14.a, b and c	RAI No.520-4183 Question No.16-300, Editorial Replaced footnote (j) with (h). It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.2-1 ware deleted according to “UAP-HF-09493”.

US-APWR DCD Chapter 16 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
3.3.2-22	Table 3.3.2-1 Item 14.c	RAI No.520-4183 Question No.16-300, Editorial “low” was capitalized. It is noted that ALLOWABLE VALUE and TRIP SETPOINT in Table 3.3.2-1 ware deleted according to “UAP-HF-09493”.
Bases		
B3.3.1-36	1st paragraph, 5th line	RAI No.520-4183 Question No.16-300, Editorial Replaced “tri” with ”trip”.
B3.3.1-36	4th paragraph, last line	RAI No.520-4183 Question No.16-300, Editorial Added “time limit”.
B3.3.1-39	ACTIONS L1 and L2	RAI No.520-4183 Question No.16-300, Editorial Deleted “Turbine Trip – Main Turbine Stop Valve Position”.
B3.3.1-44/45	T.1 and T.2	RAI No.520-4183 Question No.16-300, Editorial Editorial errors ware corrected.
B3.3.1-49	Last paragraph Item (1)	RAI No.520-4183 Question No.16-300, Editorial Added “of”.
B3.3.1-49	Last paragraph Item (3)	RAI No.520-4183 Question No.16-300, Editorial Added “Test for verification of RTB operability using the”.
B3.3.1-50	3rd paragraph, 2nd line	RAI No.520-4183 Question No.16-300, Editorial Replaced “RTS” with “PSMS”
B3.3.1-52	3rd paragraph, 1st line	RAI No.520-4183 Question No.16-300, Editorial Replaced “RTS” with “PSMS”
B3.3.1-53	2nd paragraph	RAI No.520-4183 Question No.16-300, Editorial Editorial errors ware corrected.
B3.3.2-30	Item 7.c at the top of the page	RAI No.520-4183 Question No.16-300, Editorial “coincident” and ”no” were capitalized.
B3.3.2-36	Item 11.a fourth bulleted item	RAI No.520-4183 Question No.16-300, Editorial “level” and ”no” were capitalized.
B3.3.2-40	Last paragraph	RAI No.520-4183 Question No.16-300, Editorial Replaced “without” with “including”.

US-APWR DCD Chapter 16 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
B3.3.2-44	Item 14.c	RAI No.520-4183 Question No.16-300, Editorial “low” was capitalized.
B3.3.5-7	Second paragraph from bottom	RAI No.520-4183 Question No.16-300, Editorial Replaced “Channel” with “CHANNEL”.
B3.3.6-5	Third paragraph from bottom	RAI No.520-4183 Question No.16-300, Editorial Replaced “satisfy” with “satisfies”.
B3.3.6-6	First paragraph	RAI No.520-4183 Question No.16-300, Editorial Replaced “function” with “functions”.
B3.3.6-10	Last paragraph	RAI No.520-4183 Question No.16-300, Editorial Replaced “is” with “are”.
B3.4.13-5	REFERENCES	RAI No.521-4248, Question No.14.02-120, Editorial Added “Revision 1” and replaced “1973” with “2008” in the second reference.
B3.4.15-6	REFERENCES	RAI No.521-4248, Question No.14.02-120, Editorial Added “Revision 1, May 2008” in the second reference.

1.1 Definitions

CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. CHANNEL CALIBRATION encompasses devices that are subject to drift between surveillance intervals and all input devices that are not tested through continuous automated self-testing. Refer to TADOT for output devices that are not tested through continuous automated self-testing. CHANNEL CALIBRATION confirms the accuracy of the channel from sensor to digital Visual Display Unit (VDU) readout, as described in Topical Report, “Safety I&C System Description and Design Process,” MUAP-07004 Section 4.4.2. For analog measurements CHANNEL CALIBRATION confirms the analog measurement accuracy at five calibration setpoints corresponding to 0%, 25%, 50%, 75% and 100% of the instrument range. The confirmed setpoints are monitored on the safety VDUs. For binary measurements, the CHANNEL CALIBRATION confirms the accuracy of the channel’s state change, as described in Topical Report, “Safety I&C System Description and Design Process,” MUAP-07004 Section 4.4.1. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. A CHANNEL CHECK may be conducted manually or automatically. Where the CHANNEL CHECK is conducted automatically, an alarm shall be generated when the agreement criteria are not met.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
S. Required Action and associated Completion Time for Condition N, Q, or R not met.	S.1 Be in MODE 3.	6 hours
T. Main Turbine Stop Valve Position channel inoperable	--------------------NOTE-------------------- One channel may be bypassed for up to 12 hours for surveillance testing. --	
	T.1 Place channel in trip.	12 hours DCD_16-300
	OR	18 hours
	T.2 Reduce thermal power to < P-7	

Table 3.3.1-1 (page 1 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~	~~TRIP SETPOINT~~
1. Manual Reactor Trip Initiation	1,2	3 trains	B	SR 3.3.1.4	~~NA~~	~~NA~~
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	3 trains	C	SR 3.3.1.4	~~NA~~	~~NA~~
2. High Power Range Neutron Flux						
a. high setpoint	1,2	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	~~[±4]% RTP~~	~~[109]% RTP~~
b. low setpoint	$1^{(b)}$,2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	~~[±4]% RTP~~	~~[25]% RTP~~
3. High Power Range Neutron Flux Rate						
a. Positive Rate	1,2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	~~[±2]% RTP~~	~~[10]% RTP with time constant ≥ [1] sec~~
b. Negative Rate	1,2	4	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	~~[±2]% RTP(j)~~	~~[7]% RTP with time constant ≥ [1] sec~~
4. High Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	G,H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.13	~~[±10]% RTP (j)~~	~~[25]% RTP~~

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(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

~~(j) An allowable value is not provided for time constants because time constants are digital values set in the application software. There is no drift or adjustments for these time constants.~~

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~	~~TRIP SETPOINT~~
12. Steam Generator (SG) Water Level						
a. Low	1,2	3 per SG	F	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~[±3]% of span~~	~~[13]% of span~~
b. High-High	1(e)	3 per SG	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~[±3]% of span~~	~~[70]% of span~~
13. Turbine Trip						
a. Turbine Emergency Trip Oil Pressure	1(e)	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	~~≥ [930] psig~~	~~[1000] psig(k)~~
b. Main Turbine Stop Valve Position	1(e)	1 per valve	T	SR 3.3.1.9 SR 3.3.1.12	~~≥ [1]% open~~	~~[5]% open(k)~~

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(e) Above the P-7 (Low Power Reactor Trips Block) interlock.
~~(k) Nominal Trip Setpoint~~

Table 3.3.2-1 (page 5 of 11)
Engineered Safety Feature Actuation System Instrumentation

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~	~~TRIP SETPOINT~~
5. Main Feedwater Isolation						
5A. Main Feedwater Regulation ~~v~~Valve Closure						
a. Low T_{avg}	1,2 (j),3 (j)	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[±2]°F~~	~~[564]°F~~
Coincident with Reactor Trip, P-4	Refer to Function 11.a for all P-4 requirements.					
5B. Main Feedwater Isolation						
a. Manual Initiation	1,2 (i),3 (i)	Trains A and D	F	SR 3.3.2.6	~~NA~~	~~NA~~
b. Actuation Logic and Actuation Outputs	1, 2 (i), 3(i)	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.4	~~NA~~	~~NA~~
c. High-High SG Water Level	1,2 (i),3(a) (i)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[±3]% of span~~	~~[70]% of span~~
d. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					

(a) Above the P-11 (Pressurizer Pressure) interlock.

(i) Except when all MFIVs, MFRVs, MFBRVs, and SGWFCVs are closed.

(j) Except when all MFRVs are closed.

Table 3.3.2-1 (page 10 of 11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~	~~TRIP SETPOINT~~
13. Main Control Room (MCR) Isolation						
a. Manual Initiation	1,2,3,4,(k)	3 trains including A and D(m)	M, N, O, P	SR 3.3.2.6	~~NA~~	~~NA~~
b. Actuation Logic and Actuation Output	1,2,3,4,(k)	3 trains including A and D(m)	M, N, O,P	SR 3.3.2.2 SR 3.3.2.4	~~NA~~	~~NA~~
c. MCR Outside Air Intake Radiation						
(1) MCR Outside Air Intake Gas Radiation	1,2,3,4,(k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[±6]% of span~~	~~[2E-6] μCi/cc~~
(2) MCR Outside Air Intake Particulate Radiation	1,2,3,4,(k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[±6]% of span~~	~~[8E-10] μCi/cc~~
(3) MCR Outside Air Intake Iodine Radiation	1,2,3,4,(k)	2	M, N, O, P	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[±6]% of span~~	~~[8E-10] μCi/cc~~
d. ECCS Actuation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.					

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(k) During movement of irradiated fuel assemblies within containment.

(m) Two trains of MCREFS are required to be operable (trains A and D); three trains of MCRATS are required to be operable (three out of four trains A, B, C, D).

Table 3.3.2-1 (page 11 of 11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~	~~TRIP SETPOINT~~
14. Block Turbine Bypass and Cooldown Valves						DCD_16-520/4183
a. Manual Initiation	1,2 (~~j~~h),3 (~~j~~h)	Trains A and D	F	SR 3.3.2.6	~~NA~~	~~NA~~ DCD_16-520/4183
b. Actuation Logic and Actuation Outputs	1,2 (~~j~~h),3 (~~j~~h)	Trains A and D	S,T	SR 3.3.2.2 SR 3.3.2.4	~~NA~~	~~NA~~ DCD_16-520/4183
c. Low-~~l~~Low T_{avg} Signal	1,2 (~~j~~h),3 (~~j~~h)	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	~~[2.0]°F~~	~~[553]°F~~

(~~j~~h) Except when all MSIVs are closed.

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BASES

ACTIONS (continued)

A known required inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic (for the trip functions where the required number of operable channels is three) or one-out-of three logic (for the trip functions where the required number of operable channels is four) for actuation of the two-out-of-N trips, where N is three or four (depending on the required number of operable channels). The 72 hours allowed to place the inoperable channel in the tripped condition is justified because the remaining two operable channels (for the trip functions where the required number of operable channels is three) or the remaining three operable channels (for the trip functions where the required number of operable channels is four) have automatic self-testing (as described for COT), and automatic CHANNEL CHECKS. DCD_16-300

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The number of Required Channels for the High Power Range Neutron Flux Rate is four. Four channels are required because each channel measures neutron flux in one quadrant of the core. Anomalies occurring in one core quadrant can be seen by the neutron flux detector in that quadrant and by the neutron detectors in the two adjacent quadrants, but not by the detector in the opposite quadrant. So to ensure event detection and accommodate a single failure, neutron flux detectors must be operable in all four quadrants.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is based on operating experience. DCD_16-300

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

One channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment. The 12 hours bypass limit is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the

BASES

ACTIONS (continued)

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

K.1, K.2.1, and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour.

L.1 and L.2

Condition L applies to the following reactor trip Functions:

- Low Pressurizer Pressure,
- High Pressurizer Water Level,
- Low Reactor Coolant Flow,
- Low Reactor Coolant Pump Speed,
- High-High SG Water Level, and DCD_16-300
- Turbine Trip – Turbine Emergency Trip Oil Pressure.~~, and~~ DCD_16-300

~~•Turbine Trip – Main Turbine Stop Valve Position.~~ DCD_16-300

BASES

ACTIONS (continued)

R.1 [and R.2]

Condition R applies to the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one required train inoperable, 24 hours are allowed to restore the train to OPERABLE status. The Completion Time of 24 hours is reasonable considering that in this Condition, the two remaining OPERABLE trains are adequate to perform the safety function and given the low probability of an event during this interval. The 24 hours allowed to restore the train to OPERABLE status also considers that the two remaining OPERABLE trains each have automatic self-testing as described for ACTUATION LOGIC TEST. [Required Action R.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.]

The Required Actions have been modified by a Note that allows bypassing one inoperable train up to 4 hours for surveillance testing, provided the other two trains are OPERABLE.

S.1

Condition S applies when the Required Action and associated Completion Time for Condition N, Q, or R have not been met. If the train cannot be returned to OPERABLE status, the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

Placing the unit in MODE 3 from Condition N results in Condition D entry while an RTB is inoperable.

(From Condition Q) With the unit in MODE 3, Condition D would apply to any inoperable RTB trip mechanism.

T.1 and T.2

Condition T applies to Main Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 12 hours. If placed in the tripped condition, this results in a partial trip condition requiring three additional channels to initiate a reactor trip. If the channel can not be restored to OPERR~~L~~ABLE status or placed in the trip condition, then power must be reduced below the P-7 setpoint within the next 6 hours. The 6 hours allowed for reducing power is consistent with other power reduction action completion times. DCD_16-300

BASES

ACTIONS (continued)

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. These times are justified because this is an anticipatory trip that is not~~e~~ credited in the safety analysis, and a diverse turbine trip is also initiated from the Turbine Emergency Oil Pressure.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function and Overpower ΔT Function.

A Note clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP.

[The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT. This test shall verify RTB train OPERABILITY by actuation of the two RTBs for each train to their tripped state. Each RTB may be actuated together or individually.

The RTB train test shall include three separate but overlapping tests: (1) The Undervoltage Test for verification of RTB operability using only the undervoltage trip mechanism. (2) The Shunt Trip test for verification of RTB operability using only the shunt trip mechanisms. (3) The Manual Reactor Trip Test for verification of RTB operability using the hardwired switches. The Undervoltage Test shall bypass the shunt trip mechanism, so each RTB actuates using only the undervoltage mechanism. The Shunt Trip Test shall bypass the undervoltage mechanism, so each RTB actuates using only the shunt trip mechanism. The Manual Reactor Trip Test shall actuate the RTB with both mechanisms. Figure 4.4-1 of Topical Report MUAP-07004 (Ref. 6) describes an acceptable overlapping method for conducting these three separate tests that confirms OPERABLE status.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

[The Frequency of every 62 days on a STAGGERED TEST BASIS applies to all four RTB trains. This test frequency is justified based on industry experience. The test frequency also considers the added reliability of the US-APWR RTB configuration, which includes redundant RTBs within each train and the overall two-out-of-four train configuration. Since each test actuates each RTB to its required tripped state, the STAGGERED TEST BASIS results in each RTB being tested every 248 days, and each tripping method being tested every 744 days. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The STAGGERED TEST BASES frequency of 62 days with each RTB tested every 248 days, and each trip methodology ultimately tested every 744 days is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.]

SR 3.3.1.5

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SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The ~~RTS~~PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 6 and 7. The ACTUATION LOGIC TEST is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7. [The Frequency of every 24 months is justified based on the reliability of the PSMS. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the 24 month CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the 24 month COT, the 24 month ACTUATION LOGIC TEST and the STAGGERED 62 days TADOT for the non-digital side of the output module. The Channel CALIBRATION, COT, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHAANNEL CHECK and self-testing and confirm the functioning of the self-testing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

[The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT.

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The ~~RTS~~PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing encompasses all digital Trip Setpoints and trip functions. The self-testing is described in Reference 6 and 7. The COT is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation, including digital Trip Setpoint values or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7.

A COT ensures the entire channel will perform the intended Function. A COT also ensures that the logic processing for interlocks (i.e., P-6 and P-10) is operating correctly. The combination of the COT, CHANNEL CALIBRATION, continuous self-testing and continuous CHANNEL CHECK ensures the complete P-6 and P-10 interlocks are operating correctly.

[The Frequency of 24 months is justified based on the reliability of the PSMS. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The completely continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the 24 month CHANNEL CALIBRATION for the non digital side of the inputo module, the continuous self-testing for the digital side, the 24 month COT, 24 months Actuation Logic Test and the STAGGERED 62 days TADOT for the non-digital side of the output module. The CHANNEL CALIBRATION, COT and TADOT, which are manual tests, overlap with the CHANNNEL CHECK and self-testing and confirm the functioning of the selg-testing.

The COT interval of 24 months with the self test capability is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The

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SURVEILLANCE REQUIREMENTS (continued)

result of the PSMS reliability analysis is evaluated and confirmed in the US-APWT PRA Chapter 19.]

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The Note allows a normal shutdown to proceed without a ~~delya~~delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTB closed for 4 hours this Surveillance must be performed prior to ~~over~~ 4 hours a~~g~~fter entry into MODE 3.

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SR 3.3.1.8

Performance of the CHANNEL CHECK within 4 hours after reducing power below P-6 and [once every 12 hours thereafter OR in accordance with the Surveillance Frequency Control Program] ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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c. Emergency Feedwater Isolation - High Steam Generator Water Level coincident with P-4 signal and ~~n~~No Low Main Steam Line Pressure

This signal provides protection against damaged SG overfill. There are four High Steam Generator Water Level channels in a two-out-of-four logic configuration for each Steam Generator. The ESFAS SG water level instruments provide input to the SG Water Level Control System. The interface from the safety channels in the PSMS to the PCMS is through the Signal Selector Algorithm (SSA). The SSA ensures an input failure to the control system does not result in erroneous control system action that would require the protection function actuation. Therefore, the protection function requires only two additional channels to provide the protection function actuation. Three channels total must be OPERABLE.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

High Steam Generator Water Level must be OPERABLE in MODES 1, 2 and 3 (above P-11) when the SGs are in operation. This signal may be manually blocked by the operator below the P-11 setpoint. This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. In MODES 4, 5, and 6, SGs are not in service and this Function is not required to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when RTBs have opened in two out of four RTB trains. RTB position signals from each RTB are interfaced to all PSMS trains via internal PSMS data links so that the P-4 interlock is generated independently within each train. Therefore this LCO requires three trains to be OPERABLE.

This Function allows operators to take manual control of ECCS systems after the initial phase of ECCS Actuation is complete. Once ECCS is overridden, automatic actuation of ECCS cannot occur again until the RTBs have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate MFW with coincident low T_{avg},
- Enable a manual override of ECCS Actuation and prevent ECCS reactuation,

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- EFW Isolation with coincident High SG Water ~~l~~Level and ~~n~~No Low Main Steam Line Pressure, and
- Trip the Reactor Coolant Pump with coincident ECCS Actuation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Containment Purge Isolation - ECCS Actuation

Containment Purge Isolation is also initiated by all Functions that initiate ECCS. The Containment Purge Isolation requirements for these Functions are the same as the requirements for the ECCS function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, ECCS, is referenced for all initiating Functions and requirements.

Note that all two Containment Purge Isolation trains are actuated when any two out of four ECCS - Automatic or Manual Initiation signals are actuated.

e. Containment Purge Isolation - Containment High Range Area Radiation

Containment High Range Area Radiation has four channels in a two-out-of-four logic configuration. Three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic.

The Containment Purge Isolation Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment Purge Isolation instrumentation must be OPERABLE in these MODES.

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While in MODES 5 and 6 ~~without~~ including fuel handling in progress, the Containment Purge Isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within acceptable limits.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

14. Block Turbine Bypass and Cooldown Valves

The Block Turbine Bypass and Cooldown Valves function prevents the overcooldown of the reactor coolant system when Tavg is decreased abnormally.

Block turbine bypass and cooldown valves are distributed to Trains A and D. Both trains must be OPERABLE.

a. Block Turbine Bypass and Cooldown Valves – Manual Initiation

Manual initiation of Block Turbine Bypass and Cooldown Valves can be accomplished from the main control room. There are two switches in the main control room, one for each train. This LCO requires 2 Manual Block Turbine Bypass and Cooldown Valves Actuation switches. Operation of either switch will actuate this Function.

b. Block Turbine Bypass and Cooldown Valves - Actuation Logic and Actuation Outputs

Actuation Logic and Actuation Outputs consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Block turbine bypass and cooldown valves are distributed to Trains A and D. Both trains must be OPERABLE.

