US-APWR DCD RAI Tracking Report

June 2009

Revision History

Revision	Page	Description
0	All	Original issued Including RAI responses that were submitted through January 9, 2009 for Chapter 2, 3, 5, 8, 12, 13, 14, 15 and 16
1	All	Including RAI responses that were submitted through February 27, 2009 for Chapter 3, 5, 6, 9, 10, 11, 12, 14, 16, 17 and 19
2	All	Including RAI responses that were submitted through March 31, 2009 for Chapter 3, 4, 6, 9, 10, 11, 12, 14, 17 and 18

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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") of US-APWR. The report also includes the DCD markups for the RAI responses that impacted the DCD. The related "Change ID Number" shown in the table is addressed in the corner on the right of each sheet of the DCD markups.

Notes

This report includes RAI responses that were submitted through March 31, 2009 for Chapter 3, 4, 6, 9, 10, 11, 12, 14, 17 and 18.

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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	287	03.02.01-1	2009/5/8	Y	N	N		-	DCD_03.02.01-1	TBD	
		287	03.02.01-2	2009/5/8	Y	N	N		-	DCD_03.02.02-2	TBD	
		287	03.02.01-3	2009/5/21	Y	N	N		-	DCD_03.02.02-3	TBD	
		287	03.02.01-4	2009/5/21	N	N	N		-	-	N/A	N/A
		287	03.02.01-5	2009/5/8	N	N	N		-	-	N/A	N/A
		287	03.02.01-6	2009/5/21	Y	Y	N		-	DCD_03.02.02-6	TBD	
		287	03.02.01-7	2009/5/21	Y	N	N		-	DCD_03.02.02-7	TBD	
		287	03.02.01-8	2009/5/8	N	N	N		-	-	N/A	N/A
		287	03.02.01-9	2009/5/21	Y	N	N		-	DCD_03.02.02-9	TBD	
		287	03.02.01-10	2009/5/21	Y	N	N		-	DCD_03.02.02-10	TBD	
		287	03.02.01-11	2009/5/21	N	N	N		-	-	N/A	N/A
		287	03.02.01-12	2009/5/21	N	N	N		-	-	N/A	N/A
		287	03.02.01-13	2009/5/8	Y	N	N		-	DCD_03.02.02-13	TDB	
		287	03.02.01-14	2009/5/21	Y	N	N		-	DCD_03.02.02-14	TBD	
3.2.2	System Quality Group	276	03.02.02-1	2009/4/24	Y	N	N		-	DCD_03.02.02-1	TDB	
	Classification	276	03.02.02-2	2009/4/24	Y	N	N		-	DCD_03.02.02-2	TDB	
		276	03.02.02-3	2009/5/8	Y	N	N		-	DCD_03.02.02-3	TDB	
		276	03.02.02-4	2009/4/24	Y	N	N		-	DCD_03.02.02-4	TDB	
		276	03.02.02-5	2009/5/8	Y	N	N		-	DCD_03.02.02-5	TDB	
		276	03.02.02-6	2009/5/8	N	N	N		-	-	N/A	N/A
		276	03.02.02-7	2009/5/8	N	N	N		-	-	N/A	N/A
		276	03.02.02-8	2009/5/8	N	N	N		-	-	N/A	N/A
		276	03.02.02-9	2009/4/24	Y	N	N		-	DCD_03.02.02-9	TDB	
3.3.1	Wind Loadings	215	3.3.1-01	2009/4/9	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		215	3.3.1-02	2009/4/9	N	N	N		-	-	N/A	N/A
		215	3.3.1-03	2009/4/9	N	N	N		-	-	N/A	N/A
		215	3.3.1-04	2009/4/9	Y	N	N		-	DCD_3.3.1-04	TBD	
		215	3.3.1-05	2009/4/9	Y	N	N		-	DCD_3.3.1-05	TBD	
		215	3.3.1-06	2009/4/9	N	N	N		-	-	N/A	N/A
3.3.2	Tornado Loadings	218	3.3.2-01	2009/4/9	Y	N	N		-	DCD_3.3.2-01	TBD	
		218	3.3.2-02	2009/4/9	Y	N	N		-	DCD_3.3.2-02	TBD	
		218	3.3.2-03	2009/4/9	Y	N	N		-	DCD_3.3.2-03	TBD	
		218	3.3.2-04	2009/4/9	Y	N	N		-	DCD_3.3.2-04	TBD	
3.4.1	Internal Flood Protection for	220	3.4.1-01	2009/4/8	Y	N	N		-	DCD_3.4.1-01	TBD	
	Onsite Equipment Failures	220	3.4.1-02	2009/4/23	Y	N	N		-	DCD_3.4.1-02	TBD	
		220	3.4.1-03	2009/4/8	Y	N	N		-	DCD_3.4.1-03	TBD	
		220	3.4.1-04	2009/4/23	Y	N	N		-	DCD_3.4.1-04	TBD	
		220	3.4.1-05	2009/4/23	Y	N	N		-	DCD_3.4.1-05	TBD	
		220	3.4.1-06	2009/4/23	Y	N	N		-	DCD_3.4.1-06	TBD	
		220	3.4.1-07	2009/4/23	N	N	N		-	-	N/A	N/A
		220	3.4.1-08	2009/4/23	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		220	3.4.1-09	2009/4/23	Y	N	N		-	DCD_3.4.1-09	TBD	
		220	3.4.1-10	2009/4/23	Y	N	N		-	DCD_3.4.1-10	TBD	
		220	3.4.1-11	2009/4/23	Y	N	N		-	DCD_3.4.1-11	TBD	
		220	3.4.1-12	2009/4/23	Y	N	N		-	DCD_3.4.1-12	TBD	
		220	3.4.1-13	2009/4/23	Y	N	N		-	DCD_3.4.1-13	TBD	
		220	3.4.1-14	2009/4/23	Y	N	N		-	DCD_3.4.1-14	TBD	
		220	3.4.1-15	2009/4/23	Y	N	N		-	DCD_3.4.1-15	TBD	
		220	3.4.1-16	2009/4/23	Y	N	N		-	DCD_3.4.1-16	TBD	
		220	3.4.1-17	2009/4/23	Y	N	N		-	DCD_3.4.1-17	TBD	
		220	3.4.1-18	2009/4/8	Y	N	N		-	DCD_3.4.1-18	TBD	
		220	3.4.1-19	2009/4/8	Y	N	N		-	DCD_3.4.1-19	TBD	
		220	3.4.1-20	2009/4/8	N	N	N		-	-	N/A	N/A
3.4.2	Analysis Procedures	219	3.4.2-01	2009/4/9	N	N	N		-	-	N/A	N/A
		219	3.4.2-02	2009/4/9	N	N	N		-	-	N/A	N/A
		219	3.4.2-03	2009/4/9	N	N	N		-	-	N/A	N/A
		219	3.4.2-04	2009/4/9	Y	N	N		-	DCD_3.4.2-04	TBD	
3.5.1.1	Internally Generated Missiles	127	3.5.1.1-01	2009/1/28	Y	N	N		-	DCD_3.5.1.1-01	1	2
	(Outside Containment)	127	3.5.1.1-02	2009/1/28	Y	N	N		-	DCD_3.5.1.1-02	1	2
		127	3.5.1.1-03	2009/1/28	Y	N	N		-	DCD_3.5.1.1-03	1	2
		127	3.5.1.1-04	2009/1/28	Y	Y	N		-	DCD_3.5.1.1-04	1	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		127	3.5.1.1-05	2009/1/28	Y	N	N		-	DCD_3.5.1.1-05	1	2
3.5.1.2	Internally-Generated Missiles	152	3.5.1.2-01	2009/2/4	Y	N	N		-	DCD_3.5.1.2-01	1	2
	(Inside Containment)	152	3.5.1.2-02	2009/2/4	Y	N	N		-	DCD_3.5.1.2-02	1	2
		152	3.5.1.2-03	2009/2/4	Y	N	N		-	DCD_3.5.1.2-03	1	2
3.5.1.3	Turbine Missiles	323	03.05.01.03-1/3.5.1.3-1	2009/5/20	Y	N	N		-	DCD_03.05.01.03-1/3.5.1.3-1	TBD	
		323	03.05.01.03-2/3.5.1.3-2	2009/5/20	Y	N	N		-	DCD_03.05.01.03-2/3.5.1.3-2	TBD	
		323	03.05.01.03-3/3.5.1.3-3	2009/5/20	Y	N	N		-	DCD_03.05.01.03-3/3.5.1.3-3	TBD	
		324	03.05.01.03-1/3.5.1.3-1	2009/5/20	Y	N	N		-	DCD_03.05.01.03-1/3.5.1.3-1	TBD	
		324	03.05.01.03-2/3.5.1.3-2	2009/5/20	N	N	N		-	-	N/A	N/A
		324	03.05.01.03-3/3.5.1.3-3	2009/5/20	N	N	N		-	-	N/A	N/A
		324	03.05.01.03-4/3.5.1.3-4	2009/5/20	N	N	N		-	-	N/A	N/A
		324	03.05.01.03-5/3.5.1.3-5	2009/5/20	N	N	N		-	-	N/A	N/A
		324	03.05.01.03-6/3.5.1.3-6	2009/5/20	N	N	N		-	-	N/A	N/A
		324	03.05.01.03-7/3.5.1.3-7	2009/5/20	N	N	N		-	-	N/A	N/A
3.5.1.4	Missiles Generated by	154	3.5.1.4-01	2009/2/4	Y	N	N		-	DCD_3.5.1.4-01	1	2
	Tornadoes and Extreme Winds	154	3.5.1.4-02	2009/2/4	N	N	N		-		N/A	N/A
		154	3.5.1.4-03	2009/2/4	Y	N	N		-	DCD_3.5.1.4-03	1	2
		154	3.5.1.4-04	2009/2/4	Y	N	N		-	DCD_3.5.1.4-04	1	2
		154	3.5.1.4-05	2009/2/4	Y	N	N		-	DCD_3.5.1.4-05	1	2
		357	3.5.1.4-02-S01	2009/6/4	Y	Y	N		-	DCD_3.5.1.4-02-S01	TBD	

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3.5.1.5	Site Proximity Missiles (Except Aircraft)											
3.5.1.6	Aircraft Hazards											
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	153	3.5.2-01	2009/2/4	Y	N	N		-	DCD_3.5.2-01	1	2
3.5.3	Barrier Design Procedures	221	3.5.3-01	2009/4/8	N	N	N		-	-	N/A	N/A
		221	3.5.3-02	2009/4/8	Y	N	N		-	DCD-3.5.3-02	TBD	
		221	3.5.3-03	2009/4/8	Y	N	N		-	DCD-3.5.3-03	TBD	
		221	3.5.3-04	2009/4/8	Y	N	N		-	DCD-3.5.3-04	TBD	
		221	3.5.3-05	2009/4/8	N	N	N		-	-	N/A	N/A
		221	3.5.3-06	2009/4/8	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.6.1	Plant Design for Protection	180	3.6.1-1	2009/3/3	Y	N	N		-	3.6.1-1	2	2
	Against Postulated Piping Failures	180	3.6.1-2	2009/3/3	N	N	N		-	3.6.1-2	N/A	N/A
	in Fluid Systems	180	3.6.1-3	2009/3/3	N	N	N		-	3.6.1-3	N/A	N/A
	Outside Containment	180	3.6.1-4	2009/3/3	Y	N	N		-	3.6.1-4	3	2
		180	3.6.1-5	2009/3/3	Y	N	N		-	3.6.1-5	3	2
		180	3.6.1-6	2009/3/3	Y	N	N		-	3.6.1-6	3	2
3.6.2	Determination of Rupture Locations	71	03.06.02-1	2008/10/7	N	N	N	fin.	-	-	N/A	N/A
	and Dynamic Effects Associated	71	03.06.02-2	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-2	0	2
	with the Postulated Rupture	71	03.06.02-3	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-3	0	2
	of Piping	71	03.06.02-4	2008/10/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-5	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-5	0	2
		71	03.06.02-6	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-6	0	2
		71	03.06.02-7	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-7	0	2
		71	03.06.02-8	2008/10/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-9	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-9	0	2
		71	Intro for 03.06.02-10 thru 15	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-10	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-11	2008/11/7	Y	N	N	fin.	-	DCD_03.06.02-11	0	2
		71	03.06.02-12	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-13	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-14	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-15	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-16	2008/10/7	Y	N	N	fin.	-	DCD_03.06.02-16	0	2
		71	03.06.02-17	2008/10/7	N	N	N	fin.	-	-	N/A	N/A

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		71	03.06.02-18	2008/10/7	N	N	N	fin.	-	-	N/A	N/A
		71	03.06.02-19	2008/10/7	Y	N	N	fin.	-	-	-	1
3.6.3	Leak-Before-Break	210	3.6.3-1	2009/4/9	Y	N	N		-	DCD_3.6.3-1	TBD	
	Evaluation Procedures	210	3.6.3-2	2009/4/23	N	N	N		-	-	N/A	N/A
		210	3.6.3-3	2009/4/23	N	N	N		-	-	N/A	N/A
		210	3.6.3-4	2009/4/9	Y	N	N		-	DCD_3.6.3-4	TBD	
		210	3.6.3-5	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-6	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-7	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-8	2009/4/9	Y	N	N		-	DCD_3.6.3-8	TBD	
		210	3.6.3-9	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-10	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-11	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-12	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-13	2009/4/9	N	N	N		-	-	N/A	N/A
		210	3.6.3-14	2009/4/9	N	N	N		-	-	N/A	N/A
		217	3.6.3-15	2009/3/24	Y	Y	N		-	DCD_3.6.3-15	2	2
		217	3.6.3-16	2009/4/23	Y	Y	N		-	DCD_3.6.3-16	TBD	
3.7.1	Seismic Design Parameters	211	3.7.1-1	2009/3/25	N	N	N		-	-	N/A	N/A
		211	3.7.1-2	2009/3/25	N	N	N		-	-	N/A	N/A
		211	3.7.1-3	2009/4/23	N	N	N		-	-	N/A	N/A
		211	3.7.1-4	2009/3/25	Y	N	N		-	DCD_3.7.1-4	2	2

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		211	3.7.1-5	2009/4/23	Y	N	N		-	-	N/A	N/A
		211	3.7.1-6	2009/4/23	N	N	N		-	-	N/A	N/A
		211	3.7.1-7	2009/4/23	N	N	N		-	-	N/A	N/A
3.7.2	Seismic System Analysis	212	3.7.2-1	2009/5/7	Y	N	N		-	DCD_3.7.2-2	TBD	
		212	3.7.2-2	2009/3/30	Y	N	N		-	DCD_3.7.2-2	2	2
		212	3.7.2-3	2009/5/7	Y	N	N		-	DCD_3.7.2-3	TBD	
		212	3.7.2-4	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-5	2009/5/7	Y	N	N		-	DCD_3.7.2-5	TBD	
		212	3.7.2-6	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-7	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-8	2009/3/30	Y	N	N		-	DCD_3.7.2-8	3	2
		212	3.7.2-9	2009/3/30	Y	N	N		-	DCD_3.7.2-9	2	2
		212	3.7.2-10	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-11	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-12	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-13	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-14	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-15	2009/5/7	Y	N	N		-	DCD_3.7.2-15	TBD	
		212	3.7.2-16	2009/3/30	N	N	N		-	-	N/A	N/A
		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	
		212	3.7.2-20	2009/5/7	Y	N	N		-	DCD_3.7.2-20	TBD	
		212	3.7.2-21	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-22	2009/3/30	Y	N	N		-	DCD_3.7.2-22	2	2
		212	3.7.2-23	2009/3/30	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		212	3.7.2-24	2009/5/7	Y	N	N		-	DCD_3.7.2-24	TBD	
		212	3.7.2-25	2009/3/30	N	N	N		-	-	N/A	N/A
		212	3.7.2-26	2009/3/30	Y	N	N		-	DCD_3.7.2-26	2	2
		212	3.7.2-27	2009/5/7	Y	N	N		-	DCD_3.7.2-27	TBD	
		212	3.7.2-28	2009/3/30	Y	N	N		-	DCD_3.7.2-28	2	2
3.7.3	Seismic Subsystem Analysis	213	3.7.3-01	2009/3/27	Y	N	N		-	DCD_3.7.3-01	2	2
		213	3.7.3-02	2009/3/27	N	N	N		-	-	N/A	N/A
		213	3.7.3-03	2009/4/24	Y	N	N		-	DCD_3.7.3-03	TBD	
		213	3.7.3-04	2009/4/24	Y	N	N		-	DCD_3.7.3-04	TBD	
		213	3.7.3-05	2009/3/27	Y	N	N		-	DCD_3.7.3-05	2	2
		213	3.7.3-06	2009/4/24	N	N	N		-	-	N/A	N/A
		213	3.7.3-07	2009/3/27	N	N	N		-	-	N/A	N/A
		213	3.7.3-08	2009/3/27	Y	N	N		-	DCD_3.7.3-08	2	2
		213	3.7.3-09	2009/3/27	Y	N	N		-	DCD_3.7.3-09	2	2
		213	3.7.3-10	2009/3/27	N	N	N		-	-	N/A	N/A
		213	3.7.3-11	2009/3/27	Y	N	N		-	DCD_3.7.3-11	2	2
		213	3.7.3-12	2009/4/24	Y	N	N		-	DCD_3.7.3-12	TBD	
		213	3.7.3-13	2009/3/27	Y	N	N		-	DCD_3.7.3-13	TBD	
		213	3.7.3-14	2009/4/24	N	N	N		-	-	N/A	N/A
		213	3.7.3-15	2009/4/24	N	N	N		-	-	N/A	N/A
3.7.4	Seismic Instrumentation	55	03.07.04-1	2008/9/1	Y	N	N	fin.	-	DCD_03.07.04-1	0	2
		55	03.07.04-2	2008/9/1	Y	N	N	fin.	-	DCD_03.07.04-2	0	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		-	-	-	-	-	-	-	COL3.7(15) deleted	MAP-03-001	TBD	
		-	-	-	-	-	-	-	COL3.7(18) deleted	MAP-03-002	TBD	
3.8.1	Concrete Containment	223	3.8.1-1	2009/4/14	Y	N	N		-	DCD_3.8.1-1	TBD	
		223	3.8.1-2	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-3	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-4	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-5	2009/4/14	Y	N	N		-	DCD_3.8.1-5	TBD	
		223	3.8.1-6	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-7	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-8	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-9	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-10	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-11	2009/4/14	N	N	N		-	-	N/A	N/A
		223	3.8.1-12	2009/4/24	Y	N	N		-	DCD_3.8.1-12	TBD	
			3.8.1-13	2009/4/24	Y	N	N		-	DCD_3.8.1-13	TBD	
		223	3.8.1-14	2009/4/14	Y	N	N		-	DCD_3.8.1-14	TBD	
		-	-	-	-	-	-	-	COL3.8(1) deleted	MAP-03-003	TBD	
		-	-	-	-	-	-	-	COL3.8(2) deleted	MAP-03-004	TBD	
		-	-	-	-	-	-	-	COL3.8(4) deleted	MAP-03-005	TBD	
		-	-	-	-	-	-	-	COL3.8(5) deleted	MAP-03-006	TBD	
		-	-	-	-	-	-	-	COL3.8(6) deleted	MAP-03-007	TBD	
		-	-	-	-	-	-	-	COL3.8(8) deleted	MAP-03-008	TBD	

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		-	-	-	-	-	-	-	COL3.8(9) deleted	MAP-03-009	TBD	
		-	-	-	-	-	-	-	COL3.8(12) deleted	MAP-03-010	TBD	
		-	-	-	-	-	-	-	COL3.8(13) deleted	MAP-03-011	TBD	
3.8.3	Concrete and Steel	322	3.8.3-1	2009/6/4	N	N	N		-	-	N/A	N/A
	Internal Structures	322	3.8.3-2	2009/5/21	N	N	N		-	-	N/A	N/A
	of Steel or Concrete Containments	322	3.8.3-3	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-4	2009/5/21	Y	N	N		-	DCD_3.8.3-4	TBD	
		322	3.8.3-5	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-6	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-7	2009/6/4	N	N	N		-	-	N/A	N/A
		322	3.8.3-8	2009/6/4	N	N	N		-	-	N/A	N/A
		322	3.8.3-9	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-10	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-11	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-12	2009/5/21	N	N	N		-	-	N/A	N/A
		322	3.8.3-13	2009/6/4	N	N	N		-	-	N/A	N/A
		322	3.8.3-14	2009/6/4	N	N	N		-	-	N/A	N/A
		322	3.8.3-15	2009/6/4	N	N	N		-	-	N/A	N/A
3.8.4	Other Seismic Category I Structures											
		-	-	-	-	-	-	-	COL3.8(22) revised	MAP-03-012	TBD	
3.8.5	Foundations											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.9.1	Special Topics for	296	03.09.01-1	2009/5/14	Y	N	N		-	DCD_03.09.01-1	TBD	
	Mechanical Components	296	03.09.01-2	2009/5/14	N	N	N		-	-	N/A	N/A
		296	03.09.01-3	2009/5/14	N	N	N		-	-	N/A	N/A
		296	03.09.01-4	2009/5/14	Y	N	N		-	DCD_03.09.01-4	TBD	
		296	03.09.01-5	2009/5/14	Y	N	N		-	DCD_03.09.01-5	TBD	
3.9.2	Dynamic Testing and Analysis	204	3.9.2-01	2009/3/25	Y	N	N		-	DCD_3.9.2-01	2	2
	of Systems, Structures,	204	3.9.2-02	2009/3/25	Y	N	N		-	DCD_3.9.2-02	2	2
	and Components	204	3.9.2-03	2009/3/25	Y	N	N		-	DCD_3.9.2-03	2	2
		204	3.9.2-04	2009/3/25	Y	N	N		-	DCD_3.9.2-04	2	2
		204	3.9.2-05	2009/3/25	Y	N	N		-	DCD_3.9.2-05	2	2
		204	3.9.2-06	2009/3/25	N	N	N		-	-	N/A	N/A
		204	3.9.2-07	2009/3/25	Y	N	N		-	DCD_3.9.2-07	2	2
		204	3.9.2-08	2009/3/25	Y	N	N		-	DCD_3.9.2-08	2	2
		204	3.9.2-09	2009/3/25	Y	N	N		-	DCD_3.9.2-09	2	2
		205	3.9.2-10	2009/4/30	N	N	N		-	-	N/A	N/A
		205	3.9.2-11	2009/4/30	Y	N	N		-	DCD_3.9.2-11	TBD	
		205	3.9.2-12	2009/4/30	N	N	N		-	-	N/A	N/A
		205	3.9.2-13	2009/4/30	N	N	N		-	-	N/A	N/A
		205	3.9.2-14	2009/4/30	Y	N	N		-	DCD_3.9.2-14	TBD	
		205	3.9.2-15	2009/4/30	Y	N	N		-	DCD_3.9.2-15	TBD	
		205	3.9.2-16	2009/4/30	N	N	N		-	-	N/A	N/A
		205	3.9.2-17	2009/4/30	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		205	3.9.2-18	2009/4/30	Y	N	N		-	DCD_3.9.2-18	TBD	
		272	3.9.2-19	2009/4/9	Y	N	N		-	DCD_3.9.2-19	TBD	
		272	3.9.2-20	2009/4/9	N	N	N		-	-	N/A	N/A
		272	3.9.2-21	2009/4/9	Y	N	N		-	DCD_3.9.2-21	TBD	
		272	3.9.2-22	2009/4/9	Y	N	N		-	DCD_3.9.2-22	TBD	
		272	3.9.2-23	2009/4/9	Y	N	N		-	DCD_3.9.2-23	TBD	
		272	3.9.2-24	2009/4/9	Y	N	N		-	DCD_3.9.2-24	TBD	
		272	3.9.2-25	2009/4/9	Y	N	N		-	DCD_3.9.2-25	TBD	
		272	3.9.2-26	2009/4/9	Y	N	N		-	DCD_3.9.2-26	TBD	
		272	3.9.2-27	2009/4/9	Y	N	N		-	DCD_3.9.2-27	TBD	
		272	3.9.2-28	2009/4/9	Y	N	N		-	DCD_3.9.2-28	TBD	
		272	3.9.2-29	2009/4/9	Y	N	N		-	DCD_3.9.2-29	TBD	
		272	3.9.2-30	2009/4/9	N	N	N		-	-	N/A	N/A
		272	3.9.2-31	2009/4/9	N	N	N		-	-	N/A	N/A
		272	3.9.2-32	2009/4/9	N	N	N		-	-	N/A	N/A
		272	3.9.2-33	2009/5/13	N	N	N		-	-	N/A	N/A
		272	3.9.2-34	2009/4/9	N	N	N		-	-	N/A	N/A
		272	3.9.2-35	2009/4/9	N	N	N		-	-	N/A	N/A
		214	3.9.2-34	2009/4/30	N	N	N		-	-	N/A	N/A
		214	3.9.2-35	2009/4/30	N	N	N		-	-	N/A	N/A
		214	3.9.2-36	2009/4/30	N	N	N		-	-	N/A	N/A
		214	3.9.2-37	2009/4/30	Y	N	N		-	DCD_3.9.2-37	TDB	
		214	3.9.2-38	2009/4/30	Y	N	N		-	DCD_3.9.2-38	TDB	
		214	3.9.2-39	2009/4/30	N	N	N		-	-	N/A	N/A
		214	3.9.2-40	2009/4/30	Y	N	N		-	DCD_3.9.2-40A	TBD	
		214	3.9.2-41	2009/4/30	Y	N	N		-	DCD_3.9.2-41	TBD	
		206	3.9.2-40	2009/3/27	Y	N	N		-	DCD_3.9.2-40	3	2
		206	3.9.2-41	2009/3/27	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		206	3.9.2-42	2009/3/27	Y	N	N		-	DCD_3.9.2-42	2	2
		206	3.9.2-43	2009/3/27	Y	N	N		-	DCD_3.9.2-43	2	2
			3.9.2-44						-			
			3.9.2-45						-			
			3.9.2-46						-			
			3.9.2-47						-			
			3.9.2-48						-			
			3.9.2-49						-			
		207	3.9.2-50	2009/3/27	Y	N	N		-	DCD_3.9.2-50	2	2
		207	3.9.2-51	2009/3/27	Y	N	N		-	DCD_3.9.2-51	2	2
		207	3.9.2-52	2009/3/27	Y	N	N		-	DCD_3.9.2-52	2	2
		207	3.9.2-53	2009/3/27	Y	N	N		-	DCD_3.9.2-53	2	2
		207	3.9.2-54	2009/3/27	N	N	N		-	-	N/A	N/A
		207	3.9.2-55	2009/3/27	Y	N	N		-	DCD_3.9.2-55	2	2
		207	3.9.2-56	2009/3/27	Y	N	N		-	DCD_3.9.2-56	2	2
		207	3.9.2-57	2009/3/27	Y	N	N		-	DCD_3.9.2-57	2	2
		207	3.9.2-58	2009/3/27	Y	N	N		-	DCD_3.9.2-58	2	2
		207	3.9.2-59	2009/3/27	N	N	N		-	-	N/A	N/A
			3.9.2-60						-			
			3.9.2-61						-			
			3.9.2-62						-			
			3.9.2-63						-			
			3.9.2-64						-			
			3.9.2-65						-			
			3.9.2-66						-			
			3.9.2-67						-			
			3.9.2-68						-			
			3.9.2-69						-			

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		208	3.9.2-70	2009/3/27	Y	N	N		-	DCD_3.9.2-70	3	2
3.9.3	ASME Code Class 1, 2, and 3	209	03.09.03-1	2009/4/30	N	N	N		-	-	N/A	N/A
	Components,	209	03.09.03-2	2009/4/30	Y	N	N		-	DCD_03.09.03-2	TDB	
		209	03.09.03-3	2009/4/30	Y	N	N		-	DCD_03.09.03-3	TDB	
		209	03.09.03-4	2009/4/30	Y	N	N		-	DCD_03.09.03-4	TDB	
		209	03.09.03-5	2009/4/30	Y	N	N		-	DCD_03.09.03-5	TDB	
		209	03.09.03-6	2009/4/30	Y	N	N		-	DCD_03.09.03-6	TDB	
		209	03.09.03-7	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-8	2009/4/30	Y	N	N		-	DCD_03.09.03-8	TDB	
		209	03.09.03-9	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-10	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-11	2009/4/30	Y	N	N		-	DCD_03.09.03-11	TDB	
		209	03.09.03-12	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-13	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-14	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-15	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-16	2009/4/30	Y	N	N		-	DCD_03.09.03-16	TDB	
		209	03.09.03-17	2009/4/30	Y	N	N		-	DCD_03.09.03-17	TDB	
		209	03.09.03-18	2009/4/30	Y	N	N		-	DCD_03.09.03-18	TDB	
		209	03.09.03-19	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-20	2009/4/30	Y	N	N		-	DCD_03.09.03-20	TDB	
		209	03.09.03-21	2009/4/30	N	N	N		-	-	N/A	N/A
		209	03.09.03-22	2009/4/30	Y	N	N		-	DCD_03.09.03-22	TDB	
	and Component Supports,	209	03.09.03-23	2009/4/30	Y	N	N		-	DCD_03.09.03-23	TDB	
	and Core Support Structures											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.9.4	Control Rod Drive Systems	107	1293-01	2008/12/19	Y	N	N	fin.	-	DCD_1293-01	0	2
		107	1293-02	2008/12/19	Y	N	N	fin.	-	DCD_1293-02	0	2
		107	1293-03	2008/12/19	Y	N	N	fin.	-	DCD_1293-03	0	2
		107	1293-04	2008/12/19	N	N	N	fin.	-	N/A	N/A	
		107	1293-05	2008/12/19	N	N	N	fin.	-	N/A	N/A	
		107	1293-06	2008/12/19	N	N	N	fin.	-	N/A	N/A	
		107	1293-07	2008/12/19	N	N	N	fin.	-	N/A	N/A	
		107	1293-08	2008/12/19	Y	N	N	fin.	-	DCD_1293-08	0	2
		107	1293-09	2008/12/19	N	N	N	fin.	-	N/A	N/A	
		107	1293-10	2008/12/19	Y	N	N	fin.	-	DCD_1293-10	0	2
3.9.5	Reactor Pressure Vessel Internals											
3.9.6	Functional Design, Qualification,	288	03.09.06-01	2009/5/23	Y	N	N		-	DCD_03.09.06-01	TBD	
	and Inservice Testing Programs	288	03.09.06-02	2009/5/23	Y	N	N		-	DCD_03.09.06-02	TBD	
	for Pumps, Valves,	288	03.09.06-03	2009/5/23	Y	N	N		-	DCD_03.09.06-03	TBD	
	and Dynamic Restraints	288	03.09.06-04	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-05	2009/5/23	Y	N	N		-	DCD_03.09.06-05	TBD	
		288	03.09.06-06	2009/5/23	Y	N	N		-	DCD_03.09.06-06	TBD	
		288	03.09.06-07	2009/5/23	Y	N	N		-	DCD_03.09.06-07	TBD	
		288	03.09.06-08	2009/5/23	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		288	03.09.06-09	2009/5/23	Y	N	N		-	DCD_03.09.06-09	TBD	
		288	03.09.06-10	2009/5/23	Y	N	N		-	DCD_03.09.06-10	TBD	
		288	03.09.06-11	2009/5/23	Y	N	N		-	DCD_03.09.06-11	TBD	
		288	03.09.06-12	2009/5/23	Y	N	N		-	DCD_03.09.06-12	TBD	
		288	03.09.06-13	2009/5/23	Y	N	N		-	DCD_03.09.06-13	TBD	
		288	03.09.06-14	2009/5/23	Y	N	N		-	DCD_03.09.06-14	TBD	
		288	03.09.06-15	2009/5/23	Y	N	N		-	DCD_03.09.06-15	TBD	
		288	03.09.06-16	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-17	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-18	2009/5/23	Y	N	N		-	DCD_03.09.06-18	TBD	
		288	03.09.06-19	2009/5/23	Y	N	N		-	DCD_03.09.06-19	TBD	
		288	03.09.06-20	2009/5/23	Y	N	N		-	DCD_03.09.06-20	TBD	
		288	03.09.06-21	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-22	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-23	2009/5/23	Y	N	N		-	DCD_03.09.06-23	TBD	
		288	03.09.06-24	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-25	2009/5/23	Y	N	N		-	DCD_03.09.06-25	TBD	
		288	03.09.06-26	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-27	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-28	2009/5/23	Y	N	N		-	DCD_03.09.06-28	TBD	
		288	03.09.06-29	2009/5/23	Y	N	N		-	DCD_03.09.06-29	TBD	
		288	03.09.06-30	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-31	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-32	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-33	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-34	2009/5/23	Y	N	N		-	DCD_03.09.06-34	TBD	
		288	03.09.06-35	2009/5/23	Y	N	N		-	DCD_03.09.06-35	TBD	
		288	03.09.06-36	2009/5/23	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				
		288	03.09.06-37	2009/5/23	Y	N	N		-	DCD_03.09.06-37	TBD	
		288	03.09.06-38	2009/5/23	Y	N	N		-	DCD_03.09.06-38	TBD	
		288	03.09.06-39	2009/5/23	Y	N	N		-	DCD_03.09.06-39	TBD	
		288	03.09.06-40	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-41	2009/5/23	Y	N	N		-	DCD_03.09.06-41	TBD	
		288	03.09.06-42	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-43	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-44	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-45	2009/5/23	N	N	N		-	-	N/A	N/A
		288	03.09.06-46	2009/5/23	Y	N	N		-	DCD_03.09.06-46	TBD	
		288	03.09.06-47	2009/5/23	Y	N	N		-	DCD_03.09.06-47	TBD	
		288	03.09.06-48	2009/5/23	N	N	N		-	-	N/A	N/A
3.10	Seismic/Dynamic Qual	216	USAPWR-3.10-1	2009/3/25	N	N	N		-	-	N/A	N/A
	of Mech/Elec Eqmt	216	USAPWR-3.10-2	2009/3/25	Y	N	N		-	DCD_USAPWR-3.10-2	2	2
		216	USAPWR-3.10-3	2009/4/22	Y	N	N		-	DCD_USAPWR-3.10-3	TBD	
		216	USAPWR-3.10-4	2009/3/25	N	N	N		-	-	N/A	N/A
		216	USAPWR-3.10-5	2009/4/22	N	N	N		-	-	N/A	N/A
		216	USAPWR-3.10-6	2009/3/25	Y	N	N		-	DCD_USAPWR-3.10-6	2	2
		216	USAPWR-3.10-7	2009/4/22	N	N	N		-	-	N/A	N/A
		216	USAPWR-3.10-8	2009/3/25	Y	N	N		-	DCD_USAPWR-3.10-8	2	2
		216	USAPWR-3.10-9	2009/4/22	N	N	N		-	-	N/A	N/A
		-	-	-	-	-	-	-	COL3.10(10) deleted	MAP-03-014	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
3.11	Environmental Qual											
	of Mech/Elec Eqmt											
3.12	ASME Code Class 1, 2, and 3	259	03.12-1	2009/4/17	Y	N	N		-	DCD_03.12-1	TBD	
	Piping Systems,	259	03.12-2	2009/4/17	Y	N	N		-	DCD_03.12-2	TBD	
	Piping Components	259	03.12-3	2009/4/17	Y	N	N		-	DCD_03.12-3	TBD	
	and their Associated Supports	259	03.12-4	2009/4/17	Y	N	N		-	DCD_03.12-4	TBD	
		260	03.12-5	2009/4/17	Y	N	N		-	DCD_03.12-5	TBD	
		260	03.12-6	2009/4/17	Y	N	N		-	DCD_03.12-6	TBD	
		260	03.12-7	2009/4/17	N	N	N		-	-	N/A	N/A
		260	03.12-8	2009/4/17	Y	N	N		-	DCD_03.12-8	TBD	
		260	03.12-9	2009/4/17	N	N	N		-	-	N/A	N/A
		260	03.12-10	2009/4/17	N	N	N		-	-	N/A	N/A
		260	03.12-11	2009/4/17	N	N	N		-	-	N/A	N/A
		260	03.12-12	2009/4/17	N	N	N		-	-	N/A	N/A
		260	03.12-13	2009/4/17	Y	N	N		-	DCD_03.12-13	TBD	
		260	03.12-14	2009/4/17	Y	N	N		-	DCD_03.12-14	TBD	
		260	03.12-15	2009/4/17	Y	N	N		-	DCD_03.12-15	TBD	
		260	03.12-16	2009/4/17	N	N	N		-	-	N/A	N/A
3.13	Threaded Fasteners -	273	3.13-1	2009/4/9	Y	N	N		-	DCD_3.13-1	TBD	
	ASME Code Class 1, 2, and 3	273	3.13-2	2009/4/9	Y	N	N		-	DCD_3.13-2	TBD	
		273	3.13-3	2009/4/9	Y	N	N		-	DCD_3.13-3	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		273	3.13-4	2009/4/9	Y	N	N		-	DCD_3.13-4	TBD	
		273	3.13-5	2009/4/9	Y	N	N		-	DCD_3.13-5	TBD	
		-	-	-	-	-	-	-	COL3.13(1) deleted	MAP-03-015	TBD	
		-	-	-	-	-	-	-	COL3.13(2) deleted	MAP-03-016	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
4.2	Fuel System Design	129	04.02-1	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-2	2009/1/30	Y	N	N		-	DCD_04.02-2	2	2
		129	04.02-3	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-4	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-5	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-6	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-7	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-8	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-9	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-10	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-11	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-12	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-13	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-14	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-15	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-16	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-17	2009/1/30	Y	N	N		-	DCD_04.02-17	2	2
		129	04.02-18	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-19	2009/1/30	N	N	N		-	-	N/A	N/A
		129	04.02-20	2009/1/30	N	N	N		-	-	N/A	N/A
4.3	Nuclear Design	202	04.03-1	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-2	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-3	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-4	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-5	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-6	2009/3/27	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				
		202	04.03-7	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-8	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-9	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-10	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-11	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-12	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-13	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-14	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-14A	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-14B	2009/3/27	N	N	N		-	-	N/A	N/A
		202	04.03-14C	2009/3/27	N	N	N		-	-	N/A	N/A
		256	04.03-15	2009/3/30	N	N	N		-	-	N/A	N/A
		256	04.03-16	2009/3/30	N	N	N		-	-	N/A	N/A
		256	04.03-17	2009/3/30	N	N	N		-	-	N/A	N/A
			04.03-18						-			
		256	04.03-19	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-20	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-21	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-22	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-23	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-24	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-25	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-26	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-27	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-28	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-29	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-30	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-31	2009/3/30	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				
		257	04.03-32	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-33	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-34	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-35	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-36	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-37	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-38	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-39	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-40	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-41	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-42	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-43	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-44	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-45	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-45A	2009/3/30	N	N	N		-	-	N/A	N/A
		257	04.03-45B	2009/3/30	N	N	N		-	-	N/A	N/A
4.4	Thermal and Hydraulic Design											
4.5.1	Control Rod Drive	268	4.5.1-1	2009/4/28	Y	N	N		-	DCD_4.5.1-1	TDB	
	Structural Materials	268	4.5.1-2	2009/4/28	Y	N	N		-	DCD_4.5.1-2	TDB	
		268	4.5.1-3	2009/4/28	N	N	N		-	-	N/A	N/A
		268	4.5.1-4	2009/4/28	Y	N	N		-	DCD_4.5.1-4	TDB	
		268	4.5.1-5	2009/4/28	Y	N	N		-	DCD_4.5.1-5	TDB	