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c. Block Turbine Bypass and Cooldown Valves - Low-~~l~~Low T_{avg} Signal

This function must be OPERABLE in MODES 1, 2 and 3. In MODES 4, 5, and 6, the average coolant temperature is below the low-low Tavg signal setpoint and this function is not required to be OPERABLE.

The ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 9).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

[The Frequency of 24 months is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.3.5.4

SR 3.3.5.4 is the performance of an ACTUATION LOGIC TEST. The Class 1E GTG start logic within the PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 2 and 3. The ACTUATION LOGIC TEST is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 2 and 3. [The Frequency of every 24 months is justified based on the reliability of the PSMS. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the 24 month CHANNEL CALIBRATION for the non digital sided of the input module, the continuous self-testing for the digital side, the 24 month ACTUATION LOGIC TEST, and the 24 month ESFAS and SLS TADOT for the non-digital side of the output module. The CHANNEL~~Channel~~ CALIBRATION, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing. DCD_16-520/4183

The ACTUATION LOGIC TEST interval of 24 months with the self test capability is justified in the PSMS reliability analysis. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6A.12. The result of the PSMS reliability analysis is

BASES

BACKGROUND (continued)

Rod Drive Motor-Generator sets

The Rod Drive Motor-Generator sets are the electrical power supply for the CRDMs. Tripping the Rod Drive Motor-Generator sets trip devices interrupts power to the CRDMs, which allows the control rod shutdown banks and control banks to fall into the core by gravity. There are two Rod Drive Motor-Generator sets operating in parallel. The DAS trips both Rod Drive Motor-Generator sets trip devices.

The DAS interface to the Rod Drive Motor-Generator sets is via hardwired circuit. This interface may be tested, with no reactor trip, as described in subsection 7.8.2.4. Actual tripping of the Rod Drive Motor-Generator set may be tested from the DAS. Rod Drive Motor-Generator sets may be tripped one at a time for testing.

Diverse Human System Interface Panel (DHP)

The DHP provides Manual Initiation switches for all DAS automatic actuation functions and for additional functions that are required, per the D3 Coping Analysis, to control all critical safety functions. Manual Initiation switches are not redundant. To prevent spurious actuation due to a failure of any of the above switches, a separate manual actuation permissive switch is provided. This is referred to as the “Permissive Switch for DAS HSI.”

The DHP also provides indications, per the D3 Coping Analysis, to monitor all critical safety functions.

The DHP also provides indications, per the D3 Coping Analysis, to monitor RCS Leakage.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The DAS is required to provide a diverse capability to trip the reactor and actuate the specified safety-related equipment. The DAS is not credited for mitigating accidents in the Chapter 15 safety analyses. The DAS satisfies~~y~~ Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 5). DCD_16-520/4183

The DAS LCO provides the requirements for the OPERABILITY of the DAS necessary to place the reactor in a shutdown condition and to remove decay heat in the event that required PSMS components do not function due to CCF.

A channel is OPERABLE provided the "as-found" accuracy value does not exceed its associated Allowable Value. A trip setpoint may be set more conservative than the Trip Setpoint as necessary in response to plant conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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The DAS is required to be OPERABLE in the MODES specified in Table 3.3.6-1. All functions of the DAS are required to be OPERABLE in MODES 1, 2 and 3 with the pressurizer pressure > P-11.

DAS functions are as follows:

1. Reactor Trip, Turbine Trip and Main Feedwater Isolation

   a. Manual Initiation

      The LCO requires 1 channel to be OPERABLE. This consists of the Reactor Trip, Turbine Trip and Main Feedwater Isolation - Manual Initiation switch. This function requires operation of the Permissive Switch for DAS HSI. The Permissive Switch for DAS HSI is common for all DAS Manual Initiation/Control Functions. The operator can initiate this function at any time by operation of both of these switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   b. Automatic Actuation Logic and Actuation Outputs

      This LCO requires two channels to be OPERABLE. Actuation logic consists of all circuitry housed within the DAAC, up to the Power Interface modules responsible for actuating the ESF equipment.

   c. Low Pressurizer Pressure

      There are four Low Pressurizer Pressure channels in two-out-of-four voting logic. This automatic function is automatically blocked when status signals (P-4) are received indicating that the minimum combination of the RTBs have actuated for the RT function. The LCO requires 2 Low Pressurizer Pressure channels to be OPERABLE.

   d. High Pressurizer Pressure

      There are four High Pressurizer Pressure channels in two-out-of-four voting logic. This automatic function is automatically blocked when status signals (P-4) are received indicating that the minimum combination (2-out-of-4) of the RTBs have actuated for the RT function. The LCO requires 2 High Pressurizer Pressure channels to be OPERABLE.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

Condition A applies when one or more required DAS Functions are inoperable.

If one or more required DAS functions are inoperable, 30 days are allowed to restore the Function to OPERABLE status. 30 days is reasonable because the DAS is a separate and diverse non-safety backup system. The 30 days Completion Time allows sufficient time to repair an inoperable DAS and ensures the control is repaired to provide backup protection.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 30 days, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times for Required Actions A.2 and A.3 are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

SR 3.3.6.1 is performance of CHANNEL CHECK. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

[The Frequency of 31 days is justified based on the following: Since sensor signals used by the DAS are distributed from the PSMS, the CHANNEL CHECK of the DAS sensors is included in the PSMS CHANNEL CHECK, which is conducted automatically and continuously. The isolation module of the PSMS and the indicator of the DAS, that cannot be confirmed in the continuous CHANNEL CHECK on the PSMS, are~~is~~ confirmed by this SR. These conventional analog devices, which operate only in mild environments, have a long history of proven reliability.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

[The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

[The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.] The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45 Revision 1, May 2008~~1973~~.
3. Chapter 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string. [The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. [The Frequency of 24 months 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. From the instrumentation aspects, the Frequency is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 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, Section IV, GDC 30.
2. Regulatory Guide 1.45 Revision 1, May 2008.
3. Chapter 5.

Chapter 17

US-APWR DCD Chapter 17 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ Item, table with column/row, or figure)	Description of Change
17-i	CONTENTS	Editorial: typographical error Page numbers of the following subsections are corrected: 17.4.8 ITAAC for the D-RAP 17.4.9 Combined License Information 17.4.10 References

CONTENTS

Chapter 19

US-APWR DCD Chapter 19 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
19.1-114	19.1.6.1 Third bullet (AC: Offsite power recovery) of heading description in LOOP event tree	RAI#528 19-410 Revised allowable time and recovery probability of offsite power for POS 8-1 in accordance with LPSD PRA model. Inserted the description to estimate allowable time.
19.1-122	19.1.6.2 Final bullet in description of reduction factor	RAI#528 19-410 Inserted "The failure probability … decay heat generation." as description to set allowable time for offsite power recovery" as description to estimate failure probability of offsite power.
19.1-135	19.1.6.2 Case 09 of sensitivity analysis	RAI#528 19-414 Changed from "22 times" to "2.2 times".
19.1-140	19.1.6.3.1	RAI#528 19-422 Inserted additional description for seismic at LPSD to reflect the RAI.
19.1-144	19.1.6.3.2 Final three paragraphs	RAI#528 19-407 Inserted discussion for fire PRA at LPSD.
19.1-145	19.1.6.3.3 Final three paragraphs	RAI#528 19-407 Inserted discussion for flood PRA at LPSD.
19.1-817	Table 19.1-85 Sheet 3	RAI#528 19-415 Changed D-train ESW pump status to "standby" for all success criteria.
19.1-952 to 19.1-977	Table 19.1-119	Editorial Changed total sheet number to 26 to insert new sheets in Table 19.1-119 to reflect RAI #423, #443 and #528.
19.1-954	Table 19.1-119 Sheet 3	RAI#528 19-409 Inserted design features and insights of CS/RHR pump full-flow test line stop valves.

US-APWR DCD Chapter 19 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item ,table with column/row, or figure)	Description of Change
19.1-965	Table 19.1-119 Sheet 14	RAI#528 19-417 Inserted key insights and assumptions of operator assumptions for at-power operation.
19.1-967	Table 19.1-119 Sheet 16	RAI#528 19-417 Inserted operator action for LPSD in item 16, sheet 16 of Table 19.1-130.
19.1-967	Table 19.1-119 Sheet 16	RAI#528 19-418 Inserted operator action for RHR recovery at LPSD in item 17, sheet 16 of Table 19.1-130.
19.1-969	Table 19.1-119 Sheet 18	RAI#528 19-420 Added disposition COL 13.5 (7) in item 1 of LPSD assumption.
19.1-970	Table 19.1-119 Sheet 19	RAI#528 19-408 Inserted key assumption of maintenance rule in item 14 of LPSD assumption.
19.1-971	Table 19.1-119 Sheet 20	RAI#528 19-413 Inserted key assumption of surge line flooding phenomena and new sheet (sheet 20).
19.1-978	Table 19.1-120	RAI#528 19-422 Incorporated new table “Initiating Events and Mitigation Systems during LPSD”, which is related to seismic.
19.1-979 to 19.1-999	Table 19.1-121 to Table 19.1-130	RAI#528 19-407 Incorporated new tables that show the results of external events, which are fire and flood events, during LPSD.
19.1-1078	Figure 19.1-20	RAI#528 19-419 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.

supply equipment. Following the LOOP, gas turbines, or AAC power attempt to start up and supply ac power. If the gas turbines or AAC power fail to start or run for the required mission, decay heat removal is lost.

- The frequency of a LOOP is estimated as 1.96E-01/Y. This is the frequency of the LOOP per reactor year as described in Reference 19.1-41.

- Based on a POS 8-1 duration of 56 hours (Table 19.1-82), the probability of a LOOP during POS 8-1 is:

  1.96E-01 / 8760 × 56 = 1.25E-03

- The frequency of plant shutdown for the typical analysis case is 1 shutdown / 2 years = 0.5 events per year assuming a refueling shutdown scheduled every 24 months.

Therefore, the frequency of a LOOP during POS 8-1 is:

1.25E-03 x 0.5 = 6.2E-04/Y

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 hour.~~six hours, based on time until uncover of reactor core by MAAP analysis. The probability that the LOOP duration ~~exceeds~~ does not exceed 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.

- Differences in available time for offsite power recovery are considered. Available time for offsite power recovery varies with POSs since decay heat generation and initial water level varies with POSs. For scenarios that take credit of offsite power recovery, the difference in failure probabilities of offsite power recovery compared to POS 8-1 is factored in the reduction factor. The failure probability of offsite power recovery for each POS is evaluated based on the allowable tie until uncover of reactor core evaluated by MAAP analysis, considering the POS specific initial inventory and decay heat generation.

CDFs of other POSs are given in Table 19.1-89. The overall estimate of CDF for all LPSD POSs is 2.2E-07/RY.

LOCA initiating event is significant for all POSs during low power and shutdown. For all POSs, LOCA is conservatively assumed to occur by opening of a single valve. Its frequency is higher than other initiating events that are caused by mechanical failures, hence largely contributes to the CDF. The LOCA frequencies do not vary with duration of each POSs because it is determined by human error probability. Since other initiating event frequencies vary with duration of its POS, LOCA frequencies tend to become relatively higher than other initiating events in POSs with short duration.

Significant core damage sequences for each POSs other than POS 8-1 are shown below.

(POS 3)

The top three accident sequences contribute 91 percent of the Level 1 shutdown core damage frequency of POS 3. These dominant sequences are as follows:

- LOCA initiating event, with success of leakage isolation followed by failures of RCS make-up, RCS injection and secondary side cooling, which contributes 60 percent of the CDF
- LOCA initiating event, with failures of leakage isolation and RCS injection, which contributes 17 percent of the CDF
- LOCA initiating event, with success of leakage isolation and RCS makeup followed by failures of RHR operation, RCS injection and secondary side cooling, which contributes 13 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of leakage isolation followed by failures of RCS make-up, RCS injection and secondary side cooling

  Isolation of the source of LOCA is successful. RCS makeup fails, and the RHRS as a mitigation system cannot be restored. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, gravitational injection is unavailable during this POS. Decay heat removal by SGs and injection to the RCS by charging pump or SI pumps fail, and eventually, the core is damaged.

This sensitivity study evaluates the impact of having perfect operators (i.e., setting all human error probabilities to 0.0 in the baseline shutdown core damage quantification).

This sensitivity produces a CDF of 2.8E-08/RY, which is decrease of 87 percent in the base CDF. This indicates that the operator actions are risk important at the level of plant risk obtained from the base case study.

- Case 06: All HEPs set to mean value

In this sensitivity analysis, mean HEPs, rather than lower bound value, are applied for human actions that will have frequent training. The resulting CDF is 7.9E-07/RY, which is 3.5 times of base case CDF.

- Case 07: Sensitivity to dependency of human error to CD(complete dependency)

This sensitivity study evaluates the impact of setting dependency level of human error to CD. That is, the sensitivity case most conservatively assumes that operator actions have a complete dependency on a previously failed action.

This sensitivity produces a CDF of 9.4E-06/RY, which is approximately 43 times of the base CDF. This indicates that assumption of dependency of human error provide significant impact to result of PRA during shutdown, and the operators play a significant role in maintaining a very low CDF during shutdown conditions.

- Case 08: Sensitivity to dependency of human error to ZD (zero dependency)

This sensitivity study evaluates the impact of setting dependency level of human error to ZD. That is, the sensitivity case most non-conservatively assumes that operator actions are independent absolutely between prior mitigation system and post mitigation system.

This sensitivity produces a CDF of 7.7E-08/RY, which is decrease of 65 percent in the base CDF. This indicates that assumption on dependency of human error provide meaningful sensitivity to result of PRA during shutdown.

- Case 09: Sensitivity to higher dependency of human error

This sensitivity study evaluates impact of setting higher dependency level between operator actions, which assumes that changing window on display is not effective. That is, dependency level is considered to be performed in the same location.

This sensitivity produces a CDF of 4.8E-07/RY, which is approximately ~~22~~2.2 times of the base case CDF.

19.1.6.3.1 Seismic at LPSD

The initiating events that are modeled in the internal event LPSD PRA of Subsection 19.1.6 are considerable for seismic during LPSD. According to the event trees defined by the internal event LPSD PRA, it is possible to prevent core damage if any one of mitigation systems and support systems is available. Table 19.1-120 describes the initiating events are available mitigation systems for seismic event during LPSD. For seismic, it is assumed that the SSCs of non seismic category I are not available. Only operator actions in the main control room to start-up a standby mitigation system to prevent core damage is expected in the LPSD seismic PRA.

~~For seismic,~~ SSCs for LPSD has been involved in Subsection “19.1.5.1 Seismic Risk Evaluation” and those are confirmed that the HCLPFs are greater than or equal to RLE. Seismic failures of SSCs that are assumed to directly lead to core damage, such as seismic failure of reactor building, are also included in Subsection 19.1.5.1. Thus the US-APWR has sufficient seismic margin during LPSD.

19.1.6.3.2 Internal Fire at LPSD

The scope of the internal fire PRA for LPSD at design certification phase focused on mid-loop operations since during these states the plant would be most vulnerable fire such as maintenance-induced fire. POS 8-1(mid-loop operation) is risk significant for the internal event LPSD PRA. For internal fires, risk significant POS 8-1 of LPSD has been estimated using the same methodology at power though the transient fire due to welding and cutting works and access for maintenance works have been specially reflected. The primary focus of the fire scenario development is the potential of fire damage to Yard transformers, RHRS, CVCS and its support system. Possible initiating events by internal fire at LPSD are as follows:

- LOCA
- OVDR (Loss of RHR due to over drain)
- LOOP (Loss of offsite power)

Standby states of mitigation systems for those initiators are shown in Table19.1-83. The states of out of services of POS 8-1 are similar to other POSs so that there are not more severe other POSs than POS 8-1 related to conditions of available mitigation systems. Therefore POS 8-1 is selected for internal fire at LPSD PRA.

LOCA and LOOP initiating events are potentially significant for all POSs. On the other hand, OVDR and FLWL are initiating events only considered in POSs representing mid-loop operation. Accordingly, LOCA and LOOP are significant in POSs where the RCS is full, while for POS of mid-loop operation, OVDR and/or FLWL are significant event other than LOCA and LOOP. In internal fire PRA for at-power operation, fire in the

- The impacts to LPSD mitigation systems are assumed the worst scenario.

The ~~results~~ result of CDF of POS 8-1 ~~are~~ is 1.9E-08/RY. The uncertainty range for the POS 8-1 is 1.5E-09 – 6.3E-08/RY for the 5% to 95% interval.

CDFs of other POSs for internal fire at LPSD are estimated based on the model of POS 8-1. Table 19.1-121 lists the CDF of each POS. The total CDF of internal fire at LPSD is 4.8E-08/RY. CDFs of other POSs by bounding analysis are lower than CDF of POS 8-1.

The dominant scenarios, dominant cutsets and basic event importance (FV importance and RAW) for the internal fire at LPSD (POS 8-1) are shown in Table 19.1-122, Table 19.1-123, Table 19.1-124 and Table 19.1-125, respectively. Risk by internal fire at shutdown has been very small in spite of conservative assumptions.

19.1.6.3.3 Internal Flood at LPSD

The scope of the internal flood PRA for LPSD at design certification phase focused on mid-loop operations since during these states the plant would be most vulnerable to flooding such as maintenance-induced flooding. POS 8-1(mid-loop operation) is risk significant for the internal event LPSD PRA. The primary focus of the flood scenario development is the potential of flood damage to the RHR system and its support systems. Possible initiating events by internal flood at LPSD are as follows.

- LOCA (Flood at CVCS letdown line)
- Loss of RHR (Flood at CSS/RHRS line)
- Loss of CCWS/ESWS (Flood at CCWS/ESWS line)

Standby states of mitigation systems for those initiators are shown in Table19.1-83. The states of out of services of POS 8-1 are similar to other POSs so that there are not more severe other POSs than POS 8-1 related to conditions of available mitigation systems. Therefore POS 8-1 is selected for internal flood at LPSD PRA.

Loss of CCW/ESW initiating event is significant for all POSs during low power and shutdown. As can be seen by at-power operation internal flooding PRA, the probability of consequential loss of CCW/ESW event caused by flooding is much higher than loss of other functions. In POSs where redundancy of CCW/ESW is degraded, the conditional core damage probability will increase. These features are common to all POSs and accordingly, loss of CCW/ESW is considered to be a significant initiating event.

The qualitative and quantitative steps of internal flood PRA as described in subsection 19.1.5.3 is also applied to the low power and shutdown modes.

The frequencies of internal flooding at power are also applied to the frequencies at LPSD. This assumption may be more conservative because the pressure conditions of LPSD operation are low and it may be expected that the possibility of rupture of pipe will be less.

During shutdown operations, temporary piping pressure boundaries and operator errors during maintenance may be possible initiators of internal flooding. However, the internal flood by the effect of those temporary isolation valves, such as freeze seals, are not considered from the potential initiators because the isolation valves are installed considering maintenance and CCWS has been separated individual trains.

Also flood risk at LPSD has been evaluated following conservative assumptions.

- Assumed most risk dominant POS: POS 8-1 (mid-loop operation, 55.5 hours).
- Initiating event frequencies for LPSD flood initiating events are assumed as the total flood frequencies of each flood mode (spray, flood, and major flood) at power.
- The impacts to LPSD mitigation systems are estimated assuming the worst scenario (boundary conditions of event trees).
- The flood barriers that separated the reactor building between the east side and the west are effective.
- Assumed available safety injection pumps are A and C pumps and outage safety injection pumps are B and D from the insights of flooding risk.

The CDF of the flooding risk at POS 8-1 of LPSD ~~was~~ is 1.8E-08/RY. The uncertainty range for the POS 8-1 is 4.2E-10/RY – 6.8E-08/RY for the 5% to 95% interval.

CDFs of other POSs for internal flood at LPSD are estimated based on the model of POS 8-1. Table 19.1-126 lists the CDF of each POS. The total CDF of internal flood at LPSD is 5.7E-08/RY. CDFs of other POSs by bounding analysis are lower than CDF of POS 8-1.

The dominant scenarios, dominant cutsets and basic event importance (FV importance and RAW) for the internal flood at LPSD (POS 8-1) are shown in Table 19.1-127, Table 19.1-128, Table 19.1-129 and Table 19.1-130. Important SSCs for internal flood at LPSD are RHR, CCWS and supporting power supply systems. Risk from internal flood at LPSD has been very small though it has been estimated using conservative assumptions.

19.1.7 PRA-Related Input to Other Programs and Processes

The following subsections describe PRA-related input to various programs and processes.

Table 19.1-85 Success Criteria of POS 8-1 for LPSD PRA (Example) (Sheet 3 of 4)

Success Criteria of ESWS

Initiating event identifier	Except loss of offsite power and loss of CCW/essential service water	
Success criteria	(A, B, C sub-train) ESW 1 pump/train	(D sub-train) Unavailable
	Pump A: run Pump B: run Pump C: run	Pump D: ~~outage~~standby
Mission time	24 hours	-
Operator actions	Change of strainer line by manual operation (if necessary)	-
Initiating event identifier	Loss of offsite power	
Success criteria	(A, B, C sub-train) ESW 1 pump/train	(D sub-train) Unavailable
	Pump A: run (need to restart) Pump B: run (need to restart) Pump C: run (need to restart)	Pump D: ~~outage~~standby
Mission time	24 hours	-
Operator actions	Change of strainer line by manual operation (if necessary)	-
Initiating event identifier	Loss of CCW/essential service water	
Success criteria	Unavailable	
	Pump A: run (unavailable) Pump B: run (unavailable) Pump C: run (unavailable) Pump D: ~~outage~~standby	
Mission time	-	
Operator actions	-	

Table 19.1-119 Key Insights and Assumptions (Sheet 1 of ~~23~~26)

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~~26)

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~~26)

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
- The containment spray/residual heat removal pump full-flow test line stop valves (RHS-MOV-025A/B/C/D) are locked closed.	5.4.7.2.2.3
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~~26)

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~~26)

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.	
- To cope with common cause failure of EFW pit water level sensors, a non-safety water level sensor diverse from the safety related water level sensors are installed in each EFW pit. Low water level in the EFW pit can be detected by these non-safety sensors. Accordingly, the operator can recognize the low water level in the EFW pit during EFW pump operation with high reliability.	
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

Table 19.1-119 Key Insights and Assumptions (Sheet ~~5~~ 6 of ~~23~~26)

Key Insights and Assumptions	Dispositions
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
- 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.	

Table 19.1-119 Key Insights and Assumptions (Sheet ~~6~~ 7 of ~~23~~26)

Key Insights and Assumptions	Dispositions
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~~ 8 of ~~23~~26)

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~~ 9 of ~~23~~26)

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~~ 10 of ~~24~~26)

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~~ 11 of ~~23~~26)

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~~ 12 of ~~23~~26)

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~~ 13 of ~~23~~26)

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~~ 14 of ~~23~~26)

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)
23. Action to open Unlocked motor-operated valve is performed in series through the communication between operators in electrical room and in main control room.	18.6

Table 19.1-119 Key Insights and Assumptions (Sheet ~~13~~ 15 of ~~23~~26)

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~~ 16 of ~~23~~26)

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)
16. Action to open Unlocked motor-operated valve is performed in series through the communication between operators in electrical room and in main control room.	18.6
17. In the event of decreasing RCS water level, operator actions to trip the CS/RHR pumps before cavitation and to restart the pumps after water level is restored will improve the reliability of RHR recovery. This operator action is important to reduce risk during shutdown.	COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~15~~ 17 of ~~23~~26)

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~~ 18 of ~~23~~26)

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.	COL 13.5(7)
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~~ 19 of ~~23~~26)

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)
14. Maintenance rule process is implemented to evaluate the risk of configurations being entered during shutdown. These practices assure that removing a number of related systems from service at the same time is carefully considered and virtually never done when the conditional risk impacts are high.	COL 17.6(1)

Table 19.1-119 Key Insights and Assumptions (Sheet 20 of 26)

Key Insights and Assumptions	Dispositions
15. Surge line flooding may occur if decay heat removal function is lost during plant operating states where the pressurizer manway is the only vapor release pass from the RCS. Water held up in the pressurizer can erroneous readings of water level indicators measured with reference to the pressurizer. This phenomenon can also prevent gravity injection from the SFP. Measures to prevent accident evolution caused by surge line flooding are important. Adoption of at least one of the measures listed below can reduce risk from surge line flooding event. - Installation of an temporary RCP water level sensor that measure the MCP water level with reference to pressure at the reactor vessel head vent line and cross over leg when the RCS is vented at a high elevation. - Operational procedures to perform continuous RCS injections when loss of RHR occurs under conditions where the pressurizer manway is the only vapor release pass from the RCS. The temporary water level will satisfy the following specifications. ▪ Water level can be read outside the containment vessel (CV) in order to be effective during events which involve harsh environment in the CV ▪ Tygon tubing monometer will not be used ▪ Instrumentation piping diameter will be sufficient enough to prevent delay in response	5.4.7.2.3.6 19.2.5 COL 19.3(6) COL 13.5(7)

Table 19.1-119 Key Insights and Assumptions (Sheet ~~18~~ 21 of ~~23~~26)

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~~ 22 of ~~23~~26)

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~~ 23 of ~~23~~26)

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~~ 24 of ~~23~~26)

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~~ 25 of ~~23~~26)

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~~ 26 of ~~23~~26)

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)

Table 19.1-120 Initiating Events and Mitigation Systems during LPSD

Identifier	Initiating Event Description	Mitigation Systems							POS
		LO	MC	RH	SG	SI	CV	GI	
OVDR	Loss of RHRS due to Over-drain	X	(1)	(2)	(3)	X	(4)		POS 4-1 and POS 8-1
FLML	Loss of RHRS Caused by Filling to Maintain Water Level	X	(1)	(2)	(3)	X	(4)	(5)	POS 4 and POS 8
LOCA	Loss of Coolant Accident	X	(1)	(2)	(3)	X	(4)	(5)	All POSs
LORH	Loss of RHRS Caused by Other Failures				(3)	X	(4)	(5)	All POSs
LOCS	Loss of CCWS/ESWS (7)							(5)	All POSs
LOOP	Loss of Offsite Power			X (6)	(3) (6)	X (6)	(4) (6)	(5) (6)	All POSs

(Notes)
X: The system would be functional during and after a seismic event.
(1) MC is assumed to be non-functional due to a seismic event since the refueling water auxiliary tank is not Seismic Category I.
(2) Failure of MC would lead to loss of RH.
(3) SG is not available during POS4-2, 4-3, 8-1and 8-2.
(4) CV is assumed to be non-functional due to a seismic event since the refueling water auxiliary tank is not Seismic Category I.
(5) GI is assumed to be non-functional due to a seismic event since the refueling water recirculation pumps to provide boric water from RWSP to the spent fuel pits are not Seismic Category I.
(6) In order to operate mitigating systems, GT/G is required to start and run after loss of offsite power.
(7) The plant has a seismic margin for seismically induced loss of CCWS/ESWS since the seismic capacity of CCWS/ESWS is higher than review level earthquake

(Acronyms)
LO (Isolation of Letdown Line), MC (RCS Makeup by Charging Pumps), RH (Decay Heat Removed from the RCS by the RHRS on Standby), SG(Decay Heat Removed from the RCS via SGs), SI (High Head Injection), CV (Injection by Chemical and Volume Control System), GI (Gravitational Injection)

Table 19.1-121 Core Damage Frequency of Each Initiating Event and Each POS of Internal Fire at LPSD

IE	Event Description	POS3	POS4-1	POS4-2	POS4-3	POS8-1	POS8-2	POS8-3	POS9	POS11	Total
LOCA	Loss of coolant accident	7.1E-11	N/A	N/A	N/A	N/A	N/A	N/A	1.9E-10	1.4E-09	1.6E-09
OVDR	Loss of RHR due to over drain	N/A	1.1E-09	N/A	N/A	1.7E-09	N/A	N/A	N/A	N/A	2.8E-09
FLML	Loss of RHR Caused by maintain water Level	N/A	N/A	2.3E-09	1.1E-09	N/A	1.7E-09	1.6E-09	N/A	N/A	6.6E-09
LOOP	Loss of Offsite Power	3.0E-10	5.1E-09	2.3E-09	1.7E-09	1.7E-08	2.3E-09	1.4E-09	1.3E-09	5.7E-09	3.7E-08
TOTAL		3.7E-10	6.2E-09	4.5E-09	2.8E-09	1.9E-08	4.0E-09	3.0E-09	1.5E-09	7.0E-09	4.8E-08

N/A: not applicable

Table 19.1-122 Dominant Scenarios of Internal Fire at LPSD (POS 8-1) (Sheet 1 of 2)

Rank	Fire Scenario Number	Fire Scenarios	CDF (/RY)
1	Yard	This scenario contains main transformer and reserve auxiliary transformer. Fire ignition source postulated in the switchyard are catastrophic fire, non-catastrophic fire and other fires of transformer. The fire ignition frequency is 6.4E-05/RY. The fire in this switchyard may cause LOOP (loss of offsite power), and it is anticipated that the recovery of offsite power is not easy. CCDP of this fire scenario has been estimated to 2.5E-04. Fire scenario postulated is as follows: • Fire may cause LOOP because main transformer and reserve auxiliary transformer located in switchyard may be damaged by the fire. • Offsite power cannot be recovered because the fire may damage both of main transformer and reserve auxiliary transformer. • Combination of the random failure of class 1E gas turbine generators and the failure of switchover to AAC gas turbines generators.	1.6E-08
2	FA6-101-04	FA6-101-04 has the potential of transient combustibles fire and cable fire caused by welding or cutting and so forth, whose fire ignition frequency is 4.5 E-06/RY. This scenario also contains all four train cables to class 1E bus ducts from offsite power sources. Therefore, the fire in this scenario may cause LOOP, and it may make the recovery of every power sources impossible. And, CCDP of this fire scenario has been estimated to 2.5E-04. Fire scenario is as follows: • Fire may cause LOOP because it may damage all four train cables to class 1E bus ducts from offsite power located in FA6-101-04. • Offsite power cannot be recovered because fire may damage all four train cables to class 1E bus ducts from offsite power sources. • Combination of the random failure of Gas Turbine system and the failure of changeover to AAC gas turbine generators.	1.1E-09

Table 19.1-122 Dominant Scenarios of Internal Fire at LPSD (POS 8-1) (Sheet 2 of 2)

Rank	Fire Scenario Number	Fire Scenarios	CDF (/RY)
3	FA4-101	FA4-101 consists of all zones in A/B, and many fire ignition sources are contained in this area. Fire ignition frequency of this scenario is 7.7E-05/RY. Because the cable of CVS-LCV-12A (letdown line volume control tank inlet changeover Valve) is located in this compartment, fire induced cable damage has the potential to cause "Over-drain" event. Therefore, this fire scenario has been identified and CCDP of this scenario has been estimated to 5.7E-03. Fire scenario is as follows: • Spurious operation of CVS-LCV-121A has caused "Over-drain" event. • Water feed via charging system has become impossible by the spurious closing of charging Line stop valve (A-LOOP Cold Leg AOV-1). • Failure of start-up of stand-by high head Injection pumps lead to core damage because it results in the loss of high head injection system function.	1.1E-09

Table 19.1-123 Dominant Cutsets of Internal Fire at LPSD (Sheet 1 of 2)

No.	Cutset Freq.(/RY)	Percent	Cutsets	Basic Event Description
1	3.8E-09	21.7	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			ESWCF3PMBDSWPABC-ALL	ESW PUMP A,B,C FAIL TO RE-START
			GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT
2	3.5E-09	20.0	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			CHIOO02CV212-DP3	OPERATOR FAILS TO RECOVER THE RCS WATER LEVEL BY CHARGING INJECTION WHEN THE RCS WATER LEVEL INDICATES LOW, UNDER THE CONDITION WHERE THEIR PREVIOUS TASK HAS FAILED (HE)
			GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT
			HPIOO02S-DP2	OPERATOR FAILS TO START STANDBY SI PUMP UNDER THE CONDITION OF FAILING THEIR PREVIOUS TASK (HE)
			RSSOO02RHR2	OPERATOR FAILS TO START THE STANDBY RHR PUMP (HE)
3	2.1E-09	11.9	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			CWSCF3PCBDCWPABC-ALL	CWS PUMP A,B,C FAIL TO RE-START CCF
			GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT
4	1.5E-09	8.5	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			EPSCF3DLLRDG-ALL	EMERGENCY GAS TURBINE GENERATOR (GTG A,B,C) FAIL TO RUN (>1H) CCF
			EPSOO02RDG	OPERATOR FAILS TO CONNECT ALTERNATIVE GTG TO SAFETY BUS (HE)
5	1.3E-09	7.3	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			EPSCF4CBTD6H-ALL	6.9KV AC BUS INCOMER CIRCUIT BREAKER (6HA,B,C,D) FAIL TO OPEN CCF

Table 19.1-123 Dominant Cutsets of Internal Fire at LPSD (Sheet 2 of 2)

No.	Cutset Freq.(/RY)	Percent	Cutsets	Basic Event Description
6	6.4E-10	3.6	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			EPSBTSWCCF	EPS SOFTWARE CCF
7	4.9E-10	2.8	SDFA4-101-B15	Fire Ignition Frequency (Electrical cabinets)
			GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT
			HPIOO02S	OPERATOR FAILS TO START STANDBY SI PUMP (HE)
			SDFA4-101-OVD	IE OCURRENCE PROBABILITY AT FA4-101
			SG	FAIL TO REMOVE DECAY HEAT BY STEAM GENERATOR SYSTEM
8	4.2E-10	2.4	SDFA2-308-B4	Fire Ignition Frequency (Main Control Board)
			SDFA2-308-OVD	Initiating Event (OVDR) at FA2-308
			RSPEVA	EVACUATION TO RSC Room
			HPIOO02S-R	OPERATOR FAILS TO START STANDBY SI PUMP (HE)
9	3.2E-10	1.8	SDYARD-B29	Fire Ignition Frequency (Yard transformers (Others))
			EPSCF3DLADDG-ALL	EMERGENCY GAS TURBINE GENERATOR (GTG A,B,C) FAIL TO START CCF
			EPSOO02RDG	OPERATOR FAILS TO CONNECT ALTERNATIVE GTG TO SAFETY BUS (HE)
10	2.6E-10	1.5	SDFA4-101-B21	Fire Ignition Frequency (Pumps)
			GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT
			HPIOO02S	OPERATOR FAILS TO START STANDBY SI PUMP (HE)
			SDFA4-101-OVD	IE OCURRENCE PROBABILITY AT FA4-101
			SG	FAIL TO REMOVE DECAY HEAT BY STEAM GENERATOR SYSTEM

Table 19.1-124 Basic Events (Hardware and Human Error) FV Importance of Internal Fire at LPSD (POS8-1)

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
1	GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	1.0E+00	6.9E-01	1.0E+00
2	SG	FAIL TO REMOVE DECAY HEAT BY STEAM GENERATOR SYSTEM	1.0E+00	3.2E-01	1.0E+00
3	RSSOO02RHR2	OPERATOR FAILS TO START THE STANDBY RHR PUMP (HE)	6.2E-03	2.3E-01	3.8E+01
4	ESWCF3PMBDSWPABC-ALL	ESW PUMP A,B,C FAIL TO RE-START	6.0E-05	2.3E-01	3.9E+03
5	HPIOO02S-DP2	OPERATOR FAILS TO START STANDBY SI PUMP UNDER THE CONDITION OF FAILING THEIR PREVIOUS TASK (HE)	5.5E-02	2.3E-01	4.9E+00
6	CHIOO02CV212-DP3	OPERATOR FAILS TO RECOVER THE RCS WATER LEVEL BY CHARGING INJECTION WHEN THE RCS WATER LEVEL INDICATES LOW, UNDER THE CONDITION WHERE THEIR PREVIOUS TASK HAS FAILED (HE)	1.6E-01	2.1E-01	2.1E+00
7	EPSOO02RDG	OPERATOR FAILS TO CONNECT ALTERNATIVE GTG TO SAFETY BUS (HE)	2.1E-02	1.4E-01	7.7E+00
8	CWSCF3PCBDCWPABC-ALL	CWS PUMP A,B,C FAIL TO RE-START CCF	3.3E-05	1.3E-01	3.9E+03
9	EPSCF3DLLRDG-ALL	EMERGENCY GAS TURBINE GENERATOR (GTG A,B,C) FAIL TO RUN (>1H) CCF	1.1E-03	1.1E-01	9.8E+01
10	EPSCF4CBTD6H-ALL	6.9KV AC BUS INCOMER CIRCUIT BREAKER (6HA,B,C,D) FAIL TO OPEN CCF	2.0E-05	7.8E-02	3.9E+03

Table 19.1-125 Basic Events (Hardware and Human Error) RAW of Internal Fire at LPSD (POS8-1)

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	RAW	FV
1	RTPBTSWCCF	SUPPORT SOFTWARE CCF	1.0E-07	4.0E+03	4.0E-04
2	SWSCF3PMYRSWPABC-ALL	ESSENTIAL SERVICE WATER PUMP A,B,C FAIL TO RUN (RUNNING) CCF	1.2E-07	3.9E+03	4.7E-04
3	CWSCF3PCYRCWPABC-ALL	COMPONENT COOLING WATER PUMP A,B,C FAIL TO RUN (RUNNING) CCF	6.7E-08	3.9E+03	2.6E-04
4	CWSCF3RHPRHXABC1-ALL	COMPONENT COOLING HEAT EXCHANGERS A,B,C PLUG/FOUL OR LARGE EXTERNAL LEAK CCF	3.6E-08	3.9E+03	1.4E-04
5	ESWCF3PMBDSWPABC-ALL	ESW PUMP A,B,C FAIL TO RE-START	6.0E-05	3.9E+03	2.3E-01
6	CWSCF3PCBDCWPABC-ALL	CWS PUMP A,B,C FAIL TO RE-START CCF	3.3E-05	3.9E+03	1.3E-01
7	EPSCF4CBTD6H-ALL	6.9KV AC BUS INCOMER CIRCUIT BREAKER (6HA,B,C,D) FAIL TO OPEN CCF	2.0E-05	3.9E+03	7.8E-02
8	EPSBTSWCCF	EPS SOFTWARE CCF	1.0E-05	3.9E+03	3.9E-02
9	ESWCF3CVOD602ABC-ALL	ESW C/V 602A,B,C FAIL TO RE-OPEN	3.0E-07	3.9E+03	1.2E-03
10	ESWCF3CVOD502ABC-ALL	ESW C/V 502A,B,C FAIL TO RE-OPEN	3.0E-07	3.9E+03	1.2E-03

Table 19.1-126 Core Damage Frequency of Each Initiating Event and Each POS of Internal Flood at LPSD

IE	Event Description	POS3	POS4-1	POS4-2	POS4-3	POS8-1	POS8-2	POS8-3	POS9	POS11	Total
LOCA	Loss of coolant accident	1.0E-12	1.8E-11	5.0E-12	5.0E-12	4.6E-11	5.0E-12	5.0E-12	4.6E-12	2.0E-11	1.1E-10
LOCS	Loss of CCW/ESWS	2.8E-11	8.1E-09	1.5E-10	1.9E-10	1.6E-08	3.0E-09	1.4E-08	1.2E-08	1.2E-10	5.4E-08
LORH	Loss of RHR	7.0E-14	2.4E-10	1.8E-10	1.7E-10	1.8E-09	8.3E-11	3.1E-11	2.8E-11	1.9E-13	2.5E-09
TOTAL		2.9E-11	8.4E-09	3.3E-10	3.7E-10	1.8E-08	3.1E-09	1.4E-08	1.2E-08	1.4E-10	5.7E-08