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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				
		268	4.5.1-6	2009/4/28	Y	N	N		-	DCD_4.5.1-6	TDB	
		268	4.5.1-7	2009/4/28	Y	N	N		-	DCD_4.5.1-7	TDB	
4.5.2	Reactor Internal and Core	269	4.5.2-1	2009/5/13	Y	N	N		-	DCD_4.5.2-1	TDB	
	Support Structure Materials	269	4.5.2-2	2009/5/13	Y	N	N		-	DCD_4.5.2-2	TDB	
		269	4.5.2-3	2009/5/13	Y	N	N		-	DCD_4.5.2-3	TDB	
		269	4.5.2-4	2009/5/13	N	N	N		-	-	N/A	N/A
		269	4.5.2-5	2009/5/13	Y	N	N		-	DCD_4.5.2-5	TDB	
4.6	Functional Design	316	4.6.1	2009/5/20	N	N	N		-	-	N/A	N/A
	of Control Rod Drive System	316	4.6.2	2009/5/20	Y	N	N		-	DCD_4.6.2	TDB	
		316	4.6.3	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.4	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.5	2009/5/20	Y	N	N		-	DCD_4.6.5	TDB	
		316	4.6.6	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.7	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.8	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.9	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.10	2009/5/20	N	N	N		-	-	N/A	N/A
		316	4.6.11	2009/5/20	Y	N	N		-	DCD_4.6.11	TDB	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
6.1.1	Engineered Safety Features											
	Materials											
		-	-	-	-	-	-	-	COL 6.1(1) deleted	MAP-06-001	TBD	
		-	-	-	-	-	-	-	COL 6.1(2) deleted	MAP-06-002	TBD	
		-	-	-	-	-	-	-	COL 6.1(3) deleted	MAP-06-003	1	2
		-	-	-	-	-	-	-	COL 6.1(4) deleted	MAP-06-004	1	2
		-	-	-	-	-	-	-	COL 6.1(5) deleted	MAP-06-005	TBD	
6.1.2	Protective Coating Systems (Paints)											
	Organic Materials											
6.2.1	Containment Functional Design	110	06.02.01-1	2008/12/26	N	N	N	fin.	-	-	N/A	N/A
	Organic Materials	126	06.02.01-2	2009/1/29	Y	N	N		-	DCD_06.02.01-2	1	2
		126	06.02.01-3	2009/3/19	N	N	N		-	-	N/A	N/A
		126	06.02.01-4	2009/3/19	N	N	N		-	-	N/A	N/A
		126	06.02.01-5	2009/3/19	N	N	N		-	-	N/A	N/A
		126	06.02.01-6	2009/4/21	N	N	N		-	-	N/A	N/A
		331	06.02.01-7	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-8	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-9	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-10	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-11	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-12	2009/5/26	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				
		331	06.02.01-13	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-14	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-15	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-16	2009/5/26	N	N	N		-	-	N/A	N/A
		331	06.02.01-17	2009/5/26	N	N	N		-	-	N/A	N/A
		-	-	-	-	-	-	-	COL 6.2(1) deleted	MAP-06-006	TBD	
6.2.1.2	Subcompartment Analysis	6	06.02.01.02-1	2008/6/27	Y	N	N	fin.	-	DCD_06.02.01.02-1	1	2
		111	06.02.01.02-2	2009/2/17	N	N	N		-	-	N/A	N/A
			06.02.01.02-3						-			
			06.02.01.02-4						-			
		111	06.02.01.02-5	2008/1/16	N	N	N		-	DCD_06.02.01.02-5	N/A	N/A
		111	06.02.01.02-6	2008/1/16	N	N	N		-	DCD_06.02.01.02-6	N/A	N/A
			06.02.01.02-7						-			
			06.02.01.02-8						-			
		111	06.02.01.02-9	2008/1/16	N	N	N		-	-	N/A	N/A
			06.02.01.02-10						-			
			06.02.01.02-11						-			
		111	06.02.01.02-12	2008/1/16	N	N	N		-	-	N/A	N/A
		111	06.02.01.02-13	2008/1/16	N	N	N		-	-	N/A	N/A
		111	06.02.01.02-14	2008/1/16	N	N	N		-	-	N/A	N/A
6.2.1.3	Mass and Energy Release											
	Analysis for Postulated											

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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				
	Loss-of-Coolant Accidents											
6.2.1.4	Mass and Energy Release	112	06.02.01.04-1	2008/12/26	N	N	N	fin.	-	-	N/A	N/A
	Analysis for Postulated	113	06.02.01.04-2	2009/1/15	N	N	N		-	-	N/A	N/A
	Secondary System Pipe Ruptures	114	06.02.01.04-3	2008/12/26	N	N	N	fin.	-	-	N/A	N/A
	(LOCAs)											
6.2.1.5	Min. Containment Pressure	115	06.02.01.05-1	2008/12/25	Y	N	N	fin.	-	DCD_06.02.01.05-1	1	2
	Analysis for	116	06.02.01.05-2	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
	for Emergency Core Cooling Sys.	117	06.02.01.05-3	2009/1/15	N	N	N		-	-	N/A	N/A
	Performance Capability Studies	118	06.02.01.05-4	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
		119	06.02.01.05-5	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
		120	06.02.01.05-6	2008/12/25	Y	N	N	fin.	-	DCD_06.02.01.05-6	1	2
		121	06.02.01.05-7	2008/12/25	Y	N	N	fin.	-	DCD_06.02.01.05-7	1	2
		122	06.02.01.05-8	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
6.2.2	Containment	45	06.02.02-1	2008/8/26	Y	N	N	fin.	-	DCD_06.02.02-1	-	1
	Heat Removal Systems	45	06.02.02-2	2008/8/26	N	N	N	fin.	-	-	N/A	N/A
		45	06.02.02-3	2008/8/26	N	N	N	fin.	-	-	N/A	N/A
		45	06.02.02-4	2008/8/26	N	N	N	fin.	-	-	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				
		84	06.02.02-5	2008/11/7	Y	N	N	fin.	-	DCD_06.02.02-5	1	2
		84	06.02.02-6	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		84	06.02.02-7	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		84	06.02.02-8	2008/11/7	Y	N	N	fin.	-	DCD_06.02.02-8	1	2
		84	06.02.02-9	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		85	06.02.02-10	2009/11/12	Y	N	N	fin.	-	DCD_06.02.02-10	1	2
		85	06.02.02-11	2009/11/12	N	N	N	fin.	-	-	N/A	N/A
		263	06.02.02-12	2009/3/31	Y	N	N		-	DCD_06.02.02-12	3	2
		263	06.02.02-13	2009/3/31	N	N	N		-	-	N/A	N/A
		263	06.02.02-14	2009/3/31	N	N	N		-	-	N/A	N/A
		263	06.02.02-15	2009/3/31	N	N	N		-	-	N/A	N/A
		278	06.02.02-16	2009/4/10	Y	N	N		-	DCD_06.02.02-16	TBD	
		330	06.02.02-17	2009/5/18	N	N	N		-	-	N/A	N/A
		349	06.02.02-18	2009/5/12	Y	N	N		-	DCD_06.02.02-18	TBD	
		-	-	-	-	-	-	-	COL 6.2(9) deleted	MAP-06-007	TBD	
6.2.4	Containment Isolation System	57	06.02.04-1	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-2	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-3	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-4	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-5	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-5	1	2
		57	06.02.04-6	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-7	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-8	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-8	1	2
		57	06.02.04-9	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-9	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		57	06.02.04-10	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-11	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-12	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-13	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-14	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-15	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-16	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-16	1	2
		57	06.02.04-17	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-18	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-18	1	2
		57	06.02.04-19	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-19	1	2
		57	06.02.04-20	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-20	1	2
		57	06.02.04-21	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-21	1	2
		57	06.02.04-22	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-22	1	2
		57	06.02.04-23	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-23	1	2
		57	06.02.04-24	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-25	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-26	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-27	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-28	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-29	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-30	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-31	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-31	1	2
		57	06.02.04-32	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		57	06.02.04-33	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-33	1	2
		57	06.02.04-34	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-34	1	2
		57	06.02.04-35	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-35	1	2
		57	06.02.04-36	2008/9/22	Y	N	N	fin.	-	DCD_06.02.04-36	1	2
		279	06.02.04-37	2009/4/8	Y	N	N		-	DCD_06.02.04-37	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		279	06.02.04-38	2009/4/8	Y	N	N		-	DCD_06.02.04-38	TBD	
		279	06.02.04-39	2009/4/8	Y	N	N		-	DCD_06.02.04-39	TBD	
		279	06.02.04-40	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-41	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-42	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-43	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-44	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-45	2009/4/8	N	N	N		-	-	N/A	N/A
		279	06.02.04-46	2009/4/8	Y	N	N		-	DCD_06.02.04-46	TBD	
		279	06.02.04-47	2009/4/8	Y	N	N		-	DCD_06.02.04-47	TBD	
		279	06.02.04-48	2009/4/8	Y	N	N		-	DCD_06.02.04-48	TBD	
		279	06.02.04-49	2009/4/8	Y	N	N		-	DCD_06.02.04-49	TBD	
		-	-	-	-	-	-	-	COL 6.2(6) deleted	MAP-06-008	TBD	
6.2.5	Combustible Gas Control	62	06.02.05-1	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
	in Containment	62	06.02.05-2	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-3	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-4	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-5	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-6	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-7	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-8	2008/10/1	Y	Y	N	fin.	-			
				2009/1/9	Y	N	N	fin.	-	DCD_06.02.05-8	1	2
		62	06.02.05-9	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-10	2008/10/1	N	N	N	fin.	-	-	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				
		62	06.02.05-11	2008/10/1	N	N	N	fin.	-			
				2009/1/9	Y	Y	N	fin.	-	DCD_06.02.05-11	1	2
		62	06.02.05-12	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-13	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-14	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-15	2008/10/1	N	N	N	fin.	-			
				2009/1/9	Y	N	N	fin.	-	DCD_06.02.05-15	1	2
		62	06.02.05-16	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-17	2008/10/1	Y	N	N	fin.	-	DCD_06.02.05-17		
		62	06.02.05-18	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-19	2008/10/1	N	N	N	fin.	-	-	N/A	N/A
		62	06.02.05-20	2008/10/1	Y	N	N	fin.	-	DCD_06.02.05-20	1	2
		62	06.02.05-21	2008/10/1	Y	N	N	fin.	-	DCD_06.02.05-21	1	2
		270	06.02.05-22	2009/6/5	Y	N	N		-	DCD_06.02.05-22	TBD	
		270	06.02.05-23	2009/6/5	Y	N	N		-	DCD_06.02.05-23	TBD	
		270	06.02.05-24	2009/6/5	Y	N	N		-	DCD_06.02.05-24	TBD	
		270	06.02.05-25	2009/6/5	Y	N	N		-	DCD_06.02.05-25	TBD	
		270	06.02.05-26	2009/6/5	Y	N	N		-	DCD_06.02.05-26	TBD	
		270	06.02.05-27	2009/6/5	Y	N	N		-	DCD_06.02.05-27	TBD	
		270	06.02.05-28	2009/6/5	Y	N	N		-	DCD_06.02.05-28	TBD	
		270	06.02.05-29	2009/6/5	Y	N	N		-	DCD_06.02.05-29	TBD	
		270	06.02.05-30	2009/6/5	Y	N	N		-	DCD_06.02.05-30	TBD	
		270	06.02.05-31	2009/6/5	Y	N	N		-	DCD_06.02.05-31	TBD	
		270	06.02.05-32	2009/6/5	Y	N	N		-	DCD_06.02.05-32	TBD	
		270	06.02.05-33	2009/6/5	Y	N	N		-	DCD_06.02.05-33	TBD	
		270	06.02.05-34	2009/6/5	Y	N	N		-	DCD_06.02.05-34	TBD	

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		-	-	-	-	-	-	-	COL 6.2(7) deleted	MAP-06-009	1	2
6.2.6	Containment Leakage Testing	50	06.02.06-1	2008/9/17	N	N	N	fin.	-	-	N/A	N/A
		50	06.02.06-2	2008/9/17	N	N	N	fin.	-			
				2009/1/9	Y	Y	N	fin.	-	DCD_06.02.06-2	1	2
		50	06.02.06-3	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-3	3	2
		50	06.02.06-4	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-4	3	2
		50	06.02.06-5	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-5	1	2
		50	06.02.06-6	2008/9/17	N	N	N	fin.	-	-	N/A	N/A
		50	06.02.06-7	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-7	3	2
		50	06.02.06-8	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-8	3	2
		50	06.02.06-9	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-9	1	2
		50	06.02.06-10	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-10	3	2
		50	06.02.06-11	2008/9/17	N	N	N	fin.	-	-	N/A	N/A
		50	06.02.06-12	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-12	3	2
		50	06.02.06-13	2008/9/17	Y	N	N	fin.	-	DCD_06.02.06-13	2	2
		267	06.02.06-14	2009/4/6	Y	N	N		-	DCD_06.02.06-14	TBD	
		267	06.02.06-15	2009/4/6	Y	N	N		-	DCD_06.02.06-15	TBD	
		267	06.02.06-16	2009/4/6	Y	N	N		-	DCD_06.02.06-16	TBD	
		267	06.02.06-17	2009/4/6	Y	N	N		-	DCD_06.02.06-17	TBD	
		267	06.02.06-18	2009/4/6	Y	N	N		-	DCD_06.02.06-18	TBD	
		267	06.02.06-19	2009/4/6	N	N	N		-	-	N/A	N/A
		267	06.02.06-20	2009/4/6	-	-	-	-	-	Question Deleted	-	-
		267	06.02.06-21	2009/4/6	N	N	N		-	-	N/A	N/A
		267	06.02.06-22	2009/4/6	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		-	-	-	-	-	-	-	COL 6.2(8) revised	MAP-06-010	TBD	
6.2.7	Fracture Prevention											
	of											
	Containment Pressure Boundary											
6.3	Emergency Core Cooling System											
		-	-	-	-	-	-	-	COL 6.3(3) deleted	MAP-06-011	TBD	
		-	-	-	-	-	-	-	COL 6.3(4) deleted	MAP-06-012	TBD	
		-	-	-	-	-	-	-	COL 6.3(6) deleted	MAP-06-013	TBD	
6.4	Control Room Habitability System	26	06.04-1	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		26	06.04-2	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-1	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-2	2008/9/16	Y	N	N	fin.	-	DCD_06.04-2	1	2
		49	06.04-3	2008/9/16	Y	N	N	fin.	-			
		99	06.04-3	2008/12/8	Y	N	N	fin.	-	DCD_06.04-3	1	2
		49	06.04-4	2008/9/16	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		49	06.04-5	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-6	2008/9/16	Y	N	N	fin.	-	DCD_06.04-6	1	2
		49	06.04-7	2008/9/16	Y	N	N	fin.	-	DCD_06.04-7	1	2
		49	06.04-8	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-9	2008/9/16	Y	N	N	fin.	-	DCD_06.04-9	1	2
		49	06.04-10	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-11	2008/9/16	Y	N	N	fin.	-	DCD_06.04-11	1	2
		49	06.04-12	2008/9/16	Y	N	N	fin.	-	DCD_06.04-12	3	2
		49	06.04-13	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-14	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-15	2008/9/16	Y	N	N	fin.	-	DCD_06.04-15	1	2
		49	06.04-16	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-17	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-18	2008/9/16	Y	Y	N	fin.	-	-		
				2009/1/9	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-19	2008/9/16	Y	N	N	fin.	-	DCD_06.04-19	1	2
		49	06.04-20	2008/9/16	Y	N	N	fin.	-	DCD_06.04-20	1	2
		49	06.04-21	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-22	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		49	06.04-23	2008/9/16	Y	N	N	fin.	-	DCD_06.04-23	1	2
		49	06.04-24	2008/9/16	N	N	N	fin.	-	-	N/A	N/A
		-	-	-	-	-	-	-	COL 6.4(2) revised	MAP-06-014	1	2
		-	-	-	-	-	-	-	COL 6.4(4) deleted	MAP-06-015	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
6.5.1	ESF Atmosphere Cleanup Systems	73	06.05.01-1/6.5.1-1	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-1	3	2
		73	06.05.01-1/6.5.1-2	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-3	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-4	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-5	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-5	3	2
		73	06.05.01-1/6.5.1-6	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-6	3	2
		73	06.05.01-1/6.5.1-7	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-7	3	2
		73	06.05.01-1/6.5.1-8	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-9	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-10	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-11	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-12	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-12	3	2
		73	06.05.01-1/6.5.1-13	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-13	3	2
		73	06.05.01-1/6.5.1-14	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-14	3	2
		73	06.05.01-1/6.5.1-15	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-16	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-16	3	2
		73	06.05.01-1/6.5.1-17	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-17	3	2
		73	06.05.01-1/6.5.1-18	2008/10/24	Y	N	N	fin.	-	DCD_06.05.01-18	3	2
		73	06.05.01-1/6.5.1-19	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		73	06.05.01-1/6.5.1-20	2008/10/24	N	N	N	fin.	-	-	N/A	N/A
		82	06.05.01-2	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
		300	06.05.01-3	2009/5/15	Y	N	N		-	DCD_06.05.01-3	TBD	
		300	06.05.01-4	2009/5/15	Y	N	N		-	DCD_06.05.01-4	TBD	
		300	06.05.01-5	2009/5/15	Y	N	N		-	DCD_06.05.01-5	TBD	
		300	06.05.01-6	2009/5/15	N	N	N		-	-	N/A	N/A
		300	06.05.01-7	2009/5/15	Y	N	N		-	DCD_06.05.01-7	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		-	-	-	-	-	-	-	COL 6.5(4) deleted	MAP-06-016	2	2
6.5.2	Containment Spray	234	06.05.02-1	2009/4/22	N	N	N		-	-	N/A	N/A
	as a Fission Product	234	06.05.02-2	2009/3/24	N	N	N		-	-	N/A	N/A
	Cleanup System	234	06.05.02-3	2009/4/22	N	N	N		-	-	N/A	N/A
		234	06.05.02-4	2009/3/24	N	N	N		-	-	N/A	N/A
6.5.3	Fission Product	37	06.05.03-1	2008/9/5	Y	N	N	fin.	-	DCD_06.05.03-1	1	2
	Control Systems and Structures	83	06.05.03-2	2008/11/7	N	N	N	fin.	-	-	N/A	N/A
6.5.5	Pressure Suppression Pool											
	as a											
	Fission Product Cleanup System											
6.6	Inservice Inspection and Testing	232	06.06-1	2009/3/26	Y	N	N		-	DCD_06.06-1	2	2
	of Class 2 and 3 Components											
		-	-	-	-	-	-	-	COL 6.6(1) revised	MAP-06-017	TBD	
		-	-	-	-	-	-	-	COL 6.6(2) revised	MAP-06-018	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
6.6.2	Inservice Inspection and Testing of Class 2 and 3 Components	233	06.06-2	2009/4/16	Y	N	N		-	DCD_06.06-2	TBD	
6.6.3	Inservice Inspection and Testing of Class 2 and 3 Components	241	06.06-3	2009/4/16	Y	N	N		-	DCD_06.06-3	TBD	
6.6.4	Inservice Inspection and Testing of Class 2 and 3 Components	258	06.06-4	2009/4/16	Y	N	N		-	DCD_06.06-4	TBD	

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.1.1	Criticality Safety of Fresh and	155	09.01.01.-1	2009/2/10	N	N	N		-	-	N/A	N/A
	Spent Fuel Storage and Handling	155	09.01.01-2	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-3	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-4	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-5	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-6	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-7	2009/2/10	N	N	N		-	-	N/A	N/A
		155	09.01.01-8	2009/2/10	N	N	N		-	-	N/A	N/A
		247	09.01.01-9	2009/3/30	Y	N	N		-	DCD_09.01.01-9	TBD	
		247	09.01.01-10	2009/3/30	Y	Y	N		-	DCD_09.01.01-10	TBD	
9.1.2	New and Spent Fuel Storage	132	09.01.02-01	2009/1/29	Y	N	N		-	DCD_09.01.02-01	2	2
		132	09.01.02-02	2009/1/29	N	N	N		-	-	N/A	N/A
		132	09.01.02-03	2009/1/29	N	N	N		-	-	N/A	N/A
		132	09.01.02-04	2009/1/29	Y	N	N		-	DCD_09.01.02-04	1	2
		132	09.01.02-05	2009/1/29	Y	N	N		-	DCD_09.01.02-05	1	2
		132	09.01.02-06	2009/1/29	N	N	N		-	-	N/A	N/A
		132	09.01.02-07	2009/1/29	Y	N	N		-	DCD_09.01.02-07	3	2
		132	09.01.02-08	2009/1/29	Y	N	N		-	DCD_09.01.02-08	2	2
		132	09.01.02-09	2009/1/29	Y	N	N		-	DCD_09.01.02-09	2	2
		132	09.01.02-10	2009/1/29	N	N	N		-	-	N/A	N/A
		132	09.01.02-11	2009/1/29	Y	N	N		-	DCD_09.01.02-11	1	2
		132	09.01.02-12	2009/1/29	N	N	N		-	-	N/A	N/A
		132	09.01.02-13	2009/1/29	Y	N	N		-	DCD_09.01.02-13	1	2
		132	09.01.02-14	2009/1/29	Y	N	N		-	DCD_09.01.02-14	1	2
		132	09.01.02-15	2009/1/29	Y	N	N		-	DCD_09.01.02-15	3	2
		132	09.01.02-16	2009/1/29	Y	N	N		-	DCD_09.01.02-16	3	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		132	09.01.02-17	2009/1/29	Y	N	N		-	DCD_09.01.02-17	3	2
		248	09.01.02-18	2009/3/30	Y	N	N		-	DCD_09.01.02-18	2	2
		248	09.01.02-19	2009/3/30	Y	N	N		-	DCD_09.01.02-19	2	2
		248	09.01.02-20	2009/3/30	N	N	N		-	-	N/A	N/A
9.1.3	Spent Fuel Pool Cooling and	131	09.01.03-01	2009/1/29	Y	N	N		-	DCD_09.01.03-01	1	2
	Cleanup System	132	09.01.03-02	2009/1/29	Y	N	N		-	DCD_09.01.03-02	1	2
		132	09.01.03-03	2009/1/29	Y	N	N		-	DCD_09.01.03-03	1	2
		201	09.01.03-04	2009/3/26	N	N	N		-	-	N/A	N/A
		201	09.01.03-05	2009/3/26	Y	N	N		-	DCD_09.01.03-05	2	2
		360	09.01.03-6	2009/5/27	Y	N	N		-	DCD_09.01.03-6	TBD	
9.1.4	Light Load Handling System	200	09.01.04-01	2009/4/23	Y	N	N		-	DCD_09.01.04-01	TBD	
	(Related to Refueling)	200	09.01.04-02	2009/4/23	Y	N	N		-	DCD_09.01.04-02	TBD	
		200	09.01.04-03	2009/4/23	Y	N	N		-	DCD_09.01.04-03	TBD	
		200	09.01.04-04	2009/4/23	Y	N	N		-	DCD_09.01.04-04	TBD	
		200	09.01.04-05	2009/4/23	Y	N	N		-	DCD_09.01.04-05	TBD	
		200	09.01.04-06	2009/4/23	Y	N	N		-	DCD_09.01.04-06	TBD	
		200	09.01.04-07	2009/4/23	N	N	N		-	-	N/A	N/A
		200	09.01.04-08	2009/4/23	Y	N	N		-	DCD_09.01.04-08	TBD	
		200	09.01.04-09	2009/4/23	Y	N	N		-	DCD_09.01.04-09	TBD	
		200	09.01.04-10	2009/4/23	N	N	N		-	-	N/A	N/A
		200	09.01.04-11	2009/4/23	Y	N	N		-	DCD_09.01.04-11	TBD	
		200	09.01.04-12	2009/4/23	Y	N	N		-	DCD_09.01.04-12	TBD	
		200	09.01.04-13	2009/4/23	Y	N	N		-	DCD_09.01.04-13	TBD	
		200	09.01.04-14	2009/4/23	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				
		200	09.01.04-15	2009/4/23	N	N	N		-	-	N/A	N/A
9.1.5	Overhead Heavy Load	292	9.1.5-01	2009/5/25	Y	N	N		-	DCD_9.1.5-01	TBD	
	Handling Systems	292	9.1.5-02	2009/5/25	Y	N	N		-	DCD_9.1.5-02	TBD	
		292	9.1.5-03	2009/5/25	Y	N	N		-	DCD_9.1.5-03	TBD	
		292	9.1.5-04	2009/5/25	Y	N	N		-	DCD_9.1.5-04	TBD	
		292	9.1.5-05	2009/5/25	N	N	N		-	-	N/A	N/A
		292	9.1.5-06	2009/5/25	Y	N	N		-	DCD_9.1.5-06	TBD	
		292	9.1.5-07	2009/5/25	Y	N	N		-	DCD_9.1.5-07	TBD	
		292	9.1.5-08	2009/5/25	Y	N	N		-	DCD_9.1.5-08	TBD	
		292	9.1.5-09	2009/5/25	Y	N	N		-	DCD_9.1.5-09	TBD	
		292	9.1.5-10	2009/5/25	Y	N	N		-	DCD_9.1.5-10	TBD	
		292	9.1.5-11	2009/5/25	Y	N	N		-	DCD_9.1.5-11	TBD	
		292	9.1.5-12	2009/5/25	Y	Y	N		-	DCD_9.1.5-12	TBD	
		292	9.1.5-13	2009/5/25	Y	Y	N		-	DCD_9.1.5-13	TBD	
9.2.1	Station Service Water System	203	09.02.01-1	2009/3/25	N	N	N		-	-	N/A	N/A
		203	09.02.01-2	2009/3/25	Y	N	N		-	DCD_09.02.01-2	3	2
9.2.2	Reactor Auxiliary	252	09.02.02-1	2009/3/30	N	N	N		-	-	N/A	N/A
	Cooling Water Systems	252	09.02.02-2	2009/3/30	Y	N	N		-	DCD_09.02.02-2	3	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.2.4	Potable and Sanitary Water Systems	125	09.02.04-1	2009/1/20	Y	Y	N		-	DCD_09.02.04-1	1	2
		125	09.02.04-2	2009/1/20	Y	Y	N		-	DCD_09.02.04-2	1	2
		125	09.02.04-3	2009/1/20	Y	N	N		-	DCD_09.02.04-3	1	2
9.2.5	Ultimate Heat Sink	286	09.02.05-1	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-2	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-3	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-4	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-5	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-6	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-7	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-8	2009/5/12	N	N	N		-	-	N/A	N/A
		286	09.02.05-9	2009/5/12	N	N	N		-	-	N/A	N/A
9.2.6	Condensate Storage Facilities	157	09.02.06-1	2009/2/5	Y	N	N		-	DCD_09.02.06-1	1	2
		157	09.02.06-2	2009/2/5	Y	N	N		-	DCD_09.02.06-2	1	2
		351	09.02.06-2	2009/6/9	N	N	N		-	-	N/A	N/A
9.3.1	Compressed Air System	109	09.03.01-1	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
		109	09.03.01-2	2008/12/25	Y	N	N	fin.	-	DCD_09.03.01-2	3	2
		109	09.03.01-3	2008/12/25	Y	N	N	fin.	-	DCD_09.03.01-3	3	2
		109	09.03.01-4	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
		109	09.03.01-5	2008/12/25	Y	N	N	fin.	-	DCD_09.03.01-5	1	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.3.2	Process and Post-accident	294	09.03.02-1	2009/5/13	Y	N	N		-	DCD_09.03.02-1	TBD	
	Sampling Systems	294	09.03.02-2	2009/5/13	N	N	N		-	-	N/A	N/A
		294	09.03.02-3	2009/5/13	N	N	N		-	-	N/A	N/A
		294	09.03.02-4	2009/5/13	Y	N	N		-	DCD_09.03.02-4	TBD	
		294	09.03.02-5	2009/5/13	N	N	N		-	-	N/A	N/A
		294	09.03.02-6	2009/5/13	Y	N	N		-	DCD_09.03.02-6	TBD	
		294	09.03.02-7	2009/5/13	N	N	N		-	-	N/A	N/A
		294	09.03.02-8	2009/5/13	Y	N	N		-	DCD_09.03.02-8	TBD	
		325	09.03.02-9	2009/5/19	Y	N	N			DCD_09.03.02-9	TBD	
		346	09.03.02-10	2009/6/8	N	N	N		-	-	N/A	N/A
9.3.3	Equipment and Floor	299	09.03.03-1	2009/5/13	Y	N	N		-	DCD_09.03.03-1	TBD	
	Drainage System	299	09.03.03-2	2009/5/13	Y	N	N		-	DCD_09.03.03-2	TBD	
		299	09.03.03-3	2009/5/13	Y	N	N		-	DCD_09.03.03-3	TBD	
		299	09.03.03-4	2009/5/13	N	N	N		-	-	N/A	N/A
		299	09.03.03-5	2009/5/13	N	N	N		-	-	N/A	N/A
		299	09.03.03-6	2009/5/13	Y	N	N		-	DCD_09.03.03-6	TBD	
		299	09.03.03-7	2009/5/13	Y	N	N		-	DCD_09.03.03-7	TBD	
		299	09.03.03-8	2009/5/13	Y	N	N		-	DCD_09.03.03-8	TBD	
		299	09.03.03-9	2009/5/13	Y	N	N		-	DCD_09.03.03-9	TBD	
		299	09.03.03-10	2009/5/13	Y	N	N		-	DCD_09.03.03-10	TBD	
		299	09.03.03-11	2009/5/13	Y	N	N		-	DCD_09.03.03-11	TBD	
		299	09.03.03-12	2009/5/13	Y	N	N		-	DCD_09.03.03-12	TBD	
		299	09.03.03-13	2009/5/13	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				
		299	09.03.03-14	2009/5/13	N	N	N		-	-	N/A	N/A
9.3.4	Chemical and Volume Control Syste	280	09.03.04-1	2009/4/14	N	N	N		-	-	N/A	N/A
	(PWR)	280	09.03.04-2	2009/4/14	N	N	N		-	-	N/A	N/A
	(Including Boron Recovery System)	280	09.03.04-3	2009/4/14	N	N	N		-	-	N/A	N/A
		280	09.03.04-4	2009/4/14	N	N	N		-	-	N/A	N/A
		280	09.03.04-5	2009/4/14	N	N	N		-	-	N/A	N/A
		280	09.03.04-6	2009/4/14	N	N	N		-	-	N/A	N/A
9.4.1	Control Room Area	63	09.04.01-1	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
	Ventilation System	63	09.04.01-2	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-2	1	2
		63	09.04.01-3	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-3	1	2
		63	09.04.01-4	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-5	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-6	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-7	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-8	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-8	1	2
		63	09.04.01-9	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-10	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-10	1	2
		63	09.04.01-11	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-12	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-13	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-14	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-15	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-16	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-16	1	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		63	09.04.01-17	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-18	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-19	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-19	3	2
		63	09.04.01-20	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-21	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-21	1	2
		63	09.04.01-22	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-22	3	2
		63	09.04.01-23	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-23	1	2
		63	09.04.01-24	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-24	3	2
		63	09.04.01-25	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-25	1	2
		63	09.04.01-26	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-26	1	2
		63	09.04.01-27	2008/10/3	Y	Y	N	fin.	-	DCD_09.04.01-27	3	2
		63	09.04.01-28	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-28	3	2
		63	09.04.01-29	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-30	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		63	09.04.01-31	2008/10/3	Y	N	N	fin.	-	DCD_09.04.01-31	1	2
		63	09.04.01-32	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
9.4.2	Spent Fuel Pool Area	65	09.04.02-1/9.4.2-1	2008/10/3	Y	N	N	fin.	-	DCD_09.04.02-1	3	2
	Ventilation System	65	09.04.02-1/9.4.2-2	2008/10/3	Y	N	N	fin.	-	DCD_09.04.02-2	3	2
		65	09.04.02-1/9.4.2-3	2008/10/3	Y	N	N	fin.	-	DCD_09.04.02-3	3	2
		65	09.04.02-1/9.4.2-4	2008/10/3	Y	N	N	fin.	-	DCD_09.04.02-4	3	2
		65	09.04.02-1/9.4.2-5	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		65	09.04.02-1/9.4.2-6	2008/10/3	N	N	N	fin.	-	-	N/A	N/A
		328	09.04.02-2	2009/5/21	N	N	N		-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		328	09.04.02-3	2009/5/21	Y	N	N		-	DCD_09.04.02-3	TBD	
9.4.3	Auxiliary and Radwaste Area	68	09.04.03-1/9.4.3-1	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-1	3	2
	Ventilation System	68	09.04.03-1/9.4.3-2	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		68	09.04.03-1/9.4.3-3	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-3	3	2
		68	09.04.03-1/9.4.3-4	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		68	09.04.03-1/9.4.3-5	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-5	3	2
		68	09.04.03-1/9.4.3-6	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		68	09.04.03-1/9.4.3-7	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-7	3	2
		68	09.04.03-1/9.4.3-8	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-8	3	2
		68	09.04.03-1/9.4.3-9	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-9	3	2
		68	09.04.03-1/9.4.3-10	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-10	3	2
		68	09.04.03-1/9.4.3-11	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		68	09.04.03-1/9.4.3-12	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-12	3	2
		68	09.04.03-1/9.4.3-13	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		68	09.04.03-1/9.4.3-14	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-14	3	2
		68	09.04.03-1/9.4.3-15	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-15	3	2
		68	09.04.03-1/9.4.3-16	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-16	3	2
		68	09.04.03-1/9.4.3-17	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-17	3	2
		68	09.04.03-1/9.4.3-18	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-18	3	2
		68	09.04.03-1/9.4.3-19	2008/10/8	Y	N	N	fin.	-	DCD_09.04.03-19	3	2

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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	66	09.04.04-1/9.4.4.1	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		66	09.04.04-1/9.4.4-2	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		66	09.04.04-1/9.4.4-3	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		66	09.04.04-1/9.4.4-4	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		66	09.04.04-1/9.4.4-5	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		66	09.04.04-1/9.4.4-6	2008/9/22	N	N	N	fin.	-	-	N/A	N/A
		67	09.04.04-1/9.4.4-7	2008/10/6	Y	N	N	fin.	-	DCD_09.04.04-7	1	2
		67	09.04.04-1/9.4.4-8	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		67	09.04.04-1/9.4.4-9	2008/10/6	Y	N	N	fin.	-	DCD_09.04.04-9	1	2
		67	09.04.04-1/9.4.4-10	2008/10/6	Y	N	N	fin.	-	DCD_09.04.04-10	3	2
		67	09.04.04-1/9.4.4-12	2008/10/6	Y	N	N	fin.	-	DCD_09.04.04-11	1	2
		67	09.04.04-1/9.4.4-13	2008/10/6	Y	N	N	fin.	-	DCD_09.04.04-12	1	2
		341	09.04.04-3	2009/6/1	N	N	N		-	-	N/A	N/A
9.4.5	Engineered Safety Feature	64	09.04.05-1/9.4.5-1	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
	Ventilation System	64	09.04.05-1/9.4.5-2	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-3	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-4	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-5	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-6	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-6	3	2
		64	09.04.05-1/9.4.5-7	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-7	3	2
		64	09.04.05-1/9.4.5-8	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-8	3	2
		64	09.04.05-1/9.4.5-9	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-10	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-10	3	2
		64	09.04.05-1/9.4.5-11	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-11	3	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		64	09.04.05-1/9.4.5-12	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-12	3	2
		64	09.04.05-1/9.4.5-13	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-13	3	2
		64	09.04.05-1/9.4.5-14	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-14	3	2
		64	09.04.05-1/9.4.5-15	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-15	3	2
		64	09.04.05-1/9.4.5-16	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-16	3	2
		64	09.04.05-1/9.4.5-17	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-17	3	2
		64	09.04.05-1/9.4.5-18	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-19	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-19	3	2
		64	09.04.05-1/9.4.5-20	2008/10/6	Y	N	N	fin.	-	DCD_09.04.05-20	3	2
		64	09.04.05-1/9.4.5-21	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-22	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-23	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
		64	09.04.05-1/9.4.5-24	2008/10/6	N	N	N	fin.	-	-	N/A	N/A
9.5.1	Fire Protection Program	30	09.05.01-1	2008/9/3	N	N	N	fin.	-	-	N/A	N/A
		30	09.05.01-2	2008/9/3	Y	N	N	fin.	-	DCD_09.05.01-2	1	2
		30	09.05.01-3	2008/9/3	N	N	N	fin.	-	-	N/A	N/A
		30	09.05.01-4	2008/9/3	Y	N	N	fin.	-	DCD_09.05.01-4	1	2
		30	09.05.01-5	2008/9/3	Y	Y	N	fin.	-	DCD_09.05.01-5	1	2
		30	09.05.01-6	2008/9/3	Y	Y	N	fin.	-	DCD_09.05.01-6	1	2
		30	09.05.01-7	2008/9/3	Y	Y	N	fin.	-	DCD_09.05.01-7	1	2
		30	09.05.01-8	2008/9/3	Y	N	N	fin.	-	DCD_09.05.01-8	2	2
		30	09.05.01-9	2008/9/3	Y	N	N	fin.	-	DCD_09.05.01-9	3	2
		30	09.05.01-10	2008/9/3	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		30	09.05.01-11	2008/9/3	Y	N	N	fin.	-	DCD_09.05.01-11	3	2
		30	09.05.01-12	2008/9/3	N	N	N	fin.	-	-	N/A	N/A
		30	09.05.01-13	2008/9/3	N	N	N	fin.	-	-	N/A	N/A
		87	09.05.01-14	2008/11/26	Y	N	N	fin.	-	DCD_09.05.01-14	3	2
		87	09.05.01-15	2008/11/26	Y	Y	N	fin.	-	DCD_09.05.01-15	1	2
		87	09.05.01-16	2008/11/26	N	N	N	fin.	-	-	N/A	N/A
		87	09.05.01-17	2008/11/26	Y	N	N	fin.	-	DCD_09.05.01-17	1	2
9.5.2	Communications Systems	74	09.05.02-1	2008/10/22	N	N	N	fin.	-	-	N/A	N/A
		74	09.05.02-2	2008/10/22	N	N	N	fin.	-	-	N/A	N/A
		74	09.05.02-3	2008/10/22	N	N	N	fin.	-	-	N/A	N/A
		74	09.05.02-4	2008/10/22	Y	N	N	fin.	-	DCD_09.05.02-4	2	2
		74	09.05.02-5	2008/10/22	Y	N	N	fin.	-	DCD_09.05.02-5	1	2
		139	09.05.02-6	2009/2/20	Y	N	N		-	DCD_09.05.02-6	1	2
		139	09.05.02-7	2009/2/20	Y	N	N		-	DCD_09.05.02-7	1	2
		139	09.05.02-8	2009/2/20	Y	N	N		-	DCD_09.05.02-8	1	2
		139	09.05.02-9	2009/2/20	Y	N	N		-	DCD_09.05.02-9	1	2
		139	09.05.02-10	2009/2/20	Y	N	N		-	DCD_09.05.02-10	1	2
9.5.3	Lighting Systems	34	09.05.03-1	2008/9/8	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		34	09.05.03-2	2008/9/8	Y	N	N	fin.	-	DCD_09.05.03-2	1	2
		34	09.05.03-3	2008/9/8	Y	N	N	fin.	-	DCD_09.05.03-3	1	2
		34	09.05.03-4	2008/9/8	Y	N	N	fin.	-	DCD_09.05.03-4	2	2
		34	09.05.03-5	2008/9/8	N	N	N	fin.	-	-	N/A	N/A
		34	09.05.03-6	2008/9/8	Y	N	N	fin.		DCD_09.05.03-6	2	2
		80	09.05.03-7/9.5.3-05 S02	2008/11/5	Y	N	N	fin.	-	CD_09.05.03-7(05_S02	2	2
		80	09.05.03-7/9.5.3-08 S02	2008/11/5	Y	N	N	fin.	-	CD_09.05.03-7(08_S02	2	2
		80	09.05.03-7/9.5.3-10 S02	2008/11/5	Y	N	N	fin.	-	CD_09.05.03-7(10_S02	2	2
9.5.4	Emergency Diesel Engine Fuel											
	Oil Storage and Transfer System											
9.5.5	Emergency Diesel Engine											
	Cooling Water System											
9.5.6	Emergency Diesel Engine											
	Starting System											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
9.5.7	Emergency Diesel Engine											
	Lubrication System											
9.5.8	Emergency Diesel Engine											
	Combustion Air Intake and											
	Exhaust System											

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
10.2	Turbine Generator	237	10.02-1	2009/3/25	N	N	N		-	-	N/A	N/A
		237	10.02-2	2009/3/25	N	N	N		-	-	N/A	N/A
		237	10.02-3	2009/3/25	N	N	N		-	-	N/A	N/A
		237	10.02-4	2009/3/25	Y	N	N		-	DCD_10.02-4	3	2
10.2.3	Turbine Rotor Integrity	199	10.02.03-1,10.2.3-1	2009/3/10	Y	N	N		-	DCD_10.02.03-1	2	2
		199	10.02.03-2,10.2.3-2	2009/3/10	Y	N	N		-	DCD_10.02.03-2	2	2
		199	10.02.03-3,10.2.3-3	2009/3/10	Y	N	N		-	DCD_10.02.03-3	3	2
		199	10.02.03-4,10.2.3-4	2009/3/10	Y	N	N		-	DCD_10.02.03-4	2	2
		199	10.02.03-5,10.2.3-5	2009/3/10	Y	N	N		-	DCD_10.02.03-5	2	2
		199	10.02.03-6,10.2.3-6	2009/3/10	Y	N	N		-	DCD_10.02.03-6	2	2
		199	10.02.03-7,10.2.3-7	2009/3/10	Y	N	N		-	DCD_10.02.03-7	2	2
10.3	Main Steam Supply System	329	10.3-1	2009/5/26	Y	Y	N		-	DCD_10.3-1	TBD	
		329	10.3-2	2009/5/26	N	N	N		-	-	N/A	N/A
		329	10.3-3	2009/5/26	N	N	N		-	-	N/A	N/A
10.3.6	Steam and Feedwater System Materials	250	10.03.06-1	2009/4/1	Y	N	N		-	DCD_10.03.06-1	2	2
		250	10.03.06-2	2009/4/1	Y	N	N		-	DCD_10.03.06-2	2	2
		250	10.03.06-3	2009/4/1	Y	N	N		-	DCD_10.03.06-3	TBD	
		250	10.03.06-4	2009/4/1	Y	N	N		-	DCD_10.03.06-4	2	2
		250	10.03.06-5	2009/4/1	Y	N	N		-	DCD_10.03.06-5	2	2
		250	10.03.06-6	2009/4/1	Y	Y	N		-	DCD_10.03.06-6	2	2
		250	10.03.06-7	2009/4/1	Y	N	N		-	DCD_10.03.06-7	2	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
10.4.1	Main Condensers	245	10.4.1-1	2009/3/30	Y	N	N		-	DCD_10.4.1-1	2	2
		245	10.4.1-2	2009/3/30	N	N	N		-	-	N/A	N/A
		245	10.4.1-3	2009/3/30	N	N	N		-	-	N/A	N/A
10.4.2	Main Condenser Evacuation System	246	10.4.2-1	2009/3/30	N	N	N		-	-	N/A	N/A
10.4.3	Turbine Gland Sealing System	236	10.4.3-1	2009/3/24	Y	N	N		-	DCD_10.4.3-1	2	2
		236	1..4.3-2	2009/3/24	N	N	N		-	-	N/A	N/A
10.4.4	Turbine Bypass System	159	10.4.4-1	2009/2/20	N	N	N		-	-	N/A	N/A
10.4.5	Circulating Water System											

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
10.4.6	Condensate Cleanup System	235	10.04.06-1	2009/3/25	N	N	N		-	-	N/A	N/A
		235	10.04.06-2	2009/3/25	N	N	N		-	-	N/A	N/A
		235	10.04.06-3	2009/3/25	N	N	N		-	-	N/A	N/A
		235	10.04.06-4	2009/3/25	Y	N	N		-	DCD_10.04.06-4	2	2
		235	10.04.06-5	2009/3/25	N	N	N		-	-	N/A	N/A
10.4.7	Condensate and Feedwater System	124	10.4.7-1	2008/12/25	N	N	N	fin.	-	-	N/A	N/A
				2009/6/1	Y	Y	N		-	DCD_10.4.7-1	TBD	
10.4.8	Steam Generator Blowdown System (PWR)	251	10.04.08-1	2009/4/1	Y	N	N		-	DCD_10.04.08-1	2	2
		251	10.04.08-2	2009/4/1	N	N	N		-	-	N/A	N/A
		251	10.04.08-3	2009/4/1	N	N	N		-	-	N/A	N/A
		251	10.04.08-4	2009/4/1	Y	N	N		-	DCD_10.04.08-4	2	2
		251	10.04.08-5	2009/4/1	Y	N	N		-	DCD_10.04.08-5	2	2
		251	10.04.08-6	2009/4/1	Y	N	N		-	DCD_10.04.08-6	2	2
		251	10.04.08-7	2009/4/1	Y	N	N		-	DCD_10.04.08-7	2	2
		251	10.04.08-8	2009/4/1	Y	N	N		-	DCD_10.04.08-8	TBD	
10.4.9	Auxiliary Feedwater System (PWR)	160	10.04.09-1	2009/2/20	Y	N	N		-	DCD_10.04.09-1	1	2
		160	10.04.09-2	2009/2/20	N	N	N		-	-	N/A	N/A
		160	10.04.09-3	2009/2/20	Y	N	N		-	DCD_10.04.09-3	3	2
		160	10.04.09-4	2009/2/20	Y	N	N		-	DCD_10.04.09-4	1	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		160	10.04.09-4	2009/6/1	Y	Y	N		-	DCD_10.04.09-4	TBD	
		160	10.04.09-5	2009/2/20	Y	N	N		-	DCD_10.04.09-5	1	2
		160	10.04.09-6	2009/2/20	Y	N	N		-	DCD_10.04.09-6	1	2
		160	10.04.09-7	2009/2/20	Y	N	N		-	DCD_10.04.09-7	1	2
		160	10.04.09-8	2009/2/20	Y	N	N		-	DCD_10.04.09-8	1	2
		160	10.04.09-9	2009/2/20	Y	N	N		-	DCD_10.04.09-9	1	2
		160	10.04.09-10	2009/2/20	Y	N	N		-	DCD_10.04.09-10	1	2
		160	10.04.09-11	2009/2/20	Y	N	N		-	DCD_10.04.09-11	1	2
		160	10.04.09-12	2009/2/20	N	N	N		-	-	N/A	N/A
		160	10.04.09-13	2009/2/20	Y	N	N		-	DCD_10.04.09-13	1	2
		160	10.04.09-14	2009/2/20	Y	N	N		-	DCD_10.04.09-14	1	2
		160	10.04.09-15	2009/2/20	Y	N	N		-	DCD_10.04.09-15	1	2
		160	10.04.09-16	2009/2/20	N	N	N		-	-	N/A	N/A
		160	10.04.09-17	2009/2/20	N	N	N		-	-	N/A	N/A
		160	10.04.09-18	2009/2/20	Y	N	N		-	DCD_10.04.09-18	3	2
		160	10.04.09-19	2009/2/20	Y	N	N		-	DCD_10.04.09-19	3	2
		160	10.04.09-20	2009/2/20	Y	N	N		-	DCD_10.04.09-20	3	2
		160	10.04.09-21	2009/2/20	Y	N	N		-	DCD_10.04.09-21	1	2
		160	10.04.09-22	2009/2/20	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				
11.1	Source Terms	9	11.01-1	2008/7/18	N	N	N	fin.	-	-	N/A	N/A
		29	11.01-2	2008/8/6	N	N	N	fin.	-	-	N/A	N/A
11.2	Liquid Waste Management System	164	11.02-1	2009/2/18	Y	N	N		-	DCD_11.02-1	1	2
		164	11.02-2	2009/2/18	N	N	N		-	-	N/A	N/A
		164	11.02-3	2009/2/18	Y	N	N		-	DCD_11.02-3	1	2
		164	11.02-4	2009/2/18	N	N	N		-	-	N/A	N/A
		164	11.02-5	2009/2/18	Y	N	N		-	DCD_11.02-5	1	2
		164	11.02-6	2009/2/18	N	N	N		-	-	N/A	N/A
		164	11.02-7	2009/2/18	N	N	N		-	-	N/A	N/A
		186	11.02-8	2009/3/10	Y	N	N		-	DCD_11.02-8	TBD	
		186	11.02-9	2009/3/10	Y	N	N		-	DCD_11.02-9	2	2
		186	11.02-10	2009/3/10	Y	N	N		-	DCD_11.02-10	2	2
		186	11.02-11	2009/3/10	N	N	N		-	-	N/A	N/A
		186	11.02-12	2009/3/10	N	N	N		-	-	N/A	N/A
		186	11.02-13	2009/3/10	Y	N	N		-	DCD_11.02-13	2	2
		186	11.02-14	2009/3/10	N	N	N		-	-	N/A	N/A
		186	11.02-15	2009/3/10	N	N	N		-	-	N/A	N/A
		186	11.02-16	2009/3/10	Y	N	N		-	DCD_11.02-16	2	2
		186	11.02-17	2009/3/10	N	N	N		-	-	N/A	N/A
11.3	Gaseous Waste	188	11.03-1	2009/3/10	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				
	Management System	188	11.03-2	2009/3/10	N	N	N		-	-	N/A	N/A
		188	11.03-3	2009/3/10	Y	N	N		-	DCD_11.03-3	2	2
		188	11.03-4	2009/3/10	Y	N	N		-	DCD_11.03-4	2	2
		188	11.03-5	2009/3/10	N	N	N		-	-	N/A	N/A
		189	11.03-6	2009/3/10	N	N	N		-	-	N/A	N/A
		189	11.03-7	2009/3/10	N	N	N		-	-	N/A	N/A
		189	11.03-8	2009/3/10	N	N	N		-	-	N/A	N/A
		189	11.03-9	2009/3/10	N	N	N		-	-	N/A	N/A
		189	11.03-10	2009/3/10	Y	N	N		-	DCD_11.03-10	2	2
		189	11.03-11	2009/3/10	N	N	N		-	-	N/A	N/A
								-	COL 11.3(5) deleted	MAP-11-001	TBD	
11.4	Solid Waste Management System	185	11.04-1	2009/3/11	Y	N	N		-	DCD_11.04-1	2	2
		185	11..04-2	2009/3/11	Y	N	N		-	DCD_11..04-2	2	2
		185	11.04-3	2009/3/11	Y	N	N		-	DCD_11.04-3	2	2
		185	11.04-4	2009/3/11	N	N	N		-	-	N/A	N/A
		185	11.04-5	2009/3/11	Y	N	N		-	DCD_11.04-5	2	2
		187	11.04-6	2009/3/11	Y	N	N		-	DCD_11.04-6	3	2
		187	11.04-7	2009/3/11	Y	N	N		-	DCD_11.04-7	2	2
		187	11.04-8	2009/3/11	Y	N	N		-	DCD_11.04-8	2	2
		187	11.04-9	2009/3/11	Y	N	N		-	DCD_11.04-9	2	2
		187	11.04-10	2009/3/11	Y	N	N		-	DCD_11.04-10	2	2
		187	11.04-11	2009/3/11	Y	N	N		-	DCD_11.04-11	2	2
		187	11.04-12	2009/3/11	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				
		187	11.04-13	2009/3/11	Y	N	N		-	DCD_11.04-13	2	2
		187	11.04-14	2009/3/11	N	N	N		-	-	N/A	N/A
		187	11.04-15	2009/3/11	Y	N	N		-	DCD_11.04-15	2	2
		187	11.04-16	2009/3/11	Y	N	N		-	DCD_11.04-16	2	2
		187	11.04-17	2009/3/11	Y	N	N		-	DCD_11.04-17	2	2
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	130	11.05-1	2009/1/30	N	N	N		-	-	N/A	N/A
		130	11.05-2	2009/1/30	Y	N	N		-	DCD_11.05-2	2	2
		130	11.05-3	2009/1/30	N	N	N		-	-	N/A	N/A
		130	11.05-4	2009/1/30	N	N	N		-	-	N/A	N/A
		249	11.05-5	2009/3/31	N	N	N		-	-	N/A	N/A
		249	11.05-6	2009/4/13	N	N	N		-	-	N/A	N/A
		249	11.05-7	2009/4/13	N	N	N		-	-	N/A	N/A
		249	11.05-8	2009/4/13	N	N	N		-	-	N/A	N/A
		249	11.05-9	2009/3/31	N	N	N		-	-	N/A	N/A
		249	11.05-10	2009/3/31	Y	N	N		-	DCD_11.05-10	2	2
		249	11.05-11	2009/3/31	Y	N	N		-	DCD_11.05-11	2	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
12.1	Assuring that	89	12.01-1	2008/11/26	Y	N	N	fin.	-	DCD_12.01-1	0	2
	Occupational Radiation Exposures	89	12.01-2	2008/11/26	Y	Y	N	fin.	-	DCD_12.01-2	0	2
	Are As Low As											
	Is Reasonably Achievable											
12.2	Radiation Sources	128	12.02-1	2009/1/21	N	N	N		-	-	N/A	N/A
		128	12.02-2	2009/1/21	N	N	N		-	-	N/A	N/A
		128	12.02-3	2009/1/21	N	N	N		-	-	N/A	N/A
		140	12.02-4	2009/2/6	Y	N	N		-	DCD_12.02-4	1	2
		141	12.02-5	2009/2/6	N	N	N		-	-	N/A	N/A
		141	12.02-6	2009/2/6	Y	N	N		-	DCD_12.02-6	1	2
		142	12.02-7	2009/2/6	Y	N	N		-	DCD_12.02-7	1	2
		142	12.02-8	2009/2/6	Y	N	N		-	DCD_12.02-8	1	2
		142	12.02-9	2009/2/6	N	N	N		-	-	N/A	N/A
		143	12.02-10	2009/2/6	Y	N	N		-	DCD_12.02-10	1	2
		143	12.02-11	2009/2/6	N	N	N		-	-	N/A	N/A
		144	12.02-12	2009/2/6	Y	N	N		-	DCD_12.02-12	1	2
		145	12.02-13	2009/2/6	N	N	N		-	-	N/A	N/A
		168	12.02-14	2009/3/4	Y	N	N		-	DCD_12.02-14	2	2
		169	12.02-15	2009/2/27	Y	Y	N	fin.	-	DCD_12.02-15	1	2
		179	12.02-16	2009/3/3	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				
12.3-	Radiation Protection	90	12.03-12.04-1	2008/11/26	Y	N	N	fin.	-	DCD_12.03-12.04-1	0	2
12.4	Design Features	91	12.03-12.04-2	2009/1/9	Y	Y	N	fin.	-	DCD_12.03-12.04-2	0	2
		147	12.03-12.04-3	2009/2/6	Y	N	N		-	DCD_12.03-12.04-3	1	2
		147	12.03-12.04-4	2009/2/6	Y	N	N		-	DCD_12.03-12.04-4	1	2
		147	12.03-12.04-5	2009/2/6	Y	N	N		-	DCD_12.03-12.04-5	1	2
		170	12.03-12.04-6	2009/3/4	N	N	N		-	-	N/A	N/A
		171	12.03-12.04-7	2009/3/3	N	N	N		-	-	N/A	N/A
		171	12.03-12.04-8	2009/3/3	N	N	N		-	-	N/A	N/A
		171	12.03-12.04-9	2009/3/3	Y	N	N		-	DCD_12.03-12.04-9	3	2
		172	12.03-12.04-10	2009/3/3	Y	N	N		-	DCD_12.03-12.04-10	3	2
		173	12.03-12.04-11	2009/2/27	Y	N	N		-	DCD_12.03-12.04-11	1	2
		174	12.03-12.04-12	2009/2/27	Y	N	N		-	DCD_12.03-12.04-12	1	2
		262	12.03-12.04-13	2009/5/7	Y	N	N		-	DCD_12.03-12.04-13		
		262	12.03-12.04-14	2009/5/7	Y	N	N		-	DCD_12.03-12.04-14		
		262	12.03-12.04-15	2009/5/7	Y	N	N		-	DCD_12.03-12.04-15		
		262	12.03-12.04-16	2009/5/7	N	N	N		-	-	N/A	N/A
		262	12.03-12.04-17	2009/5/7	Y	N	N		-	DCD_12.03-12.04-17		
		262	12.03-12.04-18	2009/5/7	N	N	N		-	-	N/A	N/A
		262	12.03-12.04-19	2009/5/7	N	N	N		-	-	N/A	N/A
		262	12.03-12.04-20	2009/5/7	N	N	N		-	-	N/A	N/A
12.5	Operational Radiation											
	Protection Program											

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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 -	7	14.02-1	2008/6/27	Y	N	N	fin.	-	DCD_14.02-1	-	1
	Design Certification and	12	14.02-2	2008/7/18	Y	N	N	fin.	-	DCD_14.02-2	-	1
	New License Applicants	12	14.02-3	2008/7/18	Y	N	N	fin.	-	DCD_14.02-3	-	1
		12	14.02-4	2008/7/18	Y	N	N	fin.	-	DCD_14.02-4	-	1
		12	14.02-5	2008/7/18	Y	N	N	fin.	-	DCD_14.02-5	-	1
		12	14.02-6	2008/7/18	N	N	N	fin.	-	-	N/A	N/A
		12	14.02-7	2008/7/18	Y	N	N	fin.	-	DCD_14.02-7	-	1
		27	14.02-8	2008/7/31	Y	Y	N	fin.	-	DCD_14.02-8	0	2
		27	14.02-9	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		28	14.02-10	2008/7/31	Y	N	N	fin.	-	DCD_14.02-10	-	1
		28	14.02-11	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		28	14.02-12	2008/7/31	Y	N	N	fin.	-	DCD_14.02-12	-	1
		28	14.02-13	2008/7/31	Y	N	N	fin.	-	DCD_14.02-13	-	1
		28	14.02-14	2008/7/31	Y	N	N	fin.	-	DCD_14.02-14	-	1
		28	14.02-15	2008/7/31	Y	N	N	fin.	-	DCD_14.02-15	-	1
		28	14.02-16	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		28	14.02-17	2008/7/31	Y	N	N	fin.	-	DCD_14.02-17	-	1
		28	14.02-18	2008/7/31	Y	N	N	fin.	-	DCD_14.02-18	-	1
		28	14.02-19	2008/7/31	Y	N	N	fin.	-	DCD_14.02-19	0	2
		28	14.02-20	2008/7/31	Y	N	N	fin.	-	DCD_14.02-20	-	1
		28	14.02-21	2008/7/31	N	N	N	fin.	-	-	N/A	N/A
		28	14.02-22	2008/7/31	Y	N	N	fin.	-	DCD_14.02-22	-	1
		31	14.02-23	2008/8/29	Y	Y	N	fin.	-	DCD_14.02-23	0	2
		31	14.02-24	2008/8/29	Y	N	N	fin.	-	DCD_14.02-24	0	2
		33	14.02-25	2008/9/4	Y	N	N	fin.	-	DCD_14.02-25	0	2
		33	14.02-26	2008/9/4	Y	N	N	fin.	-	DCD_14.02-26	0	2
		33	14.02-27	2008/9/4	Y	N	N	fin.	-	DCD_14.02-27	0	2
		33	14.02-28	2008/9/4	N	N	N	fin.	-	-	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				
		33	14.02-29	2008/9/4	Y	N	N	fin.	-	DCD_14.02-29	0	2
		33	14.02-30	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-31	2008/9/4	Y	N	N	fin.	-	DCD_14.02-31	0	2
		33	14.02-32	2008/9/4	Y	N	N	fin.	-	DCD_14.02-32	0	2
		33	14.02-33	2008/9/4	Y	N	N	fin.	-	DCD_14.02-33	0	2
		33	14.02-34	2008/9/4	Y	N	N	fin.	-	DCD_14.02-34	0	2
		33	14.02-35	2008/9/4	Y	N	N	fin.	-	DCD_14.02-35	0	2
		33	14.02-36	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-37	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-38	2008/9/4	N	N	N	fin.	-	DCD_14.02-38	-	1
		33	14.02-39	2008/9/4	Y	N	N	fin.	-	DCD_14.02-39	0	2
		33	14.02-40	2008/9/4	Y	N	N	fin.	-	DCD_14.02-40	0	2
		33	14.02-41	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-42	2008/9/4	Y	N	N	fin.	-	DCD_14.02-42	0	2
		33	14.02-43	2008/9/4	Y	N	N	fin.	-	DCD_14.02-43	0	2
		33	14.02-44	2008/9/4	Y	N	N	fin.	-	DCD_14.02-44	0	2
		33	14.02-45	2008/9/4	Y	N	N	fin.	-	DCD_14.02-45	0	2
		33	14.02-46	2008/9/4	Y	N	N	fin.	-	DCD_14.02-46	0	2
		33	14.02-47	2008/9/4	Y	N	N	fin.	-	DCD_14.02-47	0	2
		33	14.02-48	2008/9/4	Y	N	N	fin.	-	DCD_14.02-48	0	2
		33	14.02-49	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-50	2008/9/4	Y	N	N	fin.	-	DCD_14.02-50	0	2
		33	14.02-51	2008/9/4	Y	N	N	fin.	-	DCD_14.02-51	0	2
		33	14.02-52	2008/9/4	N	N	N	fin.	-	DCD_14.02-52	N/A	N/A
		33	14.02-53	2008/9/4	Y	N	N	fin.	-	DCD_14.02-53	0	2
		33	14.02-54	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-55	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-56	2008/9/4	Y	N	N	fin.	-	DCD_14.02-56	0	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		33	14.02-57	2008/9/4	Y	N	N	fin.	-	DCD_14.02-57	0	2
		33	14.02-58	2008/9/4	Y	N	N	fin.	-	DCD_14.02-58	0	2
		33	14.02-59	2008/9/4	Y	N	N	fin.	-	DCD_14.02-59	0	2
		33	14.02-60	2008/9/4	Y	N	N	fin.	-	DCD_14.02-60	0	2
		33	14.02-61	2008/9/4	Y	N	N	fin.	-	DCD_14.02-61	0	2
		33	14.02-62	2008/9/4	Y	N	N	fin.	-	DCD_14.02-62	0	2
		33	14.02-63	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-64	2008/9/4	Y	N	N	fin.	-	DCD_14.02-64	0	2
		33	14.02-65	2008/9/4	Y	N	N	fin.	-	DCD_14.02-65	0	2
		33	14.02-66	2008/9/4	Y	N	N	fin.	-	DCD_14.02-66	0	2
		33	14.02-67	2008/9/4	Y	N	N	fin.	-	DCD_14.02-67	0	2
		33	14.02-68	2008/9/4	Y	N	N	fin.	-	DCD_14.02-68	0	2
		33	14.02-69	2008/9/4	Y	N	N	fin.	-	DCD_14.02-69	0	2
		33	14.02-70	2008/9/4	Y	N	N	fin.	-	DCD_14.02-70	0	2
		33	14.02-71	2008/9/4	Y	N	N	fin.	-	DCD_14.02-71	0	2
		33	14.02-72	2008/9/4	Y	N	N	fin.	-	DCD_14.02-72	0	2
		33	14.02-73	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-74	2008/9/4	Y	N	N	fin.	-	DCD_14.02-74	0	2
		33	14.02-75	2008/9/4	Y	N	N	fin.	-	DCD_14.02-75	0	2
		33	14.02-76	2008/9/4	Y	N	N	fin.	-	DCD_14.02-76	0	2
		33	14.02-77	2008/9/4	Y	N	N	fin.	-	DCD_14.02-77	0	2
		33	14.02-78	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-79	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-80	2008/9/4	Y	N	N	fin.	-	DCD_14.02-80	0	2
		33	14.02-81	2008/9/4	N	N	N	fin.	-	-	N/A	N/A
		33	14.02-82	2008/9/4	Y	N	N	fin.	-	DCD_14.02-82	0	2
		33	14.02-83	2008/9/4	Y	N	N	fin.	-	DCD_14.02-83	0	2
		33	14.02-84	2008/9/4	Y	N	N	fin.	-	DCD_14.02-84	0	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		33	14.02-85	2008/9/4	Y	N	N	fin.	-	DCD_14.02-85	0	2
		58	14.02-86	2008/9/18	Y	N	N	fin.	-	DCD_14.02-86	0	2
		70	14.02-87	2008/9/25	Y	N	N	fin.	-	DCD_14.02-87	0	2
				2009/3/30	Y	Y	N	fin.		DCD_14.02-87	1	2
		78	14.02-88	2008/10/16	Y	N	N	fin.	-	DCD_14.02-88	0	2
		78	14.02-89	2008/10/16	Y	N	N	fin.	-	DCD_14.02-89	0	2
		93	14.02-90	2008/12/5	Y	Y	N	fin.	-	DCD_14.02-90	0	2
		102	14.02-91	2008/12/18	N	N	N	fin.	-	-	N/A	N/A
		102	14.02-92	2008/12/18	Y	N	N	fin.	-	DCD_14.02-92	0	2
		102	14.02-93	2008/12/18	Y	N	N	fin.	-	DCD_14.02-93	0	2
		102	14.02-94	2008/12/18	Y	N	N	fin.	-	DCD_14.02-94	0	2
		102	14.02-95	2008/12/18	N	N	N	fin.	-	-	N/A	N/A
		102	14.02-96	2008/12/18	Y	N	N	fin.	-	DCD_14.02-96	0	2
		102	14.02-97	2008/12/18	Y	N	N	fin.	-	DCD_14.02-97	0	2
		102	14.02-98	2008/12/18	Y	N	N	fin.	-	DCD_14.02-98	0	2
		102	14.02-99	2008/12/18	Y	N	N	fin.	-	DCD_14.02-99	0	2
		102	14.02-100	2008/12/18	Y	N	N	fin.	-	DCD_14.02-100	0	2
		102	14.02-101	2008/12/18	Y	N	N	fin.	-	DCD_14.02-101	0	2
		102	14.02-102	2008/12/18	Y	N	N	fin.	-	DCD_14.02-102	0	2
		102	14.02-103	2008/12/18	Y	N	N	fin.	-	DCD_14.02-103	0	2
		102	14.02-104	2008/12/18	Y	N	N	fin.	-	DCD_14.02-104	0	2
		102	14.02-105	2008/12/18	N	N	N	fin.	-	-	N/A	N/A
		102	14.02-106	2008/12/18	Y	N	N	fin.	-	DCD_14.02-106	0	2
		123	14.02-107	2008/12/18	Y	N	N	fin.	-	DCD_14.02-107	0	2
		194	14.02-108	2009/2/24	Y	N	N		-	DCD_14.02-108	1	2
				2009/4/1	Y	N	N		-	DCD_14.02-108	2	2
		243	14.02-109	2009/3/27	Y	N	N	fin.	-	DCD_14.02-109	1	2
		243	14.02-110	2009/3/27	Y	N	N	fin.	-	DCD_14.02-110	1	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		243	14.02-111	2009/3/27	Y	N	N	fin.	-	DCD_14.02-111	1	2
		243	14.02-112	2009/3/27	Y	N	N	fin.	-	DCD_14.02-112	1	2
		271	14.02-113	2009/3/30	Y	Y	N	fin.	-	DCD_14.02-113	1	2
		271	14.02-114	2009/3/30	Y	Y	N	fin.	-	DCD_14.02-114	1	2
		337	14.02-115	2009/5/18	Y	N	N		-	DCD_14.02-115	2	2
		337	14.02-116	2009/5/18	Y	N	N		-	DCD_14.02-116	2	2
		-	-	-	-	-	-	-	COL 14.2(3) deleted	MAP-14-001	0	2
14.3	Inspections, Tests, Analyses,	32	14.03-1	2008/8/29	Y	Y	N	fin.	-	DCD_14.03-1	0	2
	and Acceptance Criteria	32	14.03-2	2008/8/29	N	N	N	fin.	-	-	N/A	N/A
		32	14.03-3	2008/8/29	N	N	N	fin.	-	-	N/A	N/A
		32	14.03-4	2008/8/29	Y	N	N	fin.	-	DCD_14.03-4	3	2
		32	14.03-5	2008/8/29	N	N	N	fin.	-	-	N/A	N/A
		156	14.3-1	2009/2/5	Y	N	N	fin.	-	DCD_14.3-1	1	2
		156	14.3-2	2009/2/5	Y	N	N	fin.	-	DCD_14.3-2	1	2
14.3.2	Structural and Systems Engineering		14.03.02-1									
	Inspections, Tests, Analyses,	190	14.03.02-2	2009/3/3	Y	N	N		-	DCD_14.03.02-2	2	2
	and Acceptance Criteria	190	14.03.02-3	2009/3/3	Y	N	N		-	DCD_14.03.02-3	2	2
		190	14.03.02-4	2009/3/3	N	N	N		-	-	N/A	N/A
		190	14.03.02-5	2009/3/3	Y	N	N		-	DCD_14.03.02-5	TBD	2
		190	14.03.02-6	2009/3/3	Y	N	N		-	DCD_14.03.02-6	2	2
		190	14.03.02-7	2009/3/3	Y	N	N		-	DCD_14.03.02-7	2	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		190	14.03.02-8	2009/3/3	Y	N	N		-	DCD_14.03.02-8	2	2
14.3.3	Piping Systems and Components	242	14.03.03-1	2009/4/27	N	N	N		-	-	N/A	N/A
	and Acceptance Criteria	242	14.03.03-2	2009/4/27	N	N	N		-	-	N/A	N/A
		242	14.03.03-3	2009/4/27	Y	N	N		-	DCD_14.03.03-3	TDB	
		242	14.03.03-4	2009/4/27	Y	N	N		-	DCD_14.03.03-4	2	2
		242	14.03.03-5	2009/4/27	Y	N	N		-	DCD_14.03.03-5	TDB	
		242	14.03.03-6	2009/4/27	Y	N	N		-	DCD_14.03.03-6	TDB	
		242	14.03.03-7	2009/4/27	Y	N	N		-	DCD_14.03.03-7	TDB	
		242	14.03.03-8	2009/4/27	Y	N	N		-	DCD_14.03.03-8	TDB	
		242	14.03.03-9	2009/4/27	Y	N	N		-	DCD_14.03.03-9	TDB	
		242	14.03.03-10	2009/4/27	Y	N	N		-	DCD_14.03.03-10	TDB	
		242	14.03.03-11	2009/4/27	Y	N	N		-	DCD_14.03.03-11	TDB	
		242	14.03.03-12	2009/4/27	Y	N	N		-	DCD_14.03.03-12	TDB	
		242	14.03.03-13	2009/4/27	Y	N	N		-	DCD_14.03.03-13	TDB	
		242	14.03.03-14	2009/4/27	Y	N	N		-	DCD_14.03.03-14	TDB	
		242	14.03.03-15	2009/4/27	Y	N	N		-	DCD_14.03.03-15	TDB	
		242	14.03.03-16	2009/4/27	Y	N	N		-	DCD_14.03.03-16	TDB	
14.3.4	Reactor Systems -	191	14.03.04-1	2009/4/7	Y	N	N		-	DCD_14.03.04-1	3	2
		191	14.03.04-2	2009/4/7	Y	N	N		-	DCD_14.03.04-2	3	2
		191	14.03.04-3	2009/4/7	Y	N	N		-	DCD_14.03.04-3	3	2
		191	14.03.04-4	2009/4/7	Y	N	N		-	DCD_14.03.04-4	3	2
		191	14.03.04-5	2009/4/7	Y	N	N		-	DCD_14.03.04-5	2	2
		191	14.03.04-6	2009/4/7	Y	N	N		-	DCD_14.03.04-6	3	2
		191	14.03.04-7	2009/4/7	Y	N	N		-	DCD_14.03.04-7	3	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		191	14.03.04-8	2009/4/7	Y	N	N		-	DCD_14.03.04-8	3	2
		191	14.03.04-9	2009/4/7	Y	N	N		-	DCD_14.03.04-9	3	2
		192	14.03.04-10	2009/4/10	Y	N	N		-	DCD_14.03.04-10	3	2
		192	14.03.04-11	2009/4/10	Y	N	N		-	DCD_14.03.04-11	3	2
		192	14.03.04-12	2009/4/10	Y	N	N		-	DCD_14.03.04-12	3	2
		192	14.03.04-13	2009/4/10	Y	N	N		-	DCD_14.03.04-13	3	2
		192	14.03.04-14	2009/4/10	Y	N	N		-	DCD_14.03.04-14	3	2
		192	14.03.04-15	2009/4/10	Y	N	N		-	DCD_14.03.04-15	3	2
		192	14.03.04-16	2009/4/10	Y	N	N		-	DCD_14.03.04-16	3	2
		192	14.03.04-17	2009/4/10	Y	N	N		-	DCD_14.03.04-17	3	2
		192	14.03.04-18	2009/4/10	Y	N	N		-	DCD_14.03.04-18	3	2
		193	14.03.04-19	2009/4/9	Y	N	N		-	DCD-14.03.04-19	3	2
		193	14.03.04-20	2009/4/9	Y	N	N		-	DCD-14.03.04-20	3	2
		193	14.03.04-21	2009/4/9	Y	N	N		-	DCD-14.03.04-21	3	2
		193	14.03.04-22	2009/4/9	Y	N	N		-	DCD-14.03.04-22	3	2
		193	14.03.04-23	2009/4/9	Y	N	N		-	DCD-14.03.04-23	3	2
		193	14.03.04-24	2009/4/9	Y	N	N		-	DCD-14.03.04-24	3	2
		193	14.03.04-25	2009/4/9	Y	N	N		-	DCD-14.03.04-25	3	2
		193	14.03.04-26	2009/4/9	Y	N	N		-	DCD-14.03.04-26	3	2
		193	14.03.04-27	2009/4/9	Y	N	N		-	DCD-14.03.04-27	3	2
		193	14.03.04-28	2009/4/9	Y	N	N		-	DCD-14.03.04-28	3	2
		193	14.03.04-29	2009/4/9	Y	N	N		-	DCD-14.03.04-29	3	2
		193	14.03.04-30	2009/4/9	Y	N	N		-	DCD-14.03.04-30	3	2
		196	14.03.04-31	2009/3/5	Y	N	N		-	DCD_14.03.04-31	2	2
		196	14.03.04-32	2009/3/5	Y	N	N		-	DCD_14.03.04-32	2	2
		196	14.03.04-33	2009/3/5	Y	N	N		-	DCD_14.03.04-33	2	2
		196	14.03.04-34	2009/3/5	Y	N	N		-	DCD_14.03.04-34	2	2
		196	14.03.04-35	2009/3/5	Y	N	N		-	DCD_14.03.04-35	2	2