N/A: not applicable

Table 19.1-127 Dominant Scenarios of Internal Flood at LPSD (POS 8-1) (Sheet 1 of 5)

Rank	Flood Area		Flood Category	Fire Scenarios	CDF (/RY)
1	FA2-501-03	R/B NRCA 4F East side SGBD Water Radiation Monitor Room	Flood	Flood due to the rupture of piping on the 4F of R/B east side SGBD water radiation monitor room causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	2.5E-09
2	FA2-501-01	R/B NRCA 4F East side Non-Radioactive Zone Corridor	Flood	Flood due to the rupture of piping on the 4F of R/B east side non restricted zone corridor causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	2.5E-09

Table 19.1-127 Dominant Scenarios of Internal Flood at LPSD (POS 8-1) (Sheet 2 of 5)

Rank	Flood Area		Flood Category	Fire Scenarios	CDF (/RY)
3	FA2-407-04	R/B NRCA 3F East side Corridor	Flood	Flood due to the rupture of piping on the 3F of R/B east side non restricted zone corridor causes loss of function of both A and B trains of class 1E AC 120V panel boards, component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	8.2E-10
4	FA2-102-01	R/B NRCA B1F East side A-EFW (T/D) Pump Room	Major Flood	Major flood due to the rupture of piping in the A-EFW Pump (T/D) room on the B1F of R/B causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries by the effect of flooding propagation. Also A and B EFW pumps lose the function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	7.6E-10

Table 19.1-127 Dominant Scenarios of Internal Flood at LPSD (POS 8-1) (Sheet 3 of 5)

Rank	Flood Area		Flood Category	Fire Scenarios	CDF (/RY)
5	FA2-407-04	R/B NRCA 3F East side Corridor	Major Flood	Major flood due to the rupture of piping on the 3F of R/B east side non restricted zone corridor causes loss of function of both A and B trains of class 1E AC 120V panel boards, component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	7.5E-10
6	FA2-201-02	R/B NRCA 2F East side Corridor	Major Flood	Major flood due to the rupture of piping on the 2F of R/B east side non restricted zone corridor causes loss of function of both A and B trains of class 1E AC 120V panel boards, class 1E 6.9kV buses, component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	7.4E-10

Table 19.1-127 Dominant Scenarios of Internal Flood at LPSD (POS 8-1) (Sheet 4 of 5)

Rank	Flood Area		Flood Category	Fire Scenarios	CDF (/RY)
7	FA2-102-01	R/B NRCA B1F A-EFW (T/D) Pump Room	Flood	Flood due to the rupture of piping in the A-EFW Pump (T/D) room on the B1F of R/B causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries by the effect of flooding propagation. Also A and B EFW pumps lose the function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	6.7E-10
8	FA2-111-01	R/B NRCA B1F East side Corridor	Major Flood	Major Flood due to the rupture of piping on the B1F of R/B east side non restricted zone corridor causes loss of function of both A and B trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also B-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both A and B trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also B-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both C train and D train of safety systems for safe shutdown. This scenario results in core damage.	6.5E-10

Table 19.1-127 Dominant Scenarios of Internal Flood at LPSD (POS 8-1) (Sheet 5 of 5)

Rank	Flood Area		Flood Category	Fire Scenarios	CDF (/RY)
9	FA2-112-01	R/B NRCA B1F West side Corridor	Major Flood	Major Flood due to the rupture of piping on the B1F of R/B west side non restricted zone corridor causes loss of function of both C and D trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also C-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both C and D trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also C-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both A train and B train of safety systems for safe shutdown. This scenario results in core damage.	6.2E-10
10	FA2-501-11	R/B NRCA 4F West side Non-Radioactive Zone Corridor	Flood	Flood due to the rupture of piping on the 4F of R/B west side non restricted area corridor causes loss of function of both C and D trains of component cooling water pumps, essential chillers, and batteries, by the effect of flood propagation. Also C-EFW pump (M/D) loses function. The impacts to LPSD mitigation systems are assumed the worst scenario. It is assumed that major flood causes loss of function of both C and D trains of class 1E electrical equipments, component cooling water pumps and batteries by the effect of flood propagation. Also C-EFW pump (M/D) loses function. This scenario causes loss of component cooling water systems (LOCS) in conjunction with random failure of other side CCW system. Simultaneously, causes random failure or common cause failure of both A train and B train of safety systems for safe shutdown. This scenario results in core damage.	6.2E-10

Table 19.1-128 Dominant Cutsets of Internal Flood at LPSD (POS 8-1) (Sheet 1 of 5)

No.	Cutsets Freq.(/RY)	Percent	Cutsets	Frequency/ Probability	Basic Event Description	
1	1.4E-09	8.1	FA2-501-03-LOCS-F04	5.7E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-501-03, SGBD, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	
2	1.0E-09	5.7	FA2-501-01-LOCS-F06	4.0E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-501-01, SGBD, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	

Table 19.1-128 Dominant Cutsets of Internal Flood at LPSD (POS 8-1) (Sheet 2 of 5)

No.	Cutsets Freq.(/RY)	Percent	Cutsets	Frequency/ Probability	Basic Event Description	
3	4.6E-10	2.6	FA2-407-04-LOCS-F06	1.8E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-407-04, SGBD, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	
4	4.5E-10	2.6	FA2-501-03-LOCS-F04	5.7E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-501-03, SGBD, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWTMDADFWP	1.0E-02	DIESEL-DRIVEN FIRE SUPPRESSION PUMP (DFWP) OUTAGE	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	

Table 19.1-128 Dominant Cutsets of Internal Flood at LPSD (POS 8-1) (Sheet 3 of 5)

No.	Cutsets Freq.(/RY)	Percent	Cutsets	Frequency/ Probability	Basic Event Description	
5	4.3E-10	2.4	FA2-102-01-LOCS-M02	1.7E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-102-01, EFWS, MAJOR FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	
6	4.2E-10	2.4	FA2-407-04-LOCS-M03	1.7E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-407-04, EFWS, MAJOR FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	

Table 19.1-128 Dominant Cutsets of Internal Flood at LPSD (POS 8-1) (Sheet 4 of 5)

No.	Cutsets Freq.(/RY)	Percent	Cutsets	Frequency/ Probability	Basic Event Description	
7	4.2E-10	2.4	FA2-201-02-LOCS-M02	1.7E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-201-02, EFWS, MAJOR FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	
8	4.0E-10	2.3	FA2-501-11-LOCS-F03	1.6E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-501-11, EFWS, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	

Table 19.1-128 Dominant Cutsets of Internal Flood at LPSD (POS 8-1) (Sheet 5 of 5)

No.	Cutsets Freq.(/RY)	Percent	Cutsets	Frequency/ Probability	Basic Event Description	
9	3.8E-10	2.2	FA2-501-01-LOCS-F03	1.5E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-501-01, EFWS, FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	
10	3.8E-10	2.1	FA2-206-02-LOCS-M03	1.5E-03	PIPE FAILURE RATE PER ONE YEAR BY INTERNAL FLOODING (FA2-206-02, EFWS, MAJOR FLOOD)	Loss of CCWS/ ESWS
			POS8-1FACTOR	6.3E-03	THE FACTOR WHICH CONVERTS PIPE FAILURE RATE PER ONE YEAR INTO PIPE FAILURE RATE DURING POS8-1(55.5H)	
			RAM-LOCS-FM	1.3E-03	FAILURE PROBABILITY OF ONE TRAIN OF THE CCW SYSTEM BY THE RANDOM FAILURE.	
			ACWOO02SC	3.1E-02	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	
			GI	1.0E+00	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	

Table 19.1-129 Basic Events (Hardware and Human Error) FV Importance of Internal Flood at LPSD (POS8-1)

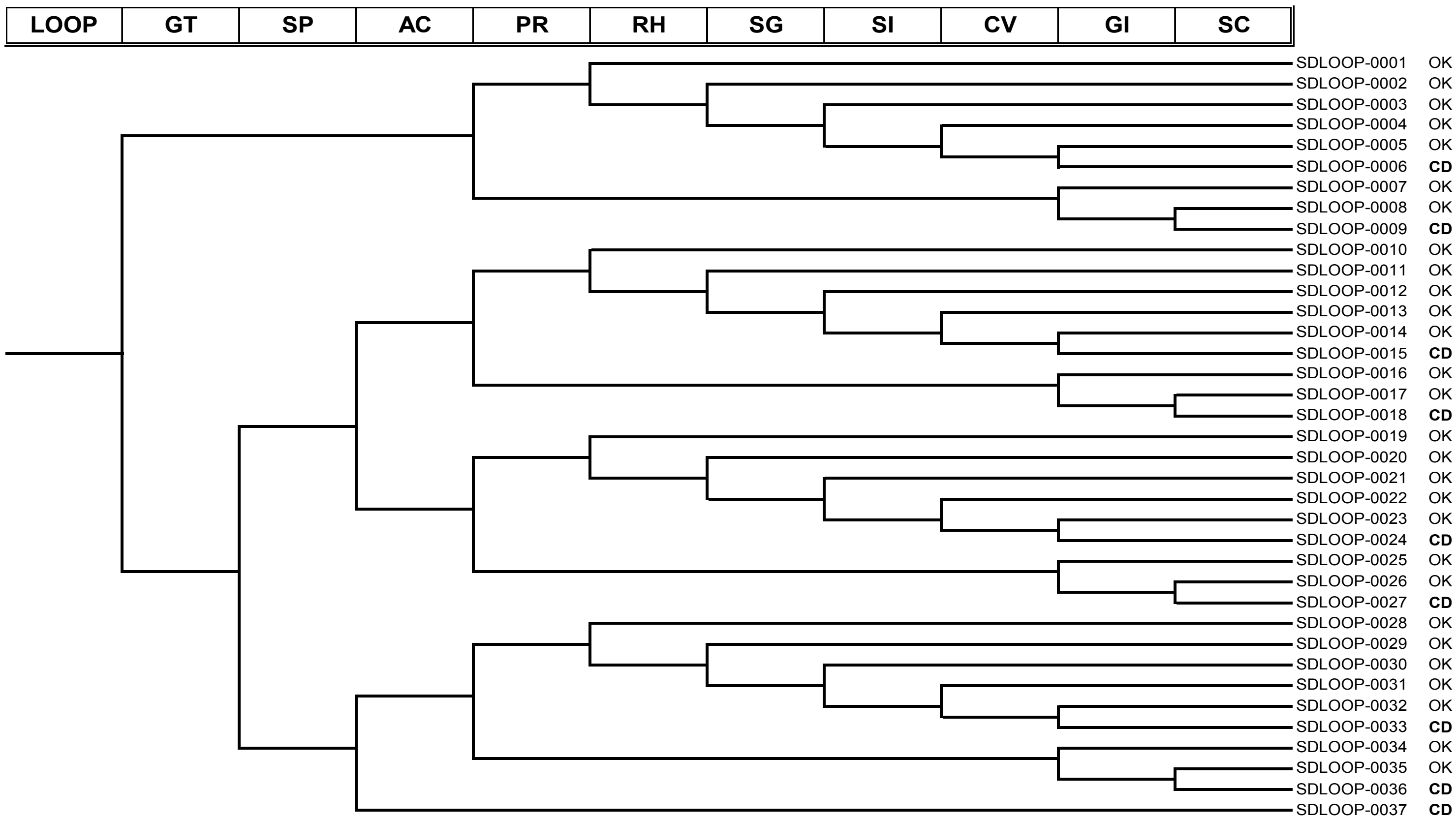
Rank	Basic Event ID	Basic Event Description	Basic Event Probability	FV Importance	RAW
1	GI	FAIL TO GRAVITY INJECTION FROM SPENT FUEL PIT	1.0E+00	1.0E+00	1.0E+00
2	ACWOO02SC	OPERATOR FAILS TO ESTABLISH THE ALTERNATIVE CCWS BY FIRE SUPPRESSION SYSTEM (HE)	3.1E-02	5.4E-01	1.8E+01
3	ACWTMDADFWP	DIESEL-DRIVEN FIRE SUPPRESSION PUMP (DFWP) OUTAGE	1.0E-02	1.1E-01	1.2E+01
4	SG	FAIL TO REMOVE DECAY HEAT BY STEAM GENERATOR SYSTEM	1.0E+00	6.8E-02	1.0E+00
5	ACWTMPZMFWP	MOTOR-DRIVEN FIRE SUPPRESSION PUMP (MFWP) OUTAGE	8.0E-03	4.8E-02	6.9E+00
6	ACWPDADDFWP	DIESEL-DRIVEN FIRE SUPPRESSION PUMP (DFWP) FAIL TO START	4.0E-03	4.6E-02	1.2E+01
7	ACWPDLRDFWP	DIESEL-DRIVEN FIRE SUPPRESSION PUMP (DFWP) FAIL TO RUN (1H<)	2.1E-03	2.4E-02	1.2E+01
8	ACWPDSRDFWP	DIESEL-DRIVEN FIRE SUPPRESSION PUMP (DFWP) FAIL TO RUN (<1H)	1.5E-03	1.7E-02	1.2E+01
9	HPIOO02S	OPERATOR FAILS TO START STANDBY SI PUMP (HE)	4.9E-03	1.7E-02	4.5E+00
10	CHIPMADCHPB-R	B-CHARGING PUMP FAIL TO START (RUNNING)	1.3E-03	1.5E-02	1.2E+01

Table 19.1-130 Basic Events (Hardware and Human Error) RAW of Internal Flood at LPSD (POS8-1) (Sheet 1 of 2)

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	RAW	FV Importance
1	SWSCF3PMYRSWPABC-ALL	ESSENTIAL SERVICE WATER PUMP A,B,C FAIL TO RUN (RUNNING) CCF	1.2E-07	2.4E+01	2.8E-06
2	CWSCF3PCYRCWPABC-ALL	COMPONENT COOLING WATER PUMP A,B,C FAIL TO RUN (RUNNING) CCF	6.7E-08	2.4E+01	1.5E-06
3	CWSCF3RHPRHXABC1-ALL	COMPONENT COOLING HEAT EXCHANGERS A,B,C PLUG/FOUL OR LARGE EXTERNAL LEAK CCF	3.6E-08	2.4E+01	8.2E-07
4	CHICF2CVOD163-ALL	SEAL WATER HEAT EXCHANGER MINIMUM FLOW LINE CHECK VALVE CVS-VLV-129A,B(163A,B) FAIL TO OPEN CCF	2.0E-06	1.8E+01	3.5E-05
5	CHICF2CVOD165-ALL	CHRGING PUMP OUTLET CHECK VALVE CVS-VLV-131A,B(165A,B) FAIL TO OPEN CCF	2.0E-06	1.8E+01	3.5E-05
6	ACWPMBDCHP-ALL	CHARGING PUMP A,B (ALTERNATE COMPONENT COOLING WATER SYSTEM) FAIL TO START (RUNNING) CCF	1.5E-04	1.8E+01	2.7E-03
7	ACWMVODCH3AB-ALL	ALTERNATE COMPONENT COOLING WATER SYSTEM FOR CHARGING PUMP MOTOR OPERATED VALVE (ACWCH3A,B) FAIL TO OPEN CCF	4.7E-05	1.8E+01	8.2E-04
8	ACWMVODCH1AB-ALL	ALTERNATE COMPONENT COOLING WATER SYSTEM FOR CHARGING PUMP MOTOR OPERATED VALVE (ACWCH1A,B) FAIL TO OPEN CCF	4.7E-05	1.8E+01	8.2E-04

Table 19.1-130 Basic Events (Hardware and Human Error) RAW of Internal Flood at LPSD (POS8-1) (Sheet 2 of 2)

Rank	Basic Event ID	Basic Event Description	Basic Event Probability	RAW	FV Importance
9	ACWMVODCH4AB-ALL	ALTERNATE COMPONENT COOLING WATER SYSTEM FOR CHARGING PUMP MOTOR OPERATED VALVE (ACWCH4A,B) FAIL TO OPEN CCF	4.7E-05	1.8E+01	8.2E-04
10	ACWMVODCH2AB-ALL	ALTERNATE COMPONENT COOLING WATER SYSTEM FOR CHARGING PUMP MOTOR OPERATED VALVE (ACWCH2A,B) FAIL TO OPEN CCF	4.7E-05	1.8E+01	8.2E-04



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

Tier 1

US-APWR DCD Tier1 Section 1.0 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
1-3	Section 1.3	Changed the definition of "As-built" to reflect the discussion with the NRC and NEI. RAI 531, 01-7
1-7	1.4.4, fifth bullet of the last paragraph	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21

US-APWR DCD Tier1 Section 2.4 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.4-7	Table 2.4.1-2, ITAAC #10	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.4-25	Table 2.4.2-5, ITAAC #8.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.4-26	Table 2.4.2-5, ITAAC #9.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.4-29	Table 2.4.2-5, ITAAC #15	Separated the ITAAC number to “15.i” and “15.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.4-50	Table 2.4.4-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.4-50	Table 2.4.4-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.4-56	Table 2.4.4-5, ITAAC #12	Separated the ITAAC number to “12.i” and “12.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.4-74	Table 2.4.5-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.4-74	Table 2.4.5-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21

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Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.4-78	Table 2.4.5-5, ITAAC #13	Separated the ITAAC number to “13.i” and “13.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.4-98	Table 2.4.6-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.4-99	Table 2.4.6-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.4-101	Table 2.4.6-5, ITAAC #15	Separated the ITAAC number to “15.i” and “15.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

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Page	Location (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.5-11	Table 2.5.1-6, ITAAC #6	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.5-11	Table 2.5.1-6, ITAAC #9	Reworded “each PSMS” to “each division of the PSMS” in the DC and AC column text for the clarification. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.5-14	Table 2.5.1-6, ITAAC #20	Deleted the redundant ITAAC (Items 9 and 20) per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.5-16	Table 2.5.1-6, ITAAC #27	Reworded “each PSMS” to “each division of the PSMS” in the DC and AC column text for the clarification. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.5-39	Table 2.5.4-2, ITAAC #3	Revised to use wording consistent with 10CFR 50.49f. Replaced “as being qualified for…” with “and that is subjected to” in the DC and AC column text. RAI 511, 03.11-21 and 03.11-24

US-APWR DCD Tier1 Section 2.6 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.6-30	Subsection 2.6.4.2	Added the following text after the first sentence of the last paragraph: The Class 1E GTG ventilation/cooling air intake and exhaust system is capable of cooling the GT including operation at 110 % of name plate. Replaced “The turbine intake openings are” with “The turbine intake and exhaust and ventilation/cooling air intake and exhaust openings are” in the last sentence of the last paragraph. RAI 505, 09.05.08-25
2.6-51	Table 2.6.8-1, ITAAC #7	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21

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Page	**Location** (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.7-15	Table 2.7.1.2-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-15	Table 2.7.1.2-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-17	Table 2.7.1.2-5, ITAAC #11	Separated the ITAAC number to “11.i” and “11.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-42	Table 2.7.1.9-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-42	Table 2.7.1.9-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-44	Table 2.7.1.9-5, ITAAC #11	Separated the ITAAC number to “11.i” and “11.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-54	Table 2.7.1.10-4, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043

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Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.7-54, 55	Table 2.7.1.10-4, ITAAC #11	Separated the ITAAC number to "11.i" and "11.ii" in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-55	Table 2.7.1.10-4, ITAAC #12	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-81	Table 2.7.1.11-5, ITAAC #5.b.ii	Replaced "inspection for the existence of the report" to "inspections and analyses for the piping" in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-81	Table 2.7.1.11-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-83	Table 2.7.1.11-5, ITAAC #11	Separated the ITAAC number to "11.i" and "11.ii" in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-101	Table 2.7.3.1-5, ITAAC #5.b.ii	Replaced "inspection for the existence of the report" to "inspections and analyses for the piping" in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043