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14.3.5	Instrumentation and Controls -	181	14.03.05-1	2009/4/6	N	N	N		-	-	N/A	
		181	14.03.05-2	2009/4/6	N	N	N		-	-	N/A	
		181	14.03.05-3	2009/4/6	Y	N	N		-	DCD-14.03.05-3	3	2
		181	14.03.05-4	2009/4/6	Y	N	N		-	DCD-14.03.05-4	3	2
		181	14.03.05-5	2009/4/6	Y	N	N		-	DCD-14.03.05-5	3	2
		181	14.03.05-6	2009/4/6	Y	N	N		-	DCD-14.03.05-6	3	2
		181	14.03.05-7	2009/4/6	N	N	N		-	-	N/A	
		181	14.03.05-8	2009/4/6	Y	N	N		-	DCD-14.03.05-8	3	2
		181	14.03.05-9	2009/4/6	Y	N	N		-	DCD-14.03.05-9	3	2
			14.03.05-10						-			
			14.03.05-11						-			
			14.03.05-12						-			
			14.03.05-13						-			
			14.03.05-14						-			
			14.03.05-15						-			
			14.03.05-16						-			
			14.03.05-17						-			
			14.03.05-18						-			
			14.03.05-19						-			
			14.03.05-20						-			
			14.03.05-21						-			
		275	14.03.05-22	2009/4/28	Y	N	N		-	DCD-14.03.05-22	TDB	
		275	14.03.05-23	2009/4/28	Y	N	N		-	DCD-14.03.05-23	TDB	
		275	14.03.05-24	2009/4/28	Y	N	N		-	DCD-14.03.05-24	TDB	
		275	14.03.05-25	2009/4/28	Y	N	N		-	DCD-14.03.05-25	TDB	
		275	14.03.05-26	2009/4/28	Y	N	N		-	DCD-14.03.05-26	TDB	

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		275	14.03.05-27	2009/4/28	-	N	N		-	-	N/A	N/A
		275	14.03.05-28	2009/4/28	Y	N	N		-	DCD-14.03.05-28	TDB	
		275	14.03.05-29	2009/4/28	Y	N	N		-	DCD-14.03.05-29	TDB	
		275	14.03.05-30	2009/4/28	N	N	N		-	-	N/A	N/A
		275	14.03.05-31	2009/4/28	N	N	N		-	-	N/A	N/A
14.3.6	Electrical Systems -		14.03.06-01						-			
	Inspections, Tests, Analyses,		14.03.06-02						-			
	and Acceptance Criteria		14.03.06-03						-			
			14.03.06-04						-			
			14.03.06-05						-			
		182	14.03.06-06	2009/4/6	Y	N	N		-	DCD-14.03.06-06	3	2
		182	14.03.06-07	2009/4/6	Y	N	N		-	DCD-14.03.06-07	3	2
		182	14.03.06-08	2009/4/6	Y	N	N		-	DCD-14.03.06-08	3	2
		182	14.03.06-09	2009/4/6	Y	N	N		-	DCD-14.03.06-09	3	2
		182	14.03.06-10	2009/4/6	Y	N	N		-	DCD-14.03.06-10	3	2
		182	14.03.06-11	2009/4/6	Y	N	N		-	DCD-14.03.06-11	3	2
		182	14.03.06-12	2009/4/6	Y	N	N		-	DCD-14.03.06-12	3	2
		182	14.03.06-13	2009/4/6	Y	N	N		-	DCD-14.03.06-13	3	2
		182	14.03.06-14	2009/4/6	Y	N	N		-	DCD-14.03.06-14	3	2
14.3.7	Plant Systems -	54	14.03.07-1/14.3.7.3.1-1	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
	Inspections, Tests, Analyses,	54	14.03.07-1/14.3.7.3.1-2	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
	and Acceptance Criteria	54	14.03.07-1/14.3.7.3.1-3	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.01-3	3	2
		54	14.03.07-2/14.3.7.3.2-1	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-2	2008/9/19	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		54	14.03.07-2/14.3.7.3.2-3	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-4	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-5	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-5	1	2
		54	14.03.07-2/14.3.7.3.2-6	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-6	1	2
		54	14.03.07-2/14.3.7.3.2-7	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-8	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-8	1	2
		54	14.03.07-2/14.3.7.3.2-9	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-10	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-11	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-11	2	2
		54	14.03.07-2/14.3.7.3.2-12	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-13	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-14	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-14	1	2
		54	14.03.07-2/14.3.7.3.2-15	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-15	1	2
		54	14.03.07-2/14.3.7.3.2-16	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-16	1	2
		54	14.03.07-2/14.3.7.3.2-17	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-17	3	2
		54	14.03.07-2/14.3.7.3.2-18	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-18	0	2
		54	14.03.07-2/14.3.7.3.2-19	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-20	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-21	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.02-21	0	2
		54	14.03.07-2/14.3.7.3.2-22	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-2/14.3.7.3.2-23	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-1	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-2	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-3	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-4	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-5	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-6	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.04-6	1	2
		54	14.03.07-3/14.3.7.3.4-7	2008/9/19	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		54	14.03.07-3/14.3.7.3.4-8	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-9	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.04-9	1	2
		54	14.03.07-3/14.3.7.3.4-10	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.04-10	0	2
		54	14.03.07-3/14.3.7.3.4-11	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.04-11	3	2
		54	14.03.07-3/14.3.7.3.4-12	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.04-12	0	2
		54	14.03.07-3/14.3.7.3.4-13	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-14	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-3/14.3.7.3.4-15	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-1	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-2	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-3	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-4	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-4	0	2
		54	14.03.07-5/14.3.7.3.6-5	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-5	1	2
		54	14.03.07-5/14.3.7.3.6-6	2008/9/19	Y	Y	N	fin.	-	DCD_14.03.07.03.06-6	3	2
		54	14.03.07-5/14.3.7.3.6-7	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-7	3	2
		54	14.03.07-5/14.3.7.3.6-8	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-8	1	2
		54	14.03.07-5/14.3.7.3.6-9	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-9	2	2
		54	14.03.07-5/14.3.7.3.6-10	2008/9/19	N	N	N	fin.	-			
				2009/1/9	Y	N	N	fin.	-	DCD_14.03.07.03.06-10	2	2
		54	14.03.07-5/14.3.7.3.6-11	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-12	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-13	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-13	2	2
		54	14.03.07-5/14.3.7.3.6-14	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-14	3	2
		54	14.03.07-5/14.3.7.3.6-15	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-16	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-16	1	2
		54	14.03.07-5/14.3.7.3.6-17	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-17	1	2
		54	14.03.07-5/14.3.7.3.6-18	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-19	2008/9/19	N	N	N	fin.	-	-	N/A	N/A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		54	14.03.07-5/14.3.7.3.6-20	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-20	3	2
		54	14.03.07-5/14.3.7.3.6-21	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-5/14.3.7.3.6-22	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-22	3	2
		54	14.03.07-5/14.3.7.3.6-23	2008/9/19	Y	N	N	fin.	-	DCD_14.03.07.03.06-23	0	2
		54	14.03.07-6/14.3.7.3.7-1	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-6/14.3.7.3.7-2	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-6/14.3.7.3.7-3	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		54	14.03.07-6/14.3.7.3.7-4	2008/9/19	N	N	N	fin.	-	-	N/A	N/A
		183	14.03.07-7	2009/4/6	Y	N	N			DCD-14.03.07-7	3	2
		183	14.03.07-8	2009/4/6	Y	N	N			DCD-14.03.07-8	3	2
		183	14.03.07-9	2009/4/6	Y	N	N			DCD-14.03.07-9	3	2
		183	14.03.07-10	2009/4/6	Y	N	N			DCD-14.03.07-10	3	2
		183	14.03.07-11	2009/4/6	Y	N	N			DCD-14.03.07-11	3	2
		183	14.03.07-12	2009/4/6	Y	N	N			DCD-14.03.07-12	3	2
		183	14.03.07-13	2009/4/6	Y	N	N			DCD-14.03.07-13	3	2
		183	14.03.07-14	2009/4/6	Y	N	N			DCD-14.03.07-14	3	2
		183	14.03.07-15	2009/4/6	N	N	N			-	N/A	N/A
		184	14.03.07-16	2009/4/9	Y	N	N		-	DCD_14.03.07-16	3	2
		184	14.03.07-17	2009/4/9	Y	N	N		-	DCD_14.03.07-17	3	2
		184	14.03.07-18	2009/4/9	Y	N	N		-	DCD_14.03.07-18	3	2
		184	14.03.07-19	2009/4/9	Y	N	N		-	DCD_14.03.07-19	3	2
		184	14.03.07-20	2009/4/9	Y	N	N		-	DCD_14.03.07-20	3	2
		184	14.03.07-21	2009/4/9	Y	N	N		-	DCD_14.03.07-21	3	2
		184	14.03.07-22	2009/4/9	Y	N	N		-	DCD_14.03.07-22	3	2
		184	14.03.07-23	2009/4/9	Y	N	N		-	DCD_14.03.07-23	3	2
		184	14.03.07-24	2009/4/9	Y	N	N		-	DCD_14.03.07-24	3	2
		184	14.03.07-25	2009/4/9	Y	N	N		-	DCD_14.03.07-25	3	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		184	14.03.07-26	2009/4/9	Y	N	N		-	DCD_14.03.07-26	3	2
		184	14.03.07-27	2009/4/9	Y	N	N		-	DCD_14.03.07-27	3	2
		184	14.03.07-28	2009/4/9	N	N	N		-	-	N/A	N/A
		184	14.03.07-29	2009/4/9	Y	N	N		-	DCD_14.03.07-29	3	2
		184	14.03.07-30	2009/4/9	N	N	N		-	-	N/A	N/A
		184	14.03.07-31	2009/4/9	Y	N	N		-	DCD_14.03.07-31	3	2
		184	14.03.07-32	2009/4/9	Y	N	N		-	DCD_14.03.07-32	3	2
		184	14.03.07-33	2009/4/9	Y	N	N		-	DCD_14.03.07-33	3	2
		184	14.03.07-34	2009/4/9	Y	N	N		-	DCD_14.03.07-34	3	2
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	195	14.03.10-01	2009/3/5	Y	N	N		-	DCD_14.03.10-01	2	2
		195	14.03.10-02	2009/3/5	Y	N	N		-	DCD_14.03.10-02	2	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
14.3.11	Containment Systems -	51	14.03.11-1	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-1	0	2
	Inspections, Tests, Analyses,	51	14.03.11-2	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-2	3	2
	and Acceptance Criteria	51	14.03.11-3	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-3	3	2
		51	14.03.11-4	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-5	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-5	0	2
		51	14.03.11-6	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-6	3	2
		51	14.03.11-7	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-8	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-8	3	2
		51	14.03.11-9	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-10	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-11	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-11	0	2
		51	14.03.11-12	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-12	0	2
		51	14.03.11-13	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-14	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-14	0	2
		51	14.03.11-15	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		51	14.03.11-16	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-16	3	2
		51	14.03.11-17	2008/9/18	Y	N	N	fin.	-	DCD_14.03.11-17	0	2
		198	14.03.11-18	2009/4/9	Y	N	N		-	DCD_14.03.11-18	3	2
		198	14.03.11-19	2009/4/9	N	N	N		-	-	N/A	N/A
		198	14.03.11-20	2009/4/9	Y	N	N		-	DCD_14.03.11-20	3	2
		198	14.03.11-21	2009/4/9	Y	N	N		-	DCD_14.03.11-21	3	2
		198	14.03.11-22	2009/4/9	Y	N	N		-	DCD_14.03.11-22	3	2
		198	14.03.11-23	2009/4/9	Y	N	N		-	DCD_14.03.11-23	3	2
		198	14.03.11-24	2009/4/9	Y	N	N		-	DCD_14.03.11-24	3	2
		198	14.03.11-25	2009/4/9	Y	N	N		-	DCD_14.03.11-25	3	2
		198	14.03.11-26	2009/4/9	Y	N	N		-	DCD_14.03.11-26	3	2

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		198	14.03.11-27	2009/4/9	Y	N	N		-	DCD_14.03.11-27	3	2
		222	14.03.11-28	2009/4/23	Y	N	N		-	DCD_14.03.11-28	3	2
		222	14.03.11-29	2009/4/23	Y	N	N		-	DCD_14.03.11-29	3	2
		222	14.03.11-30	2009/4/23	Y	N	N		-	DCD_14.03.11-30	3	2
		222	14.03.11-31	2009/4/23	Y	N	N		-	DCD_14.03.11-31	3	2
		222	14.03.11-32	2009/4/23	Y	N	N		-	DCD_14.03.11-32	3	2
		222	14.03.11-33	2009/4/23	N	N	N		-	-	N/A	N/A
		222	14.03.11-34	2009/4/23	Y	N	N		-	DCD_14.03.11-34	3	2
		222	14.03.11-35	2009/4/23	Y	N	N		-	DCD_14.03.11-35	3	2
		222	14.03.11-36	2009/4/23	Y	N	N		-	DCD_14.03.11-36	3	2
		222	14.03.11-37	2009/4/23	Y	N	N		-	DCD_14.03.11-37	3	2
14.3.12	Physical Security Hardware -	52	14.03.12-1	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
	Inspections, Tests, Analyses,	52	14.03.12-2	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
	and Acceptance Criteria	52	14.03.12-3	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-4	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-5	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-6	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-7	2008/9/18	Y	N	N	fin.	-	DCD_14.03.12-7	3	2
		52	14.03.12-8	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-9	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-10	2008/9/18	N	N	N	fin.	-	-	N/A	N/A
		52	14.03.12-11	2008/9/18	Y	N	N	fin.	-	DCD_14.03.12-11	3	2
		261	14.03.12-12	2009/4/3	N	N	N		-	-	N\A	N\A
		261	14.03.12-13	2009/4/3	N	N	N		-	-	N\A	N\A

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No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		261	14.03.12-14	2009/4/3	N	N	N		-	-	N\A	N\A
		261	14.03.12-15	2009/4/3	N	N	N		-	-	N\A	N\A
		261	14.03.12-16	2009/4/3	N	N	N		-	-	N\A	N\A
		261	14.03.12-17	2009/4/3	N	N	N		-	-	N\A	N\A
		261	14.03.12-18	2009/4/3	N	N	N		-	-	N\A	N\A

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16.1	General, Plant Sys,	133	16-12	2009/1/29	Y	Y	N		-	DCD_16-12	1	2
	Refueling, & Adm Ctrls:	133	16-13	2009/1/29	Y	Y	N		-	DCD_16-13	1	2
	Technical Specifications	133	16-14	2009/1/29	Y	Y	N		-	DCD_16-14	1	2
		133	16-15	2009/1/29	N	N	N		-	-	N/A	N/A
		133	16-16	2009/1/29	Y	Y	N		-	DCD_16-16	1	2
		133	16-17	2009/1/29	Y	Y	N		-	DCD_16-17	1	2
		133	16-18	2009/1/29	Y	Y	N		-	DCD_16-18	1	2
		133	16-19	2009/1/29	Y	Y	N		-	DCD_16-19	1	2
		133	16-20	2009/1/29	N	N	N		-	-	N/A	N/A
		161	16-115	2009/2/20	Y	Y	N		-	DCD_16-115	1	2
		161	16-116	2009/2/20	Y	Y	N		-	DCD_16-116	1	2
		161	16-117	2009/3/19	N	N	N		-	-	N/A	N/A
		161	16-118	2009/2/20	Y	Y	N		-	DCD_16-118	1	2
		161	16-119	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-120	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-121	2009/2/20	Y	Y	N		-	DCD_16-121	1	2
		161	16-122	2009/2/20	Y	Y	N		-	DCD_16-122	1	2
		161	16-123	2009/2/20	Y	Y	N		-	DCD_16-123	1	2
		161	16-124	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-125	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-126	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-127	2009/2/20	Y	Y	N		-	DCD_16-127	2	2
		161	16-128	2009/2/20	Y	Y	N		-	DCD_16-128	1	2
		161	16-129	2009/2/20	Y	Y	N		-	DCD_16-129	1	2
		161	16-130	2009/2/20	Y	Y	N		-	DCD_16-130	2	2
		161	16-131	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-132	2009/2/20	Y	Y	N		-	DCD_16-132	1	2
		161	16-133	2009/2/20	N	N	N		-	-	N/A	N/A

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		161	16-134	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-135	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-136	2009/2/20	Y	Y	N		-	DCD_16-136	1	2
		161	16-137	2009/2/20	N	N	N		-	-	N/A	N/A
		161	16-138	2009/2/20	Y	Y	N		-	DCD_16-138	1	2
		161	16-139	2009/2/20	Y	Y	N		-	DCD_16-139	2	2
		161	16-140	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-141	2009/2/20	Y	Y	N		-	DCD_16-141	1	2
		162	16-142	2009/2/20	Y	Y	N		-	DCD_16-142	1	2
		162	16-143	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-144	2009/2/20	Y	Y	N		-	DCD_16-144	1	2
		162	16-145	2009/2/20	Y	Y	N		-	DCD_16-145	1	2
		162	16-146	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-147	2009/2/20	Y	Y	N		-	DCD_16-147	1	2
		162	16-148	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-149	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-150	2009/2/20	Y	Y	N		-	DCD_16-150	1	2
		162	16-151	2009/2/20	Y	Y	N		-	DCD_16-151	1	2
		162	16-152	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-153	2009/2/20	Y	Y	N		-	DCD_16-153	1	2
		162	16-154	2009/2/20	Y	Y	N		-	DCD_16-154	1	2
		162	16-155	2009/2/20	Y	Y	N		-	DCD_16-155	2	2
		162	16-156	2009/2/20	N	N	N		-	-	N/A	N/A
		162	16-157	2009/2/20	Y	Y	N		-	DCD_16-157	1	2
		166	16-158	2009/3/18	Y	Y	N		-	DCD_16-158	TBD	
		166	16-159	2009/3/18	Y	Y	N		-	DCD_16-159	TBD	
		166	16-160	2009/3/18	Y	Y	N		-	DCD_16-160	TBD	
		166	16-161	2009/3/18	Y	Y	N		-	DCD_16-161	TBD	

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		166	16-162	2009/3/18	Y	Y	N		-	DCD_16-162	TBD	
		166	16-163	2009/3/18	Y	Y	N		-	DCD_16-163	TBD	
		166	16-164	2009/3/18	Y	Y	N		-	DCD_16-164	TBD	
		166	16-165	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-166	2009/3/18	Y	Y	N		-	DCD_16-166	TBD	
		166	16-167	2009/3/18	Y	Y	N		-	DCD_16-167	TBD	
		166	16-168	2009/3/18	Y	Y	N		-	DCD_16-168	TBD	
		166	16-169	2009/3/18	Y	Y	N		-	DCD_16-169	TBD	
		166	16-170	2009/3/18	Y	Y	N		-	DCD_16-170	TBD	
		166	16-171	2009/3/18	Y	Y	N		-	DCD_16-171	TBD	
		166	16-172	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-173	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-174	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-175	2009/3/18	Y	Y	N		-	DCD_16-175	TBD	
		166	16-176	2009/3/18	Y	Y	N		-	DCD_16-176	TBD	
		166	16-177	2009/3/18	Y	Y	N		-	DCD_16-177	TBD	
		166	16-178	2009/3/18	Y	Y	N		-	DCD_16-178	TBD	
		166	16-179	2009/3/18	Y	Y	N		-	DCD_16-179	TBD	
		166	16-180	2009/3/18	Y	Y	N		-	DCD_16-180	TBD	
		166	16-181	2009/3/18	Y	Y	N		-	DCD_16-181	TBD	
		166	16-182	2009/3/18	Y	Y	N		-	DCD_16-182	TBD	
		166	16-183	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-184	2009/3/18	Y	Y	N		-	DCD_16-184	TBD	
		166	16-185	2009/3/18	Y	Y	N		-	DCD_16-185	TBD	
		166	16-186	2009/3/18	Y	N	N		-	DCD_16-186	TBD	
		166	16-187	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-188	2009/3/18	N	N	N		-	-	N/A	N/A
		166	16-189	2009/3/18	Y	Y	N		-	DCD_16-189	TBD	

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		166	16-190	2009/3/18	Y	Y	N		-	DCD_16-190	TBD	
		166	16-191	2009/3/18	Y	Y	N		-	DCD_16-191	TBD	
		166	16-192	2009/3/18	Y	Y	N		-	DCD_16-192	TBD	
		166	16-193	2009/3/18	Y	N	N		-	DCD_16-193	TBD	
		166	16-194	2009/3/18	Y	Y	N		-	DCD_16-194	TBD	
		166	16-195	2009/3/18	Y	N	N		-	DCD_16-195	TBD	
		167	16-196	2009/3/23	Y	Y	N		-	DCD_16-196	TBD	
		167	16-197	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-198	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-199	2009/3/23	Y	N	N		-	DCD_16-199	TBD	
		167	16-200	2009/3/23	Y	Y	N		-	DCD_16-200	TBD	
		167	16-201	2009/3/23	Y	Y	N		-	DCD_16-201	TBD	
		167	16-202	2009/3/23	Y	Y	N		-	DCD_16-202	TBD	
		167	16-203	2009/3/23	Y	Y	N		-	DCD_16-203	TBD	
		167	16-204	2009/3/23	Y	N	N		-	DCD_16-204	TBD	
		167	16-205	2009/3/23	Y	Y	N		-	DCD_16-205	TBD	
		167	16-206	2009/3/23	Y	Y	N		-	DCD_16-206	TBD	
		167	16-207	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-208	2009/3/23	Y	Y	N		-	DCD_16-208	TBD	
		167	16-209	2009/3/23	Y	Y	N		-	DCD_16-209	TBD	
		167	16-210	2009/3/23	Y	Y	N		-	DCD_16-210	TBD	
		167	16-211	2009/3/23	Y	Y	N		-	DCD_16-211	TBD	
		167	16-212	2009/3/23	Y	Y	N		-	DCD_16-212	TBD	
		167	16-213	2009/3/23	Y	N	N		-	DCD_16-213	TBD	
		167	16-214	2009/3/23	Y	Y	N		-	DCD_16-214	TBD	
		167	16-215	2009/3/23	Y	Y	N		-	DCD_16-215	TBD	
		167	16-216	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-217	2009/3/23	N	N	N		-	-	N/A	N/A

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		167	16-218	2009/3/23	Y	Y	N		-	DCD_16-218	TBD	
		167	16-219	2009/3/23	Y	Y	N		-	DCD_16-219	TBD	
		167	16-220	2009/3/23	Y	Y	N		-	DCD_16-220	TBD	
		167	16-221	2009/3/23	Y	Y	N		-	DCD_16-221	TBD	
		167	16-222	2009/3/23	Y	Y	N		-	DCD_16-222	TBD	
		167	16-223	2009/3/23	Y	Y	N		-	DCD_16-223	TBD	
		167	16-224	2009/3/23	Y	Y	N		-	DCD_16-224	TBD	
		167	16-225	2009/3/23	Y	Y	N		-	DCD_16-225	TBD	
		167	16-226	2009/3/23	Y	Y	N		-	DCD_16-226	TBD	
		167	16-227	2009/3/23	Y	Y	N		-	DCD_16-227	TBD	
		167	16-228	2009/3/23	Y	Y	N		-	DCD_16-228	TBD	
		167	16-229	2009/3/23	Y	Y	N		-	DCD_16-229	TBD	
		167	16-230	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-231	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-232	2009/3/23	Y	Y	N		-	DCD_16-232	TBD	
		167	16-233	2009/3/23	Y	Y	N		-	DCD_16-233	TBD	
		167	16-234	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-235	2009/3/23	Y	Y	N		-	DCD_16-235	TBD	
		167	16-236	2009/3/23	Y	N	N		-	DCD_16-236	TBD	
		167	16-237	2009/3/23	Y	N	N		-	DCD_16-237	TBD	
		167	16-238	2009/3/23	Y	Y	N		-	DCD_16-238	TBD	
		167	16-239	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-240	2009/3/23	Y	Y	N		-	DCD_16-240	TBD	
		167	16-241	2009/3/23	Y	Y	N		-	DCD_16-241	TBD	
		167	16-242	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-243	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-244	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-245	2009/3/23	Y	Y	N		-	DCD_16-245	TBD	

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		167	16-246	2009/3/23	Y	N	N		-	DCD_16-246	TBD	
		167	16-247	2009/3/23	Y	Y	N		-	DCD_16-247	TBD	
		167	16-248	2009/3/23	Y	Y	N		-	DCD_16-248	TBD	
		167	16-249	2009/3/23	Y	Y	N		-	DCD_16-249	TBD	
		167	16-250	2009/3/23	Y	Y	N		-	DCD_16-250	TBD	
		167	16-251	2009/3/23	Y	N	N		-	DCD_16-251	TBD	
		167	16-252	2009/3/23	Y	Y	N		-	DCD_16-252	TBD	
		167	16-253	2009/3/23	Y	Y	N		-	DCD_16-253	TBD	
		167	16-254	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-255	2009/3/23	Y	Y	N		-	DCD_16-255	TBD	
		167	16-256	2009/3/23	Y	Y	N		-	DCD_16-256	TBD	
		167	16-257	2009/3/23	Y	Y	N		-	DCD_16-257	TBD	
		167	16-258	2009/3/23	Y	Y	N		-	DCD_16-258	TBD	
		167	16-259	2009/3/23	Y	Y	N		-	DCD_16-259	TBD	
		167	16-260	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-261	2009/3/23	Y	Y	N		-	DCD_16-261	TBD	
		167	16-262	2009/3/23	Y	Y	N		-	DCD_16-262	TBD	
		167	16-263	2009/3/23	Y	Y	N		-	DCD_16-263	TBD	
		167	16-264	2009/3/23	Y	Y	N		-	DCD_16-264	TBD	
		167	16-265	2009/3/23	Y	Y	N		-	DCD_16-265	TBD	
		167	16-266	2009/3/23	Y	Y	N		-	DCD_16-266	TBD	
		167	16-267	2009/3/23	Y	Y	N		-	DCD_16-267	TBD	
		167	16-268	2009/3/23	Y	Y	N		-	DCD_16-268	TBD	
		167	16-269	2009/3/23	Y	N	N		-	DCD_16-269	TBD	
		167	16-270	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-271	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-272	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-273	2009/3/23	N	N	N		-	-	N/A	N/A

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		167	16-274	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-275	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-276	2009/3/23	Y	N	N		-	DCD_16-276	TBD	
		167	16-277	2009/3/23	Y	Y	N		-	DCD_16-277	TBD	
		167	16-278	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-279	2009/3/23	Y	Y	N		-	DCD_16-279	TBD	
		167	16-280	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-281	2009/3/23	Y	Y	N		-	DCD_16-281	TBD	
		167	16-282	2009/3/23	Y	Y	N		-	DCD_16-282	TBD	
		167	16-283	2009/3/23	Y	Y	N		-	DCD_16-283	TBD	
		167	16-284	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-285	2009/3/23	Y	Y	N		-	DCD_16-285	TBD	
		167	16-286	2009/3/23	Y	Y	N		-	DCD_16-286	TBD	
		167	16-287	2009/3/23	Y	Y	N		-	DCD_16-287	TBD	
		167	16-288	2009/3/23	Y	Y	N		-	DCD_16-288	TBD	
		167	16-289	2009/3/23	Y	Y	N		-	DCD_16-289	TBD	
		167	16-290	2009/3/23	Y	Y	N		-	DCD_16-290	TBD	
		167	16-291	2009/3/23	Y	Y	N		-	DCD_16-291	TBD	
		167	16-292	2009/3/23	Y	Y	N		-	DCD_16-292	TBD	
		167	16-293	2009/3/23	Y	Y	N		-	DCD_16-293	TBD	
		167	16-294	2009/3/23	Y	Y	N		-	DCD_16-294	TBD	
		167	16-295	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-296	2009/3/23	N	N	N		-	-	N/A	N/A
		167	16-297	2009/3/23	Y	Y	N		-	DCD_16-297	TBD	
16.2	SLs, Reactivity,											

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	Core Op Limits, & Special Ops:											
	Technical Specifications											
16.3	Instrumentation:	36	01-1	2008/8/22	Y	N	N	fin.	-	DCD_16.3_01-1	-	1
	Technical Specifications	72	16-1	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-2	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-3	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-4	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-5	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-6	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-7	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-8	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-9	2008/10/8	Y	Y	N	fin.	-	DCD_16-9	0	2
		72	16-10	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
		72	16-11	2008/10/8	N	N	N	fin.	-	-	N/A	N/A
16.4	CS & ECCS: Technical Specificatior	135	16-48	2009/2/4	Y	Y	N		-	DCD_16-48	1	2
		135	16-49	2009/2/4	Y	Y	N		-	DCD_16-49	1	2
		135	16-50	2009/2/4	Y	Y	N		-	DCD_16-50	1	2
		135	16-51	2009/2/4	Y	Y	N		-	DCD_16-51	1	2
		135	16-52	2009/2/4	Y	Y	N		-	DCD_16-52	1	2
		135	16-53	2009/2/4	N	N	N		-	-	N/A	N/A
		135	16-54	2009/2/4	Y	Y	N		-	DCD_16-54	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	cc	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		135	16-55	2009/2/4	Y	Y	N		-	DCD_16-55	1	2
		135	16-56	2009/2/4	Y	Y	N		-	DCD_16-56	1	2
		135	16-57	2009/2/4	Y	Y	N		-	DCD_16-57	1	2
		135	16-58	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-66	2009/2/4	Y	Y	N		-	DCD_16-66	1	2
		146	16-67	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-68	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-69	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-70	2009/2/4	Y	Y	N		-	DCD_16-70	1	2
		146	16-71	2009/2/4	Y	Y	N		-	DCD_16-71	1	2
		146	16-72	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-73	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-74	2009/2/4	Y	N	N		-	DCD_16-74	1	2
		146	16-75	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-76	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-77	2009/2/4	Y	Y	N		-	DCD_16-77	1	2
		146	16-78	2009/2/4	Y	Y	N		-	DCD_16-78	1	2
		146	16-79	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-80	2009/2/4	Y	Y	N		-	DCD_16-80	1	2
		146	16-81	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-82	2009/2/4	Y	Y	N		-	DCD_16-82	1	2
		146	16-83	2009/2/4	Y	Y	N		-	DCD_16-83	1	2
		146	16-84	2009/2/4	Y	Y	N		-	DCD_16-84	1	2
		146	16-85	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-86	2009/2/4	Y	Y	N		-	DCD_16-86	1	2
		146	16-87	2009/2/4	Y	Y	N		-	DCD_16-87	1	2
		146	16-88	2009/2/4	Y	Y	N		-	DCD_16-88	1	2
		146	16-89	2009/2/4	Y	Y	N		-	DCD_16-89	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	cc	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		146	16-90	2009/2/4	N	N	N		-	-	N/A	N/A
		146	16-91	2009/2/4	Y	Y	N		-	DCD_16-91	1	2
		146	16-92	2009/2/4	Y	Y	N		-	DCD_16-92	1	2
		146	16-93	2009/2/4	Y	Y	N		-	DCD_16-93	1	2
		146	16-94	2009/2/4	Y	Y	N		-	DCD_16-94	1	2
		146	16-95	2009/2/4	Y	Y	N		-	DCD_16-95	1	2
		146	16-96	2009/2/4	Y	Y	N		-	DCD_16-96	1	2
		146	16-97	2009/2/4	Y	Y	N		-	DCD_16-97	1	2
		146	16-98	2009/2/4	N	N	N		-	-	N/A	N/A
X		146	16-99	2009/2/4	Y	Y	N		-	DCD_16-99	2	2
		158	16-100	2009/2/20	Y	Y	N		-	DCD_16-100	1	2
		158	16-101	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-102	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-103	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-104	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-105	2009/2/20	Y	Y	N		-	DCD_16-105	1	2
		158	16-106	2009/2/20	Y	Y	N		-	DCD_16-106	1	2
		158	16-107	2009/2/20	Y	Y	N		-	DCD_16-107	2	2
		158	16-108	2009/2/20	Y	Y	N		-	DCD_16-108	1	2
		158	16-109	2009/2/20	Y	Y	N		-	DCD_16-109	1	2
		158	16-110	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-111	2009/2/20	Y	Y	N		-	DCD_16-111	1	2
		158	16-112	2009/2/20	Y	Y	N		-	DCD_16-112	1	2
		158	16-113	2009/2/20	N	N	N		-	-	N/A	N/A
		158	16-114	2009/2/20	Y	Y	N		-	DCD_16-114	1	2
16.5	Containment Systems:	136	16-59	2009/2/4	Y	Y	N		-	DCD_16-59	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	cc	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
	Technical Specifications	136	16-60	2009/2/4	Y	Y	N		-	DCD_16-60	1	2
		136	16-61	2009/2/4	Y	Y	N		-	DCD_16-61	1	2
		136	16-62	2009/2/4	N	N	N		-	-	N/A	N/A
		136	16-63	2009/2/4	Y	N	N		-	DCD_16-63	1	2
		136	16-64	2009/2/4	Y	Y	N		-	DCD_16-64	1	2
		136	16-65	2009/2/4	Y	Y	N		-	DCD_16-65	1	2
16.6	Electrical Power Sys:	134	16-21	2009/2/4	Y	Y	N		-	DCD_16-21	1	2
	Technical Specifications	134	16-22	2009/2/4	Y	Y	N		-	DCD_16-22	1	2
		134	16-23	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-24	2009/2/4	Y	Y	N		-	DCD_16-24	1	2
		134	16-25	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-26	2009/2/4	Y	Y	N		-	DCD_16-26	1	2
		134	16-27	2009/2/4	Y	Y	N		-	DCD_16-27	1	2
		134	16-28	2009/2/4	Y	Y	N		-	DCD_16-28	1	2
		134	16-29	2009/2/4	Y	Y	N		-	DCD_16-29	1	2
		134	16-30	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-31	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-32	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-33	2009/2/4	Y	Y	N		-	DCD_16-33	1	2
		134	16-34	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-35	2009/2/4	Y	Y	N		-	DCD_16-35	1	2
		134	16-36	2009/2/4	Y	Y	N		-	DCD_16-36	1	2
		134	16-37	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-38	2009/2/4	Y	Y	N		-	DCD_16-38	1	2
		134	16-39	2009/2/4	Y	Y	N		-	DCD_16-39	1	2
		134	16-40	2009/2/4	Y	Y	N		-	DCD_16-40	1	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	cc	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		134	16-41	2009/2/4	Y	Y	N		-	DCD_16-41	1	2
		134	16-42	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-43	2009/2/4	Y	Y	N		-	DCD_16-43	1	2
		134	16-44	2009/2/4	N	N	N		-	-	N/A	N/A
		134	16-45	2009/2/4	Y	Y	N		-	DCD_16-45	1	2
		134	16-46	2009/2/4	Y	Y	N		-	DCD_16-46	1	2
		134	16-47	2009/2/4	Y	Y	N		-	DCD_16-47	2	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
17.1	Quality Assurance											
	During the Design and											
	Construction Phases											
17.4	Reliability Assurance Program (RAP)	101	17.04-1	2008/12/12	Y	N	N	fin.	-	DCD_17.04-1	3	2
		101	17.04-2	2008/12/12	Y	N	N	fin.	-	DCD_17.04-2	3	2
		101	17.04-3	2008/12/12	Y	N	N	fin.	-	DCD_17.04-3	3	2
		101	17.04-4	2008/12/12	Y	N	N	fin.	-	DCD_17.04-4	2	2
		101	17.04-5	2008/12/12	Y	N	N	fin.	-	DCD_17.04-5	3	2
		101	17.04-6	2008/12/12	Y	N	N	fin.	-	DCD_17.04-6	1	2
		101	17.04-7	2008/12/12	Y	N	N	fin.	-	DCD_17.04-7	1	2
		101	17.04-8	2008/12/12	Y	N	N	fin.	-	DCD_17.04-8	1	2
		101	17.04-9	2008/12/12	Y	N	N	fin.	-	DCD_17.04-9	3	2
		101	17.04-10	2008/12/12	Y	N	N	fin.	-	DCD_17.04-10	1	2
		101	17.04-11	2008/12/12	Y	N	N	fin.	-	DCD_17.04-11	1	2
		101	17.04-12	2008/12/12	Y	N	N	fin.	-	DCD_17.04-12	1	2
		101	17.04-13	2008/12/12	Y	N	N	fin.	-	DCD_17.04-13	2	2
		101	17.04-14	2008/12/12	Y	N	N	fin.	-	DCD_17.04-14	1	2
		101	17.04-15	2008/12/12	Y	N	N	fin.	-	DCD_17.04-15	3	2
		101	17.04-16	2008/12/12	N	N	N	fin.	-	-	N/A	N/A
		101	17.04-17	2008/12/12	N	N	N	fin.	-	-	N/A	N/A
		101	17.04-18	2008/12/12	Y	N	N	fin.	-	DCD_17.04-18	3	2
		150	17.04-19	2009/3/10	Y	N	N		-	DCD_17.04-19	2	2
		150	17.04-20	2009/3/10	N	N	N		-	-	N/A	N/A
		150	17.04-21	2009/2/6	Y	N	N		-	DCD_17.04-21	1	2
		150	17.04-22	2009/2/6	Y	N	N		-	DCD_17.04-22	1	2
		150	17.04-23	2009/3/10	Y	N	N		-	DCD_17.04-23	2	2

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SRP Section		DCD RAI Response							Other Drivers	Change ID Number for DCD forthcoming Revision	DCD Tracking Report Revision	DCD Revision
No.	Title	RAI No.	Question No.	Response Date	Impact on DCD	Impact on COLA	Impact on PRA	Response Status				
		150	17.04-24	2009/3/10	Y	N	N		-	DCD_17.04-24	2	2
		150	17.04-25	2009/2/6	Y	N	N		-	DCD_17.04-25	1	2
		150	17.04-26	2009/2/6	Y	N	N		-	DCD_17.04-26	1	2
		150	17.04-27	2009/2/6	Y	N	N		-	DCD_17.04-27	1	2
		150	17.04-28	2009/2/6	Y	N	N		-	DCD_17.04-28	1	2
		150	17.04-29	2009/2/6	Y	N	N		-	DCD_17.04-29	3	2
		150	17.04-30	2009/3/10	Y	N	N		-	DCD_17.04-30	2	2
		150	17.04-31	2009/2/6	N	N	N		-	-	N/A	N/A
		150	17.04-32	2009/2/6	Y	N	N		-	DCD_17.04-32	1	2
		150	17.04-33	2009/2/6	Y	N	N		-	DCD_17.04-33	1	2
		150	17.04-34	2009/2/6	Y	N	N		-	DCD_17.04-34	1	2
		150	17.04-35	2009/2/6	N	N	N		-	-	N/A	N/A
		175	17.04-36	2009/3/3	Y	N	N		-	DCD_17.04-36	2	2
		175	17.04-37	2009/4/3	Y	N	N		-	DCD_17.04-37	2	2
		175	17.04-38	2009/4/3	Y	N	N		-	DCD_17.04-38	2	2
		175	17.04-39	2009/3/3	Y	N	N		-	DCD_17.04-39	2	2
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants											
17.6	Maintenance Rule	137	17.06-1	2009/1/21	Y	N	N		-	DCD_17.06-1	1	2

Chapter 1

ACRONYMS AND ABBREVIATIONS (Continued)

BTU	british thermal unit
BWR	boiling water reactor
BWROG	boiling water reactor owners' group
C/V	containment vessel
CAGI	Compressed Air and Gas Institute
CAGS	compressed air and gas system
CAMS	containment atmosphere monitoring system
CAOC	constant axial offset control
CAS	central alarm station
CASS	compressed air supply system
CAV	cumulative absolute velocity
CBP	computer-based procedure
CBS	condenser water box vacuuming priming system
CCDP	conditional core damage probability
CCF	common cause failure
CCFP	conditional containment failure probability
CCTV	closed ~~captioned~~ circuit television
CCW	component cooling water
CCWP	component cooling water pump
CCWS	component cooling water system
CCWT	component cooling water train
CD	complete dependence
CDF	core damage frequency
CDR	Certified Design Report
CDS	condensate system
CEDE	committed effective dose equivalent
CET	containment event tree
CF	core flooding
CFCS	containment fan cooler system
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CGS	compressed gas supply system
CHF	critical heat flux
CHP	charging pump
CHR	cooling water/hot water return
CHS	containment hydrogen monitoring and control system

Chapter 3

3.6.1.2.2 Basic Protection Measures

3.6.1.2.2.1 Separation

Separation by distance, compartments or enclosures is used as much as practicable to protect redundant safety-related systems and trains. Deliberate separation protects against the dynamic effects of postulated pipe failures of the systems and components. Redundant safety systems and components are arranged to prevent the loss of the safety function as a result of a postulated pipe failure.