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Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.7-102	Table 2.7.3.1-5, ITAAC #7.i	Revised “the capability of” to “the heat removal capability of” in the ITA column text. Revised “the as-built ESWS provides adequate cooling water to the CCW heat exchangers and essential chiller units of the ECWS during all plant operating conditions,…..” to “the heat removal capability of the as-built CCW heat exchangers and essential chiller units of the ECWS is greater than or equal to the design value for all plant operating conditions,…..” in the AC column text to provide the consistent wording. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-103	Table 2.7.3.1-5, ITAAC #12	Separated the ITAAC number to “12.i” and “12.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-125	Table 2.7.3.3-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-125	Table 2.7.3.3-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-127	Table 2.7.3.3-5, ITAAC #12	Separated the ITAAC number to “12.i” and “12.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-133	2.7.3.5.1	Replaced with “Not applicable” under “Equipment to be Qualified for Harsh Environments”. RAI 511, 03.11-24

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Page	**Location** (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.7-144	Table 2.7.3.5-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-146	Table 2.7.3.5-5, ITAAC #12	Separated the ITAAC number to “12.i” and “12.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

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Page	**Location** (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	**Description of Change**
2.7-177	Table 2.7.5.1-3, ITAAC #8	Separated the ITAAC number to “8.i” and “8.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
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
2.7-206	Table 2.7.5.2-3, ITAAC #8	Separated the ITAAC number to “8.i” and “8.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

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Page	**Location** (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.7-217	2.7.5.3.2	Deleted the following text for the clarification: Table 2.11.2-2 specifies the ITAAC for the non-ECWS piping system and components that are part of the CIS that supplies cooling water to the containment fan cooler unit and CRDM cooling unit cooling coils. Revised the last sentence as the following for the clarification: The ITAAC associated with the equipment, components and piping of the CVVS and non-ECWS that also comprise a portion of the CIS, are described in Table 2.11.2-2. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-218	Table 2.7.5.3-1, ITAAC #2	Recovered ITAAC number 2 as “deleted” to be consistent with the former RAI response. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-219	2.7.5.4.1.1, Location and Functional Arrangement	Added the following text as the last paragraph under “Location and Functional Arrangement”: The auxiliary building HVAC system and containment low volume purge system are cross tied. This crosstie allows the exhaust flow from the auxiliary building HVAC system to be redirected to the containment low volume purge manually upon a high radiation alarm in the auxiliary building HVAC ductwork. RAI 483, 09.04.03-08
2.7-220	2.7.5.4.1.1, Key Design Features	Added the following bullets to the end of the bullets under “Key Design Features”: • The auxiliary building HVAC system and containment low volume purge system are cross connected to allow the exhaust from the radiological controlled areas to be filtered by the containment low volume purge exhaust filtration units. • Airborne radioactivity is monitored inside the exhaust air duct from the controlled areas. RAI 483, 09.04.03-08

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Page	Location (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.7-229	Table 2.7.5.4-3, ITAAC #7	Separated the ITAAC number to “7.i” and “7.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-230	Table 2.7.5.4-3, ITAAC #9, #10	Added new ITAAC as Items 9 and 10 RAI 483, 09.04.03-08
2.7-248	Table 2.7.6.3-5, ITAAC #6.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-249	Table 2.7.6.3-5, ITAAC #10	Separated the ITAAC number to “10.i” and “10.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-256, 257	Table 2.7.6.4-2, ITAAC #7, 8, 9	Separated the reference ITAAC (i.e., Reference to Table 2.11.2-2) to the individual ITAAC. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.7-278	Table 2.7.6.7-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.7-278, 279	Table 2.7.6.7-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.7-281	Table 2.7.6.7-5, ITAAC #13	Separated the ITAAC number to “13.i” and “13.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

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Page	Location (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.7-298	2.7.6.12.1, Key Design Features	Replaced the first paragraph with the following: The potable and sanitary water system is designed with no interconnection to systems that could potentially introduce contaminants including radiological contaminates into the system. CP RAI 126, 09.02.04-1 (UAP-HF-10046)
2.7-299	2.7.6.12.1, Interface Requirements	Changed “Interfaces Requirements” to “Interface Requirements” in the heading. Replaced “The PSWS are interface systems” with “There are no safety-related interfaces with systems outside of the certified design. CP RAI 126, 09.02.04-1 (UAP-HF-10046)
2.7-306	Table 2.7.6.13-3, ITAAC #3	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21

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Page	**Location** (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.8-2	Table 2.8-1, ITAAC #2	Added the applicable ITAAC number of Table 2.7.6.13-3 in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

US-APWR DCD Tier1 Section 2.10 Rev. 2, Tracking Report Rev. 2 Change List

Page	**Location** (e.g., subsection with paragraph/sentence/ item, table with column/row, or figure)	Description of Change
2.10-2	Table 2.10-1, ITAAC #4	Revised “Table 2.7.6.10-1” to “Table 2.7.6.10-1, ITAAC Items 2, 3, and 4” to specify the applicable ITAAC numbers. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

US-APWR DCD Tier1 Section 2.11 Rev. 2, Tracking Report Rev. 2 Change List

Page	Location (e.g., subsection with paragraph/sentence/item, table with column/row, or figure)	Description of Change
2.11-23	Table 2.11.2-2, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.11-23, 24	Table 2.11.2-2, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.11-27	Table 2.11.2-2, ITAAC #11.b	Separated the ITAAC number to “11.b.i” and “11.b.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)
2.11-41	Table 2.11.3-5, ITAAC #5.b.ii	Replaced “inspection for the existence of the report” to “inspections and analyses for the piping” in ITA column text per the conference call on 12/14/2009. Editorial changes, UAP-HF-10043
2.11-41	Table 2.11.3-5, ITAAC #6.a	Revised to use wording consistent with 10CFR 50.49f. RAI 511, 03.11-21
2.11-44	Table 2.11.3-5, ITAAC #12	Separated the ITAAC number to “12.i” and “12.ii” in the ITA and AC column text. Added new ITAAC for the testing of the as-built RSC control in the ITA and AC column text. Editorial changes, UAP-HF-10043 (conference call on 12/14/2009)

Analysis means a calculation, mathematical computation, or engineering/ technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar SSCs.

As-built means the physical properties of a structure, system, or component~~the SSC~~ following ~~the~~ completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, ~~D~~determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing do not alter the properties.

ASME Code means Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Column line is the designation applied to a plant reference grid used to define the locations of building walls and columns. Column lines may not represent the centerline of walls and columns.

Containment, when this term is used as “the containment,” means the containment vessel or, as it is sometimes referred to, the prestressed concrete containment vessel.

Design commitment means that portion of the design description that is verified by ITAAC.

Design description means that portion of the design that is certified.

Design plant grade means the elevation of the soil around the nuclear island assumed in the design (i.e., “plant grade” or “finished grade level”) in relation to plant structures to which other plant elevations are correlated and which is set at 2’-7”.

Division (for electrical systems or equipment) is the designation applied to a given safety-related system (or set of components) that is (are) physically, electrically, and functionally independent from other redundant sets of components.

Division (for mechanical systems or equipment) is the designation applied to a specific set of safety-related components within a system.

Exists, when this term is used in the acceptance criteria, means that the item is present and consistent with the design description.

Functional arrangement (for a system) means the physical arrangement of systems and components to provide the function for which the system is intended as described in the ITAAC design description and shown in the specified figures.

Harsh environment means the limiting environmental conditions resulting from a design basis accident.

Inspection means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare the SSC condition to one or more design commitments. Examples include walkdowns, configuration checks, measurements of dimensions, or nondestructive examinations.

Operate means the actuation and running of the equipment.

under internal pressures that are experienced during service. The appropriate inspection, test, and analysis is typically stated as follows: a hydrostatic test will be conducted on those components of the system required to be hydrostatically tested by the ASME Code, and preoperational non-destructive examination will be conducted on those components of the system for which inspections are required by the ASME Code. The acceptance criterion is typically that the results of the hydrostatic test of the ASME Code components of the system conform to the requirements in Section III of the ASME Code.

- The design commitment on welding typically states that pressure boundary welds associated with the ASME Code components of the system meet the requirements of Section III of the ASME Code. The appropriate inspection, test, and analysis are usually non-destructive tests of the as-built pressure boundary welds as specified in Section III of the ASME Code. The acceptance criteria indicate that the ASME Code Section III requirements are met for non-destructive examination of the as-built pressure boundary welds.
- The design commitment on seismic qualification typically states that the specified seismic Category I components can withstand design basis seismic loads and continue to serve their safety function. Type tests and/or analyses of the seismic Category I components are specified to verify the design commitment. The acceptance criteria typically indicate that the results of the type tests and/or analyses concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
- The design commitment on environmental qualification typically indicates that the specified components can maintain functional operability under all service conditions, including design basis accidents. The appropriate inspection, test, and analysis typically involve inspections of the components and the associated wiring, cabling, and terminations located in a harsh environment, along with type tests, ~~and/or~~ analyses, or a combination of type tests and analyses. The acceptance criteria indicates that the results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment as being qualified for a harsh environment can withstand the environmental conditions.
- The design commitment on motor-operated valves typically states that the specified valves open, close, or both open and close under differential pressure, fluid flow, and temperature conditions. The appropriate inspection, test, or analysis typically involves tests of the installed valves under system preoperational conditions. The acceptance criteria typically entail the appropriate operation upon receipt of the actuating signal.

1.4.5 Implementation of Inspections, Tests, and Analyses

Although ITAAC are identified separately for each design commitment, this practice does not mean that a separate inspection, test, or analysis is required for each design commitment. A single inspection, test, or analysis may suffice for verification of multiple design commitments.

Table 2.4.1-2 Reactor System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The reactor internals withstand flow-induced vibration.	9. The flow-induced vibration test will be performed to measure the vibration response in the pre-operational test on the first US-APWR unit, with associated pre-test and post-test inspections.	9. The results of the flow-induced vibration test show that the alternative stress is acceptably low in comparison with the limit for high cycle fatigue in the ASME code. No structural damage or change is observed in post-test inspections.
10. The Class 1E equipment identified in Table 2.4.1-1 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	10.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	10.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.4.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	10.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	10.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.4.1-1 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
11. The Class 1E equipment, identified in Table 2.4.1-1, is powered from their respective Class 1E division.	11. A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	11. The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.4.1-1 under test.
12. Separation is provided between the Class 1E divisions for the equipment identified in Table 2.4.1-1 as Class 1E/qualified and non-Class 1E divisions.	12. Inspections of the as-built Class 1E divisional cables will be performed.	12. Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

	8.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.4.2-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	8.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.2-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.

Table 2.4.2-5 Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.a The Class 1E equipment identified in Table 2.4.2-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	9.a.i Type tests~~, and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	9.a.i The results of the type tests~~, and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.4.2-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	9.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	9.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.4.2-2 as being qualified for a harsh environment are bounded by type tests~~, and/or~~ analyses, or a combination of type tests and analyses.
9.b The Class 1E equipment, identified in Table 2.4.2-2, is powered from their respective Class 1E division.	9.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	9.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.4.2-2 under test.
9.c Separation is provided between RCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.	9.c Inspections of the as-built Class 1E divisional cables will be performed	9.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 2.4.2-5 Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 7 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. RSC alarms displays and controls are identified in Table 2.4.2-4.	15.i Inspections of the as-built RSC alarms, displays and controls will be performed.	15.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.4.2-4.
	15.ii Tests of the as-built RSC controls will be performed.	15.ii Controls exist to operate each as-built RSC control function identified in Table 2.4.2-4.
16. Each of the as-built piping identified in Table 2.4.2-3 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the piping.	16. Inspections of the as-built piping will be performed based on the evaluation report for LBB or the protection from dynamic effects of a pipe break, as specified in Section 2.3.	16. The LBB acceptance criteria are met by the as-built piping and piping materials, or the protection is provided for the dynamic effects of the piping break.
17. Controls exist in the MCR to start and stop the pressurizer heaters identified in Table 2.4.2-4.	17. Tests will be performed on the as-built pressurizer heaters listed in Table 2.4.2-4 using controls in the as-built MCR.	17. Controls exist in the as-built MCR to start and stop the as-built pressurizer heaters identified in Table 2.4.2-4.

Table 2.4.4-5 Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 10)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b Each of the seismic Category I piping, including supports, identified in Table 2.4.4-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.4.4-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.4-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.4.4-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.4-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.4.4-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests,~~ and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests,~~ and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.4.4-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.4.4-2 as being qualified for a harsh environment are bounded by type tests,~~ and/or~~ analyses, or a combination of type tests and analyses.

Table 2.4.4-5 Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 10 of 10)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. MCR alarms and displays of the parameters identified in Table 2.4.4-4 can be retrieved in the MCR.	11. Inspections will be performed for retrievability of the ECCS parameters in the as-built MCR.	11. MCR alarms and displays identified in Table 2.4.4-4 can be retrieved in the as-built MCR.
12. RSC alarms, displays and controls are identified in Table 2.4.4-4.	12.i Inspections of the as-built RSC alarms, displays and controls will be performed.	12.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.4.4-4.
	12.ii Tests of the as-built RSC controls will be performed.	12.ii Controls exist to operate each as-built RSC control function identified in Table 2.4.4-4.
13. Each of the as-built piping identified in Table 2.4.4-3 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	13. Inspections of the as-built piping will be performed based on the evaluation report for LBB or the protection from dynamic effects of a pipe break, as specified in Section 2.3.	13. The LBB acceptance criteria are met by the as-built piping and pipe materials, or the protection is provided for the dynamic effects of the piping break.
14.a The materials of construction of the ASME Code Section III, Class 1 components, identified in Table 2.4.4-2, are in accordance with ASME Code requirements.	14.a Inspection of the certified material test reports will be performed.	14.a The materials of construction of the ASME Code Section III, Class 1 components identified in Table 2.4.4-2 conform to the requirements of the ASME Code.
14.b The materials of construction of the ASME Code Section III, Class 1 piping, identified in Table 2.4.4-3, are in accordance with ASME Code requirements.	14.b Inspection of the certified material test reports will be performed.	14.b The materials of construction of the ASME Code Section III, Class 1 piping identified in Table 2.4.4-3 conform to the requirements of the ASME Code.

Table 2.4.5-5 Residual Heat Removal System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.4.5-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.5-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.4.5-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests,~~ and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests,~~ and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.4.5-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.4.5-2 as being qualified for a harsh environment are bounded by type tests,~~ and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table 2.4.5-2, is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.4.5-2 under test.
6.c Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 2.4.5-5 Residual Heat Removal System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 8 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. RSC alarms, displays and controls are identified in Table 2.4.5-4.	13.i Inspections of the as built RSC alarms, displays and controls will be performed.	13.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.4.5-4.
	13.ii Tests of the as-built RSC controls will be performed.	13.ii Controls exist to operate each as-built RSC control function identified in Table 2.4.5-4.
14. Each of the as-built piping identified in Table 2.4.5-3 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	14. Inspections of the as-built piping will be performed based on the evaluation report for the LBB or the protection from dynamic effects of a pipe break, as specified in Section 2.3.	14. The LBB acceptance criteria are met by the as-built piping and pipe materials, or protection is provided for the dynamic effects of the piping break.
15.a The materials of construction of the ASME Code Section III, Class 1 components, identified in Table 2.4.5-2, are in accordance with ASME Code requirements.	15.a Inspection of the certified material test reports will be performed.	15.a The materials of construction of the ASME Code Section III, Class 1 components identified in Table 2.4.5-2 conform to the requirements of the ASME Code.
15.b The materials of construction of the ASME Code Section III, Class 1 piping, identified in Table 2.4.5-3, are in accordance with ASME Code requirements.	15.b Inspection of the certified material test reports will be performed.	15.b The materials of construction of the ASME Code Section III, Class 1 piping identified in Table 2.4.5-3 conform to the requirements of the ASME Code.

Table 2.4.6-5 Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a The seismic Category I equipment, identified in Table 2.4.6-2, is designed to withstand seismic design basis loads without loss of safety function.	5.a.i Inspections will be performed to verify that the seismic Category I as-built equipment and valves identified in Table 2.4.6-2 are located in the containment or reactor building.	5.a.i The as-built seismic Category I as-built equipment identified in Table 2.4.6-2 is located in the containment or reactor building.
	5.a.ii Type tests and/or analyses of the seismic Category I equipment will be performed.	5.a.ii The results of the type tests and/or analyses conclude that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	5.a.iii An inspection will be performed on the as-built equipment including anchorage.	5.a.iii The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
5.b Each of the seismic Category I piping, including supports, identified in Table 2.4.6-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.4.6-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.6-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.4.6-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.4.6-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.

Table 2.4.6-5 Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.a The Class 1E equipment identified in Table 2.4.6-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.4.6-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.4.6-2 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table 2.4.6-2, is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.4.6-2 under test.
6.c Separation is provided between CVCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
7. Deleted.	7. Deleted.	7. Deleted.
8.a The CVCS provides makeup capability to maintain the RCS volume.	8.a A test of the as-built CVCS will be performed to measure the makeup flow rate.	8.a Each as-built CVCS charging pump provides a flow rate of greater than or equal to 160 gpm.

Table 2.4.6-5 Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Controls exist in the MCR to start and stop the pumps identified in Table 2.4.6-4.	11. Tests will be performed on the as-built pumps in Table 2.4.6-4 using controls in the as-built MCR.	11. Controls exist in the as-built MCR to start and stop the as-built pumps listed in Table 2.4.6-4.
12. MCR alarms and displays of the parameters identified in Table 2.4.6-4 can be retrieved in the MCR.	12. Inspections will be performed for retrievability of the CVCS parameters in the as-built MCR.	12. MCR alarms and displays identified in Table 2.4.6-4 can be retrieved in the as-built MCR.
13. RSC alarms, displays and controls are identified in Table 2.4.6-4.	13.i Inspections of the as-built RSC alarms, displays and controls will be performed.	13.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.4.6-4.
	13.ii Tests of the as-built RSC controls will be performed.	13.ii Controls exist to operate each as-built RSC control function identified in Table 2.4.6-4.
14.a The materials of construction of the ASME Code Section III, Class 1 components, identified in Table 2.4.6-2, are in accordance with ASME Code requirements.	14.a Inspection of the certified material test reports will be performed.	14.a The materials of construction of the ASME Code Section III, Class 1 components identified in Table 2.4.6-2 conform to the requirements of the ASME Code.
14.b The materials of construction of the ASME Code Section III, Class 1 piping, identified in Table 2.4.6-3, are in accordance with ASME Code requirements.	14.b Inspection of the certified material test reports will be performed.	14.b The materials of construction of the ASME Code Section III, Class 1 piping identified in Table 2.4.6-3 conform to the requirements of the ASME Code.

Table 2.5.1-6 RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The Class 1E equipment identified in Table 2.5.1-1 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	6.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.5.1-1 as being qualified for a harsh environment can withstand the environmental conditions.
	6.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.5.1-1 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
7. The RPS, ESFAS, SLS, safety VDU processor, and safety VDU are qualified to meet the electromagnetic conditions that would exist before, during, and following a design basis accident, with respect to its location in the facility, without loss of safety function for the time required to perform the safety function.	7. Type tests and/or analyses will be performed on the equipment.	7. A report exists and concludes that the RPS, ESFAS, SLS, safety VDU processor, and safety VDU are qualified to meet the electromagnetic conditions that would exist before, during, and following a design basis accident, with respect to its location in the facility, without loss of safety function for the time required to perform the safety function.
8. The Class 1E equipment listed in Table 2.5.1-1 is located in a facility area that provides protection from natural phenomena hazards such as tornadoes, and accident related hazards such as missiles, pipe breaks and flooding.	8. An inspection of the as-built equipment location will be performed.	8. The as-built equipment listed in Table 2.5.1-1 is located in a plant area that provides protection from natural phenomena hazards such as tornadoes, and accident related hazards such as missiles, pipe breaks and flooding.
9. The Class 1E equipment listed in Table 2.5.1-1 is powered from two safety related power sources: the first source is its respective Class 1E division and the second source is from another division to ensure reliable power to each division of the PSMS.	9. Inspection of the as-built equipment will be performed.	9. The Class 1E equipment listed in Table 2.5.1-1 is powered from two safety related power sources: the first source is its respective Class 1E division and the second source is from another division to ensure reliable power to each division of the PSMS.