A multi-step process is used to develop the placement of safety-related systems and components which consider the following means for separation.

- Wherever practical, locate safety-related systems away from high-energy piping
- Locate redundant safety systems in separate compartments
- If necessary, enclose specific components required to function as a result of a postulated pipe failure
- Design drainage routing and flood control to maintain adequate separation from equipment required to function as a result of a postulated pipe failure

Each of the four safety trains are separated into four quadrants around the outside of the PCCV. Each train is isolated by physical barriers as well as isolating the radiological control area from the non-radiological control area of the R/B. The concrete walls are designed to prevent events on one safety train from impacting another train. The segregation also includes segregation of fluid containing SSCs of a train from the electrical SSCs of the same train to the extent practical. In general, cable trays are routed at higher elevations than piping. Chases are provided between the cable trays and piping to maintain the electrical/mechanical separation if required. Physically, individual train equipment within the four quadrants is located to provide the maximum separation between the same equipment of the other three trains within the confines of the R/B footprint. This separation minimizes the probability of an event affecting more than one of the safety trains at a given time. Where components must cross between isolating barriers, the penetrations are located above flood levels to the extent possible. In addition, penetration seals maintain compartment to compartment separation.

3.6.1.2.2.2 Barriers and Shields

Where physical separation is not sufficent to protect safety-related systems and components from postulated pipe failures, structural elements such as walls, floors, columns, and foundations are designed to serve as protective barriers and shields whenever possible. Other barriers, deflectors or shields are provided where additional protection is required. The barriers, including compartments as applicable, are designed to withstand loading generated by postulated jet forces and pipe whip impact forces in combination with loadings associated with seismic Category I requirements, within the respective design load limits for the structures. Refer to Subsection 3.6.2.4 for additional discussion on the design of barriers, deflectors and shields.

Portions of the containment internal structure provide a series of protective barriers. The reactor coolant loops (RCLs) are shielded from the containment liner by the secondary

3.6.3.4 Analytical Methods and Criteria

The method and criteria used for LBB analysis are consistent with the guidelines in NUREG-1061 (Reference 3.6-23) and Standard Review Plan 3.6.3, Rev. 1 (Reference 3.6-4).

LBB BACs are prepared for each applicable piping system. These curves provide the design guidelines meeting the allowable standards for stress limits and LBB acceptance criteria. The critical location having the highest stress point from piping analysis is determined and compared to the BAC. The maximum stress location must be on or below the BAC to satisfy the LBB criteria.

The bounding analysis methods are described in Appendix 3B. Preparation of BAC provides an evaluation method meeting the requirements and guidelines of the NRC documents.

Piping analysis boundary is from one terminal end or anchor to the other terminal end or anchor. Connection to a larger pipe or a component of larger diameter is generally considered a terminal end. LBB evaluation is based on the fracture mechanics of cracks and analysis of break mechanism which compares the selected leakage cracks with critical crack sizes. This analysis method is outlined below.

Crack stability is demonstrated by leak detection analysis on the assumption that postulated circumferential cracks are limited if the stresses are on or below the "LBB BAC."

3.6.3.4.1 Leak Detection Capability

Leakage flaws are postulated for piping identified in Subsection 3.6.3.1 as following. Sizes of postulated flaws are sufficiently large so that leaks can be detected by a sufficient margin. Leak rate of 10 times the capability of the leak detector is postulated for normal operating load combinations.

Rated detection capability of the leak detector for reactor coolant in the containment is ~~1.0~~ 0.5 gpm within one hour of detector response time. The methods used for the reactor coolant are the containment sump water levels, inventory balance, and the radiation in the environment of containment. The method to detect leaks from the main steam pipe in containment is the containment sump water level. The condensate water flow rate of containment air cooler, containment environmental pressures, and temperatures also suggest the possibility of leakage.

3.6.3.4.2 Stability and Critical Crack Sizes

The local and global break mechanisms are evaluated, as required, to provide a margin to the break size and load. Local mode of breaks deals with the behaviors of crack tips: slowdown, start, development, and instability. Mechanisms of local breaks are evaluated by using J integration method for ferritic steel pipes. Global break mode deals with the behaviors of all cross sections: initial yield, strain hardening, and plastic hinge formation. Global break mechanisms (critical loading method) are evaluated for the stainless steel pipes not containing the casting materials and shielded metal arc weld. From these evaluations, the critical crack sizes are determined, so that the cracks larger than critical crack size have unstable features of growth.

3.6.3.4.3 Allowable Standards

Crack size margin is determined by comparing the crack sizes determined above to the critical crack size. The critical crack size is determined by adding maximum individual loads by absolute summation. The margin of two applies to the leakage crack size compared to ~~margin between~~ the critical crack ~~flaw~~ size with the size of flaw large enough so that the leakage from the flaw during normal operation is 10 times greater than the minimum leakage the detection system is capable of sensing ~~and the 10-gpm leakage size flaw~~. The margin of 1.0 on the load is used, since the loads are added by absolute sum.

3.6.3.4.4 Bounding Analysis Methods

BACs are developed for each different combination of material type, pipe size, pressure and temperature. These curves provide “Maximum crack stress” versus “Corresponding stress meeting LBB standards.” These curves are used to satisfy the requirements for LBB.

Critical location is the maximum stress location determined by the results of pipe stresses.

At all critical locations, loads related to maximum stress calculation are added by using absolute sums. Loads are combined as shown below.

$$|\,Pressure\,| + |\,dead\ load\,| + |\,thermal\ (100\%\ power)^*\,| + |\,SSE\,|$$

* Including applicable (thermal) stratification loads

Standard stresses are calculated according to the following combinations of loads by using arithmetical sum method at critical location.

$$Pressure + dead\ load + thermal\ (100\%\ power)^*$$

* Including applicable (thermal) stratification loads

Stresses by longitudinal force and bending moment are calculated by the following formula.

$$\sigma = F/A + M/Z$$

where

σ = Stress

F = Longitudinal force

M = Bending moment

A = Cross-sectional area

Z = Section modulus

DCD_3.7.1-04

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 ~~[which corresponds to rock material for the purpose of defining input motion in accordance with Section 1.2 of ASCE 4-98 (Reference 3.7-9)]~~, the site response analysis ~~has to~~ 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.

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, ~~V~~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

Analyses of seismic category I and II subsystems are primarily performed using equivalent static load analysis or modal response spectra analysis. The input seismic loads are defined by ISRS that are obtained from the time history analyses of the major seismic category I buildings and structures. Seismic subsystems are discussed in Subsection 3.7.3, and the modal response spectra and equivalent static load analysis methods are discussed in Subsection 3.7.3.1.

Seismic anchor motions are taken into consideration for all seismic analysis methods used in the design of seismic category I and seismic category II SSCs. All analysis approaches have been based on linear elastic analysis of SSCs, with allowable stresses within the elastic limits for seismic loads and load combinations as delineated in Section 3.8. Except in limited cases where permitted by code, inelastic behavior is not considered for seismic loads and load combinations in performing the plant design, however, limited inelastic and nonlinear behavior for seismic loading conditions may be used for site-specific COL designs, future operability analyses or as-built evaluations, as permitted in SRP 3.7.2 (Reference 3.7-16). Nonlinear and inelastic behavior is considered for certain loads and load combinations involving impact and impulsive loading, as discussed in Subsection 3.8.4.

3.7.2.2 Natural Frequencies and Responses

~~As discussed further below in Subsection 3.7.2.3, the seismic analysis and design of the R/B, PCCV, and containment internal structure and their common basemat are based on a coupled model that consists of a detailed RCL lumped mass stick model coupled with a combined R/B-PCCV-containment internal structure lumped mass stick model. The seismic analysis of the RCL-R/B-PCCV-containment internal structure coupled model is the subject of a Technical Report (Reference 3.7-18). The results obtained from the seismic analysis of the coupled model are presented in the report and reconciled, as necessary, with those results obtained from the current seismic analysis.~~

The ~~current~~ seismic analysis and structural design of the R/B, PCCV, and containment internal structure and their common basemat are based on a combined lumped mass stick model consisting of three lumped mass stick models (for each of the three structures) that are all rigidly cross-connected at the surface of the common basemat, as discussed further below in Subsection 3.7.2.3. The natural frequencies and modal responses for the combined R/B-PCCV-containment internal structure model (which includes the masses of the RCL system but is not coupled with the RCL lumped mass stick model) are presented in Appendix 3H. As discussed in a Technical Report (Reference 3.7-18), seismic analysis of the R/B, PCCV, and containment internal structure and their common basemat is also performed on a coupled model that consists of a detailed RCL lumped mass stick model coupled with a combined R/B-PCCV-containment internal structure lumped mass stick model. The results obtained from the seismic analysis of the coupled model are reconciled, as necessary, with those results obtained from the seismic analysis of the un-coupled model. The seismic analysis of the coupled model is also used to develop ISRS as discussed in Subsection 3.7.2.5.

It should be noted that the results obtained from the seismic analysis of the lumped mass stick models are obtained considering the potential effects of SSI. The site-independent SSI analyses, which are discussed further in Subsection 3.7.2.4, consider four generic subgrade conditions: (1) soft soil with shear wave velocity

Since there will not be any seismic category I SSCs contained within seismic category II buildings, the development of ISRS is not necessary.

Using the computer program NASTRAN (Reference 3.7-20), detailed FE models are developed for the major seismic category I and seismic category II structures, primarily to be utilized as static analysis models for structural design based on loads and load combinations as described in Section 3.8. However, the NASTRAN FE models are also used for validation of the dynamic lumped mass stick models and the seismic analysis results, as discussed later in this section. Final results obtained from analysis of the NASTRAN FE models are validated by comparison to the results of separate ANSYS (Reference 3.7-21) FE model analyses.

3.7.2.3.2 R/B, PCCV, and Containment Internal Structure Lumped Mass Stick Models

~~The seismic analysis and design of the R/B, PCCV, and containment internal structure and their common basemat are based on a model that consists of a detailed RCL system lumped mass stick model coupled with a combined R/B-PCCV-containment internal structure lumped mass stick model. The seismic analysis of the RCL-R/B-PCCV-containment internal structure coupled model is addressed in a separate Technical Report (Reference 3.7-18). The results obtained from the seismic analysis of the RCL-R/B-PCCV-containment internal structure coupled model are compared to the current seismic analysis of the R/B, PCCV, and containment internal structure and their common basemat and design adjustments due to the reconciliation are made, as necessary, and addressed in the Technical Report.~~

The ~~current~~ seismic analysis and structural design of the R/B, PCCV, and containment internal structure and their common basemat is based on a combined lumped mass stick model consisting of three lumped mass stick models (for each of the three structures) that are all rigidly cross-connected at the surface of the common basemat. Included in this combined model is the calculated mass of the RCL seismic subsystem, which is conservatively rounded up by 20% and distributed proportionately to the appropriate model nodes based on the mass distribution of the RCL system. This is considered a conservative approach for the seismic analysis and design of the R/B, PCCV, and containment internal structure and their basemat because the round-up compensates for uncertainties in the mass distribution and potential effects due to coupling of the RCL subsystem, such as shifts or changes in natural frequency and response modes. Appendix 3H presents the detailed model descriptions, seismic analysis results, and the associated tables and figures that are particular and specific to the uncoupled RCL approach ~~currently~~ used for the R/B, PCCV, and containment internal structure. Similarly, Appendix 3C presents the analytical methods and modeling approaches ~~currently~~ used for the uncoupled RCL seismic subsystem analysis. Seismic analysis of the R/B, PCCV, and containment internal structure and their common basemat is also performed using a coupled model that consists of a detailed RCL lumped mass stick model coupled with the combined R/B-PCCV-containment internal structure lumped mass stick model, as documented in a separate Technical Report (Reference 3.7-18). The results obtained from the seismic analysis of the coupled model are reconciled, as necessary, with those results obtained from the seismic analysis of the un-coupled model.

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 Subsection 3.3.17 of ASCE 4-98 (Reference 3.7-9), the LB and UB values for initial soil shear modulii (G_s) are established as follows:

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

where C_v is a variation factor. ASCE 4-98 (Reference 3.7-9) mandates that value of C_v must be greater than 0.5. When insufficient data are available to address uncertainties in properties of deep soil layers, C_v must be greater than 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). 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 ~~is to~~ 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. 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 applicability for the specific site conditions. The strain-compatible soil profiles for the site-specific verification SSI analyses of the major seismic category I structures can be obtained from the results of the site response analyses that are performed to calculate site-amplification factors for the development of GMRS, as described in Subsection 3.7.1.1.

The depth of the water table must be considered when developing the P-wave velocities of the submerged subgrade materials. Significant variations in the water table elevation and significant variations of the subgrade properties in the horizontal direction are addressed by using additional sets of site profiles.

To assure the proper comparability, the site-specific verification SSI analyses must use the same verified and validated lumped mass stick models of the building super-structure as those used for the US-APWR standard plant design. FE analyses are employed to evaluate the flexibility of the basemat and the embedded portion of the building. The floor slabs located at and above the ground surface are assumed absolutely rigid. In order to verify the converted structural model, a site-specific SSI analysis is performed with hard rock site profile that simulates fixed base conditions. The results of the SSI analysis with hard rock site profile are to match closely with the results from the analysis of fixed base stick model. In accordance with requirements of Section 1.2 of RG 1.61 (Reference 3.7-15), the lower OBE damping values in Table 3.7.3-1(b) are assigned to the structural model as complex damping.

The results for 5% damping ISRS at major floor locations and soil pressures on the basement exterior walls that are obtained from all considered soil cases are enveloped and broadened. The plots, tables, and digitized data are then documented for review and comparison with the corresponding results from site-independent analyses. The COL Applicant is to verify that the results of the site-specific SSI analysis for the broadened ISRS and basement walls lateral soil pressures are enveloped by the US-APWR standard design.

The analyses use input soil properties derived from geotechnical investigations of the site that are compatible to the strains generated in the subgrade by the input design earthquake. The uncertainties and variations of the subgrade properties are considered using the methodology previously described for the development of the strain-compatible site profiles for the site-specific SSI analysis of the major seismic category I structures. The control motions are developed from site-specific FIRS that are described in Subsection 3.7.1.1 and applied to the models at the bottom of the basemat.

In accordance with Section 1.2 of RG 1.61 (Reference 3.7-15), the lower OBE damping values in Table 3.7.3-1(b) are assigned to the structural model used for development of ISRS if the site-specific SSE is not large enough to use the damping values and Table 3.7.3-1(a), and OBE design loads. ISRS do not need to be generated for seismic category II buildings and structures which do not contain or support safety-related SSCs, such as the T/B and A/B.

Simplified SSI modeling approaches, such as a lumped parameter model, can be employed for the site-specific seismic response analyses of seismic category I and II buildings and structures that are not part of the US-APWR standard design if it is demonstrated that for the specific site conditions the following applies:

- The basemats are much stiffer than the supporting subgrade
- The SSI impedance functions remain relatively constant in the range of frequencies important for the design
- The consideration of basemat embedment yields conservative results

In accordance with SRP 3.7.2 (Reference 3.7-16), Section II.4, fixed base response analysis can be performed if the basemats are supported by subgrades having a shear wave velocity of 8,000 ft/s or higher, under the entire surface of the foundation. ~~In accordance with Subsection 3.3.1.1 of ASCE 4-98 (Reference 3.7-9), fixed base response analysis can be performed if the basemats are supported by subgrades that meet the following condition.~~

~~The frequency of the system consisting of subgrade stiffness (SSI impedance) and the combined lumped mass inertia of the whole super-structure and the basemat (i.e., by assuming the super-structure and the basemat are absolutely rigid) are twice the frequency obtained from the fixed base modal analysis of the superstructure.~~

3.7.2.5 Development of Floor Response Spectra

ISRS for the major seismic category I structures as well as design spectra for the RCL system are required to be developed from the results of the site-independent seismic analysis of the coupled RCL-R/B-PCCV-containment internal structure lumped mass

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- The maximum spectral acceleration at each frequency obtained from the seismic analysis of any general subgrade conditions is selected for the envelope.
- The enveloped ISRS are smoothed and broadened by +/-15%. The valleys in the enveloped ISRS are filled when necessary to capture potential shifts in the seismic response caused by soil properties that are different from, but bounded by, the four generic soil conditions of the standard plant. ~~The broadened response spectra method discussed in Subsection 3.7.3.1 is used or a~~Alternatively in some locations, the peak shifting method described in Subsection 3.7.3.1 can be used instead of the broadened response spectra method.

ISRS are not required for non-seismic category I building structures, such as the AC/B, A/B and T/B, since no safety-related systems and components are present in non-seismic category I buildings and structures. The design, installation, and mounting of non safety-related systems and components in these buildings are based on the applicable site-specific building codes and standards.

3.7.2.6 Three Components of Earthquake Motion

As previously discussed in Subsection 3.7.1.1, the seismic analyses of the major seismic category I structures are based on one set of three mutually orthogonal artificial time histories, with each of the three directional components being statistically independent of the other two. The acceleration time histories of the horizontal H1 and H2 components of the earthquake are applied in N-S direction and E-W directions respectively. The acceleration time history V is applied in the vertical direction.

The three components of the earthquake are applied on the lumped mass stick model separately and the maximum accelerations of the response in the three orthogonal directions are calculated at each lumped mass point. The maximum responses of interest of SSCs obtained from the responses of each of the three components of motion are then combined using SRSS or the Newmark 100%-40%-40% method in accordance with RG 1.92, Rev.2 (Reference 3.7-27). The combined maximum accelerations, obtained through the process described previously in Subsection 3.7.2, are then used as basis for development of the SSE loads used for the design of structural members, components and connections of US-APWR standard plant. These SSE design loads are applied as static loads on the detailed FE model in conjunction with other design loads and load combinations.

The development of the ISRS uses the SRSS method to combine the responses from the three components of the earthquake motion.

Although the above approach has been used for seismic analysis of the major seismic category I structures, seismic responses of other seismic systems and subsystems due to the three components of earthquake motion can be combined using any one of the following methods in accordance with RG 1.92, Rev.2 (Reference 3.7-27):

(i) The peak responses due to the three earthquake components from the response spectra and equivalent static analyses are combined using the SRSS method.

(ii) The peak responses due to the three earthquake components are combined directly, using the Newmark combination method that assumes that when the peak response from one component occurs, the responses from the other two components are 40% of the peak (100%-40%-40% method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus) are considered.

(iii) The time-history of the responses from the three earthquake components that are applied simultaneously can be combined algebraically at each time step to obtain the combined response time-history. The design seismic loads are selected from the maximum values or the most critical combination of values extracted from the time history results representing the responses directly related to the design of the particular member considering sign reversals, such as the relevant internal forces or stresses in the member. Due to the uncertainties introduced by phasing effects, the design does not use time history results for other responses, such as accelerations or displacements at points in time that are indirectly related to the basic design inputs.

3.7.2.7 Combination of Modal Responses

As previously discussed, the seismic responses of the major seismic category I structures lumped mass stick models are obtained utilizing the direct integration method of time history analyses. As described in Subsection 3.7.2.1, the damping matrix is not proportional to the stiffness and mass matrix of the combined soil-structure model, so the decomposition of the equations of motion in generalized coordinates is not possible. Therefore, the response of the major seismic category I structures are obtained directly by integrating the equations of motions presented in Equation 3.7.2-1 without performing modal decomposition and subsequent modal superposition.

When the modal superposition time history analyses or response spectra analyses are used for seismic design of other seismic category I and seismic category II systems and subsystems, all necessary modes are included in order to capture a minimum of 90% of the cumulative mass of the building or structure being analyzed. In modal superposition, only modes with frequencies less than the frequencies defining the ZPA response participate in the modal solution. The modal contribution of the residual rigid response for modes with frequencies greater than ZPA frequency is accounted for by using the missing mass method. As permitted by RG 1.92, Rev.2 (Reference 3.7-27), the missing mass contribution, scaled to the instantaneous input acceleration, is treated as an additional mode in the algebraic summation of modal responses at each time step. The missing mass contribution is considered for all DOF.

When the response spectra method of analysis is used (see Subsection 3.7.3.1 for a discussion of response spectra methods of analysis), modal responses have been combined by one of the RG 1.92, Rev.2 (Reference 3.7-27), methods, or by the 10% grouping method described below. In some applications, the more conservative modal combination methods contained in Rev.1 of RG 1.92 (Reference 3.7-28) are also used, as permitted in Revision 2 of RG 1.92 (Reference 3.7-27).

For the ~~10%~~ grouping method, the total unidirectional seismic response for subsystems is obtained by combining the individual modal responses using the SRSS method for frequencies spaced more than 10%.

For subsystems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen so that the differences between the frequencies of the first mode and the last mode in the group do not exceed 10% of the lower frequency.

The combined total response for systems having such closely spaced modal frequencies is obtained by adding to the SRSS of all modes the product of the responses of the modes in each group of closely spaced modes ~~times appropriate coupling factors~~.

This can be represented mathematically as follows:

$$\cancel{R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{k=Mj}^{Nj-1} \sum_{\ell=k+1}^{Nj} R_k \; R_\ell \; \varepsilon_{k\ell}}$$

$$R^2 = \sum_{k=1}^{N} R_k^{\ 2} + \sum_{q=1}^{P} \sum_{l=i}^{j} \sum_{m=i}^{j} \left| R_{lq} \cdot R_{mq} \right| \qquad l \neq m$$

where

$R_{\cancel{T}}$	=	total unidirectional response
R_k	=	the peak value of the response due to the kth mode
$\cancel{R_j}$ R_{lq}, R_{mq}	=	~~absolute value of response of mode l~~ are the modal responses, R_l and R_m within the qth group
N	=	total number of modes considered
~~S~~ P	=	number of groups of closely spaced modes
$\cancel{M_j}$ i	=	lowest modal number associated with group ~~j~~ q of closely spaced modes
$\cancel{N_j}$ j	=	highest modal number associated with group ~~j~~ q of closely spaced modes
~~ε_{kl}~~	~~=~~	~~coupling factor, defined as follows:~~

$$\cancel{\varepsilon_{k\ell} = \left(1 + \frac{(w'_k - w'_\ell)^2}{(\beta'_k w_k + \beta'_\ell w_\ell)^2}\right)^{-1}}$$

~~and,~~

$$\cancel{w'_k = w_k \left[1 - (\beta_k)^2\right]^{1/2}}$$

$$\cancel{\beta'_k = \beta_k + \frac{2}{w_k t_d}}$$

~~where~~

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~~ω_k = frequency of closely spaced mode k~~

~~β_k = fraction of critical damping in closely spaced mode k~~

~~t_d = duration of the earthquake~~

Alternatively, a more conservative ten percent grouping method can be used in the seismic response spectra analyses. The groups of closely spaced modes are chosen so that the difference between two frequencies (the first and last mode in a group) is no greater than 10%. Therefore,

$$\cancel{R_T^2 = \sum_{i=1}^{N} R_i^2 + 2\sum \varepsilon_{kl} R_k R_l}$$

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

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

~~where~~

~~ω_k = first circular frequency in the group~~

~~ω_l = last circular frequency in the group~~

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

The 10% grouping method is more conservative than the grouping method because the same mode can appear in more than one group.

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

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

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

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

The plan dimension of each PS/B is nominally 111'-6" x 66'-0" between centerlines of exterior walls. Each PS/B is a reinforced concrete structure with one floor level under ground and the main floor level above ground. The design philosophy of the PS/Bs is stated as follows.

- The east and west PS/Bs are nearly identical structurally, and one bounding analysis is performed to represent both.
- Reinforced concrete structure of the PS/Bs is designed by ACI 349 code (Reference 3.7-31).
- The SSE load condition is the same as for the R/B.
- The design of the PS/Bs is based on a static analysis utilizing a three-dimensional FE model, and a seismic dynamic analysis using a three-dimensional lumped mass model.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

ISRS are generated for all US-APWR seismic category I structures.

To account for variations in the structural frequencies due to the uncertainties in parameters, such as material and mass properties of the structures, damping values, soil properties, SSI analysis techniques, and the seismic modeling methods, the computed ISRS are smoothed and the peaks associated with the structural frequencies are broadened by ±15% in accordance with RG 1.122 (Reference 3.7-26).

As described in Subsection 3.7.2.4, the seismic analyses of standard plant seismic category I buildings include four generic supporting media intended to bound the varying subgrade conditions and to account for the variation in SSI analysis techniques and seismic modeling methods. The soil properties bound shear wave velocities ranging from 1,000 ft/s, which is the LB of what is considered seismically competent material, up to an UB of 6,500 ft/s corresponding to rock. The seismic models also consider a fixed-base analysis considering a hard rock support medium with a shear wave velocity of 8,000 ft/s. This wide range of supporting media conditions captures SSI and other seismic response effects, including the resulting variances in ISRS. Further, valleys between the peaks of the standard plant design ISRS are filled in, if necessary to account for variations of site-specific soil properties within the range of supporting media conditions considered in the standard plant design.

3.7.2.10 Use of Constant Vertical Static Factors

The plant design does not utilize constant vertical static factors in the seismic design. The vertical component of the seismic motion is obtained using one of the analysis methods described in Subsection 3.7.2.1. The vertical component is combined with the horizontal components of the seismic motion as described in Subsection 3.7.2.6.

3.7.2.11 Method Used to Account for Torsional Effects

The seismic analyses of seismic category I buildings and structures incorporate the torsional DOF in the mathematical models, as discussed in Subsection 3.7.2.3.

The torsional effect is included in accordance with SRP 3.7.2 Section II (Reference 3.7-16) in the design of all seismic category I and II structures by use of the following process:

- Computation of the horizontal mass properties on each floor elevation of the building lumped mass stick models, and the corresponding nodal accelerations.
- Computation of the accidental eccentricity by determining the distance between the center of mass at each floor with respect to its center of rigidity, computed separately for each floor level, as required by ASCE 4 (Reference 3.7-9) Subsection 3.1.1(d).
- The torsional moments due to eccentricities of the masses at each floor elevation are assumed to act in the same direction on each structure unless otherwise demonstrated in the analysis. Both positive and negative values are considered in order to capture worst case effects.

For member design only, an additional building torsion (accidental torsion) equal to story shear force with a moment arm of 5% of the plan dimension of the floor perpendicular to the direction of the applied motion, as stipulated in ASCE 4-98 (Reference 3.7-9) Subsection 3.1.1 (e), is applied in the resultant force calculations. As explained in ASCE 4-98 Subsection 3.3.1.2 (a), this accounts for effects of non-vertically incident or incoherent waves.

The methods and approaches used to capture torsional effects in seismic category I buildings are described further in Subsection 3.7.2.3.

3.7.2.12 Comparison of Responses

The major seismic category I structures are analyzed using time history analysis methods.

As described in Subsection 3.7.1.1, the time history analyses are based on design ground motion time histories which have been artificially synthesized and meet the requirements of “Acceptance Criteria, Design Time History Option 1: Single Set of Time Histories, Approach 2”, NUREG-0800, SRP 3.7.1, Section II (Reference 3.7-10). ~~As required by Approach 2, the response spectra obtained from the artificial ground motion time histories have been compared with the target response spectra to assure that the spectra derived from the time histories match/envelope the CSDRS with an approximate mean based fit.~~ Since only a time history analysis method is used, comparison of the responses between the response spectrum method and a time history analysis method, as per SRP Section 3.7.2.II.12 (Reference 3.7-16), is not applicable.

3.7.2.13 Methods for Seismic Analysis of 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.

3.7.2.14 Determination of Dynamic Stability of Seismic Category I Structures

aboveground tanks, and the like, which are exterior to the R/B, PCCV, PS/Bs, and the ESWPT.

Each non-category I system and component is designed to be isolated from any seismic category I systems and components by either a constraint or barrier, or is remotely located with regard to the seismic category I systems and components. If it is not feasible or practical to isolate the seismic category I systems and components, adjacent non-category I systems and components are analyzed for the same seismic input motion that is applicable to the seismic category I systems and components. In this case, the analysis demonstrates position retention of the non-category I subsystems and components, with no adverse interaction effects on seismic category I SSCs. For non-category I systems and components attached to seismic category I systems and components, the dynamic effects of the non-category I subsystems and components are simulated in the modeling of the seismic category I systems and components. The attached non-category I systems and components, up to the first anchor beyond the interface, are designed in such a manner that during an earthquake of SSE intensity, the structural integrity and safety functions of the seismic category I systems and components are not jeopardized.

Seismic and dynamic qualification of mechanical and electrical equipment and subsystems performed by testing is discussed in Section 3.10 and Appendix 3D. Mechanical subsystems include mechanical equipment, piping, vessels, tanks, heat exchangers, valves, and instrumentation tubing and tubing supports. The seismic analysis of mechanical subsystems is addressed in Sections 3.9 and 3.12. The RCL analysis is discussed in Appendix 3C.

A list of seismic category I mechanical and fluid systems, components, and equipment is given in Table 3.2-2. Seismic analysis of civil structural items related to those subsystems is discussed in this subsection.

3.7.3.1 Seismic Analysis Methods

Modal response spectra analysis, time history analysis, or equivalent static load analysis methods may be used for seismic analysis of seismic category I subsystems. The methods are the same as those discussed in Subsection 3.7.2.1 and conform to the requirements of SRP 3.7.1 and SRP 3.7.2 (References 3.7-10 and 3.7-16).

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.

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.

3.7.3.1.1 Equivalent Static Load Method of Analysis

The equivalent static load method involves the use of equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location. Loads, stresses, or deflections obtained using the equivalent static load methods are adjusted to account for the relative motion between points of support when significant.

3.7.3.1.2 Single DOF or Rigid Structures and Components

For rigid structures and components, or for cases where the response is such that the system has a single DOF, the following procedures may be used:

- For rigid SSCs (fundamental frequency greater than 50 Hz), an equivalent seismic load is defined for the direction of excitation as the product of the component mass and the ZPA value obtained from the applicable ISRS.
- A rigid component (fundamental frequency greater than 50 Hz), whose support can be represented by a flexible spring, can be modeled as a single DOF model in the direction of excitation (horizontal or vertical directions). The equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value corresponding to the natural frequency of the support from the applicable ISRS. If the frequency is not determined, the peak acceleration from the applicable ISRS times a factor of 1.5 is used. Supports which have been determined to have natural frequencies less than the frequency corresponding to the peak floor acceleration (i.e., whose natural frequencies are to the left of spectra peak on an acceleration versus the frequency spectra plot) also utilize the peak acceleration times a factor of 1.5.
- If the structure, equipment, or component has a distributed mass whose dynamic response is single mode dominant, the equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the component natural frequency from the applicable ISRS times a factor of 1.5, with exceptions noted as follows. A factor of less than 1.5 may be used if justified, such as using a factor of 1.0 when the component natural frequency is in the rigid range (greater than 50 Hz), such that no dynamic amplification will occur. A factor of 1.0 is used for structures or equipment that can be represented as simply supported, fixed-simply supported, or fixed-fixed beams as discussed in References 3.7-36 and 3.7-37. In accordance with ASCE 4-98, Subsection 3.2.5.2 (Reference 3.7-9), for cantilever beams with uniform mass distribution, the equivalent-static-load base shear is determined using the peak acceleration, and the base moment is determined using the peak acceleration times a factor of 1.1. If the frequency of a structure, equipment, or component is not determined, the peak acceleration from the

applicable ISRS times a factor of 1.5 is used. Any structures, equipment, or components which have been determined to have natural frequencies less than the frequency corresponding to the peak floor acceleration (i.e., whose natural frequencies are to the left of spectra peak on an acceleration versus the frequency spectra plot) also utilize the peak acceleration times a factor of 1.5 unless otherwise justified.

3.7.3.1.3 Multiple DOF Response

This procedure applies to piping, instrumentation tubing, conduit, cable trays, HVAC, and other structural subsystems consisting of multiple spans. The equivalent static load method of analysis can be used for the design of piping systems, and the instrumentation and supports that have significant responses at several vibrational frequencies. In this case, a static load factor of 1.5 is applied to the peak accelerations of the applicable ISRS, unless a lower value is justified. For runs with axial supports, the acceleration value of the mass of piping in its axial direction may be limited to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. The relative motion between support points is also considered.

3.7.3.1.4 Modal Response Spectra Analysis

The methods of modal response spectra analysis that have been utilized for the design of seismic category I and II SSCs are the envelope broadened response spectra method, the peak shifting method, the uniform support motion method and the independent support motion method, described in the following subsections.

3.7.3.1.5 Envelope Broadened Response Spectra Method

The envelope broadened response spectra method is based on the utilization of the ISRS that are developed for the US-APWR seismic category I structures and buildings. The envelope broadened response spectra method is discussed in Subsection 3.7.2.5. ~~The ISRS are developed by filling in the valleys between all peaks, and broadening the peaks, of the theoretical ISRS that are developed from the time history seismic analyses methods and models discussed in Subsection 3.7.2.~~

3.7.3.1.6 Seismic Response Spectra Peak Shifting

The peak shifting method may be used in place of the broadened spectra method. It determines the natural frequencies $(f_e)_n$ of the system to be qualified in the broadened range of the maximum spectra acceleration peak. If no equipment or piping system natural frequencies exists in the ±15% interval associated with the maximum spectra acceleration peak, then the interval associated with the next highest spectra acceleration peak is selected and used in the following procedure.

Consider all *N* natural frequencies in the interval:

$$f_j - 0.15f_j \leq (f_e)_n \leq f_j + 0.15f_j$$

where

and high frequency modes to obtain the resultant forces, moments, displacements, accelerations, and support loads. The total seismic responses are combined by the SRSS method for all three earthquake directions.

3.7.3.1.7.2 Independent Support Motion Method

When there is more than one supporting structure, the independent support motion method for seismic response spectra may be used.

Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the ~~SRSS~~ absolute sum method. The analysis of piping systems with multiple supports and independent inputs will be consistent with the recommendation provided in Section 2.4 of NUREG-1061, Volume 4 (Reference 3.7-46), which describes independent support motion (ISM) methodology, sequence of combination, and high frequency modes. If the ISM method is utilized, the criteria presented in NUREG-1061 related to the ISM method are required to be followed according to SRP subsection 3.7.2.II, item 9 (Reference 3.7-16) as provided under SRP Acceptance Criteria. The displacement response in the modal coordinate becomes:

$$\cancel{q_i = [\sum_{j=1}^{N} (P_{ij}\ d_{ij})^2]^{1/2}}$$

$$q_i = \Sigma P_{ij}\, d_{ij}$$

A support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

3.7.3.1.7.3 Analysis of Seismic Subsystems versus Qualification by Testing

For the purpose of seismic and dynamic qualification of civil structure-related SSCs by *analysis* using the methods described above in this section, the rigid range is defined as having a natural frequency greater than 50 Hz. This is consistent with the CSDRS defined in Subsection 3.7.1.1. However, for the purpose of *testing* equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment to be tested is sensitive to the response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz. This approach is further clarified in the following paragraphs.

Historically, there have been occurrences of ground motions which have caused an exceedance of a plant's design spectra in the high frequency range, where high frequency is defined as 10 Hz or greater. Based on this nuclear plant operating experience, the high frequency response motion exceedances were found to be non-damaging to passive civil structure-related components such as those addressed in the section above, which are typically qualified by analysis. However, nuclear industry experience has found that certain SSCs, in particular components such as relays and other electrical and instrumentation and control devices whose output signals could be affected by high frequency excitation, are potentially sensitive to high frequency motion and can be damaged by high frequency exceedances of the design spectra.

DCD_3.7.3-09
DCD_3.7.3-13

3.7.3.2 Procedures Used for Analytical Modeling

Seismic subsystems are defined as those systems that are not analyzed in conjunction with basemats and subgrade, as previously discussed in Subsection 3.7.2. The procedures used for analytical modeling of subsystems ~~may be the same as those used for the major seismic category I and II building structures described previously in Subsection 3.7.2.3. These procedures~~ include the use of mathematical computer models comprised of nodes and elements used to represent connections and members. Depending on the complexity of the subsystem, the models may be lumped mass stick models or FE models. The models contain ~~are~~ sufficient~~ly~~ detail~~ed~~ and DOFs to represent the ~~overall~~ structural and seismic response of the subsystem, and are incorporated into the overall building model when required by the coupling criteria discussed in Subsection 3.7.2.3.4. Depending on the complexity of the seismic subsystem, structure, or component being analyzed, detailed member design may be performed by hand calculations using the results of the overall building structural and seismic analyses. Alternatively, the computer model may be sufficiently detailed to be used for the design calculation of the individual members. In all cases, the computer programs used for analytical modeling of subsystems are verified and validated in accordance with ANSI/ASME NQA-1-2004 (Reference 3.7-23) requirements.

3.7.3.3 Analysis Procedure for Damping

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The SSE damping values to be used in the dynamic analysis for various seismic category I and II subsystems and their related supports are shown in Table 3.7.3-1(a). The damping values are based on RG 1.61 (Reference 3.7-15) and ASCE Standard 4-98 (Reference 3.7-9). The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7% of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining NRC review for acceptance on a case-by-case basis.

For subsystems that are composed of different material types, the composite modal damping approach with either the weighted mass or stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems.

~~Composite modal damping for coupled building and piping systems is used for piping systems that are coupled to the RCL and the containment internal structure. Alternatively, Rayleigh damping with direct integration may be used. Seismic analysis of the RCL is addressed in a separate Technical Report (Reference 3.7-18), and piping systems coupled to the RCL are also addressed therein.~~

Piping systems are analyzed for SSE using 4% damping. Alternately, frequency-dependent damping values may be utilized as noted and described in Tables 3.7.3-1(a) and 3.7.3-1(b). The seismic analysis of piping and other mechanical subsystems is addressed if further detail in Sections 3.9 and 3.12.

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, 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 ~~is to~~ 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.

COL3.7(9) The COL Applicant is to assure that the design or location of any site-specific seismic category I SSCs, for example buried yard piping 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) It is the responsibility of the COL Applicant to confirm the masses and frequencies of the PCCV polar crane and fuel handling crane and to determine if coupled site-specific analyses are required.

COL3.7(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.

COL3.7(13) The COL Applicant is to set the value of the OBE that serves as the basis for defining the criteria for shutdown of the plant, according to the site specific conditions.

3.7-38 Independent Support Motion (ISM) Method of Modal Spectra Seismic Analysis, Task Group on Independent Support Motion as Part of the PVRC Technical Committee on Piping Systems, December 1989.

3.7-39 Seismic Instrumentation, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, United States Nuclear Regulatory Commission Standard Review Plan 3.7.4, Revision 2, March 2007.

3.7-40 Nuclear Power Plant Instrumentation for Earthquakes, Regulatory Guide 1.12, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, March 1997.

3.7-41 Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions, Regulatory Guide 1.166, U.S. Nuclear Regulatory Commission, Washington, DC, March 1997.

3.7-42 Standardization of the Cumulative Absolute Velocity, Electric Power Research Institute TR-100082, December 1991.

3.7-43 A Criterion for Determining Exceedance of the Operating Basis Earthquake, Electric Power Research Institute NP-5930, July 1988.

3.7-44 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable, Regulatory Guide 8.8, Rev.3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

3.7-45 Guidelines for Nuclear Plant Response to an Earthquake, Electric Power Research Institute NP-6695, December 1989.

3.7-46 Evaluation of Other Dynamic Loads and Load Combinations, NUREG-1061, Volume 4, U.S. Nuclear Regulatory Commission Piping Review Committee, December, 1984.

- Results from other similar verified programs

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis is not used for the US-APWR.

3.9.1.4 Considerations for the Evaluation for the Faulted Condition

Analytical methods used to evaluate faulted condition (Level D loading) for ASME Code, Section III (Reference 3.9-1), Class 1, 2, and 3 components are described in Subsection 3.9.3.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The testing of piping vibration, thermal expansion, and dynamic effects occurs during three phases which make up the initial test program (ITP) as discussed in Subsection 14.2.1.2: construction acceptance testing, pre-operational testing, and hot functional (startup) testing. The ITP is implemented to verify that the piping and piping restraints for all ASME Code, Section III (Reference 3.9-1) Class 1, 2, and 3 piping systems, other high-energy piping systems inside seismic Category I structures, high energy portions of systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, and seismic Category I portions of moderate-energy piping systems located outside containment will remain within acceptable limits when subjected to piping vibrations and dynamic transients such as those caused by in-line component trips. ~~The~~ Other SSCs for which preoperational and startup testing is performed are identified in Subsection 14.2.1~~, which includes Class 1, 2, and 3 piping systems required by the ASME Code, Section III (Reference 3.9-1) to undergo a pre-operational test program~~. When applicable, instrumentation lines are included up to the first support in each of three orthogonal directions from the process pipe or equipment connection point.

The construction test phase involves checking the as-built piping systems, supports, and associated components for correct installation. The piping, pipe supports, and equipment supports are checked for proper assembly and design settings. The cold settings and cold gaps are recorded for the major system pipe supports, whip restraints, equipment, and equipment supports. The major piping systems checked are the reactor coolant, the RHR, the main steam, and the feedwater systems.

The purpose of pre-operational test program is to assure ASME Code, Section III (Reference 3.9-1), Class 1, 2, and 3, and other high-energy or seismic category I piping systems meet functional design requirements and that piping vibrations are within acceptable levels.

Thermal monitoring of the systems is also performed as required during pre-operational testing to assure that thermal predicted movements meet the required design considerations within appropriate support and whip restraint gaps. Excessive thermal deflections are noted and checked against as-analyzed piping results.

A check of snubber operability is made by recording hot and cold positions and comparing these positions to calculated hot and cold positions. The list of snubbers on systems, which experience sufficient thermal movement to measure snubber travel from cold to hot position, will be provided as part of the ITP plan. In addition, the ITP plan will include the procedure necessary to verify snubber operability when no snubber piston movement is noted.

During the hot functional (startup) testing, the piping systems which are specified as part of the ITP are operated to check the performance characteristics of critical pumps, valves, controls, and auxiliary equipment. The flow modes of operation and transients that are performed during the test include pump trips and valve closures. In the case of the RCS heatup tests, transients that are applied to the system include the RCP start, RCP trip, the operation of pressure-relieving valves, and closure of a turbine stop valve. ~~The~~ Additional requirements of startup testing are outlined in Subsection 14.2.1.2.3.

3.9.2.1.1 System Vibration and System Dynamic Effects Tests

Transient-induced vibrations and steady-state vibrations are possible within operational conditions. Transient-induced vibrations are a dynamic response to a transient, time-dependent forcing function such as fast valve closure. Steady-state vibrations are a constant presence, usually flow-induced.

The general requirements for vibration and dynamic effects testing of piping systems are specified in “Initial Test Programs for Water-Cooled Nuclear Power Plants”, RG 1.68 (Reference 3.9-12). More specific vibration testing requirements are defined in “Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems”, ASME OM (Reference 3.9-13). Detailed test specifications ~~are~~ written in accordance with ~~this standard~~ applicable portions of RG 1.68 and ASME OM are included in the test program description, and address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria. If vibration is noted beyond the acceptable levels, corrective restraints are to be designed, incorporated in the piping system analysis, and installed. The ITP plan will identify which measurement technique (e.g., visual observation, remote monitoring, or local measurements) is to be used for each of the piping systems covered by the vibration and dynamic effects testing program, the methods used to anticipate piping movements and deflections, and any computer codes used in the analysis including whether the computer codes have been reviewed and approved by the NRC. The ITP plan will include a list of locations in the specific piping systems that are selected for visual inspection and other measurements during the vibration, thermal expansion, and dynamic effects testing program. In addition, acceptance criteria for the deflection, pressure, and/or other appropriate criteria to be obtained during the tests will be included in the ITP plan to determine if the stress and fatigue limits are within design levels.

These tests are used to validate that the piping, components, restraints, and supports of specified high-energy and moderate-energy systems have been designed to withstand the dynamic effects of transient and steady-state flow-induced vibration and anticipated operational transient conditions.

modes. System thermal expansion tests are developed in accordance with the guidance of ASME OM (Reference 3.9-13), Part 7. If piping system restraints are determined during the test to be inadequate or are damaged, corrective restraints are to be installed and another test performed to determine whether the thermal motion has been reduced to an acceptable level. The detailed description of the thermal motion monitoring program will be included as part of the ITP plan. The thermal motion monitoring program will include verification of snubber movement, adequate clearances and gaps, the acceptance criteria, and how the motion is to be measured.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

3.9.2.2.1 Seismic Qualification Testing

Seismic category I mechanical equipment and supports are designed to safely withstand the effects of postulated earthquakes combined with appropriate effects of normal and accident conditions without loss of intended safety-related function. Seismic qualification is performed by either analysis, testing or by a combination of both testing and analysis. The methods for seismic qualification of safety-related mechanical equipment by testing is performed in accordance with the recommendations of “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations”, ANSI/IEEE Std 344-1987 (Reference 3.9-15), as endorsed by NRC, RG 1.100, Rev.2, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (Reference 3.9-16). The seismic qualification testing methods for safety-related mechanical equipment are described in Subsection 3.10.2.

3.9.2.2.2 Seismic System Analysis Methods

The seismic system and subsystem analysis methods (including response spectrum analysis, time history analysis, and equivalent static load analysis) are discussed in Subsections 3.7.2 and 3.7.3. The method of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Std 344-1987 (Reference 3.9-15) and Subsections 3.10.2 and 3.10.3. The majority of mechanical equipment is supplied by vendors that are required to provide a seismic qualification report that meets the design specification provided in the purchase order.

The stiffness of the seismic subsystem anchorage must be determined and the assumptions made in the seismic analysis must be verified as accurately reflecting the mounting condition.

Two separate models are used for the RCL seismic analysis. One for RCL seismic analysis, which consists of the use of stick mass spring model of SG, RCP, Reactor Pressure Vessel, loop piping and buildings. The other is used for seismic analysis of internal components of the SG. RCL seismic analysis is described in Appendix 3C. The SG seismic analysis is performed considering internal components.

responses without the core are also the same or slightly larger than those with the core. This is because the flow rate increases with the elimination of fuel assemblies and the subsequent pressure loss. Thus, in the preoperational test of the prototype plant, the results of vibration measurements after core loading are bounded by the measurements before core loading, and only measurements before core loading are necessary.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

3.9.2.4.1 Background

The first operational US-APWR reactor internals are classified as Prototype in accordance with RG 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 based on the designation of Regulatory Guide 1.20. The first COL Applicant is to commit to implement a pre-operational vibration assessment program and to prepare the final report consistent with guidance of RG 1.20 for a prototype. Subsequent COL Applicant need only provide information in accordance with the applicable portion of position C.3 of RG 1.20 for Non-Prototype internals.

Following the recommendation of Regulatory Guide 1.20 (Reference 3.9-21), a pre-operational vibration measurement program is developed for the first operational US-APWR reactor internals. Data will be acquired only during the hot functional test, before core loading. Analysis (Subsection 3.9.2.3) shows that the responses under normal operating condition with fuel assemblies in the core are almost the same or slightly smaller than those under hot functional test conditions without the core. Detailed information, including discussions about other effects with or without the core, is described in Subsection 3.4.3 of Reference 3.9-22.

Subsequent to the completion of the vibration assessment program for the first US-APWR reactor internals, the vibration analysis program will be used to qualify subsequent US-APWR under the criteria for non-prototype category I.

The needs for flow-induced vibration, measurement testing, of steam generator internals is discussed in Subsection 5.4.2.1.2.10.

3.9.2.4.2 Measurement Program

Measurements will be performed during the pre-operational test to confirm the vibration characteristics and structural integrity of the Prototype US-APWR reactor internals.

The acquired data will be used to confirm that unexpected, abnormal vibrations do not occur, and that the vibration responses are sufficiently small compared to an acceptance criterion based on the design fatigue curves in the ASME Code, Section III.

Instrumentation consisting of strain gages, accelerometers, pressure transducers and displacement transducers will be installed on selected components. Accelerometers and displacement transducers will be used to measure the responses of the reactor internals. Strain gages will be used to directly measure the strains at key connecting points, and dynamic pressure transducers will be used to measure the pressure fluctuations at selected locations. Some of the specific measurement locations are described below.

- Core barrel: Strains in the core barrel flange will be measured with strain gages. Shell-mode responses will be measured with accelerometers mounted on the wall of the core barrel.
- Lower core support plate: An accelerometer mounted near the center of the lower core support plate will be used to measure the vertical response of this component.
- Neutron reflector: Shell mode responses and vertical motions will be measured by accelerometers. Relative displacement between the core barrel and the neutron reflector will be measured by displacement sensors. The vibration responses of the tie rod will be measured by strain gages.
- Secondary core support assembly: Vibration responses will be measured by strain gages mounted on the diffuser plate support columns.
- RCCA guide tubes and upper support columns: Beam mode responses due to the cross-flow in the upper plenum will be measured by strain gages and accelerometers.
- Upper core support: The vertical response will be measured by an accelerometer mounted near the center of the upper core support plate. Horizontal responses will be measured by strain gages installed on the upper core support skirt.

Details for the data acquisition and reduction system, including redundancy, are described in Subsection 4.1 of Reference 3.9-22.

3.9.2.4.3 Inspection Program

The internal components of all US-APWR plants will be inspected before and after the hot functional test. The reactor internals will not be considered adequate and pass the comprehensive vibration assessment program unless no structural damage or change is observed.

3.9.2.4.4 Acceptance Criteria

The acceptance criteria of the pre-operational flow-induced vibration testing for reactor internals are as follows.

- Vibration measurement

  The measured rms vibration amplitudes will be multiplied by 4.5 to convert them into 0-peak values. The corresponding 0-peak stresses in key connecting components will be calculated from the measured vibration amplitudes or strains. These stresses must show sufficient safety margins based on the design fatigue curves in the ASME Code, Section III, Appendix-I.

- Inspection

  No structural damage or change is observed in the post-hot functional test inspection.

3.9.2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

NUREG-0800, SRP 3.9.2, Rev. 3 (Reference 3.9-25), requires that the Design Control Document (DCD) provide a detailed discussion of the reactor internals, design criteria and dynamic analyses methodology for the combined seismic and postulated pipe rupture events under ASME Level D (faulted) service conditions. The results of theanalyses are required to meet the stress limits of the ASME Code, Section III, Subsection NG (Ref. 3.9-1) for Core Support Structures (CSSs), and the functional requirements of the reactor internals design specification. Meeting the requirements of the ASME Code, Section III (Ref. 3.9-1) and the design specification should provide assurance of the structural and functional integrity of the reactor internals under ASME Level D service condition combined the loads of seismic and pipe rupture events.

Both seismic and LOCA dynamic analysis models are three dimensional, non-linear finite element (FE) mathematical models representing the reactor vessel and its internals in six degree of freedoms. The general purpose FE computer code, ANSYS (Ref 3.9-7) is used as the basis for the modeling.

The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that most dominant frequencies are represented in the seismic-LOCA response. This forms the bases for establishing any directional decoupling and system structural partitioning in the seismic-LOCA system models. Detailed discussion of the seismic and LOCA system models are described in Reference 3.9-58.

3.9.2.5.1 Seismic Analysis Methodology and Acceptance Criteria

The seismic analysis methodology is based on two separate mathematical models and uses general purpose FE computer code. The first model is a three-dimensional non-linear dynamic FE computer model representing the reactor internals and the support system and is used to determine the maximum accelerations, displacements, and loadings that are used as input to the second model. The second computer model or models are three-dimensional static FE computer programs that are used to determine the maximum seismic stress intensities and displacements. These models and description are described in Reference 3.9-58.

The maximum stresses from the static FE model for both SSE and LOCA events are combined by the SRSS and the results are compared to the service limit stress intensities in the ASME Code, Section III, Subsection NG (Ref. 3.9-1). The maximum displacements and loads are compared to the allowable limits in the design specification. The details of the seismic dynamic computer model are discussed below.

The pre-processing input of the seismic mathematical computer model comes from the design drawings and is the bases of the geometrical and material representation and connectivity of the reactor internals components and interfacing components. This includes the representation of the RV support system, inlet and outlet piping nozzles, Control Rod Drive Mechanism (CRDM) system, integrated head support system, in-core instrumentation support system, and fuel assembly nozzles and grids.

Figures 3.9-3 and 3.9-4 represent a typical mathematical model of the reactor internals used for seismic analysis. The physical geometry and material properties (density, modulus of elasticity, Poisson's ratio) of the reactor internals are represented by beams elements. The reactor internals and interfacing structures are connected or represented by mass inertia effect, stiffness matrices, and hydro-dynamic matrices, springs, and/or impact elements including gap and damping (including coexistence of viscous and Coulomb damping).

The nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that the most dominant frequencies are represented in the seismic response. Dominant frequencies are identified by comparing the frequency response of the reactor internals with the expected responses based on experience and measurements.

Fluid-structure interaction effects are accounted for by matrices developed for that purpose. The hydrodynamic masses are calculated for the following locations in the seismic analysis model:

(1) Between the reactor vessel and core barrel in two horizontal directions

(2) Between the core barrel and neutron reflector in two horizontal directions

(3) Between the upper core support and reactor vessel head in vertical direction

In the LOCA dynamic analysis, the hydrodynamic mass matrices between the reactor vessel and core barrel were deleted because the hydrodynamics mass effect were included in the pressure force as the output of blow-down analysis code, MULTIFLEX.

The effects of flow upon both the lumped mass and flexibility properties in the LOCA dynamic system model were also accounted because they were included in the MULTIFLEX results which were used as input to the LOCA dynamic system model.

The reactor internals seismic input can either be from in-structure response spectra or in-structure time-history accelerations which is obtained from the analysis results described in Subsection 3.8.3. This model employs the design response spectra of the building-RCL coupled model based on modified input from RG 1.60, Rev. 1 (Ref. 3.9-27) as described in Section 3.7. This model is used in determining the effect of vibratory motion for SSE and 1/3 SSE seismic conditions.

Additional loading input to the seismic analysis are vertical pressure loadings converted to nodal point external loads, and the vertical weights of the reactor internals and interfacing components by input of density on the beams with spring effects or mass nodal points.

The reactor internals static computer models are used to determine the reactor internals component stresses and displacements. The structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe SSE in combination with the LOCA events.

3.9.2.5.2 Pipe Rupture Analysis Methodology and Acceptance Criteria

The pipe rupture design basis methodology is similar to the seismic methodology, wherein, a dynamic computer model is used to determine the maximum accelerations, displacements, and loadings, and the reactor internals static computer models are used to determine the reactor internals component stresses and displacements. However, instead of a response spectra or time-history for the seismic input, a time-history computer code is used to determine the pipe rupture loads-time history on the dynamic computer model nodes and elements. The details of the pipe rupture dynamic model and pressure input loads are discussed below.

The LOCA dynamic computer model is a three-dimensional FE model that defines the geometry, material properties, and nodal point connections and elements.

The mathematical model for LOCA dynamic effects includes reactor internals and dynamically-related piping stiffness, RV supports, interfacing components, and fluid-structure interaction effects.

The mathematical models in Figures 3.9-3 and Figure 3.9-4 are used for the LOCA dynamic system analysis, which include such structural characteristics as the flexibility, mass inertia effects, geometric configuration, spring, and impact elements including gap and damping (including coexistence of viscous and Coulomb damping). The effects of flow upon the mass and flexibility properties of the system are accounted for in the model. These models are described in References 3.9-22 and 3.9-58.

The design input for the pipe rupture event is defined by the postulated Leak-Before-Break (LBB) pipe rupture as discussed in Subsection 3.6.2 of the DCD. A time-history forcing function on the reactor internals comes from pipe rupture that are enveloped by the most limiting blow-down hydraulic loads.

The MULTIFLEX computer code is used for the blowdown analysis in the hydraulic load evaluation of the postulated LOCA accident. The MULTIFLEX is a computer program which calculates the transient of pressure, flow rate and density during the initial phase of the blowdown in a complex system such as the primary coolant system of a Pressurized Water Reactor (PWR). The MULTIFLEX code includes mechanical structure models and their interactions with thermal-hydraulic system (Ref.3.9-9).

The general characteristics of the MULTIFLEX code are shown in the following.

- The complex system is modeled with one-dimensional hydraulic piping.
- The flow conditions within the system are calculated by solving the one-dimensional equations of mass, momentum and energy conservation using the method of characteristics.
- The MULTIFLEX code includes heat transfer models of the core and the SG, and also simulates various boundary conditions of the PWR system including the core.
- The calculated results of the MULTIFLEX code (pressure, flow rate, and so on) are used in the RV internals load evaluation and the RCL mechanical load evaluation.

The methods and procedures for the LOCA dynamic system analysis is based on the computer program code used in the LOCA analysis. The computer code incorporates the governing equations of motion and the computational scheme for deriving results. Asymmetric LOCA loads for the reactor internals are considered for the LOCA dynamic system analysis. More details about the methods of LOCA dynamic system analysis is described in Section 6 of Reference 3.9-58.

The outputs of the LOCA response analysis are time-history accelerations, displacement (absolute and relative), and loadings (forces and moments). The maximum loadings and displacements are input into reactor internals component static FE models and the maximum stress intensities and displacements are compared to the ASME Code, Section III (Ref. 3.9-1) and the allowable interface load and displacement limits (Ref. Table 3.9-2). The criteria for acceptance of the LOCA loads and displacements are discussed in Section 3.9.2.5.

The LOCA dynamic system analyses results confirm that the structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe LOCA in combination with the SSE.

3.9.2.5.3 Structural Design Adequacy Criteria for Level D Combined Loadings

The most severe dynamic loadings and displacements of the pipe rupture event are combined with the SSE event and the resulting stresses are compared with the limits of the ASME Level D service limits for acceptability.

The locations to evaluate the stress and fatigue are described in Reference 3.9-58.

In addition to the ASME Code, Section III (Ref. 3.9-1) stress criteria, there are functional requirements as listed in Table 3.9-2 to be met as follows:

(a) The allowable horizontal load of the guide tube should not impede insertion of the control rod assemblies after the LOCA event.

(b) The upper core barrel displacement is not to impede the down comer emergency core cooling flow after the LOCA event.

(c) The reaction loads at the RV connections are not to exceed allowable values of the interface load.

(d) The maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube.

(e) The upper core barrel permanent displacement should not prevent loss of function of the control rod assembly by radial inwardly deforming the upper guide tube.

(f) The core barrel and upper guide tube are stable during a pipe rupture break.

The structural design and sizing of the US-APWR reactor internals are based upon current 4-loop plants. However, the pipe break sizes of current 4-loop plants were based on the largest LOCA loads that resulted from either a 1.0 ft^2 single-ended cold leg break or a double-ended hot leg break, whereas, LBB is applied to determine the break condition for the US-APWR design input. The magnitude of blowdown hydraulic loads applying LBB is smaller than either of the loads for the large cold leg or hot leg breaks. Consequently, stresses and deflections of reactor internals under faulted conditions meet the ASME Code, Section III (Reference 3.9-1) stress and deflection limits including are provided in the corresponding EQSDSs. An EQSDS is developed for every item of instrumentation and electrical equipment classified as seismic category I. Section 3.11 and Appendix 3D provides the environmental conditions of the electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, to be demonstrated before, during and after a seismic event. The equipment qualification file and EQSDS identify the test response spectrum (TRS) and the Required Response Spectra (RRS) for the seismic qualification. The TRS is required to envelope the RRS for qualification of equipment.

The performance requirements for seismic category I active mechanical components are defined in the corresponding equipment specifications along with the system functional requirements as described in Section 3.2, Section 3.9, and in the sections describing the various systems. Subsection 3.10.2.2 and Section 3.9 discuss additional requirements for active pumps, valves, and dampers and these requirements are included in the EQSDSs contained in the equipment qualification file. For other seismic category I mechanical components, the performance requirements are to maintain structural integrity under seismic and other concurrent applicable loading conditions. The demonstration of meeting the performance requirements is included in the EQSDSs for each mechanical component.

3.10.1.3 Performance Criteria

The qualification of safety-related components to safely withstand seismic loadings in combination with other concurrent dynamic loading effects demonstrates that safety-related seismic category I instrumentation and electrical equipment, and mechanical equipment, including active pumps, valves and dampers, are capable of performing their designated safety-related function(s) under the postulated SSE, as defined in Subsection 3.7.1, in combination with other concurrent loadings. Deformation of supports and structures is acceptable at the SSE levels, provided that their designated safety-related functional performance is not compromised and does not compromise the safety-related function of other equipment.

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

The recommended guidance and requirements in IEEE Std 344-1987 (Reference 3.10-6) and RG 1.100 (Reference 3.10-7) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment. The methods and guidance in “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants”, ASME QME-1-200~~2~~7 (Reference 3.10-12), including Appendix QR-A with exceptions to be provided in a future revision of RG 1.100 (Reference 3.10-7), are also used for seismic qualification of active mechanical equipment.

The US-APWR seismic category I active mechanical and electrical equipment are seismically qualified in accordance with IEEE standards to safely withstand the SSE effects in combination with other applicable dynamic and static loads.

The design limits, load combinations associated with normal operations, postulated accident, and specified seismic and other transient events, and methods for combining dynamic responses for mechanical equipment are described in Subsection 3.9.3. The dynamic loads considered in testing of instrumentation and electrical equipment, are seismic loads, hydrodynamic, and vibratory loads, as applicable, as discussed in Section 3.11.

Recent seismic research, including recently published attenuation relations, indicates that earthquakes in the central and eastern United States have more energy content in the high-frequency range than earthquakes in the western United States. Therefore, the COL Applicant is to investigate if site-specific in-structure response spectra generated for the COL application may exceed the standard US-APWR design's in-structure response spectra in the high-frequency range. Accordingly, the COL Applicant is to consider the functional performance of vibration-sensitive components, such as relays and other instrument and control devices whose output could be affected by high frequency excitation.

The potential failure modes of the high frequency-sensitive component types and assemblies are considered in order to demonstrate the suitability of the equipment for high-frequency seismic environments. The generic failure modes involving inadvertent change of state, contact chatter, signal change/drift, and connection problems due to high frequency effects are the main focus of the high frequency qualification testing. High frequency failures resulting from improper design of mounting, inadequate design connections and fasteners, mechanical misalignment/binding of parts and the rare case of failure of a component part, will result from the same structural failure modes as those experienced during low frequency content spectra qualification testing in accordance with IEEE Std 344-1987 (Reference 3.10-6). Because the safety-related equipment will experience higher stresses and deformations when subjected to the low frequency excitation, these failure modes are more likely to occur under the low frequency testing. Failure modes related to improper mounting, inadequate securing of connections, poor quality joints (cyclic strain effects), etc., are precluded by quality assurance inspection and process/design controls.

Potentially high frequency sensitive components include: electro-mechanical relays; electro-mechanical contactors; circuit breakers; auxiliary contacts; control switches; transfer switches; process switches and sensors; potentiometers; and digital/solid-state devices (mounting and connections only).

Acceptable methods for resolving high frequency concerns not already addressed by certified design qualification where site-specific in-structure response spectra generated for the COL application results in high frequency exceedances of the standard design in-structure response spectra include: review existing equipment qualification test data for adequate high frequency input motion; review circuits containing potentially sensitive items for inappropriate system actions due to intermediacy or set point drifts; or screening test to confirm equipment does not have high frequency vulnerabilities.

If existing test data are not available and a system and control logic review indicates that inadvertent change of state or intermediacy must be considered, then one of the following high frequency screening tests are used to demonstrate lack of sensitivity to high frequency vibrations in the 25-50 Hz range where the function is monitored during the screening test followed by post test functional testing: sine sweep (fast linear rate, traditional log rate); sine beat at ~~1/3~~ 1/6 octave spacing; band-limited white noise; or, random multifrequency time history.

The above testing is not a qualification test but is intended to determine if equipment is potentially sensitive to high-frequency excitation. If the screening tests determine that equipment is potentially sensitive to high-frequency excitation ("screened-in"), then full-scale qualification testing including testing over the range of high-frequency exceedances is required to assure that ~~high frequency sensitive~~ unacceptable components are not present in the set of qualified certified design equipment and functional systems.

In conjunction with the above, for the purpose of qualification of equipment by analysis, the rigid range is defined as having a natural frequency greater than 50 Hz. For the purpose of testing equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment, to be tested, is sensitive to response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz.

The US-APWR utilizes the following methods for seismic qualification of equipment based on the type, size, shape, and complexity of the equipment configuration, whether the safety function can be assessed in terms of operability or structural integrity alone, and the reliability of the conclusions:

- Predict the equipment's performance by analysis
- Test the equipment under simulated seismic conditions
- Qualify the equipment by a combination of test and analysis

The US-APWR seismic category I equipment is qualified to show that it can perform its safety-related function during and after a postulated earthquake. The seismic qualification considers interfaces and the effects of the amplification within the equipment due to the interfaces and supporting structure. The function of the equipment is dependent on the equipment itself and the system in which it is to function. The safety-related function is determined as that required both during and after a postulated earthquake, which could be different. For example, an electrical device may be required to have no spurious operations during the postulated earthquake or to perform an active function both during, and after, the postulated earthquake, or it may be required to survive during the postulated earthquake and perform an active function after the postulated earthquake, or any combination of these. Another device may only be required to maintain structural integrity during and after the postulated earthquake.

The functionality of mechanical and electrical equipment during and after a postulated earthquake of magnitude up to and including the SSE for static and dynamic loads from normal, Anticipated Operational Occurrence and accident load conditions is assured by tests and/or analyses. The horizontal and vertical SSE RRS curves developed at the damping of interest, as discussed in Subsections 3.7.1 and 3.7.3, form the basis for the seismic qualification of the equipment. The equipment is demonstrated to withstand the equivalent effect of five OBE excitations followed by one SSE for qualification without loss of structural integrity and functionality, as required.

With the elimination of the OBE from design considerations, two alternatives exist that essentially maintain the requirements provided in IEEE Std 344-1987 (Reference 3.10-6) to qualify equipment with the equivalent of five OBE events followed by one SSE event of the postulated earthquakes, and whether the equipment is to be used in one application or many (proof testing or generic testing). Equipment is conservatively tested considering the multidirectional effects of the postulated earthquakes.

The types of test to be used are single frequency and multifrequency. The seismic and dynamic test inputs are provided by the in-structure floor response spectra identified with the building elevation derived from the SSE and developed by the time-history modal analysis method or direct integration method for various damping values as described in Subsection 3.7.3.

Multi-frequency testing provides a broadband test motion that is appropriate for producing a simultaneous response from modes of a multi-degree-of-freedom system whose malfunction may be caused by modal interaction. Multi-frequency testing is the preferred method since the seismic and dynamic load excitation generally has broad frequency content.

Single-frequency testing, such as sine beats, is used when the seismic ground motion is filtered due to one predominant structural mode; when the resulting floor motion may consist of one predominant frequency; when it can be demonstrated that the anticipated response of the equipment is adequately represented by one mode; or, when the input has sufficient intensity and duration to excite the relevant modes to the required magnitude, such that the TRS envelopes the corresponding spectra.

For the seismic and dynamic portion of the loads, the test input motions are applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it is demonstrated that the equipment response is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An alternate method is to test with the vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. This type of testing must be repeated with the equipment rotated 90 degrees horizontally.

Components that have been previously tested to IEEE Std 344-1971 prior to submittal of the DCD are reevaluated to justify the appropriateness of the input motion and requalify the equipment, if necessary. The COL Applicant is to requalify the component using biaxial test input motion unless the applicant provides justification for using a single-axis test input motion. Guidelines for qualifying components are included in the procedures of the US-APWR Equipment Environmental Qualification Program (Reference 3.11-3).

The equipment to be tested is mounted in a manner that simulates the intended service mounting, and the fixture design is such that it does not cause any extraneous dynamic coupling to the test component.

The dynamic coupling effect of electrical connections, conduit, sensing lines, and any other interfaces are considered and included in the test unless otherwise justified. The method chosen for testing depends upon the nature of the expected vibration environment and also on the nature of the equipment.

Seismic testing is performed in the proper sequence as indicated in “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations”, IEEE Std 323-1974 (Reference 3.10-15) and the testing identifies and accounts for significant

3.10-2 General Design Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

3.10-3 Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix B.

3.10-4 Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs. SECY-93-087, United States Regulatory Commission, April 2, 1993.

3.10-5 Earthquake Engineering Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix S, U.S. Nuclear Regulatory Commission, Washington, DC.

3.10-6 IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987.

3.10-7 Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants. Regulatory Guide, 1.100, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 1988.

3.10-8 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. Institute of Electrical and Electronics Engineers (IEEE) Std 344 -2004.

3.10-9 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment. NUREG-0800, Standard Review Plan 3.10, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.10-10 Boiler and Pressure Vessel Code. “Section III, Division 1, Nuclear Power Plant Components,” American Society of Mechanical Engineers.

3.10-11 Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants. Regulatory Guide 1.148, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.

3.10-12 Qualification of Active Mechanical Equipment Used in Nuclear Power Plants. American Society of Mechanical Engineers (ASME) QME-1-200~~2~~7.

3.10-13 Damping Values for Seismic Design of Nuclear Power Plants. Regulatory Guide 1.61, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.10-14 Guidance for Seismic Qualifications of Class 1 Electric Equipment for Nuclear Power Generating Stations. Institute of Electrical and Electronics Engineers (IEEE) Std 344-1971.

where, subscripts indicate the following loads.

SSE = Inertia load due to safe-shutdown earthquake (SSE)

SAM = Seismic anchor motion load due to SSE

(2) Safety factor

The safety factors required for the LBB evaluation of the US-APWR by the Reference 3B-2 are associated with the following items.

i. Leak rate ten times as large as detectable leak rate

ii. Critical crack length/leakage crack length $\geqq 2$

iii. Safety factor for the maximum load = 1 for absolute sum

1.4 for algebraic sum

The leak detection system for US-APWR is designed to detect a leak rate of ~~1~~ 0.5 gpm. Consequently, the leak rate for the LBB evaluation is ~~10~~ at least 5 gpm based on Item i mentioned above. The applied load is evaluated by absolute sum; therefore, the safety factor of 1.0 is used for the maximum load.

LBB evaluation procedure to satisfy the above three safety factors is shown in Figure 3B-6. The procedure is as follows:

a. Obtain the crack opening area corresponding to the applied loads under normal operation.

b. Calculate leakage crack length L_L from the crack opening area, the leak rate ten times as large as the detectable leak rate and the leak rate based on thermal hydraulics model.

c. Calculate the critical crack length L_C from the fracture mechanics analysis of the applied load under the maximum load condition.

d. If the critical crack length L_C is twice as large as or larger than the leakage crack length L_L, restraint is unnecessary because the leak is detectable before pipe rupture.

3B.3.1 Generation of BAC

3B.3.1.1 BAC Methodology

The BAC methodology is an LBB assessment diagram (Reference. 3B-3) used to satisfy the three safety factors identified in the previous section. In the BAC diagram, $\sigma_{nor}=P_m+P_b$, the sum of the membrane stress and the bending stress under normal operation is plotted along the abscissa, and $|\sigma_{max}| = |P_{m\_max}| + |P_{b\_max}|$, the absolute sum of the membrane stress and the bending stress under the maximum load is plotted along the ordinate. The plotting procedure on the diagram is as follows.

Item 1 Determine the leakage crack length with a leak rate 10 times as large as the detectable leak rate by applying abscissa's stress $\sigma_{nor} = P_m + P_b$.

Figure 3H.3-3 presents the comparison between results obtained for the containment internal structure from the lumped mass stick model versus those obtained from the detailed FE model with respect to static deformations. The comparison shows the correlation of the stiffness properties of the two models.

Figure 3H.3-4 presents the comparison between the 5% damping ISRS obtained for the containment internal structure at various locations and elevations from the fixed-base lumped mass stick model, versus those obtained from the fixed-base FE model. The comparison shows the correlation of the dynamic responses obtained from the two models.

Table 3H.3-10 presents the results of the PS/B time history analyses by using the methodology described in Subsection 3.7.2.

Tables 3H.3-11 through 3H.3-14 provide resulting maximum displacement at various mass nodes for shear wave velocities.

3H.4 REFERENCES

3H-1 Dynamic Analysis of the Coupled RCL-R/B-PCCV-Containment Internal Structure Lumped Mass Stick Model, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008 ~~MHI Technical Report, Later~~.

3H-2 Seismic Analysis of Safety-Related Nuclear Structures. American Society of Civil Engineers, ASCE 4-98, Reston, Virginia, 2000.

3H-3 Combining Responses and Spatial Components in Seismic Response Analysis. Regulatory Guide 1.92, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, July 2006.

3H-4 Investigation of Out-of-Plane Flexibility of Floor Slabs with Respect to Development of ISRS, MHI Technical Report, Later.

Chapter 4

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Table 4.1-2 Analytical Techniques Summary

(sheet 1 of 2)

Design Category	Analysis Techniques/Approach	Primary Code (s) Used	Referenced Sections
Fuel			
Key parameters such as rod internal pressure, cladding oxidation, fuel temperatures and cladding stress, strain	Fuel performance models using thermal, fission gas release, corrosion and hydrogen uptake based on extensive empirical data	FINE	Subsection 4.2.3 Subsection 4.4.2.11
Loads, stress and deflection of fuel assembly components	Static and dynamic analyses of fuel assembly components for events occurred during normal operation, AOOs and postulated accidents	Finite element method codes such as ANSYS or ABAQUS FINDS	Subsection 4.2.3
Nuclear			
Few-group microscopic and macroscopic cross-sections	Collapse of fine-group data and spatial homogenization, performed by 2D current coupling collision probability methods (CCCP)	PARAGON	Subsection 4.3.3.1
3D power distributions, peaking factors, fuel depletion, boron concentrations, reactivity coefficients, control rod worth, transient fission product behavior (Xe, Sm)	2-group diffusion theory applied with a nodal expansion method (NEM)	ANC	Subsection 4.3.2 4.3.3.1
Vessel Irradiation			
Fast neutron flux	Discrete ordinates Sn transport methodology	DORT	Subsection 4.3.2.8
Criticality			
Criticality of reactor, fuel assemblies, new and spent fuel racks, fuel handling	Monte-Carlo methodology	MCNP	Subsection 4.3.2.6 4.3.3.2

strain, fatigue, fuel temperature, rod internal pressure, etc., using ~~applying~~ the fuel densification model and the swelling model.

4.2.1.2.3 Chemical Properties

Chemical compatibility of the fuel with other fuel rod materials and the reactor coolant is discussed in detail in Appendix B of Reference 4.2-6.

4.2.1.3 Fuel Rod Performance

4.2.1.3.1 Analytical Models

The FINE code incorporates all of the basic fuel rod performance models required to evaluate in-reactor fuel behavior. The FINE code fuel performance models are described in detail in Chapter 4 of Reference 4.2-6. A summary of these models is provided in Subsection 4.2.3. The uncertainties in the fuel performance models and in the fuel fabrication are taken into account to obtain conservative evaluations of the fuel rod performance.