Table 2.5.1-6 RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 5 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
17.a The PSMS is designed to facilitate the timely recognition, location, replacement, repair and adjustment of malfunctioning components or modules.	17.a An inspection of the as-built PSMS will be performed.	17a. The as-built PSMS is designed to facilitate the timely recognition, location, replacement, repair and adjustment of malfunctioning components or modules.
17. b A single channel or division of the PSMS can be bypassed to allow on-line testing, maintenance or repair without impeding the safety function.	17. b Tests will be performed to confirm the as-built channel or division bypass capabilities and to confirm the function of the bypass interlock logic.	17. b A single channel or division of the as-built PSMS can be bypassed to allow on-line testing, maintenance or repair without impeding the safety function.
18. The PSMS automatically removes operating bypasses when permissive conditions are not met.	18. A test of the as-built PSMS will be performed.	18. The as-built PSMS automatically removes operating bypasses when permissive conditions are not met.
19. The PSMS setpoints are determined using a methodology based on proven nuclear industry standards. This methodology provides allowance for uncertainties between analytical limits and device setpoints.	19. An inspection will be performed to define the as-built PSMS setpoints in accordance with the acceptable methodology.	19. The as-built PSMS setpoints are determined using the acceptable methodology, which provides allowance for uncertainties between analytical limits and device setpoints based on proven nuclear industry standards.
20. ~~Each division of the PSMS and field equipment listed in Table 2.5.1-1 is supplied from two safety-related Class 1E power sources. Either power source is sufficient to power each division of the PSMS.~~Deleted.	20. ~~A test of the as-built equipment will be performed.~~Deleted.	20. ~~Each division of the as-built PSMS and field equipment listed in Table 2.5.1-1 is supplied from two safety-related Class 1E power sources. Either power source is sufficient to power each division of the as-built PSMS.~~ Deleted.
21. The PSMS logic is designed to fail to a safe state such that loss of electrical power to a division of PSMS results in a reactor trip condition for that division. Loss of electrical power does not result in ESF actuation.	21. A test will be performed by disconnecting the electrical power to each division of the as-built PSMS.	21. Each division of the as-built PSMS will fail to a safe state upon loss of electrical power to the division (i.e., results in a reactor trip condition for that division), and loss of electric power does not result in ESF actuation.

Table 2.5.1-6 RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 7 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
26. A signal selector algorithm (SSA) is provided in the PCMS for the monitoring variables as listed in Table 2.5.1-5 to ensure the PCMS does not take an erroneous control action that results in a condition which requires RT or ESF action to consider a single instrument channel failure or a single RPS train failure.	26. An inspection of the as-built SSA functional arrangement will be performed.	26. The as-built PSMS and PCMS conform to the functional arrangement of the SSA functions as described in the design description and Table 2.5.1-5.
27. Input sensors from each division of the PSMS are compared continuously in the PCMS to detect abnormal deviations for checking the operational availability of each division of the PSMS input sensor that may be required for a safety function during reactor operation.	27. An inspection of the as-built PSMS and PCMS functions will be performed.	27. The input sensors from each division of the as-built PSMS are compared continuously in the as-built PCMS to detect abnormal deviations.
28. The spatially dependent sensors that are required for protective actions are identified in Table 2.5.1-2 and Table 2.5.1-3.	28. An inspection of the as-built spatially dependent sensors required for protective actions will be performed.	28. The as-built PSMS includes the minimum number and locations of spatially dependent sensors that are required for protective actions as identified in Table 2.5.1-2 and Table 2.5.1-3.
29a. ESF systems are automatically initiated from signals that originate in the RPS.	29a. A test of the as-built PSMS will be performed.	29a. As-built ESF systems are automatically initiated from signals that originate in the as-built RPS.
29b. Manual actuation of ESF systems is carried out through a diverse signal path that bypasses the RPS.	29b. A test of the as-built PSMS will be performed.	29b. Manual actuation of the as-built ESF systems is carried out through a diverse signal path that bypasses the as-built RPS.

Table 2.5.4-2 Information Systems Important to Safety Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Information systems important to safety (PAM, BISI, alarms, SPDS) are appropriately displayed and alarmed in the MCR, RSR, TSC and EOF, as appropriate.	1. A test will be performed to demonstrate alarm, display and control capabilities for information systems important to safety.	1. The as-built information systems important to safety (PAM, BISI, alarms, SPDS) are appropriately displayed and alarmed in the MCR, RSR, TSC and EOF, as appropriate.
2. Information and controls for credited manual operator actions are provided in the MCR.	2. A test of the as-built PSMS and PCMS will be performed.	2. The information and controls for credited manual operator actions are provided in the as-built MCR.
3. The field instrumentation for the PAM variables identified in Table 2.5.4-1 and that is subjected to~~as being qualified for~~ a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	3.i Type tests,~~ and/or~~ analyses, or a combination of type tests and analyses will be performed on the field instrumentation located in a harsh environment.	3.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the field instrumentation for the PAM variables identified in Table 2.5.4-1 and that is subjected to ~~as being qualified for~~ a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	3.ii Inspections will be performed on the as-built field instrumentation and the associated wiring, cables, and terminations located in a harsh environment.	3.ii The as-built field instrumentation and the associated wiring, cables, and terminations identified in Table 2.5.4-1 and that is subjected to~~as being qualified for~~ a harsh environment are bounded by type tests,~~ and/or~~ analyses, or a combination of type tests and analyses.
4. The functional arrangement of the information systems important to safety is as described in the Design Description and as shown in Figure 2.5.4-1	4. An inspection of the as-built information systems important to safety will be performed.	4. The as-built information systems important to safety conform to the functional arrangement as described in the Design Description and as shown in Figure 2.5.4-1.

Alarms are provided in the MCR for low fuel oil level in the fuel oil storage tanks and low and high level in the fuel oil day tanks.

System logic involves the fuel oil transfer pump starting automatically on a fuel oil day tank low level signal and stopping automatically on a fuel oil day tank high-level signal. There are no system interlocks.

Each fuel oil transfer pump is powered from its respective Class 1E division. Separation is provided between Class 1E divisions and between Class 1E divisions and the non-Class 1E division.

If a safety-related mechanical component in the EPS support systems is not designed to ASME Code Section III, then quality of the component is demonstrated and documented (e.g. seismic design, testing and qualification).

The stored air starting system is capable of providing starting air to each of the four Class 1E EPSs without requiring replenishment.

The safety-related portions of starting air system components are designed to seismic Category I standards. These portions are designed to meet the requirements of the ASME Code. Section III.

Alarms are provided in the MCR for low pressure in air receivers.

Each lubrication oil tank provides a seven day supply of lube oil to its respective Class 1E EPS.

Lubrication oil is circulated by main shaft driven pump during EPS operation.

Alarms are provided in the MCR for low pressure and high temperature of lubrication oil system.

The Class 1E GTG combustion air intake and exhaust system is capable of supplying an adequate quantity of combustion air to the GT and of disposing the exhaust gases without creating an excessive backpressure on the GT when operating at 110% of nameplate rating. The Class 1E GTG ventilation/cooling air intake and exhaust system is capable of cooling the GT including operation at 110% of nameplate rating. The turbine intake and exhaust and ventilation/cooling air intake and exhaust openings are above the roof of the power source buildings (PS/B), and the portion of the piping/ducts above the roof is protected by a guard structure against precipitation and tornado missiles.

2.6.4.3 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.4-1 describes the ITAAC for the Class 1E EPS and the FOS systems.

Table 2.6.8-1 Containment Electrical Penetration Assemblies Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 2)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The back up circuit protection device for each EPA circuit is sized to ensure mechanical integrity of the EPA for postulated overload and short-circuit conditions, during normal and accident conditions.	6.i An analysis will be performed to verify the back up circuit protection device for each EPA circuit is sized to ensure mechanical integrity of the EPA for postulated overload and short-circuit conditions, during normal and accident conditions.	6.i The back up circuit protection device for each EPA circuit is sized to ensure mechanical integrity of the EPA for postulated overload and short-circuit conditions, during normal and accident conditions.
	6.ii An inspection will be performed to verify ratings of the back-up circuit protection device for each as-built EPA circuit bound the requirements of the analysis.	6.ii The ratings of the back-up circuit protection device for each as-built EPA circuit bound the requirements of the analysis.
7. Each EPA as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	7.i Type tests,~~ and/or~~ analyses, or a combination of type tests and analyses will be performed on the EPAs located in a harsh environment.	7.i The results of the type tests,~~ and/or~~ analyses, or a combination of type tests and analyses conclude that each EPA as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	7.ii Inspection will be performed on each as-built EPA located in a harsh environment.	7.ii Each as-built EPA as being qualified for a harsh environment is bounded by type tests,~~ and/or~~ analyses, or a combination of type tests and analyses.

Table 2.7.1.2-5 Main Steam Supply System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.2-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.2-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.7.1.2-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.1.2-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.1.2-2 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table 2.7.1.2-2, is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.1.2-2 under test.
6.c Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 2.7.1.2-5 Main Steam Supply System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.d After loss of motive power, the remotely operated valves, identified in Table 2.7.1.2-2, assume the indicated loss of motive power position.	9.d Tests of the as-built valves will be performed under the conditions of loss of motive power.	9.d Upon loss of motive power, each as-built remotely operated valve identified in Table 2.7.1.2-2 assumes the indicated loss of motive power position.
9.e The MSIVs identified in Table 2.7.1.2-2 perform an active safety function to change position as indicated in the table.	9.e.i Tests or type tests of the MSIVs will be performed to demonstrate the capability of the valve to operate under its design conditions.	9.e.i Each MSIV changes position as indicated in Table 2.7.1.2-2 under design conditions.
	9.e.ii Tests of the as-built MSIVs will be performed under pre-operational flow, differential pressure, and temperature conditions.	9.e.ii Each as-built MSIV changes position as indicated in Table 2.7.1.2-2 under pre-operational test conditions.
10. MCR alarms and displays of the parameters identified in Table 2.7.1.2-4 can be retrieved in the MCR.	10. Inspections will be performed for retrievability of the MSS parameters in the as-built MCR.	10. The MCR alarms and displays identified in Table 2.7.1.2-4 can be retrieved in the as-built MCR.
11. RSC alarms, displays, and controls are identified in Table 2.7.1.2-4.	11.i Inspections of the as-built RSC alarms, displays and controls will be performed.	11.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.1.2-4.
	11.ii Tests of the as-built RSC controls will be performed.	11.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.1.2-4.
12. Each of the as-built piping identified in Table 2.7.1.2-3 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	12. Inspections of the as-built piping will be performed based on the evaluation report for LBB or the protection from dynamic effects of a pipe break, as specified in Section 2.3.	12. The LBB acceptance criteria are met by the as-built piping and pipe materials, or the protection is provided for the dynamic effects of the piping break.

Table 2.7.1.9-5 Condensate and Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b Each of the seismic Category I piping, including supports, identified in Table 2.7.1.9-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.9-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.9-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed ~~for the existence of a report verifying~~to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.9-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.9-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.7.1.9-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.1.9-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.1.9-2 as being qualified for a harsh environment are bounded by type tests ~~and/or~~, analyses, or a combination of type tests and analyses.

Table 2.7.1.9-5 Condensate and Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.a The valves, identified in Table 2.7.1.9-2 perform an active safety function to change position as indicated in the table.	9.a.i Tests or type tests of air operated valves and MFIVs will be performed that demonstrate the capability of the valve to operate under its design conditions.	9.a.i Each air operated valve and each MFIV changes position as indicated in Table 2.7.1.9-2 under design condition.
	9.a.ii Tests of the as-built air operated valves and MFIVs will be performed under pre-operational flow, differential pressure, and temperature conditions.	9.a.ii Each as-built air operated valve and each as-built MFIV changes position as indicated in Table 2.7.1.9-2 under the pre-operational test conditions.
	9.a.iii Tests of the as-built check valves with active safety functions identified in Table 2.7.1.9-2 will be performed under preoperational test pressure, temperature, and fluid flow conditions.	9.a.iii Each as-built check valve changes position as indicated in Table 2.7.1.9-2.
9.b After loss of motive power, the remotely operated valves, identified in Table 2.7.1.9-2, assume the indicated loss of motive power position.	9.b Tests of the as-built valves will be performed under the conditions of loss of motive power.	9.b Upon loss of motive power, each as-built remotely operated valves identified in Table 2.7.1.9-2 assumes the indicated loss of motive power position.
10. MCR alarms and displays of the parameters identified in Table 2.7.1.9-4 can be retrieved in the MCR.	10. Inspections will be performed for retrievability of the CFS parameters in the as-built MCR.	10. MCR alarms and displays identified in Table 2.7.1.9-4 can be retrieved in the as-built MCR.
11. RSC alarms, displays and controls are identified in Table 2.7.1.9-4.	11.i Inspections of the as-built RSC alarms displays and controls will be performed.	11.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.1.9-4.
	11.ii Tests of the as-built RSC controls will be performed.	11.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.1.9-4.

Table 2.7.1.10-4 Steam Generator Blowdown System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b Each of the seismic Category I piping, including supports, identified in Table 2.7.1.10-2 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.10-2 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.10-2 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.10-2 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.10-2 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6. The Class 1E equipment, identified in Table 2.7.1.10-1, is powered from their respective Class 1E division.	6. A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6. The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.1.10-1 under test.
7. Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	7. Inspections of the as-built Class 1E divisional cables will be performed.	7. Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
8. After loss of motive power, the air-operated valves, identified in Table 2.7.1.10-1, assume the indicated loss of motive power position.	8. Tests of the as-built valves will be performed under the conditions of loss of motive power.	8. Upon loss of motive power, each as-built remotely operated valves identified in Table 2.7.1.10-1 assumes the indicated loss of motive power position.
9. Each mechanical division of the SGBDS except for piping (Division A&B and C&D pairs) is physically separated from the other divisions with the exception of reactor building exterior and inside the containment.	9. Inspections of the as-built SGBDS will be performed.	9. Each mechanical division of the as-built SGBDS except for piping is physically separated from other mechanical divisions of the as-built SGBDS by structural barriers with the exception of reactor building exterior and inside the containment.
10. MCR alarms and displays of the parameters identified in Table 2.7.1.10-3 can be retrieved in the MCR.	10. Inspections will be performed for retrievability of the SGBDS parameters in the as-built MCR.	10. The MCR alarms and displays identified in Table 2.7.1.10-3 can be retrieved in the as-built MCR.
11. RSC alarms, displays and controls are identified in Table 2.7.1.10-3.	11.i Inspections of the as-built RSC alarms, displays and controls will be performed.	11.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.1.10-3.

Table 2.7.1.10-4 Steam Generator Blowdown System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 5 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	11.ii Tests of the as-built RSC controls will be performed.	11.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.1.10-3.
12. The Class 1E equipment identified in Table 2.7.1.10-1 as being qualified for a harsh environment are designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	12.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	12.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.1.10-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	12.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	12.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.1.10-1 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
13.a Controls exist in the MCR to open and close the remotely operated valves identified in Table 2.7.1.10-3.	13.a Tests will be performed on the as-built remotely operated valves listed in Table 2.7.1.10-3 using controls in the as-built MCR.	13.a Controls exist in the as-built MCR to open and close the as-built remotely operated valves listed in Table 2.7.1.10-3.
13.b The valves identified in Table 2.7.1.10-1 as having PSMS control perform an active safety function after receiving a signal from PSMS.	13.b Test will be performed on the as-built remotely operated valves listed in Table 2.7.1.10-1 using simulated signals.	13b The as-built remotely operated valves identified in Table 2.7.1.10-1 as having PSMS control perform the active safety function identified in the table after receiving a simulated signal.

Table 2.7.1.11-5 Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.a.iii Inspections will be performed on the as-built equipment including anchorage.	5.a.iii The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
5.b Each of the seismic Category I piping, including supports, identified in Table 2.7.1.11-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.11-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.11-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.1.11-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.1.11-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.7.1.11-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests,~~ and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.1.11-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.1.11-2 as being qualified for a harsh environment are bounded by type tests,~~ and/or~~ analyses, or a combination of type tests and analyses.

Table 2.7.1.11-5 Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.a The motor-operated valves and check valves, identified in Table 2.7.1.11-2, perform an active safety function to change position as indicated in the table.	9.a.i Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	9.a.i Each motor-operated valve changes position as indicated in Table 2.7.1.11-2 under design conditions.
	9.a.ii Tests of the as-built motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	9.a.ii Each as-built motor-operated valve changes position as indicated in Table 2.7.1.11-2 under pre-operational test conditions.
	9.a.iii Tests of the as-built check valves with active safety functions identified in Table 2.7.1.2-2 will be performed under pre-operational test pressure, temperature, and fluid flow conditions.	9.a.iii Each as-built check valve changes position as indicated in Table 2.7.1.11-2.
9.b After loss of motive power, the remotely operated valves, identified in Table 2.7.1.11-2, assume the indicated loss of motive power position.	9.b. Tests of the as-built valves will be performed under the conditions of loss of motive power.	9.b Upon loss of motive power, each as-built remotely operated valve identified in Table 2.7.1.11-2 assumes the indicated loss of motive power position.
10. MCR alarms and displays of the parameters identified in Table 2.7.1.11-4 can be retrieved in the MCR.	10. Inspections will be performed for retrievability of the EFWS parameters in the as-built MCR.	10. MCR alarms and displays identified in Table 2.7.1.11-4 can be retrieved in the as-built MCR.
11. RSC alarms, displays and controls are identified in Table 2.7.1.11-4.	11.i Inspections of the as-built RSC alarms, displays and controls will be performed.	11.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.1.11-4.
	11.ii Tests of the as-built RSC controls will be performed.	11.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.1.11-4.
12. Each EFW pump delivers at least the minimum flow required for removal of core decay heat using the SGs against a SG pressure up to the set pressure of the first stage of main steam safety valve plus 3 percent.	12 A test of each as-built EFW pump will be performed to determine system flow vs. SG pressure under preoperational condition. Analyses will be performed to convert the test results to the design conditions.	12 From the result of analyses, any two of the as-built EFW pumps deliver at least 705 gpm to ~~the~~ any two of the four~~two~~ SGs against a SG pressure up to the set pressure of the first stage of main steam safety valve plus 3 percent.

Table 2.7.3.1-5 Essential Service Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 5)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.b The ASME Code Section III piping, identified in Table 2.7.3.1-3, retains its pressure boundary integrity at its design pressure.	4.b A hydrostatic test will be performed on the as-built piping required by the ASME Code Section III to be hydrostatically tested.	4.b The results of the hydrostatic test of the as-built piping identified in Table2.7.3.1-2 as ASME Code Section III conform to the requirements of the ASME Code Section III.
5.a The seismic Category I equipment identified in Table 2.7.3.1-2 is designed to withstand seismic design basis loads without loss of safety function.	5.a.i Inspections will be performed to verify that the seismic Category I as-built equipment identified in Table 2.7.3.1-2 is installed in the location identified in Table 2.7.3.1-1.	5.a.i The seismic Category I as-built equipment identified in Table 2.7.3.1-2 is installed in the location identified in Table 2.7.3.1-1.
	5.a.ii Type tests and/or analyses of the seismic Category I equipment will be performed.	5.a.ii The results of the type tests and/or analyses conclude that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	5.a.iii Inspections will be performed on the as-built equipment including anchorage.	5.a.iii The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
5.b Each of the seismic Category I piping, including supports, identified in Table 2.7.3.1-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.3.1-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.3.1-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.3.1-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.3.1-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.7.3.1-2 is powered from their respective Class 1E division.	6.a A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.a The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.3.1-2 under test.