4.2.1.3.2 Mechanical Design Limits

4.2.1.3.2.1 Cladding Collapse

On the basis of the extensive operating experience of Mitsubishi fuel in Japan, it is concluded that cladding collapse does not occur for pre-pressurized fuel rods with initial fuel pellet density of 95%TD or greater. Maintaining initial pellet density of 95%TD or greater, with current fabrication processes, results in a fuel pellet that is relatively stable with respect to fuel densification, so that any pellet-pellet gaps are too small for cladding collapse to occur. In addition, the initial helium pressurization reduces the differential between the system pressure and the rod internal pressure and provides additional resistance to cladding collapse. Therefore, cladding flattening does not occur. Cladding collapse is discussed in detail in References 4.2-6 and 4.2-7.

4.2.1.3.2.2 Rod Internal Pressure

The fuel rod internal pressure remains below the lowest of the following three rod internal pressure design limits.

- No cladding liftoff during normal operation
- No reorientation of the hydrides in the radial direction in the cladding
- A description of any additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents

These limits are described more completely in References 4.2-6 and 4.2-7.

Chapter 6

the environment is below limits. Thus, the US-APWR fission product removal (three systems) and control (containment) systems are as follows:

- main control room (MCR) heating, ventilation, and air conditioning (HVAC) System
- annulus emergency exhaust system
- containment spray system
- containment vessel

The annulus emergency exhaust system is separate and distinct from the control room habitability system, which is presented in Section 6.4. The plant ventilation systems for Class-1E electrical rooms, safeguard component areas emergency feed pump areas, and the emergency power sources are presented in Chapter 9, Subsection 9.4.5~~1~~. The containment spray for containment cooling is presented in Chapter 6, Subsection 6.2.2.

6.0.6 Inservice Inspection (ISI) of Class 2 and Class 3 Components

Regular and periodic examinations, tests, and inspections of pressure retaining components (and supports) are required by 10CFR50.55a(g) (Ref. 6.0-1). Section 6.6 discusses the ISI and testing programs to address these requirements.

6.0.7 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

6.0.8 References

6.0-1 Codes and Standards, 10CFR50.55a, Nuclear Regulatory Commission, U.S., Washington, D.C., January 2007 Edition.

6.0-2 General Design Criteria for Nuclear Power Plants, 10CFR50 Appendix A, Nuclear Regulatory Commission, U.S., Washington, D.C., January 2007 Edition.

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MAP-06-016

- In-place aerosol leak tests are performed in accordance with Section 10 of ASME N510-1989 (Ref. 6.5-5) on the HEPA filters initially, periodically, after filter replacement (full or partial), after suspected water intrusion, and following painting, fire, or chemical release in any area served by the annulus emergency exhaust system if such a release may affect filter performance

6.5.1.6 Instrumentation Requirements

The ECCS actuation signal automatically actuates the annulus emergency exhaust system.

6.5.1.6.1 Radiation Monitors

Four area radiation monitors are located in containment. The containment radiation monitors detect high radiation and actuate an alarm in the MCR. Radiation monitoring is discussed in Chapter 12, Subsection 12.3.4.

6.5.1.6.2 Flow Rate

The total combined flow rate from the penetration areas is stored by the process computer in the MCR. The annulus emergency exhaust filtration unit fan train A and train B outlet flow rate is also stored by the process computer.

The annulus emergency exhaust filtration unit fan outlet air high and low flow alarms are provided in the MCR.

6.5.1.6.3 Pressure

The pressure in the penetration areas and safeguard component areas are stored by the process computer in the MCR.

The differential pressure across the high efficiency filter and HEPA filter in each train is indicated locally and alarmed in the MCR.

6.5.1.7 Materials

The ESF filter system materials are specified to resist premature failure of the annulus emergency exhaust system or any other ESF system due to radiolytic and pyrolytic decomposition products according to the environmental conditions in which the ESF filter systems are installed. The ESF filter system materials are chosen in accordance with the requirement of RG 1.52 (Ref. 6.5-1) and ASME AG-1-2003 (Ref. 6.5-4).~~The COL Applicant is responsible to provide an as-built list of material used in or on the ESF filter systems by their commercial names, quantities (estimate where necessary), and chemical composition and show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other ESF.~~

leakage rate. The potential containment bypass leakage rate is assumed to be due to leakage from containment isolation valves installed in piping, which penetrate both the primary containment and penetration areas and is determined based on valve design limitations. As a result, the potential containment bypass leakage is considered to be much less than 10%. However, the leakage fraction to the penetration areas in dose evaluations that are discussed in Chapter 15 is credited as 50%, that is, including a conservative margin assumed for the evaluation.

These systems limit the maximum radiation dose to less than the criteria of RG 1.183 (Ref. 6.5-3). The radiological consequences following a design basis accident are presented in Chapter 15, Subsection 15.4.8 and 15.6.5.
~~The US-APWR design does not utilize a secondary containment. This subsection is not applicable to the US-APWR.~~

6.5.4 Ice Condenser as a Fission Product Cleanup System

The US-APWR containment is a prestressed, post-tensioned concrete structure described in Subsection 3.8.1. The US-APWR design does not include an ice condenser-type containment design.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

The US-APWR containment is a prestressed, post-tensioned concrete structure described in Subsection 3.8.1. The US-APWR design is not a pressure suppression pool-type containment design.

6.5.6 Combined License Information

Any utility that references the US-APWR certified design for construction and operation is specifically responsible for the following:

COL 6.5(1) *Deleted*

COL 6.5(2) *Deleted*

COL 6.5(3) *Deleted*

COL 6.5(4) *Deleted* ~~*The COL Applicant is responsible to provide an as-built list of material used in or on the ESF filter systems by their commercial names, quantities (estimate where necessary), and chemical composition and show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other ESF.*~~

6.5.7 References

6.5-1 U.S. Nuclear Regulatory Commission, Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-

between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. The ISI program contains information addressing areas subject to inspection, method of inspection, and extent and frequency of inspection. The program covers the high-energy fluid systems described in Chapter 3, Subsections 3.6.1 and 3.6.2.

The COL Applicant is responsible for preparing an augmented inservice inspection program for high-energy fluid system piping. The preservice inspection program addresses the equipment and examination techniques to be used.

As noted in Subsection 6.6.2, the design and installed arrangement of US-APWR Class 2 and 3 components provide clearance adequate to conduct Code-required examinations.

6.6.9 Combined License Information

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

COL 6.6(1) The COL Applicant is responsible for the preparation of a preservice inspection program (non-destructive baseline examination) and an Inservice inspection program for ASME Code Section III Class 2 and 3 systems, components (pumps and valves), piping, and supports in accordance with 10 CFR50.55a(g), including selection of specific examination techniques and preparing appropriate inspection procedures.

COL 6.6(2) The COL Applicant is responsible for preparing an augmented inservice inspection program for high-energy fluid system piping.

6.6.10 References

6.6-1. Inservice Inspection Requirements, Title 10, code of Federal Regulations, 10 CFR 50.55a(g), January 2007.

6.6-2. Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler & Pressure Vessel Code, Division 1, Section XI, American Society of Mechanical Engineers, ~~July 2006~~2001 Edition with 2003 Addenda.

6.6-3. U.S. Nuclear Regulatory Commission, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Regulatory Guide 1.147, Rev. 14, August 2005.

Chapter 9

stainless steel and the spent fuel storage rack cell consists of stainless steel with boron. Metamic is selected as neutron absorbing material.~~Borated stainless steel or proven boron absorbers such as Boral and Metamic are being considered.~~ The steel plate thickness and boron content are conservatively set to a minimum. Performance effectiveness of the neutron absorbing materials in the racks is taken into consideration.

- The rack cell array is either assumed to be infinite in the lateral direction or is assumed to be surrounded by a conservatively chosen reflector, whichever is appropriate for the design:

  New fuel storage rack

- A finite rack cell array and the surrounding concrete reflectors are used in the calculations.

  Spent fuel storage rack

- Basically, an infinite rack array in the lateral direction is used in the calculations. However, in the sensitivity study for determining uncertainty, the analysis model depends on the type of tolerance.

- Uncertainties are appropriately determined either by using worst-case conditions or by performing sensitivity studies. The uncertainties considered are material composition, fabrication tolerances of the fuel and rack, and the fuel location within the rack cell, as follows:

  - Steel plate thickness and its boron content are directly set to minimum so as to maximize $K_{eff.}$

  - Other uncertainties are considered less effective and independent and are therefore statistically combined with the analysis code bias uncertainty.

The criticality evaluation is performed in accordance with Section 5.1 of ANSI/ANS-8.17-2004. Section 5 describes the following relationships.

$$k_p \leq k_c - \Delta k_p - \Delta k_c - \Delta k_m,$$

If the various uncertainties are independent,

$$k_p \leq k_c - (\Delta k_p^2 + \Delta k_c^2)^{1/2} - \Delta k_m.$$

where:

k_p	is the calculated K_{eff}
k_c	is the mean K_{eff} derived from the code validation
Δk_p	is an allowance; calculation, tolerances

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handling crane is described in detail in Subsection 9.1.5. Failure modes of the fuel handling machine are described in Subsection 9.1.4.

The spent fuel rack is designed as a moderate density storage arrangement which provides adequate natural coolant circulation to remove the residual decay heat from spent fuel stored in the spent fuel rack, in combination with the SFPCS described in Subsection 9.1.3. SFP nuclear safety and criticality are discussed in Subsection 9.1.1.

Equipment classifications of the US-APWR are described in Section 3.2. Subsection 9.1.1 describes criticality analysis of new and spent fuel storage. The spent fuel area ventilation system is described in Subsection 9.4.3, and the radiation monitoring system and shielding is discussed in Section 12.3.

9.1.2.2 Facilities Description

9.1.2.2.1 New Fuel Storage

The approximately 18 feet deep dry, unlined reinforced concrete new fuel storage pit is designed to provide support for the new fuel storage rack. The new fuel storage pit is designed to maintain its structural integrity following a SSE and to perform its intended function following a postulated event such as fire, internal/external missiles, or pipe break. The walls surrounding the fuel handling area and new fuel storage pit protect the fuel from missiles generated inside the R/B. The fuel handling area does not contain a credible source of missiles. The R/B is a seismic category I structure and is described in Subsection 1.2.1.7.1. Subsection 3.8.4 describes the structural design of the new fuel storage area and Section 3.5 discusses missile sources and protection.

The structure of the new fuel storage pit supports the weight of the new fuel rack at the floor level. The new fuel storage rack, as shown in Figure 9.1.2-1, consists of individual vertical cells interconnected to each other at several elevations, and supported by the pit walls with a grid structure near the top and bottom elevations. The rack module is not anchored to the pit floor, but supported by lateral bracing attached to the pit wall. The new fuel storage pit is covered by solid lids and an access platform. For each cell, the lids are normally closed and prevent misloading of a new fuel assembly in the space between the cells. The access platform provides passage between racks for inspection of the new fuel. Both the lids and access platform are designed not to fall or collapse in the event of the SSE.

The new fuel storage pit is provided with a ~~manually operated~~ drain system, which is connected to the R/B sump to prevent the new fuel pit from being flooded by an unanticipated release of water. The design of the drain piping system includes a check valve to prevent~~s~~ backflow into the new fuel pit storage area through the drain system. The new fuel rack storage cells are each designed with an opening at the bottom of each of the four sides, which can drain such unanticipated release of water. These openings are sized the same as the openings at the botoom of the spent fuel storage rack cells.

Center-to-center spacing of the new fuel rack array is 16.9 inches as shown in Figure 9.1.2-1, which provides a minimum separation between adjacent fuel assemblies. This design is sufficient to maintain a subcritical array even in the event of the new fuel storage

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pit being flooded with unborated water, fire extinguishing aerosols or during any design basis event. Additionally the design of the rack is such that a fuel assembly cannot be inserted into a location other than a location designed to receive an assembly, and an assembly cannot be inserted into a full location. Surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel.

9.1.2.2.2 Spent fuel storage

The SFP, including its integrally attached liner, is designed as seismic category I and is located within the seismic category I reactor building fuel handling area. The spent fuel storage pit and its liner are designed for loads and load combinations addressed in DCD Subsection 3.8.4.3 and Table 3.8.4-3. Applicable loads include but are not limited to dead, live, hydrostatic, hydrodynamic, seismic, normal operating, accident thermal, and spent fuel assembly drop loads. The spent fuel storage pit and its liner are designed to maintain their structural integrity and remain leak tight under all applicable design loads and load combinations. The walls of the SFP are an integral part of the seismic category I reactor building structure. The facility is protected from the effects of natural phenomena such as earthquakes (Section 3.7~~.2~~), wind and tornados (Section 3.3), floods (Section 3.4), and external missiles (Section 3.5). The facility is designed to maintain its structural integrity following a SSE and to perform its intended function following a postulated event such as a fire. Refer to Subsection 1.2.4.1 for further discussions of the reactor building fuel handling area.

The SFP is approximatery 47 feet deep of the reinforced concrete lined with stainless steel plate. The SFP normal water level is approximatery 1 ft -2 in. below the operating floor with approximatery 400,000 gallons of borated water. This water level allows a spent fuel assembly to be transferred with at least 133 inches of water shielding above the top of the fuel assembly for personell protection.The Spent Fuel Pit (SFP) is lined with stainless steel. The liner surface will have a 2B or higher finish, selected to minimize accumulation of corrosion and fission products, and also provide easy maintenance and decontamination. This liner surface is smooth and non-porous to avoid buildup of radioactive material.

Penetrations for the drain and makeup lines are located to preclude the draining of the SFP due to a break in a line or failure of a pump to stop. The connection for the SFP pumps' suction is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the SFPCS is unavailable.

Pipes which discharge into the spent fuel pool include a siphon break between the normal water level and the level of the SFP pumps' suction connection.

The capability to makeup to the SFP is provided by a Quality Group C, seismic category I makeup system, as discussed in Subsection 9.1.3.

A liner leakage collection system is provided to collect possible leakage from liner plate welds on the pit walls and floor. The stainless steel liners are welded to the C-shape embedment in the pit walls and floors, and the embedment are interconnected and drain to a collection point which is monitored to determine whether leakage is occurring.

The refueling canal is connected on one side to the SFP. On its opposite side, the refueling canal connects to the spent fuel cask loading pit and to the fuel inspection pit. A weir and gate provide physical isolation of the refueling canal from each of the three pits. All the gates are normally closed and only opened as required.

The SFP is not connected to the equipment drain system (Subsection 9.3.3) to preclude unanticipated drainage.

SFP water level and temperature gauges, and an area radiation monitor in the fuel handling area are provided with alarms to the main control room (MCR) and locally.

Normal auxiliary building (A/B) HVAC system provides ventilation for the fuel handling area to maintain the atmospheric pressure in this area slightly negative with respect to outside the building.

The spent fuel racks are composed of individual vertical cells, and several tiers of grid structures which interconnect each cell to rigidly maintain the cell array configuration. The racks are supported laterally by the grid structure, and each rack module is vertically supported by 4 legs on the pit floor without anchoring. Additionally, each rack cell is vertically supported by 4 legs on the pit floor without anchoring. The grid structures are designed such that a fuel assembly cannot be inserted between the cells, or any other locations around the racks.

Moderate density racks containing neutron absorbing material are provided in the SFP. Center-to-center spacing of the rack array is 11.1 ~~in~~ to maintain the required degree of subcriticality as shown in Figure 9.1.2-~~.~~2~~-1~~.

Materials used in rack construction are compatible with the SFP environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pit environment. ~~Borated stainless steel or proven boron absorbers such as Boral and~~ Metamic is selected ~~are being considered for~~ the neutron absorbing material. ~~Venting of the neutron absorbing material, if necessary, will be considered in the detailed design of the spent fuel storage racks.~~ ~~A~~ Following program for monitoring the effectiveness of neutron poison by incorporating basic tests assures that the subcriticality requirements of the stored fuel array are maintained.

Purpose of Surveillance Program

The purpose of the surveillance program is to characterize certain properties of the Metamic with the objective of providing data necessary to assess the capability of the Metamic panels in the racks to continue to perform their intended function. The surveillance program is also capable of detecting the onset of any significant degradation with ample time to take such corrective action as may be necessary.

The Metamic surveillance program depends primarily on representative coupon samples to monitor performance of the absorber material without disrupting the integrity of the storage system. The principal parameters to be measured are the thickness (to monitor for swelling) and B-10 loading (to monitor for the continued presence of boron in the Metamic).

Coupon Surveillance Program

Coupon Description

The coupon measurement program includes coupons suspended on a mounting (called a tree), placed in a designated cell, and surrounded by spent fuel. Coupons are removed from the array on a prescribed schedule and certain physical measurements from which the stability and integrity of the Metamic in the fuel storage racks may be inferred.

The coupon surveillance program uses a tree with a total of 10 test coupons. In mounting the coupons on the tree, the coupons are positioned axially within the central eight feet (approximate) of the active fuel zone where the gamma flux is expected to be reasonably uniform.

The coupons will be taken from the same lot as that used for construction of the racks. Each coupon will be carefully pre-characterized prior to insertion in the pool to provide reference initial values for comparison with measurements made after irradiation. As a minimum, the surveillance coupons will be pre-characterized for weight, dimensions (especially thickness) and B-10 loading.

Surveillance Coupon Testing Schedule

To assure that the coupons will have experienced a slightly higher radiation dose than the Metamic in the racks, the coupon tree will be surrounded (to the extent possible and subject to other NRC requirements related to distributing “hot” fuel throughout the spent fuel pool) by freshly-discharged fuel assemblies after each refueling. At the scheduled test date, the coupon tree will be removed and a coupon removed for evaluation. Effort will be made to surround the coupon tree with freshly discharged fuel (subject to the limitations already mentioned) during each refueling discharge. The recommended coupon measurement schedule is shown in the following table.

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RECOMMENDED COUPON MEASUREMENT SCHEDULE	
Coupon	Years note 1
1	2
2	4
3	7
4	11
5	16
6	22
7	29
8	37
9	46
10	60

Note 1 The years pertain to those after the first loading of spent fuel into the spent fuel storage racks.

Evaluation of the coupons removed will provide information of the effects of the radiation, thermal and chemical environment of the pool and by inference, comparable information on the Metamic panels in the racks. Coupons, which have not been destructively analyzed by wet-chemical processes, may optionally be returned to the storage pool and remounted on the tree. They will then be available for subsequent investigation of defects, should any be found.

Measurement Program

The coupon measurement program is intended to monitor changes in physical properties of the Metamic absorber material by performing the following measurements on the preplanned schedule:

- Visual Observation and Photography
- Neutron Attenuation
- Dimensional Measurements (length, width and thickness)

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・Weight and Specific Gravity

Surveillance Coupon Acceptance Criteria

Of the measurements to be performed on the Metamic surveillance coupons, the most important are (1) the neutron attenuation [note2] measurements (to verify the continued presence of the boron) and (2) the thickness measurement (as a monitor of potential swelling). Acceptance criteria for these measurements are as follows:

Note 2 Neutron attenuation measurements are a precise instrumental method of chemical analysis for Boron-10 content using a nondestructive technique in which the percentage of thermal neutrons transmitted through the panel is measured and compared with predetermined calibration data. Boron-10 is the nuclide of principal interest since it is the isotope responsible for neutron absorption in the Metamic panel.

・A decrease of no more than 5% in Boron-10 content, as determined by neutron attenuation, is acceptable. This is tantamount to a requirement for no loss in boron within the accuracy of the measurement.

・An increase in thickness at any point should not exceed 10% of the initial thickness at that point.

Changes in excess of either of these two criteria requires investigation and engineering evaluation, which may include early retrieval and measurement of one or more of the remaining coupons to provide corroborative evidence that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation shall be performed to identify further testing or any corrective action that may be necessary.

The remaining measurement parameters serve a supporting role and should be examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in measurement schedule. These include (1) visual or photographic evidence of unusual surface pitting, corrosion or edge deterioration, or (2) unaccountable weight loss in excess of the measurement accuracy. ~~The surveillance program relies on representative coupon samples to monitor performance of the absorber material without disrupting the integrity of the storage system. The coupons are hung in the spent fuel pit so that they receive dosage comparable to the rack poison panels. The coupons are periodically removed from the pool and examined for their physical appearance. After establishing that the coupon is indeed intact, it may be returned to the pool.~~

~~The coupons used in the surveillance program are taken from the poison material production lot. The surveillance program uses a predetermined number of test coupons that simulate the actual in-service conditions of the poison material in the storage racks.~~

~~Each coupon is pre-characterized prior to insertion in the pool to provide reference initial values for comparison with measurements made after irradiation. Archive samples of the poison material will also be retained for later comparison with the irradiated coupons.~~

~~A "tree" of coupons is mounted in a designated storage cell, located such that the freshly discharged fuel will always be in the surrounding cells. Coupons would be "pulled" and analyzed at preset intervals, to be determined as part of the surveillance program development effort. Based on the results of the initial surveillance coupon measurements, the future schedule will be determined as necessary.~~

~~The COL applicant is to provide a program for monitoring the effectiveness of neutron present in the neutron absorbing panel.~~ Design of the spent fuel storage facility is in accordance with Regulatory Guide 1.13.

The SFP is also provided with an array of 12 storage spaces for damaged fuel assembly containers. These racks do not contain the neutron absorber and the center-to-center spacing of this array is 24 inches.

No overhead crane, except the light load fuel handling machine, pass over the SFP. The fuel handling machine is designed to withstand seismic category I loads to preclude its fall or collapse due to an SSE.

9.1.2.2.3 New Fuel Storage Rack and Spent Fuel Storage Rack Design

The fuel storage facilities are designed to meet the guidelines of ANS 57.2 (Ref. 9.1.7-7) and ANS 57.3 (Ref. 9.1.7-9). Structural design and stress analysis of the new and spent fuel storage racks are evaluated in accordance with the seismic category I requirements of Regulatory Guide 1.29.

The dynamic and stress analyses are performed and described in the technical report (Ref. 9.1.7-8). Loads and load combinations considered in the structural design and stress analysis are shown in Table 9.1.2-1 based on SRP Section 3.8.4, Appendix D.

Uplift force analysis is also performed for new and spent fuel racks design, and described in the technical report (Ref. 9.1.7-8). Each rack is evaluated for withstanding a maximum uplift force of 4,400 pounds based on the lifting capacity of the suspension hoist and the fuel handling machine. Structural analysis is performed to verify that resultant stress in the critical part of the rack is within acceptable stress limits and deformation of the rack array is limited to maintain a subcritical array.

Fuel assembly drop analysis is performed for each fuel rack to maintain a subcritical array. Drop weight is determined from the maximum weight handled for each rack and drop height is determined from the higher value of 2 ft or the design height for handling fuel above each rack. The analysis is also provided in the technical report (Ref. 9.1.7-8)

9.1.2.3 Safety Evaluation

9.1.2.3.1 New Fuel Racks

The new fuel rack, being a seismic category I structure, is designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the SSE event.

In addition to the limit devices, the control system is designed to include safety devices, which will assure the OHLHS returns to and/or maintains a secure holding position of critical loads in the event of a system fault. These safety devices are in addition to and separate from the control devices used for normal operation of the OHLHS. Emergency stop buttons are strategically placed at various locations to de-energize the OHLHS independent of the system controls. The overload sensing system is designed to be reset when switching the OHLHS between maximum critical load operations and design rate load operations. This resetting is performed remotely from the system controls and is governed by the OHLHS administrative procedures defined in Subsection 13.5.1.

The OHLHS driver control systems are designed using a combination of electrical and mechanical components. The control systems take into account the hoisting (raising and lowering) of the complete range of loads from the load hook itself up to and including the rated load in conjunction with the inertia of moving components, such as the motor armature, shafting and coupling, gear reducer, drum, etc. In general, the OHLHS is not contemplated to be used to lift individual spent fuel elements. The control system has been designed to be adaptable to include manual interlocks, which will preclude trolley and/or bridge movement while a spent fuel assembly is being hoisted free of the reactor vessel or a storage rack. The manual interlocks are controlled by the administrative control procedures defined in Subsection 13.5.1.

Instrumentation is installed within the motor control circuits to detect and react to malfunctions such as excessive electric current, excessive motor temperature, overspeed, overload, and overtravel. Control devices are installed to absorb the kinetic energy of the rotating components and arrest the hoisting movement should the load line or one of the dual revving systems fail, or should an overload and/or overspeed condition occur.

The drives are designed to conform to ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), (Ref. 9.1.7-20) with respect to hoist speed, specifically Section 5331 of ASME NOG-1.

The complete operating control system, along with emergency control features is located in the cab on the OHLHS. Additional wireless remote control stations are also provided for remote operations of the OHLHS. The wireless remote control stations have the same control, including emergency, features as the cab mounted controls. The configuration of the controls stations are in accordance with Section 2-1.13 of ANSI/ASME B30.2, Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist, (Ref. 9.1.7-22). The individual control stations are interlocked to permit only one station to be operable at a time.

9.1.6 Combined License Information

COL 9.1(1) *~~The COL Applicant is to provide a program for monitoring the effectiveness of neutron poison present in the neutron absorbing panel.~~Deleted*

COL 9.1(2) *Deleted*

COL 9.1(3) *Deleted*

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Table 9.1.2-2 Light Load Drop Condition for New and Spent Fuel Rack

	Drop object	Drop weight	Drop Situation	Drop height above rack top
Case-1N	a fuel assembly plus new fuel handling tool	2,000 lbs	Straight Incline	4.2 feet above rack top
Case-2N			Straight	4.2 feet above rack bottom with empty cell
Case-1S	a fuel assembly plus spent fuel handling tool	2,450 lbs	Straight Incline	2 feet above rack top
Case-2S			Straight	2 feet above rack bottom with empty cell

Table 9.1.3-1 Recommended Spent Fuel Pit Water Chemistry Speciation

~~Analysis~~Parameter		Unit	Standard value	Limit~~ed~~ value	Sampling Frequency
1	Boron	ppm	—	≧4000	1/Week
2	Chloride ion	ppm	≦0.05	≦0.15	1/Week
3	Fluoride ion	ppm	≦0.05	≦0.15	1/Week
4	Sulfate	ppm	≦0.05	≦0.15	1/Week
5	Silica	ppm	≦1.00	≦1.00	1/Month
6	Turbidity	ppm	≦0.50	-	1/Week
7	Gamma Isotopics	μCi/ml	-	-	Continuous

Table 9.1.3-2 Spent Fuel Pit Design Parameters

SFP storage capacity	10 years spent fuel plus one core
SFP water volume (below normal water level)	400,000 gal
Boron concentration of water (ppm)	4000

9.4.1.1 Design Bases

9.4.1.1.1 Safety Design Bases

The MCR HVAC System is designed to:

- Exclude entry of airborne radioactivity into the CRE and remove radioactive material from the CRE environment such that radiation dose to MCR personnel is within the GDC 19 (Chapter 6, Section 6.4).~~(10 CFR 50, Appendix A).~~
- Support and maintain CRE habitability and permit personnel occupancy and proper functioning of instrumentation during normal and design basis accidents, assuming a single active failure (Chapter 6, Section 6.4).
- Withstand the effects of adverse environmental conditions.
- Withstand the effects of tornadoes and tornado missiles.
- Withstand the effects of seismic events. The MCR HVAC system equipment and the associated ductwork are designed to seismic category I requirements.
- Provide the MCR personnel protection by detecting and preventing the introduction of smoke into the CRE by automatically aligning the system to the emergency isolation mode (Chapter 6, Section 6.4).
- Automatically switch from normal operating mode to emergency pressurization mode upon the MCR isolation signal (Chapter 7).
- ~~Automatically switch from normal operating mode to the emergency isolation mode upon the detection of smoke in the outside air intakes (Chapter 6, Section 6.4).~~

The emergency filtration units are designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.52 (Ref. 9.4.8-3).

Proper MCR personnel protection against toxic gases is described in Chapter 6, Section 6.4.

9.4.1.1.2 Power Generation Design Bases

The MCR HVAC System is designed to:

- Maintain the CRE under proper ambient conditions (Table 9.4-1) to assure personnel comfort during normal operation and to support the continuous operation of the plant control and instrumentation equipment and components.

- The system can withstand the effects of adverse environmental conditions.
- The system can withstand the effects of tornado depressurization and tornado-generated missiles.

9.4.5.1.1.1 Annulus Emergency Exhaust System

The annulus emergency exhaust system is designed to satisfy the following design basis.

- The emergency exhaust filtration units are designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.52 (Ref. 9.4.8-3).
- The system is designed to mitigate the consequences of postulated accidents by removing the airborne radioactive material that may leak from containment.
- The system remains functional during and after a design basis accident~~ and have the capability to retain radioactive material after the system is taken out of service~~.
- The system maintains a negative pressure in the penetration and safeguard comportment areas relative to the adjacent areas (Chapter 6, Section 6.5.1).

9.4.5.1.1.2 Class 1E Electrical Room HVAC System

The Class 1E electrical room HVAC system is designed to satisfy the following design basis.

- Maintain proper operating environmental conditions within the Class 1E electrical rooms (Table 9.4-1) during normal and design basis accident.
- Maintain the hydrogen concentration below 2% by volume of Class 1E battery room.

9.4.5.1.1.3 Safeguard Component Area HVAC system

During normal plant operation, the safeguard component areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the safety-related redundant isolation dampers automatically isolate the supply and exhaust line of the auxiliary building HVAC system. The safeguard component area HVAC system is designed to satisfy the following design basis:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to support the operation of the control and instrumentation equipment and components in the individual safeguard component areas during a design basis accident or LOOP.

9.4.5.1.1.4 Emergency Feedwater Pump Area HVAC System

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- The system can withstand the effects of adverse environmental conditions.
- The system can withstand the effects of tornado depressurization and tornado-generated missiles.

9.4.5.1.1.1 Annulus Emergency Exhaust System

The annulus emergency exhaust system is designed to satisfy the following design basis.

- The emergency exhaust filtration units are designed and constructed in accordance with ASME standard N509 (Ref. 9.4.8-1), AG-1 (Ref. 9.4.8-2) and with the recommendations of RG 1.52 (Ref. 9.4.8-3).
- The system is designed to mitigate the consequences of postulated accidents by removing the airborne radioactive material that may leak from containment.
- The system remains functional during and after a design basis accident~~ and have the capability to retain radioactive material after the system is taken out of service~~.
- The system maintains a negative pressure in the penetration and safeguard comportment areas relative to the adjacent areas (Chapter 6, Section 6.5.1).

9.4.5.1.1.2 Class 1E Electrical Room HVAC System

The Class 1E electrical room HVAC system is designed to satisfy the following design basis.

- Maintain proper operating environmental conditions within the Class 1E electrical rooms (Table 9.4-1) during normal and design basis accident.
- Maintain the hydrogen concentration below 2% by volume of Class 1E battery room.

9.4.5.1.1.3 Safeguard Component Area HVAC system

During normal plant operation, the safeguard component areas are served by the auxiliary building HVAC system (Section 9.4.3). During a design basis accident or LOOP, the safety-related redundant isolation dampers automatically isolate the supply and exhaust line of the auxiliary building HVAC system. The safeguard component area HVAC system is designed to satisfy the following design basis:

- Provide and maintain proper environmental conditions within the required temperature range (Table 9.4-1) to support the operation of the control and instrumentation equipment and components in the individual safeguard component areas during a design basis accident or LOOP.

9.4.5.1.1.4 Emergency Feedwater Pump Area HVAC System

Chapter 10

- Hydrogen gas purity
- Generator ampere, voltage, and power

Additional generator protective devices are listed in Table 10.2-3.

10.2.2.3.4 Plant Loading and Load Following

The T/G control system has the same loading and load following characteristics as the control system described in Section 7.7.

10.2.2.3.5 Inspection and Testing Requirements

Major system components are readily accessible for inspection and are available for testing during normal plant operation. Turbine trip circuitry is tested prior to unit startup. To test control valves with minimal disturbance, the load is reduced to that capable of being carried with one control valve closed.

10.2.3 Turbine Rotor Integrity

Turbine rotor integrity is provided by the integrated combination of material selection, rotor design, fracture toughness requirements, tests, and inspections. This combination results in a very low probability of a condition that could result in a rotor failure.

10.2.3.1 Materials Selection

Fully integral turbine rotors are made from ladle refined, vacuum deoxidized Ni-Cr-Mo-V alloy steel by processes that maximize the cleanliness and toughness of the steel. The lowest practical concentrations of residual elements are obtained through the melting process. The turbine rotor material complies with the chemical property limits of ASTM A470 (Reference 10.2-5), Classes 5, 6, and 7. The specification for the rotor steel has lower limitations than indicated in the ASTM standard (Reference 10.2-5) for phosphorous, sulphur, aluminum~~,~~ and antimony~~, tin, argon, and copper~~. This material has the lowest fracture appearance transit temperatures (FATT) and the highest Charpy V-notch energies obtainable on a consistent basis from water-quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Charpy tests and tensile tests in accordance with ASTM, A370 (Reference 10.2-6)~~ and/or the equivalent are required from the forging supplier~~.

The production of steel for the turbine rotors starts with the use of high-quality, low residual element scrap. An oxidizing electric furnace is used to melt and dephosphorize the steel. Ladle furnace refining is then used to remove oxygen, sulphur, and hydrogen from the rotor steel. The steel is then further degassed using a process whereby steel is poured into a mold under vacuum to produce an ingot with the desired material properties. This process minimizes the degree of chemical segregation since silicon is not used to deoxidize the steel.

10.2.3.2 Fracture Toughness

conservative factor to correct for the imperfect nature of a flaw as an ultrasonic reflector, as compared to the calibration reflector. The resulting area is the corrected flaw area. For an acceptable design, the allowable initial flaw area must be greater than or equal to the corrected flaw area.

For rotor contour~~ or for flaws near the rotor bore (for bored rotors)~~, a surface connected elliptical crack is assumed. The flaw is assumed to be orientated normal to the maximum principle stress direction.

The beginning-of-life FATT for the high pressure and low pressure rotor is specified in the material specification for the specific material alloy selected. Both high pressure and low-pressure turbines operate at a temperature at which temperature embrittlement is insignificant. The beginning-of-life FATT is not expected to shift during the life of the rotor due to temperature embrittlement.

Minimum material toughness is provided by specification of the maximum FATT and minimum upper shelf impact energy for the specific material alloy selected. There is not a separate material toughness (K_{IC}) requirement for US-APWR rotors.

10.2.3.2.2 Rotor Fatigue Analysis

A fatigue analysis is performed for the turbine rotors to show that cumulative usage is acceptable for expected transient conditions including normal plant startups, load following cycling, and other load changes. A margin is provided by assuming a conservatively high number of turbine start and stop cycles. The turbine rotors in operating nuclear power plants were designed using this methodology and have had no history of fatigue crack initiation due to duty cycles.

In addition to the low cycle fatigue analysis for transient events, an evaluation for high cycle fatigue is performed. This analysis considers loads due to gravity bending and bearing elevation misalignment. The local alternating stress is calculated at critical rotor locations considering the bending moments due to the loads described above. The maximum alternating stress is less than the smooth bar endurance strength modified by a size factor.

The T/G is supported by a reinforced concrete foundation, which is designed so that the vertical deflection of beams, girders and columns/column-wall should not impose additional alternating stress on the T/G or shaft train considering the following factors:

- Condenser vacuum load
- Normal torque load
- Thermal load due to machine expansion-contraction
- Load due to temperature increase of the deck
- Piping load

The dynamic response of the T/G foundation including vibration amplitude and natural frequency analysis are analyzed to confirm that no additional alternating stress is imposed on the T/G shaft train.

10.2.3.3 Preservice Inspection

Preservice inspections for turbine rotors include the following:

- Rotor forgings are rough machined with a minimum stock allowance prior to heat treatment.

- Each rotor forging is subjected to a 100-percent volumetric (ultrasonic) examination. Each finish-machined rotor is subjected to a surface magnetic particle and visual examination. Results of the above examination are evaluated by use of criteria that are more restrictive than those specified for Class 1 components in ASME Code, Section III and V (Reference 10.2-7 and 10.2-8). These criteria include the requirement that subsurface ultrasonic~~sonic~~ indications are either removed or evaluated to verify that they do not grow to a size which compromises the integrity of the unit during the service life of the unit.

- Finish-machined surfaces are subjected to a magnetic particle examination. No magnetic particle flaw indications are permissible in ~~bores (if present) or other~~ highly stressed regions.

- Each fully bladed turbine rotor assembly is spin tested at 120 percent overspeed, the maximum anticipated design overspeed at a load rejection from full load.

Rotor areas which require threaded holes are not subjected to a magnetic particle examination of the threaded hole. The number of threaded holes is minimized, and threaded holes are not located in high stress areas.

10.2.3.4 Turbine Rotor Design

The turbine assembly is designed to withstand normal conditions and anticipated transients, including those resulting in turbine trip, without loss of structural integrity. The design of the turbine assembly meets the following criteria:

- The design overspeed of the turbine is 5% above the highest anticipated speed resulting from a loss of load.

- The combined stresses of the low-pressure turbine rotor at design overspeed due to centrifugal forces and thermal gradients do not exceed 0.75 of the minimum specified yield strength of the material at design overspeed.

- The turbine shaft bearings are able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in turbine trip.

- Each rotor, stationary and the rotating blade path component is inspected visually and by magnetic particle testing on its accessible surfaces. Magnetic Particle ~~Ultrasonic~~ inspection of the side entry blade grooves is conducted. These inspections are conducted at intervals equal or less than~~of about~~ 10 years for both high-pressure and low-pressure turbines.

- A 100 percent surface examination of couplings and coupling bolts is performed.

- The fluorescent penetrant examination is conducted on nonmagnetic components.

- At least one main steam stop valve, one main steam control valve, one reheat stop valve, and one intercept valve are dismantled approximately every 3 years during scheduled refueling or maintenance shutdowns. A visual and surface examination of the valve internals is conducted. If unacceptable flaws or excessive corrosion are found in a valve, the other valves of the same type are inspected. Valve bushings are inspected and cleaned and bore diameters are checked for proper clearance.

- Main stop valves, control valves, reheat stop and intercept valves may be tested with the turbine online. The DEH control test panel is used to stroke or partially stroke the valves.

- Extraction nonreturn valves are tested prior to each startup.

- Turbine valve testing is performed at quarterly intervals. The quarterly testing frequency is based on nuclear industry experience that turbine-related tests are the most common cause of plant trips at power. Plant trips at power may lead to challenges of the safety-related systems. Evaluations show that the probability of turbine missile generation with a quarterly valve test is less than the evaluation criteria.