Table 2.7.3.1-5 Essential Service Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 5)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.b Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.b Inspections of the as-built Class 1E divisional cables will be performed.	6.b Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
7. The ESWS components identified in Table 2.7.3.1-2 provide adequate cooling water to the CCW heat exchangers and the essential chiller units of the ECWS during all plant operating conditions, including normal plant operating, abnormal and accident conditions.	7.i An inspection for the existence of a report that determines the heat removal capability of the as-built ESWS will be performed.	7.i A report exists and concludes that the heat removal capability of the as-built ~~ESWS provides adequate cooling water to the~~ CCW heat exchangers and the essential chiller units of the ECWS is greater than or equal to the design value for~~during~~ all plant operating conditions, including normal plant operating, abnormal and accident conditions.
	7.ii Tests will be performed to confirm that the as-built ESWS pumps can provide flow to the CCW heat exchangers and the essential chiller units of the ECWS.	7.ii The as-built ESWS pumps identified in Table 2.7.3.1-2 are capable of achieving their design flow rate.
8. Controls exist in the MCR to open and close the remotely operated valves identified in Table 2.7.3.1-2.	8. Tests will be performed on the as-built remotely operated valves listed in Table 2.7.3.1-2 using controls in the as-built MCR.	8. Controls exist in the as-built MCR to open and close the as-built remotely operated valves listed in Table 2.7.3.1-2.
9.a The remotely operated and check valves, identified in Table 2.7.3.1-2, perform an active safety function to change position as indicated in the table.	9.a.i Tests or type tests of the remotely operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	9.a.i Each remotely operated valve changes position as indicated in Table 2.7.3.1-2 under design conditions.
	9.a.ii Tests of the as-built remotely operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	9.a.ii Each as-built remotely operated valve changes position as indicated in Table 2.7.3.1-2 under pre-operational test conditions.

Table 2.7.3.1-5 Essential Service Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 5 of 5)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	9.a.iii Tests of the as-built check valves will be performed under preoperational flow, differential pressure, and temperature conditions.	9.a.iii Each as-built check valve changes position as indicated in Table 2.7.3.1-2.
9.b Upon the receipt of an ESWP start signal, the essential service water discharge valve opens automatically. Each pump's discharge valve is interlocked to close when the pump is not running or is tripped. The valve starts to open after the respective pump starts.	9.b A test of each as-built interlock for the essential service water discharge valve will be performed using a simulated test signal.	9.b The ESW discharge valve closes when its respective pump is not running. Upon the receipt of a simulated ESWP start signal, the as-built discharge valve for the respective pump starts to open automatically after the pump starts. The valve closes when the pump is tripped.
9.c After loss of motive power, the remotely operated valves, identified in Table 2.7.3.1-2, assume the indicated loss of motive power position.	9.c Tests of the as-built valves will be performed under the conditions of loss of motive power.	9.c Upon loss of motive power, each as-built remotely operated valve identified in Table 2.7.3.1-2 assumes the indicated loss of motive power position.
10.a Controls exist in the MCR to start and stop the pumps identified in Table 2.7.3.1-4.	10.a Tests will be performed on the as-built pumps listed in Table 2.7.3.1-4 using controls in the as-built MCR.	10.a Controls exist in the as-built MCR to start and stop the as-built pumps listed in Table 2.7.3.1-4.
10.b The pumps identified in Table 2.7.3.1-2 as having PSMS control perform an active safety function after receiving a signal from PSMS.	10.b Tests will be performed on the as-built pumps listed in Table 2.7.3.1-2 using simulated signals.	10.b The as-built pumps identified in Table 2.7.3.1-2 as having PSMS control perform the active safety function identified in the table after receiving a simulated signal.
11. MCR alarms and displays of the parameters identified in Table 2.7.3.1-4 can be retrieved in the MCR.	11. Inspections will be performed for retrievability of the ESWS parameters in the as-built MCR.	11. The MCR alarms and displays identified in Table 2.7.3.1-4 can be retrieved in the as-built MCR.
12. RSC alarms, displays, and controls are identified in Table 2.7.3.1-4.	12.i Inspections of the as-built RSC alarms, displays and controls will be performed.	12.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.3.1-4.
	12.ii Tests of the as-built RSC controls will be performed.	12.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.3.1-4.

Table 2.7.3.3-5 Component Cooling Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.3.3-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.3.3-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The applicable Class 1E equipment identified in Table 2.7.3.3-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.3.3-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii An inspection will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.3.3-2, as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment identified in Table 2.7.3.3-2 is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.3.3-2 under test.
6.c Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be performed.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 2.7.3.3-5 Component Cooling Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.b After loss of motive power, the remotely operated valves, identified in Table 2.7.3.3-2, assume the indicated loss of motive power position.	9.b Tests of the as-built valves will be performed under the conditions of loss of motive power.	9.b Upon loss of motive power, each as-built remotely operated valve identified in Table 2.7.3.3-2 assumes the indicated loss of motive power position.
10.a Controls exist in the MCR to start and stop the pumps identified in Table 2.7.3.3-4.	10.a Tests will be performed on the as-built pumps listed in Table 2.7.3.3-4 using controls in the as-built MCR.	10.a Controls exist in the as-built MCR to start and stop the as-built pumps listed in Table 2.7.3.3-4.
10.b The pumps identified in Table 2.7.3.3-2 as having PSMS control perform an active safety function after receiving a signal from PSMS.	10.b Test will be performed on the as-built pumps listed in Table 2.7.3.3-2 using simulated signals.	10.b The as-built pumps identified in Table 2.7.3.3-2 as having PSMS control perform the active safety function identified in the table after receiving a simulated signal.
11. MCR alarms and displays of the parameters identified in Table 2.7.3.3-4 can be retrieved in the MCR.	11. Inspections will be performed for retrievability of the CCWS parameters in the as-built MCR.	11. The MCR alarms and displays identified in Table 2.7.3.3-4 can be retrieved in the as-built MCR.
12. RSC alarms, displays and controls are identified in Table 2.7.3.3-4.	12.i Inspections of the as-built RSC alarms, displays and controls will be performed.	12.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.3.3-4.
	12.ii Tests of the as-built RSC controls will be performed.	12.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.3.3-4.
13. The CCW pumps have sufficient net positive suction head (NPSH).	13. Tests to measure the as-built CCW pump suction pressure will be performed. Inspections and analyses to determine NPSH available to each pump will be performed.	13. The as-built system meets the design, and the analysis confirms that the NPSH available exceeds the required NPSH.

The ECWS provides chilled water required for the safety-related HVAC systems during all plant conditions, including normal plant operations, abnormal and accident conditions.

Alarms, Displays, and Controls

Table 2.7.3.5-4 identified alarms, displays, and controls associated with the ECWS that are located in the MCR.

Logic

Upon receipt of the ECCS actuation signal, the ECWS automatically starts or, if in operation, continue to operate. The ECWS automatically starts in case of loss of off-site power.

Interlocks

The starting of the essential chilled water pumps and the detection of the ESWS flows are a prerequisite for the chiller unit startup.

Class 1E Electrical Power Sources and Divisions

The ECWS equipment identified in Table 2.7.3.5-2 as Class 1E is powered from their respective Class 1E divisions, and separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

Equipment to be Qualified for Harsh Environments

Not applicable.~~The equipment identified in Table 2.7.3.5-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function.~~

Interface Requirements

There are no safety-related interfaces with systems outside of the certified design.

Numeric Performance Values

When necessary to demonstrate satisfaction of a design commitment, numeric performance values for selected components have been specified as ITTAC acceptance criteria in Table 2.7.3.5-5.

2.7.3.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.3.5-5 describes the ITAAC for the ECWS.

Table 2.7.3.5-5 Essential Chilled Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.3.5-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.3.5-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment, identified in Table 2.7.3.5-2, is powered from their respective Class 1E division.	6.a A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.a The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.3.5-2 under test.
6.b Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.b Inspections of the as-built Class 1E divisional cables will be performed.	6.b Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
7. The ECWS components identified in Table 2.7.3.5-2 remove heat from various cooling coils during all plant operating conditions, including normal plant operating, abnormal and accident conditions.	7.i An inspection for the existence of a report that determines the heat removal capability of the as-built ECWS will be performed.	7.i A report exists and concludes that the heat removal capability of the as-built ECWS is greater than or equal to the design values for all plant operating conditions, including normal plant operating, abnormal and accident conditions.
	7.ii Tests will be performed to confirm that the as-built ECWS pumps identified in Table 2.7.3.5-2 provide flow to the ECWS cooling unit.	7.ii The as-built ECWS pumps identified in Table 2.7.3.5-2 are capable of achieving their design flow rate.
8. The remotely operated valves identified in Table 2.7.3.5-2 as having PSMS control perform an active safety function after receiving a signal from PSMS.	8. Test will be performed on the as-built remotely operated valves listed in Table 2.7.3.5-2 using simulated signals.	8. The as-built remotely operated valves identified in Table 2.7.3.5-2 as having PSMS control perform the active function identified in the table after receiving a simulated signal.

Table 2.7.3.5-5 Essential Chilled Water System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. MCR alarms and displays of the parameters identified in Table 2.7.3.5-4 can be retrieved in the MCR.	11. Inspections will be performed for retrievability of the ECWS parameters in the as-built MCR.	11. The MCR alarms and displays identified in Table 2.7.3.5-4 can be retrieved in the as-built MCR.
12. RSC alarms displays and controls are identified in Table 2.7.3.5-4.	12.i Inspections of the as-built RSC alarms, displays and controls will be performed	12.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.3.5-4.
	12.ii Tests of the as-built RSC controls will be performed.	12.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.3.5-4.
13. The ECWS pumps have sufficient net positive suction head (NPSH).	13. Tests to measure the as-built ECWS pump suction pressure will be performed. Inspections and analysis to determine NPSH available to each pump will be performed.	13. The as-built system meets the design, and the analysis confirms that the NPSH available exceeds the required NPSH.

Table 2.7.5.1-3 Main Control Room HVAC System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. MCR alarms and displays of the MCR HVAC system parameters identified in Table 2.7.5.1-2 can be retrieved in the MCR.	7. Inspections will be performed for retrievability of the MCR HVAC system parameters in the as-built MCR.	7. MCR alarms and displays, identified in Table 2.7.5.1-2, can be retrieved in the as-built MCR.
8. RSC alarms, displays and controls are identified in Table 2.7.5.1-2.	8.i Inspections of the as-built RSC alarms, displays and controls will be performed.	8.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.5.1-2.
	8.ii Tests of the as-built RSC controls will be performed.	8.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.5.1-2.

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.d The safeguard component area HVAC system, emergency feedwater pump area HVAC system, and the safety related component area HVAC system air handling unit fans identified in Table 2.7.5.2-1 start after receiving a high temperature signal.	6.d Tests of the as-built safeguard component area HVAC system, emergency feedwater pump area HVAC system, and the safety related component area HVAC system air handling unit fans will be performed using a simulated signal.	6.d The as-built safeguard component area HVAC system, emergency feedwater pump area HVAC system, and the safety related component area HVAC system air handling unit fans identified in Table 2.7.5.2-1 start after receiving a high temperature signal.
7. MCR alarms and displays of the ESFVS parameters identified in Table 2.7.5.2-2 can be retrieved in the MCR.	7. Inspections will be performed for retrievability of the ESFVS parameters in the as-built MCR.	7. MCR alarms and displays identified in Table 2.7.5.2-2 can be retrieved in the as-built MCR.
8. RSC alarms, displays and controls are identified in Table 2.7.5.2-2.	8.i Inspections of the as-built RSC alarms, displays and controls will be performed.	8.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.5.2-2.
	8.ii Tests of the as-built RSC controls will be performed.	8.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.5.2-2.

There is no important system operation.

Alarms, Displays, and Controls

There are no important alarms, displays, and controls.

Logic

There is no logic needed for direct safety functions related to the reactor cavity cooling system.

Interlocks

There are no interlocks needed for direct safety functions related to the reactor cavity cooling system.

Class 1E Electrical Power Sources and Divisions

Not applicable.

Equipment to be Qualified for Harsh Environments

Not applicable.

Interface Requirements

There are no safety-related interfaces with systems outside of the certified design.

Numeric Performance Values

Not applicable.

2.7.5.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.5.3-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the CVVS. ~~Table 2.11.2-2 specifies the ITAAC for the non-ECWS piping system and components that are part of the CIS that supplies cooling water to the containment fan cooler unit and CRDM cooling unit cooling coils.~~ The ITAAC associated with the ~~CVVS~~ equipment, components and piping of the CVVS and non-ECWS ~~and~~ that also comprise a portion of the CIS, are described in Table 2.11.2-2.

Table 2.7.5.3-1 Containment Ventilation System Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the CVVS is as described in the Design Description of this Subsection 2.7.5.3.1.	1. Inspections of the as-built CVVS will be performed.	1. The as-built CVVS conforms with the functional arrangement as described in Design Description of this Subsection 2.7.5.3.1.
2. Deleted.	2. Deleted.	2. Deleted.

2.7.5.4 Auxiliary Building Ventilation System (ABVS)

The ABVS is designed to provide proper environmental conditions throughout all areas of the reactor building, the power source building, the auxiliary building and the access building during normal plant operation.

The ABVS includes:

- Auxiliary building HVAC system
- Non-Class 1E electrical room HVAC system
- Main steam / feedwater piping area HVAC system
- Technical support center HVAC system

2.7.5.4.1 Design Description

2.7.5.4.1.1 Auxiliary Building HVAC System

System Purpose and Functions

The auxiliary building HVAC system is designed to provide conditioning air to maintain the proper environmental conditions for areas housing mechanical and electrical equipment (including area housing ESF equipment) in the reactor building, power source building, auxiliary building and access building during normal plant operation. With the exception of the isolation dampers, the auxiliary building HVAC system is a non safety-related system.

Location and Functional Arrangement

The major components of auxiliary building HVAC system are located in the auxiliary building. The auxiliary building HVAC system consists of supply and exhaust systems. The supply system has two 50% capacity air handling units, both air handling units are connected to a common air distribution ductwork supplying air to served areas. The exhaust system has three 50% capacity exhaust fans.

The ABVS exhaust flow is aligned to plant vent stack and is capable of providing dilution flow to gaseous effluent stream prior to release.

The auxiliary building HVAC system and containment low volume purge system are cross tied. This crosstie allows the exhaust flow from the auxiliary building HVAC system to be redirected to the containment low volume purge manually upon a high radiation alarm in the auxiliary building HVAC ductwork.

Key Design Features

The key design features of the auxiliary building HVAC system are reflected in the system design bases, which include:

- The auxiliary building HVAC system has the capability to close the safety-related, seismic Category I isolation dampers of the penetration and safeguard component areas during a design basis accident, as shown in Figure 2.7.5.2-1 and Figure 2.7.5.2-3.

- The auxiliary building HVAC system has the capability to close safety-related, seismic Category I isolation dampers to prevent the back flow from the annulus emergency exhaust system during a design basis accident, as shown in Figure 2.7.5.2-1.

- The auxiliary building HVAC system ductwork is supported as required to prevent adverse interaction with other safety-related systems during a seismic event.

- The auxiliary building HVAC system provides conditioning air to maintain the proper environmental conditions for the areas it serves during normal plant condition.

- The ABVS has the non-safety related capability of providing dilution flow to the gaseous stream prior to its release from the plant vent stack.

- The auxiliary building HVAC system and containment low volume purge system are cross connected to allow the exhaust from the radiological controlled areas to be filtered by the containment low volume purge exhaust filtration units.

- Airborne radioactivity is monitored inside the exhaust air duct from the controlled areas.

The ventilation system has fire dampers to limit the spread of fire and combustion products. The fire dampers are capable of closing against full airflow.

Seismic and ASME Code Classifications

Only the auxiliary building HVAC system isolation dampers identified in Table 2.7.5.4-1 are qualified as seismic Category I. The system components are not designed or constructed to ASME Code Section III requirements.

System Operation

The important aspects of system operation are specified under "Logic".

Alarms, Displays, and Controls

Table 2.7.5.4-2 identifies alarms, displays, and controls associated with the system that are located in the MCR.

Logic

The isolation dampers identified in Table 2.7.5.4-1 operate upon receipt of the ECCS actuation signal.

Table 2.7.5.4-3 Auxiliary Building Ventilation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 3)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a The isolation dampers identified in Table 2.7.5.4-1 as having PSMS control, perform an active safety function after receiving a signal from PSMS.	4.a Tests will be performed on the as-built isolation dampers in Table 2.7.5.4-1 using a simulated signal.	4.a Each as-built isolation dampers identified in Table 2.7.5.4-1 as having PSMS control, perform the active function identified in the table after receiving a simulated signal.
4.b After loss of motive power, the isolation dampers identified in Table 2.7.5.4-1, assume the closed position.	4.b Tests of the as-built isolation dampers will be performed under the conditions of loss of motive power.	4.b Upon loss of motive power, each as-built isolation damper identified in Table 2.7.5.4-1 assumes the closed position.
4.c The fire dampers in ductwork that penetrates fire barrier that are required to protect safe-shutdown capability close fully when called upon to do so.	4.c Tests of the as-built fire dampers will be performed.	4.c Each as-built fire damper in ductwork that penetrates fire barrier that are required to protect safe-shutdown capability close under design air flow conditions.
5. Controls exist in the MCR to close the remotely operated isolation dampers identified in Table 2.7.5.4-2.	5. Tests will be performed on the as-built remotely operated isolation dampers listed in Table 2.7.5.4-2 using controls in the MCR.	5. Controls exist in the as-built MCR to open and close the as-built remotely operated valves listed in Table 2.7.5.4-2.
6. MCR alarms and displays of the parameters identified in Table 2.7.5.4-2 can be retrieved in the MCR.	6. Inspections will be performed for retrievability of the as-built ABVS parameters in the as-built MCR.	6. MCR alarms and displays identified in Table 2.7.5.4-2 can be retrieved in the as-built MCR.
7. RSC alarms, displays and controls are identified in Table 2.7.5.4-2.	7.i Inspections of the as-built RSC alarms, displays and controls will be performed.	7.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.5.4-2.
	7.ii Tests of the as-built RSC controls will be performed.	7.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.5.4-2.

Table 2.7.6.3-5 Spent Fuel Pit Cooling and Purification System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I equipment, identified in Table 2.7.6.3-1 is designed to withstand seismic design basis loads without loss of safety function.	5.i Inspections will be performed to verify that the seismic Category I as-built equipment identified in Table 2.7.6.3-1 is located in the containment and reactor building.	5.i The seismic Category I as-built equipment identified in Table 2.7.6.3-1 is located in the containment and reactor building.
	5.ii Type tests and/or analyses of seismic Category I equipment will be performed.	5.ii The results of the type tests and/or analyses conclude that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	5.iii Inspections will be performed on the as-built equipment including anchorage.	5.iii The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6. Each of the seismic Category I piping, including supports, identified in Table 2.7.6.3-2 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	6.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.6.3-2 are supported by a seismic Category I structure(s).	6.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.6.3-2 is supported by a seismic Category I structure(s).
	6.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.6.3-2 can withstand combined normal and seismic design basis loads without a loss of its safety function.	6.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.6.3-2 can withstand combined normal and seismic design basis loads without a loss of its safety function.
7.a The Class 1E equipment identified in Table 2.7.6.3-1 is powered from their respective Class 1E division.	7.a A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	7.a The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.7.6.3-1, under test.
7.b Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	7.b Inspections of the as-built Class 1E divisional cables will be performed.	7.b Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

Table 2.7.6.3-5 Spent Fuel Pit Cooling and Purification System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The SFPCS components identified in Table 2.7.6.3-1 remove the decay heat generated by the spent fuel assemblies in the SFP during all plant operating conditions, including normal plant operating, abnormal and accident conditions.	8.a An inspection for the existence of a report that determines heat removal capacity of the as-built heat exchangers will be performed.	8.a A report exists and concludes that the product of the overall heat transfer coefficient and heat exchanger area of the SFP heat exchangers identified in Table 2.7.6.3-1 is greater than or equal to the design values for all plant operating conditions, including normal plant operating, abnormal and accident conditions.
	8.b Tests will be performed to confirm that the as-built SFP pumps can provide flow to the SFP heat exchangers.	8.b The as-built SFP pumps identified in Table 2.7.6.3-1 are capable of achieving their design flow rate.
9. MCR displays of the parameters identified in Table 2.7.6.3-3 can be retrieved in the MCR.	9. Inspections will be performed for the retrievability of the SFPCS parameters in the as-built MCR.	9. MCR displays identified in Table 2.7.6.3-3 can be retrieved in the as-built MCR.
10. RSC displays and controls are identified in Table 2.7.6.3-3.	10.i Inspections of the as-built RSC displays and controls will be performed.	10.i Displays and controls exist on the as-built RSC as identified in Table 2.7.6.3-3.
	10.ii Tests of the as-built RSC controls will be performed.	10.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.6.3-3.
11. Controls exist in the MCR to start and stop the pumps identified in Table 2.7.6.3-3.	11. Tests will be performed on the as-built pumps in Table 2.7.6.3-3 using controls in the as-built MCR.	11. Controls exist in the as-built MCR to start and stop the as-built pumps listed in Table 2.7.6.3-3.
12. The check valves, identified in Table 2.7.6.3-1, perform an active safety function to change position as indicated in the table.	12. Tests of the as-built check valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	12. Each as-built check valve changes position as indicated in Table 2.7.6.3-1 under pre-operational test conditions.

Table 2.7.6.4-2 Light Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 3)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The fuel handling machine utilizes electrical interlocks, limit switches, and mechanical stops to: a) prevent damage to a fuel assembly, b) assure appropriate radiation shielding depth below the water level in the reactor cavity. c) monitor the fuel assembly load for imparted loads greater than the nominal weight of the fuel assembly.	6. Test of the as-built electrical interlocks, limit switches, and mechanical stops of the as-built fuel handling machine will be performed, including: a) Operating the open controls of the gripper while suspending a dummy fuel assembly. b) Attempting to raise a dummy fuel assembly above a preset height. c) Attempting to lift a dummy assembly that is heavier than the nominal fuel assembly.	6. The as-built fuel handling machine utilizes electrical interlocks, limit switches, and mechanical stops to: a) prevent damage to a fuel assembly, b) assure appropriate radiation shielding depth below the water level in the reactor cavity. c) monitor the fuel assembly load for imparted loads greater than the nominal weight of the fuel assembly.
7. The fuel transfer tube is designed and constructed in accordance with ASME Code Section III requirements.	7.i Inspections will be conducted of the fabrication and installation of as-built fuel transfer tube.	7.i Design documentation exists and concludes that the as-built fuel transfer tube is fabricated, installed, and inspected in accordance with ASME Code Section III requirements.
	7.ii Analysis will be conducted to reconcile the as-designed and as-built information with the ASME design documentation.	7.ii The analysis concludes that the as-built fuel transfer tube is reconciled with the design documents.
8. Pressure boundary welds of the fuel transfer tube meet ASME Code Section III requirements for non-destructive examination of welds.	8. Inspections of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	8. The ASME Code Section III code reports exist and conclude that the ASME Code Section III requirements are met for non-destructive examination of the as-built pressure boundary welds.
9. The fuel transfer tube retains the pressure boundary integrity at the design pressure.	9. A pressure test will be performed on the as-built fuel transfer tube required to be hydrostatically examined by applicable ASME code.	9. The results of the pressure test of the as-built fuel transfer tube conform with the requirements in the applicable ASME Code.

~~7. ITAAC for fuel transfer tube: Refer to DCD Tier 1 Table 2.11.2-2, ITAAC Items 1, 2b, 3b, and 4b.~~

Table 2.7.6.7-5 Process and Post-accident Sampling System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b Each of the seismic Category I piping, including supports, identified in Table 2.7.6.7-3 is designed to withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, identified in Table 2.7.6.7-3 are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.6.7-3 is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.7.6.7-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.7.6.7-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Tables 2.7.6.7-1 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.7.6.7-1 as being qualified for a harsh environment withstands the environmental conditions that would exist before, during, and following a design basis event without loss of their safety function, for the time required to perform the safety function.

	6.a.ii An inspection will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.6.7-1 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment identified in Table 2.7.6.7-1 is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment, identified in Table 2.7.6.7-1, under test.

Table 2.7.6.7-5 Process and Post-accident Sampling System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of 6)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.a Controls exist in the MCR to close remotely operated valves identified in Table 2.7.6.7-1.	10.a Tests will be performed on the as-built remotely operated valves identified in Table 2.7.6.7-1 using the controls in the MCR.	10.a Controls exist in the as-built MCR to open and close the as-built remotely operated valves identified in Table 2.7.6.7-1.
10.b The valves identified in Table 2.7.6.7-1 as having PSMS control perform an active safety function after receiving a signal from PSMS.	10.b Tests will be performed on the as-built remotely operated valves listed in Table 2.7.6.7-1 using simulated signals.	10.b The as-built remotely operated valves identified in Table 2.7.6.7-1 as having PSMS control, perform the active function identified in the table after receiving a simulated signal.
11. After loss of motive power, the remotely operated valves identified in Table 2.7.6.7-1 assume the indicated loss of motive power position.	11. Tests of the as-built valves will be performed under the conditions of loss of motive power.	11. After loss of motive power, each as-built remotely operated valve identified in Table 2.7.6.7-1 assumes the indicated loss of motive power position.
12. MCR alarms and displays of the parameters identified in Table 2.7.6.7-4 can be retrieved in the MCR.	12. Inspections will be performed for retrievability of the PSS parameters in the as-built MCR.	12. MCR alarms and displays identified in Table 2.7.6.7-4 can be retrieved in the as-built MCR.
13. RSC alarms, displays and controls are identified in Table 2.7.6.7-4.	13.i Inspections of the as-built RSC alarms, displays and controls will be performed.	13.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.7.6.7-4.
	13.ii Tests of the as-built RSC controls will be performed.	13.ii Controls exist to operate each as-built RSC control function identified in Table 2.7.6.7-4.

2.7.6.12 Potable and Sanitary Water Systems (PSWS)

2.7.6.12.1 Design Description

System Purpose and Functions

The PSWS is not a safety-related system. The PSWS provides water for domestic use and human consumption and to collect site sanitary waste for treatment, dilution and discharge during normal operation.

Location and Functional Arrangement

The system serves all the areas in the turbine building, reactor building, auxiliary building, access building, firehouse and future facilities.

Key Design Features

The potable and sanitary water system ~~layout~~ is designed with no interconnection ~~and/or sharing between the~~to systems that could potentially introduce contaminants including radiological contaminates into the system~~, or between the units, to prevent contamination due to potential radioactivity, or due to backflow, making water unfit for human consumption~~.

The sanitary drainage system collects sanitary waste from various plant areas such as restrooms and locker room etc., and carries the wastewater for processing to the treatment facility. The sanitary drainage system does not serve any facilities in the radiological controlled areas.

Seismic and ASME Code Classifications

The PSWS is non-seismic category and is not designed to ASME Code Section III requirements.

System Operation

There is no important system operation.

Alarms Displays and Controls

There are no important alarms, displays, and controls.

Logic

There is no logic needed for direct safety functions related to the PSWS.

Interlocks

There are no interlocks needed for direct safety functions related to the PSWS.

Table 2.7.6.13-3 Area Radiation and Airborne Radioactivity Monitoring Systems Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 3)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The Class 1E radiation monitors identified in Table 2.7.6.13-1 as being designed for harsh environment are designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	3.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E radiation monitor located in a harsh environment.	3.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E radiation monitors identified in Table 2.7.6.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	3.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	3.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.7.6.13-1 as being qualified for a harsh environment are bounded by type test, ~~and/or~~ analyses, or a combination of type tests and analyses.
4.a The Class 1E radiation monitors identified in Table 2.7.6.13-1 are powered from their respective Class 1E division.	4.a A test will be performed on each division of the as-built radiation monitors by providing a simulated test signal only in the Class 1E division under test.	4.a The simulated test signal exists at the as-built Class 1E radiation monitors, identified in Table 2.7.6.13-1, under test.
4.b Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	4.b Inspections of the as-built Class 1E divisional cables will be performed.	4.b Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
5. Each division of Class 1E radiation monitors identified in Table 2.7.6.13-1 is physically separated from the other divisions.	5. Inspections of the as-built Class 1E radiation monitors will be performed.	5. Each division of the Class 1E radiation monitors identified in Table 2.7.6.13-1 is physically separated from other divisions.

Table 2.8-1 Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.a Shielding walls and floors listed in Table 2.2-2 are provided to maintain the maximum radiation levels specified in Table 2.8-2.	1.a Inspections of the as-built shielding walls and floors thicknesses will be performed. ~~Refer to Table 2.2-4 ITAAC Item 1.~~	1.a The as-built shielding walls and floors listed in Table 2.2-2 are consistent with the designed concrete wall and floor thicknesses. ~~Refer to Table 2.2-4 ITAAC Item 1.~~
1.b Shielding walls and floors in the auxiliary building are provided to maintain the maximum radiation levels specified in Table 2.8-2.	1.b Inspections of the as-built shielding walls and floors thicknesses will be performed.	1.b The as-built shielding walls and floors in the auxiliary building are consistent with the designed concrete wall and floor thicknesses.
2. Area radiation and airborne radioactivity monitoring systems are provided to monitor radioactivity concentrations.	2. Refer to Table 2.7.6.13-3 ITAAC Item 1.	2. Refer to Table 2.7.6.13-3 ITAAC Item 1.
3. Ventilation flow for the radioactive controlled area is provided to control the concentrations of airborne radioactivity specified in 10 CFR 20 Appendix B.	3. Tests of the as-built containment purge system and auxiliary building HVAC system will be performed.	3. The as-built containment purge system and auxiliary building HVAC provide ventilation flow to control the concentrations of airborne radioactivity specified in 10 CFR 20 Appendix B.

Table 2.8-2 Radiation Zone Designations

Zone	Dose Rate
I	≤0.25 mrem/h
II	≤1.0 mrem/h
III	≤2.5 mrem/h
IV	≤15.0 mrem/h
V	≤100.0 mrem/h
VI	≤1.0 rem/h
VII	≤10.0 rem/h
VIII	≤100.0 rem/h
IX	≤500.0 rad/h
X	>500.0 rad/h

2.10.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.10-1 describes ITAAC for emergency planning.

Table 2.10-1 Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The TSC floor space is at least 1875 ft^2 (75 ft^2 for each of at least 25 persons).	1. An inspection of the as-built TSC floor area will be performed.	1. The as-built TSC has at least 1875 ft^2 of floor space.
2. The TSC is located close to the MCR.	2. An inspection will be performed for the location of the as-built TSC relative to the as-built MCR.	2. Walking between the as-built TSC and MCR takes no more than 2 minutes.
3. The TSC provides a habitable workspace environment.	3. See Table 2.7.5.4-3, ITAAC Item 8.	3. See Table 2.7.5.4-3, ITAAC Item 8.
4. Adequate emergency communications systems are in place.	4. See Table 2.7.6.10-1, ITAAC Items 2, 3, and 4 and Table 2.9-1, ITAAC Items 7.k and 7.l.	4. See Table 2.7.6.10-1, ITAAC Items 2, 3, and 4 and Table 2.9-1, ITAAC Items 7.k and 7.l.

Table 2.11.2-2 Containment Isolation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a The seismic Category I equipment is designed to withstand seismic design basis loads without loss of safety function.	5.a.i Inspections will be performed to verify that the seismic Category I as-built equipment are located in the containment and the reactor building.	5.a.i The seismic Category I as-built equipment is located in the containment and the reactor building.
	5.a.ii Type tests and/or analyses of seismic Category I equipment will be performed.	5.a.ii The results of the type tests and/or analyses concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	5.a.iii Inspections will be performed the as-built equipment including anchorage.	5.a.iii The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
5.b Each of the seismic Category I piping, including supports, is designed to withstand combined normal and seismic design basis loads without loss of its safety function.	5.b.i Inspections will be performed to verify that the as-built seismic Category I piping, including supports, are supported by a seismic Category I structure(s).	5.b.i Reports(s) document that each of the as-built seismic Category I piping, including supports, is supported by a seismic Category I structure(s).
	5.b.ii Inspections and analyses will be performed to verify ~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.11.2-1 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses conclude that the Class 1E equipment identified in Table 2.11.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform

		the safety function.

Table 2.11.2-2 Containment Isolation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	6.a.ii An inspection will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.11.2-1 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table 2.11.2-1, is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.11.2-1 under test.
6.c Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	6.c Inspections of the as-built Class 1E divisional cables will be conducted.	6.c Physical separation or electrical isolation is provided between the as-built cables of Class 1E divisions and between Class 1E divisions and non-Class 1E cables.
7. CIS isolates containment upon receipt of a containment isolation signal.	7. Tests will be performed to verify that the as-built containment isolation air operated valves and motor operated valves close on receipt of an isolation signal.	7. The as-built containment isolation air operated valves and motor operated valves close on receipt of an isolation signal.
8.i The RCS CIVs close within the containment isolation response time.	8.i Tests will be performed to verify as-built RCS CIVs close within the isolation response times.	8.i The following as-built RCS CIVs close within the required times: ≤ 15 seconds RCS-AOV-132, RCS-AOV-138, RCS-AOV-147, RCS-AOV-148
8.ii The WMS CIVs listed in Table 2.11.2-1 close within the containment isolation response time.	8.ii Tests will be performed to verify as-built WMS CIVs listed in Table 2.11.2-1 close within the isolation response times.	8.ii The as-built WMS CIVs listed in Table 2.11.2-1 close within 15 seconds.

Table 2.11.2-2 Containment Isolation System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 7 of 8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.xiv The CCWS CIVs close within the containment isolation response time.	8.xiv Tests will be performed to verify as-built CCWS CIVs close within the isolation response times.	8.xiv The following as-built CCWS CIVs close within the required times: ≤ 20 seconds NCS-MOV-511, NCS-MOV-517 ≤ 40 seconds NCS-MOV-402 A, B, NCS-MOV-436 A,B, NCS-MOV-438 A,B, NCS-MOV-531, NCS-MOV-537
8.xv The PSS CIVs close within the containment isolation response time.	8.xv Tests will be performed to verify as-built PSS CIVs close within the isolation response times.	8.xv The following as-built PSS CIVs close within the required times: ≤ 15 seconds PSS-AOV-003, PSS-MOV-006, PSS-MOV-013, PSS-MOV-023, PSS-MOV-031 A,B PSS-AOV-062 A,B,C,D, PSS-AOV-063
9. The systems penetrating containment retain their containment inventory during containment isolation.	9. Tests will be performed to verify the as-built containment isolation valve leakage in accordance with 10 CFR 50, Appendix J, Type C tests.	9. The as-built containment isolation valve leakage is within design limits and is less than the allowable leakage rate specified in 10 CFR 50, Appendix J.
10. Controls exist in the MCR to open and close the remotely operated valves identified in Table 2.11.2-3.	10. Tests will be performed on the as-built remotely operated valves listed in Table 2.11.2-3 using controls in the as-built MCR.	10. Controls exist in the as-built MCR operate to open and close the as-built remotely operated valves listed in Table 2.11.2-3.
11.a MCR alarms and displays of the parameters identified in Table 2.11.2-3 can be retrieved in the MCR.	11.a Inspections will be performed for retrievability of the CIS parameters in the as-built MCR.	11.a The as-built MCR alarms and displays identified in Table 2.11.2-3 can be retrieved in the as-built MCR.
11.b RSC alarms, displays and controls are identified in Table 2.11.2-3.	11.b.i Inspections of the as-built RSC alarms, displays and controls will be performed.	11.b.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.11.2-3.
	11.b.ii Tests of the as-built RSC controls will be performed.	11.b.ii Controls exist to operate each as-built RSC control function identified in Table 2.11.2-3.

Table 2.11.3-5 Containment Spray System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5.b.ii Inspections and analyses will be performed to verify ~~for the existence of a report verifying~~ that the as-built seismic Category I piping, including supports, identified in Table 2.11.3-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.	5.b.ii A report exists and concludes that each of the as-built seismic Category I piping, including supports, identified in Table 2.11.3-3 can withstand combined normal and seismic design basis loads without a loss of its safety function.
6.a The Class 1E equipment identified in Table 2.11.3-2 as being qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.	6.a.i Type tests, ~~and/or~~ analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment located in a harsh environment.	6.a.i The results of the type tests, ~~and/or~~ analyses, or a combination of type tests and analyses concludes that the Class 1E equipment identified in Table 2.11.3-2 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis event without loss of safety function for the time required to perform the safety function.
	6.a.ii Inspections will be performed on the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	6.a.ii The as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.11.3 -2 as being qualified for a harsh environment are bounded by type tests, ~~and/or~~ analyses, or a combination of type tests and analyses.
6.b The Class 1E equipment, identified in Table 2.11.3-2, is powered from their respective Class 1E division.	6.b A test will be performed on each division of the as-built equipment by providing a simulated test signal only in the Class 1E division under test.	6.b The simulated test signal exists at the as-built Class 1E equipment identified in Table 2.11.3-2 under test.

Table 2.11.3-5 Containment Spray System Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 7 of 7)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.c An interlock is provided for each division of CS/RHR to preclude the simultaneous opening of both the RHR discharge line containment isolation valves and the corresponding containment spray header containment isolation valve.	10.c Tests will be performed on each as-built interlock for the RHR discharge line containment isolation valves and the containment spray header containment isolation valve.	10.c Each as-built interlock for the RHR discharge line containment isolation valves and the corresponding containment spray header containment isolation valve preclude the simultaneous opening of both the RHR discharge line containment isolation valves and the corresponding containment spray header containment isolation valve.
10.d An interlock is provided for each division of CS/RHR to allow opening of the containment spray header containment isolation valve only if either or both of the corresponding two in-series CS/RHR pump hot leg isolation valves are closed.	10.d Tests will be performed on each as-built interlock for the containment spray header containment isolation valves and CS/RHR pump hot leg isolation valves.	10.d Each as-built interlock for the containment spray header containment isolation valve and corresponding two in-series CS/RHR pump hot leg isolation valves will allow opening of the containment spray header containment isolation valve only if either or both of the corresponding two in-series CS/RHR pump hot leg isolation valves are closed.
11. MCR alarms and displays of the parameters identified in Table 2.11.3-4 can be retrieved in the MCR.	11. Inspections will be performed for retrievability of the CSS parameters in the as-built MCR.	11. MCR alarms and displays identified in Table 2.11.3-4 can be retrieved in the as-built MCR.
12. RSC alarms, displays and controls are identified in Table 2.11.3-4.	12.i Inspections of the as-built RSC alarms, displays and controls will be performed.	12.i Alarms, displays and controls exist on the as-built RSC as identified in Table 2.11.3 -4.
	12.ii Tests of the as-built RSC controls will be performed.	12.ii Controls exist to operate each as-built RSC control function identified in Table 2.11.3-4.