- Extraction nonreturn valves are tested locally by stroking the valve full open with air, then equalizing air pressure, allowing the spring closure mechanism to close the valve. Closure of each valve is verified by direct observation of the valve arm movement.

The Combined License Applicant is to develop turbine maintenance and inspection procedure and then to implement prior to fuel load. Plant startup procedure including warm-up time will be completed therein.

10.2.4 Evaluation

Components of the turbine-generator are conventional and typical of those which have been extensively used in other nuclear power plants. Instruments, controls, and protective devices are provided to confirm reliable and safe operation. Redundant, fast actuating controls are installed to prevent damage resulting from overspeed and/or full load rejection. The control system initiates turbine trip upon reactor trip. Automatic low-pressure exhaust hood water sprays are provided to prevent excessive hood temperatures. Exhaust casing rupture diaphragms are provided to prevent low-pressure cylinder overpressure in the event of loss of condenser vacuum. The diaphragms are flange mounted and designed to maintain atmospheric pressure within the condenser and turbine exhaust housing while passing full flow.

Component	Alloy/Carbon Steel
Pipe	ASME B31.1
Fittings	ANSI B16.9
	ANSI B16.11
	ANSI B16.28
Flanges	ANSI B16.5

Material specifications for the MSS and CFS piping and components are listed in Tables 10.3.2-3 and 10.3.2-4.

Nondestructive inspection of ASME Code Section III (Reference 10.3-6), Class 2 and 3 components is addressed in Section 6.6.

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).

- ~~Austenitic stainless steel components conform with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel" (Reference 10.3-17), and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel" (Reference 10.3-18).~~

- 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 ~~steel is implemented at preheat temperatures specified~~ 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. ~~Controls in the welding procedures are stated with respect to carbon or low-alloy steel components.~~ The minimum preheat temperatures for carbon steel and low alloy materials conform to the recommendations in ASME ~~with~~ Section III, Appendix D, Article D-1000~~, of the ASME Code~~ (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.

- ~~Cast austenitic stainless steel materials are inspected by volumetric methods.~~

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.

Pipe schedules/wall thicknesses are selected taking into consideration expected corrosion over the design life of the plant. Corrosion allowances meet the requirements of ASME section III (Reference 10.3-6) for safety class piping and ASME B31.1 (Reference 10.3-7) for non safety class piping.

Piping design and layout minimizes bends and elbows. Pipe sizes are selected to have velocities within industry recommended values.

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 is to address preparation of an FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam.

10.3.7 Combined License Information

COL 10.3(1) FAC monitoring program
The Combined License Applicant ~~is to address preparation~~will provide a description of ~~a~~the FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam. The description will be address consistency with Generic Letter 89-08 and NSAC-202L-R3 and will provide a milestone schedule for implementation of the program.

COL 10.3(2) Safety and relief valve information
The Combined License Applicant is to address the actual throat area of the MSSV.

10.3.8 References

10.3-1 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50, Appendix A.

10.3-2 Station Blackout, Regulatory Guide 1.155 Rev.0, August 1988.

10.3-3 Loss of all alternating current power, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50.63.

10.3-4 Protection Against Low-Trajectory Turbine Missiles, Regulatory Guide 1.115 Rev.1, July 1977.

10.3-5 Tornado Design Classification, Regulatory Guide 1.117 Rev.1, April 1978.

10.3-6 Rules for Construction of Nuclear Facility Components, ASME Boiler and Pressure Vessel Code. Division 1, Section III, 2007.

10.3-7 Power Piping, ASME B31.1.

10.3-8 Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, Section XI, Division 1.

10.3-9 Codes and standards, NRC Regulations Title 10, Code of Federal Regulations, 10CFR Part 50.55a.

10.3-10 Rules for Design of Safety Valve Installations, ASME Boiler and Pressure Vessel Code Division 1, Section III, Non-mandtory Appendix O.

10.3-11 U.S. Nuclear Regualtory Commission, Monitoring of Secondary Side Water Chemistry in PWR. Steam Generators, NUREG-0800 Branch Technical Position MTEB 5-3.

10.3-12 Design Fatigue Curves, ASME Boiler and Pressure Vessel Code Division I, Section III, MANDATORY APPENDIX I.

10.3-13 MATERIALS PART A Ferrous Material Specifications, ASME Boiler and Pressure Vessel Code, Section II, 2007.

10.3-14 MATERIALS PART B Nonferrous Material Specifications, ASME Boiler and Pressure Vessel Code, Section II, 2007.

10.3-15 MATERIALS PART C Specifications for Welding Rods, Electrodes, and Filter Metals, ASME Boiler and Pressure Vessel Code, Section II, 2007.

10.3-16 DESIGN AND FABRICATION CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1, Regulatory Guide 1.84 Rev.26, July 1989.

10.3-17 Deleted~~NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL, Regulatory Guide 1.36.~~

Table 10.3.2-3 Main Steam and Feedwater Piping Design Data

Main Steam Piping

Segment	Material specification	Nominal OD	ASME Class
SG outlet to containment penetration	SA-333, Grade 6 (Seamless)	32 inch	Section III, Class 2
Containment penetration to MSIV	SA-333, Grade 6 (Seamless)	32 inch	Section III, Class 2
MSIV to main steam/feedwater piping area wall	SA-333, Grade 6 (Seamless)	32 inch	Section III, Class 3
Fittings	SA-181, Gr. 70 or SA-333, Grade 6 (Seamless)	32 inch	Note: Material Spec. for fittings, flanges and valves is same between ASME Section III Class 2 and 3.
Flanges	SA-508 Class 1, Class 900		
Valves (Globe, Gate, Check)	SA-352, Grade LCB		
Main steam steam/feedwater piping area wall to equalization piping	ASTM A-672 Grade~~.~~ B60	32 inch	B31.1
Equalization piping	ASTM A-672 Grade~~.~~ B60	28 inch & 42 inch	
Lines to TSV	ASTM A-672 Grade~~.~~ B60	32 ~~28~~ inch & 30 inch	
Fittings	ASTM A-105, A-672 Grade B60	28 inch, 32 inch, 42 inch	
Flanges	ASTM A-105		
Valves (Globe, Gate, Check)	ASTM A-181 Grade 70 or ASTM A-216 Grade WCB, Class 900		

Feedwater Piping

Segment	Material specification	Nominal OD	ASME Class
Feedwater pump outlet to~~—~~ feedwater pump discharge equalization piping	ASTM A-672 Grade~~,~~ B60	22 inch	B31.1
Feedwater pump discharge equalization piping	ASTM A-672 Grade~~,~~ B60	36 inch	
Feedwater pump discharge equalization piping to feedwater heaters 6/7	ASTM A-672 Grade~~,~~ B60	26 inch	
Feedwater heaters 6/7 outlet to feedwater heater 7 discharge equalization piping	ASTM A-672 Grade~~,~~ B60	26 inch	
Feedwater heater 7 discharge equalization piping	ASTM A-672 Grade~~,~~ B60	36 inch	
Fittings	ASTM A-105	22 inch, 26 inch & 36 inch	
Flanges	ASTM A-105		
Valves (Globe, Gate, Check)	ASTM A-181 Grade 70, or ASTM A-216 Grade WCB, Class 900		
Feedwater heater 7 discharge ~~E~~equalization piping to ~~MFIV~~main/steam feedwater piping area wall	~~S~~A-335~~,~~ Grade P22 (Seamless)	18 inch	B31.1
Fittings	ASTM A-182 Grade F22, ASTM A-336 Grade F22 or ASTM A-335 Grade P22	18 inch	
Flanges	ASTM A-182 Grade F22		
Valves (Globe, Gate, Check)	ASTM A-182 Grade F22, or ASTM A-217 Grade WC9		
Main/steam feedwater piping area wall to MFIV	SA-335 Grade P22 (Seamless)	18 inch	Section III, Class 3
MFIV to SG	SA-335~~,~~ Grade P22 (Seamless)	16 inch	Section III, Class 2
Fittings	SA-182 Grade F22 or SA-336 Grade F22 or SA-335 Grade P22	16 inch & 18 inch	Note: Material Spec. for fittings, flanges and valves is same between ASME Section III Class 2 and 3.
Flanges	SA-182 Grade F22		
Valves (Globe, Gate, Check)	SA-182 or SA-217 Grade WC9		

Table 10.3.2-4 Main Steam Branch Piping Design Data (2.5-INCH AND LARGER)

Segment	Material specification	Nominal OD	ASME Class
Main steam piping to MSRV/MSDV	SA-~~106~~333~~,~~ Grade~~.~~ ~~B~~6 (Seamless)	6 inch	Section III, Class 2
MSRV/MSDV discharge piping to main steam/feedwater piping area wall	~~ASTM~~ SA-106 Grade~~.~~ A ~~(Welded)~~	6 inch & 12 inch	Section III, Class 3
Main steam piping to MSSV	SA-333 Grade 6 (Seamless)	6 inch	Section III, Class 2
MSSV discharge piping to main steam/feedwater piping area wall	~~ASTM~~ SA-106 Grade~~.~~ A (Welded)	12 inch 16 inch	Section III, Class 3
Main steam piping to turbine-driven EFW pump turbine steam isolation valve	SA-~~106~~333~~,~~ Grade ~~B~~6 (Seamless)	6 inch	Section III, Class 2
EFW pump turbine steam isolation valve to turbine-driven EFW pump steam turbine	SA-333 Grade 6 (Seamless)	6 inch	Section III, Class 3
Fittings	SA-105 or SA-333 Grade 6	6 inch	Note: Material Spec. for fittings, flanges and valves is same between ASME Section III Class 2 and 3.
Flanges	SA-105	12 inch	
Valves (Globe, Gate, Check)	SA-216 Grade WCB, Class 900 or SA-181 Grade 70	16 inch	
MSRV/MSDV discharge piping outside main steam/feedwater piping area	A-106 Grade A	12 inch	B31.1
MSSV discharge piping outside main steam/ feedwater piping area	A-106 Grade A	16 inch	
Fittings	ASTM A-105 or A-106 Grade A	12 inch	
Flanges	ASTM A-105 (or equal), Class 150	16 inch	
Valves (Globe, Gate, Check)	ASTM A-105, ASTM A-216, Grade WCB (or equal), Class 150		
Reheating steam to moisture separator reheater	ASTM A-387 Grade 22	46 inch	B31.1
Fittings	ASTM A-387 Grade 22		
Flanges	ASTM A-182 F22		
Valves (Globe, Gate, Check)	ASTM A-182 F22 or ASTM A-217 Grade WC9, Class 900		
Moisture separator reheater steam to LP turbine	ASTM A-672 Grade. C60	46 inch	B31.1
Fittings	ASTM A-105		
Flanges	ASTM A-181 Grade 70 or A-216 Grade WCB, Class 900		
Valves (Globe, Gate, Check)	A-216 Grade WCB, Class 900		

Table 10.3.2-5 ASME Material Specifications with Filler Metal Specifications and Classification

ASME Material Specification	Mat'l P #	Mat'l Group #	Tensile Strength (ksi)	Filler Metal (Note 1, 6, 7, 8)			
				GTAW (Note 2)		SMAW (Note 3)	
				Specification	Classification	Specification	Classification
A-333 Grade 6	1	1	60	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-672 Grade B60	1	1	60	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-335 Grade P22	5A	1	60	AWS A5.28	ER90S-B3	AWS A5.5	E901X-B3
SA-182 F22	5A	1	Note 4	AWS A5.28	ER90S-B3	AWS A5.5	E901X-B3
SA-217 Grade WC9	5A	1	70	AWS A5.28	ER90S-B3	AWS A5.5	E901X-B3
SA-508 Grade 1	1	2	70	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-352 Grade LCB	1	1	65	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-106 Grade B	1	1	60	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-106 Grade A	1	1	48	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-387 Grade 22	5A	1	Note 5	AWS A5.28	ER90S-B3	AWS A5.5	E901X-B3
SA-672 Grade C60	1	1	60	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-181 Grade 70	1	2	70	AWS A5.18	ER70S-X	AWS A5.1	E701X
SA-105	1	2	70	AWS A5.18	ER70S-X	AWS A5.1	E701X

Notes:

1. Filler metal specifications and classifications were given for GTAW and SMAW only (the most likely welding processes). Filler metal information can be provided for other welding processes if required.
2. GTAW – Gas Tungsten Arc Welding process (Tig)
3. SMAW – Shielded Metal Arc Welding process (Stick)
4. Class 1 has a tensile strength of 60 ksi and Class 3 has a tensile strength of 75 ksi.
5. Class 1 has a tensile strength of 60 ksi and Class 2 has a tensile strength of 75 ksi.
6. In some cases filler metal classifications contain an “X” in the classification number. The welding procedure specification would need to be consulted to determine the complete classification number before ordering filler metal.
7. ASTM and ASME material specifications both recognize the AWS filler metal specifications. However, ASME has adopted SFA as their own prefix for the AWS specification (i.e. AWS A5.18 becomes SFA-5.18).
8. The filler metal specifications and classifications shown assume that the base metal is being joined to itself.

Table 10.3.2-6 Main Steam and Feedwater Piping Fluid Data

Segment	Fluid	Flow rate	Temperature	Pressure
Main steam piping (32 inch piping, ASME Class 2 or 3)	Steam (Two-phase with 0.1 % moisture)	5,050,000 lb/hr	541.2 degF	972 psia
Feedwater piping (18 inch or 16 inch piping, ASME Class 2 or 3)	Water (Single-phase)	5,050,000 lb/hr	456.7 degF	972 psia
Steam generator blowdown piping (4 inch or 3 inch piping, ASME Class 2 or 3)	Water (Single-phase)	50,500 lb/hr	456.7 degF	972 psia

The condenser shells are located below the turbine building operating floor and are rigidly supported on the turbine foundation. An expansion connection is provided between each low-pressure turbine exhaust opening and the steam inlet connections of the condenser. Four low-pressure feedwater heaters are located in the neck area of each condenser shell. Nozzles are provided at the bottom of condenser hotwell for instrumentation and control and leak detection connections.

10.4.1.2.1 System Operation

During normal power operation, exhaust steam from the low-pressure turbines is directed into the main condenser shells. The condenser also receives system flows from feedwater heater vents and drains and gland steam condenser drain.

The hotwell level controller provides automatic makeup or rejection of condensate to maintain a normal level in the condenser hotwells. On low level, the makeup control valves open and admit condensate to the hotwell from the condensate storage tank. On high-water level, the condensate reject control valves open to divert water from the condensate pump discharge to the condensate storage tank. This rejection automatically stops when the hotwell level reaches normal operating range.

Air inleakage and noncondensable gases contained in the turbine exhaust steam are collected in the condenser and removed by the main condenser air removal system. The main condenser evacuation system is discussed further in Subsection 10.4.2.

To protect the condenser shells and turbine outer casings from overpressurization, steam relief blowout diaphragms are provided in the low-pressure turbine outer casings. Pressure transmitters are provided on the condenser shells to detect the loss of the condenser vacuum. Pressure transmitters generate a turbine trip signal upon detecting the condenser pressure above its setpoint.

The main condenser is capable of accepting up to 68 percent of rated load main steam flow from the turbine bypass system. Operation of the turbine bypass system is discussed in Subsection 10.4.4.

In the event of a high condenser pressure or trip of all circulating water pumps, or trip of all condensate pumps, the turbine bypass valves are prohibited from opening.

Perforated distribution piping or baffle plates are installed to protect the condenser tubes, feedwater heaters located in the condenser neck, and other condenser components from turbine bypass steam or high-temperature drains entering the condenser shell.

The main condenser interfaces with the tube leak detection system as discussed in Subsection 9.3.2 to permit sampling of the condensate in the condenser hotwell. Should circulating water in-leakage occur, these provisions permit determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. This is performed by isolating the circulating water system from the affected water box. Plant power is reduced as necessary. The water box is then drained and the affected tubes are either repaired or plugged.

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.

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. Excess steam flow from high-pressure turbine glands is handled by the gland spillover control valve which discharges to the No. 1 feedwater heaters.

The gland seal condenser is monitored for shell side pressure and internal liquid level.

Pressure indication with an appropriate alarm is provided for monitoring the operation of

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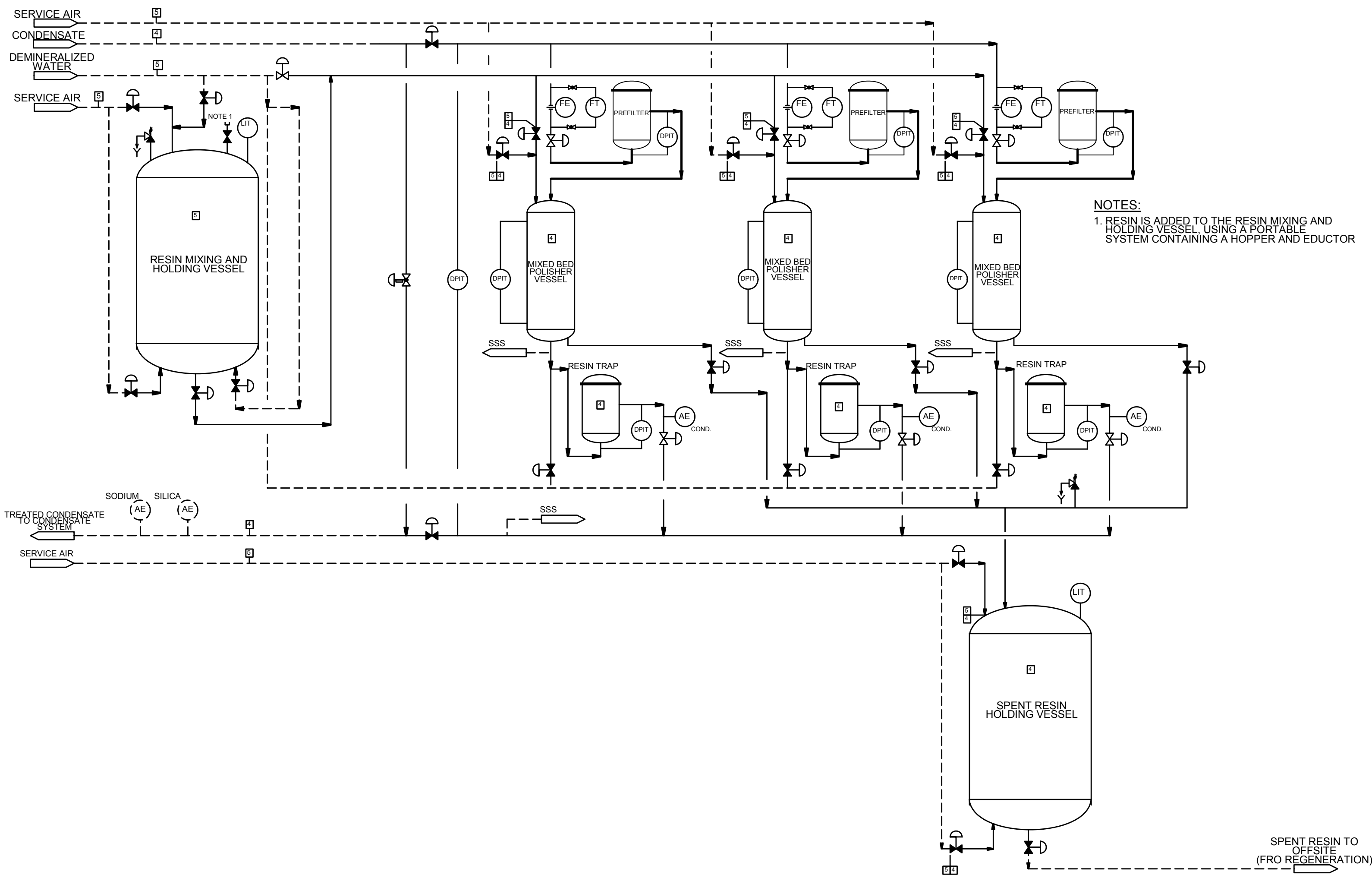


Figure 10.4.6-1 Condensate Polishing System Piping and Instrumentation Diagram

the auxiliary building and the turbine building (T/B).

The SGBDS consists of a flash tank, regenerative heat exchangers, non-regenerative coolers, filters, demineralizers, piping, valves and instrumentation. The flash tank, regenerative heat exchangers and non-regenerative coolers are provided to cool the blowdown water with heat recovery, while the filters and demineralizers are provided to purify the blowdown water.

One blowdown line per steam generator is provided. The blowdown from each steam generator flows independently to the flash tank. The blowdown water from the flash tank flows via one common line to regenerative heat exchangers and non regenerative coolers. Blowdown is split into two trains ahead of the heat exchangers. Common discharge from the coolers flows to the filters and demineralizers, where the flow is split into two trains. The purified water from the demineralizers flows to the condenser via a common discharge line.

The blowdown line from each steam generator is provided with two flow paths, a line for purifying blowdown water used during normal plant operation and a line for discharging the blowdown water to the WWS or the condenser used during startup and abnormal water chemistry conditions.

~~The blowdown water is drawn from a location above the tube sheet of each steam generator where impurities are expected to accumulate.~~ The US-APWR SG's utilize a "peripheral" blowdown system arrangement. In this arrangement, blowdown holes are drilled from approximately 7 inches below the secondary surface of the tubesheet and intersect with the peripheral groove on the secondary face of the tubesheet. This arrangement is shown as Figure 10.4.8-3 and facilitates effective sludge removal from the tubesheet. The blowdown from each steam generator is depressurized by a throttle valve located downstream of the isolation valves. The throttle valves can be manually adjusted to control the blowdown rate.

The depressurized blowdown water flows to the flash tank, where water and flashing vapor are separated. The vapor is diverted to deaerator and the water is transferred to regenerative and non-regenerative heat exchangers for further cooling. During plant startup when the pressure in the flash tank is low, the vapor is diverted to condenser. The condensate and feedwater system (CFS) provides the condensate in regenerative heat exchanger(s) to recover thermal energy.

The turbine component cooling water system (TCS) cools blowdown water in the non-regenerative heat exchanger to protect the demineralizer resin prior to purifying the blowdown water. The impurities from the cooled blowdown water are removed by the inlet filters, demineralizers and outlet strainers. SG blowdown demineralizers consist of two cation demineralizers and two mixed bed demineralizers. The purified water is returned to the condenser.

A local grab sample point which is provided downstream of each demineralizers, a radiation monitor downstream of demineralizers outlet strainers and a radiation monitor in the sample line measures impurities concentration and the radioactivity level in the power.

During normal operation, including SG tube leakage and condenser tube leakage within allowable limits and with low impurities, the CFS water chemistry is high in pH.

SG blowdown water is cooled in series of regenerative heat exchanger and non-regenerative cooler, purified by SG blowdown demineralizers and discharged to the condenser hotwell. After the initiation of purification in the SG blowdown demineralizers, all condensate water bypasses CPS.

The radioactive spent resins are transferred to solid waste management system (SWMS) for disposal, when SG tube leakage exceeds allowable limits and resins are non recyclable. During normal operation without SG tube leakage, non-radioactive spent resins are transferred to a non-radioactive spent resin holding vessel (SRHV) in CPS. These resins are shipped to an off-site facility for regeneration.

10.4.8.2.2.3 Plant Shutdown

In this mode, the reactor is brought from no-load power operating temperature and pressure to a cold shutdown.

High blowdown rates (up to 3% MSR at rated power) may be used to reduce the solids contents in the steam generators and maintain secondary water chemistry within allowable limits. The blowdown water is returned to the condenser or to the WWS.

10.4.8.2.2.4 Steam Generator Drain

The SGBDS is used to drain the steam generators. In this mode, the blowdown drain water is directed to the condenser or to the WWS. The COL Applicant is to describe the nitrogen or equivalent system design for Steam Generator drain.

10.4.8.2.2.5 Abnormal Operation

(1) Condenser Tube leakage

The CPS goes into service and maintains the condensate water quality. SG blowdown water can be purified by the SG blowdown demineralizers to support purification of CPS or diverted directly to the condenser.

(2) SG blowdown lines isolation signals

When SG blowdown line isolation signals are generated, SG blowdown lines are isolated. Following recovery from this event, SG blowdown water is initially routed directly to the condenser and when the blowdown water quality is acceptable, SG blowdown demineralizers start purifying the blowdown water.

(3) Abnormal water chemistry condition

When the impurities concentrations at the outlet of the SG blowdown demineralizers increase beyond the predetermined limit, blowdown water is diverted to the condenser of to the WWS.

(4) Malfunction in SGBDS

SGBDS lines are automatically isolated, in the case of a malfunction of regenerative heat exchangers, non-regenerative coolers, flash tank vent line and the detection of one of the following conditions:

- The outlet temperature of Non-regenerative coolers is equal to or higher than the predetermined set point.
- High Pressure in the SG blowdown flash tank
- High-high water level in the SG blowdown flash tank

In these cases, the blowdown water is diverted to the condenser and the CPS is placed in operation, as required.

Similarly when high pressure or an abnormally high water level is detected in the deaerator, the SG blowdown water is diverted without purifying to the condenser and the CPS is placed in operation, as required.

After the condition is restored, the SGBDS is placed in service.

(5) SG Tube Leakage

In the case of primary to secondary SG tube leakage within allowable tube leak rate, as specified in the plant technical specifications, blowdown water continues to be purified with SG blowdown demineralizers to remove the radioactivity entering from leaking SG tube(s).

To minimize the contaminated radioactive spent resin, the secondary water is switched to low pH AVT operation. Spent resin is transferred to SWMS for disposal.

When the SG tube leak exceeds the allowable limits, the SG blowdown lines are automatically isolated upstream of the SG blowdown demineralizers by the SG blowdown return water radiation monitor high signal.

10.4.8.2.3 Component Description

Component design parameters are provided in Table 10.4.8-1. The US-APWR SG Blowdown system design specifies low-alloy steel and stainless steel for most of the piping and components in order to preclude the need for the application of the FAC monitoring program. For any portion of the piping and /or components that are made of carbon steel, the SGBDS relies on the control of , the water chemistry as described in DCD Subsection 10.3.5, Water Chemistry. This control minimizes the potential for flow accelerated corrosion in the carbon steel piping and/or components. In addition, for the carbon steel lines and components, the FAC monitoring program will be conducted, which includes the inspection of the wall thickness of carbon steel piping and/or components and replacement if required. The FAC monitoring program is a COL item, (see the FAC monitoring program COL Item 10.3 (1) in Chapter 10.3.2).

(1) SG blowdown Flash Tank

SG blowdown flash tank is located in the T/B. During normal operation, maximum 1% MSR at rated power blowdown water is separated into flashing vapor and saturated liquid in this tank by lowering pressure and temperature in this tank.

(2) SG blowdown regenerative heat exchangers

SG blowdown regenerative heat exchangers are located in the T/B. Two-50 percent capacity blowdown regenerative heat exchanger trains are provided. Blowdown water from the flash tank is cooled in the regenerative heat exchanger(s) by the the condensate from the CFS to recover thermal energy from the blowdown water. The heated condensate is discharged into the deaerator.

(3) SG blowdown non-regenerative coolers

SG blowdown non-regenerative coolers are located in the T/B. Two-50 percent capacity non-regenerative blowdown cooler trains are provided. Blowdown water from the regenerative heat exchanger discharge flows to non-regenrative cooler(s) and is cooled by the cooling water from the TCS.

(4) SG blowdown filters

SG blowdown filters are located in the auxiliary building, ahead of SG blowdown demineralizers. These filters remove impurities from the blowdown water to protect the demineralizers. These filters consist of Two-100% capacity filters.

(5) SG blowdown demineralizers

SG blowdown demineralizers are located in the auxiliary building. The SG blowdown demineralizers purify the blowdown water during normal operation. The demineralizers consist of two cation beds and two mixed beds. Each bed has 100 percent ion exchange capability. Any cation bed can be used with any mixed bed.

(6) SG blowdown isolation valves

These valves isolate blowdown line upon receipt of an isolation signal. Two valves in series per SG located outside the containment in the reactor building are provided. See Table 10.4.8-1.

- Failure modes and effects analysis, as listed in Table 10.3.3-1, concludes that no single failure coincident with loss of offsite power compromises system's safety functions.

- High and moderate energy pipe break locations and its effects are discussed in Section 3.6.

- Coolant chemistry specifications to demonstrate compatibility with SG tube primary to secondary system pressure boundary material are addressed in Subsection 10.3.5. Preserving these specifications is accordingly able to ~~ensures~~ control secondary water chemistry needed to maintain the integrity of the SG tube materials. Furthermore the description of the bases for the selected chemistry limit and secondary coolant chemistry program for steam generator blowdown sample are specified in Subsection 10.3.5.

10.4.8.4 Inspection and Tests

The SGBDS and components are tested in accordance with the plant procedures, during the initial testing and operation program. Since the SGBDS is in continuous use during normal plant operation and essential parameters are monitored, the satisfactory operation of the system and components demonstrate system operability. The safety-related components (piping and valves) are designed and located to permit preservice and inservice inspections to the extent practical.

Additional description of inspection and tests is provided in Chapter 14.

10.4.8.5 Instrumentation Applications

Pressure, flow, temperature and radiation instrumentation monitor and control the system operation.

High pressure and high water level in the blowdown flash tank closes the upstream flow control valve.

Flow elements located downstream of the isolation valves measure and control blowdown flow from each steam generator.

Temperature instrumentation monitors the temperature of the blowdown fluid upstream and downstream of the heat exchangers and the fluid temperature is limited below predetermined value into demineralizers. A high temperature signal upstream of SG blowdown demineralizers isolates the flow to the demineralizers. A setpoint temperature of 130 °F is set to isolate the flow to the demineralizers.

The SG blowdown return water radiation monitor, located in the piping downstream of the demineralizers, detects the presence of radioactivity in SGBDS. Upon detection of the significant levels of radioactivity, the blowdown flow is diverted to the LWMS.

A high radiation signal of the SG blowdown return water radiation monitor close the SGBDS isolation valves.

The SG blowdown water radiation monitor in the blowdown sample line continuously monitors SG tube leakage. Upon detection of the significant levels of radioactivity, the

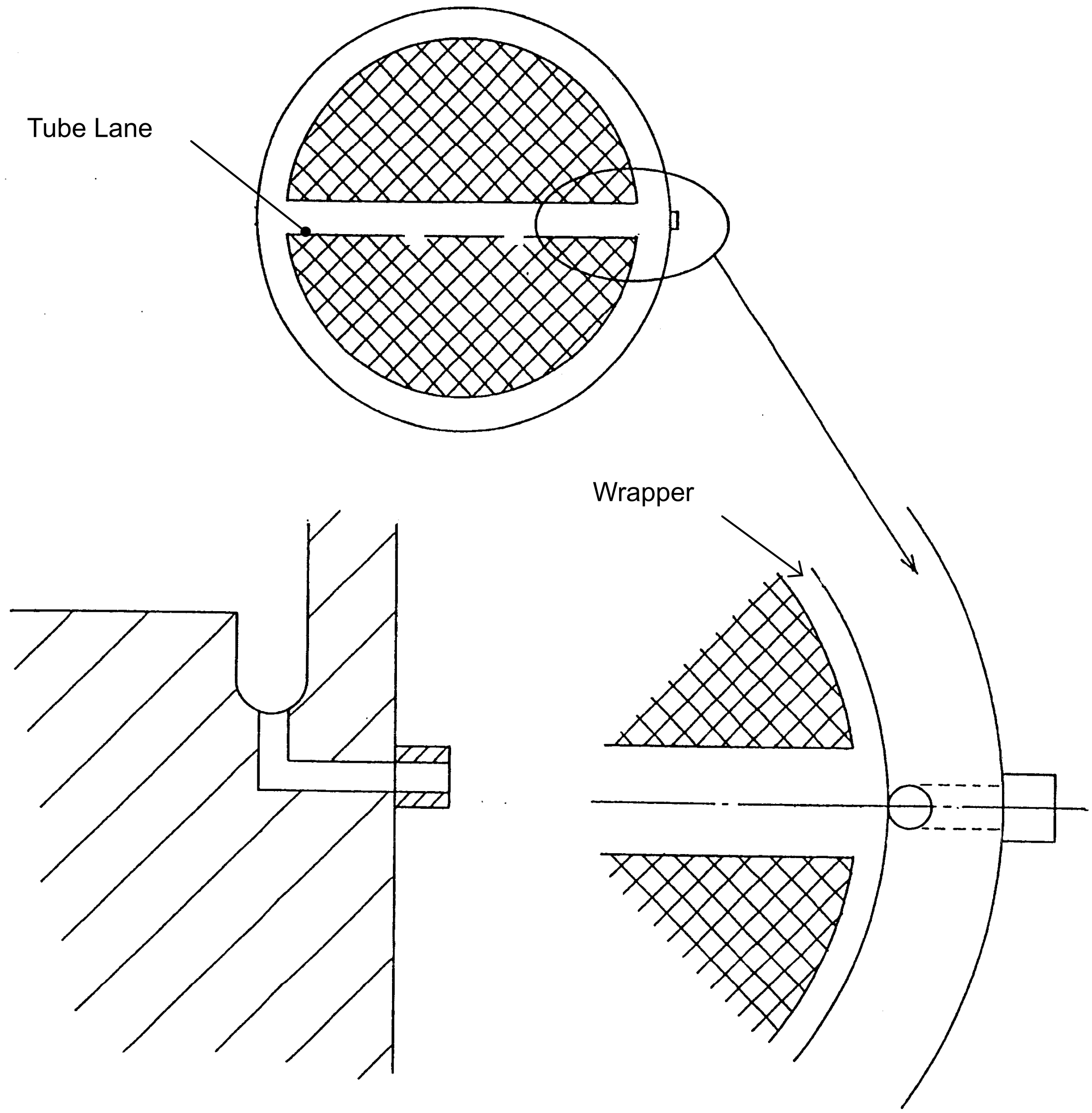


Figure 10.4.8-3 Concept of Peripheral Blowdown

Chapter 11

the environment, facilitates decommissioning, and minimizes the generation of radioactive waste.

Chemical wastes are collected and pH adjusted. The waste is neutralized prior to being pumped to waste holdup tanks for further processing or transferred to a container for disposal. Figure 11.2-1 provides flexibility to process chemical effluent either way.

The CVDT provides storage of reactor coolant pump (RCP) seal leakages, letdown water, inside containment valve leakages, and accumulator (ACC) drainage. The liquid collected is normally sent to the chemical and volume control system (CVCS) for processing. Nitrogen gas is used as a blanket in the tank to exclude oxygen and air.

11.2.1.5 Site-Specific Cost-Benefit Analysis

The LWMS is designed for use at any site. The design is flexible so that site-specific requirements such as preference of technologies, the degree of automated operation, and radioactive liquid waste storage can be incorporated with minor modifications to the design.

RG1.110(Ref. 11.2-21) outlines compliance with 10 CFR 50, Appendix I (Ref. 11.2-2) numerical guidelines for offsite radiation doses as a result of radioactive liquid effluents during normal operations, including AOOs. The cost-benefit numerical analysis as required by 10 CFR 50, Appendix I, Section II, Paragraph D (Ref. 11.2-2) demonstrates that the addition of items of reasonably demonstrated technology will not provide a more favorable cost benefit.

The COL applicant is to perform a site-specific cost-benefit analysis to demonstrate compliance with the regulatory requirements

11.2.1.6 Mobile or Temporary Equipment

The LWMS is designed with permanently installed equipment (i.e., tanks, filters, activated charcoal filter, ion exchange columns, and pumps). The LWMS does not include the use of mobile or temporary equipment. However, a space is provided inside the A/B to accommodate future installation of mobile or temporary equipment. Process and utility piping and electrical connections are provided to forward liquid waste to future mobile system or temporary equipment, at the discretion of the facility operation. Treated liquid can be returned to the waste monitor tanks for sampling, recycling, and/or release. The COL applicant is responsible for ensuring that mobile and temporary liquid radwaste processing equipment and its interconnection to plant systems conforms to regulatory requirements and guidance such as 10 CFR 50.34a(Ref.11.2-5),10CFR 20.1406(Ref.11.2-7) and RG1.143(Ref.11.2-3).

11.2.2 System Description

The boundary of the liquid waste processing system starts at the ~~building sumps and ends at the isolation valve of the discharge lines to a tank or the discharge header.~~interface valves for each of the input streams potentially containing radioactive material from other plant systems as indicated in Figure11.2-1. For many of these streams, the boundary of the LWMS starts at the respective building sump tank discharge line. The boundary of the liquid waste processing system ends at the isolation valve of the discharge lines to a tank or the discharge header.

The liquid waste processing system equipment drainage and floor drainage processing subsystem consists of four WHTs, two waste holdup tank pumps, two liquid filters, an activated charcoal filter, four ion exchange columns, two waste monitor tanks, and two waste monitor tank pumps to collect treated fluid for analysis. A process flow diagram is presented in Figure 11.2-1(Sheet 1 of 3). The WHTs and waste monitor tanks and their associated pumps are located in the A/B. The filters and ion exchange columns are located at an elevation of 3'-7" in the A/B. Layout drawings of the A/B are presented in Figures 11.5-2a through ~~j~~k.

For the purpose of this Design Control Document (DCD), process flow diagrams with process equipment, flow data, tank batch capabilities, and key control instrumentation are provided to indicate process design, method of operation, and release monitoring. Piping and instrumentation diagrams (P&IDs) are to be included in the combined license application (COLA) after the preferred process control and operating methods are established.

The four WHTs are divided into two sets: two are designed to collect high-quality liquid from equipment drainage and the other two are designated to collect liquid from floor drainage. A common header with an isolation valve is provided to segregate the collection from equipment drainage and floor drainage, however the WHTs can be used interchangeably in the event that excess equipment drainage and/or excess floor drainage waste is generated in anticipated operations.

Two filters are connected in parallel to provide redundancy. Normally, one filter is used while the other one is on standby or being maintained.

The carbon filter is sized to handle the entire effluent inventory. It is used to remove organics which could foul the ion exchange columns. The carbon filter is designed to operate occasionally and only when there is a high level of organic contaminants. It is expected that the carbon filter medium will not need to be replaced frequently. However, in case of severe fouling, the carbon can be replaced in a similar manner as the spent resin.

Four ion exchange columns are provided to operate in separate trains: two columns in series each with mixed resins for optimum performance. During normal operation, including AOOs, only one of the two trains of columns is required to operate, while the other set is on standby. When high nuclide concentration is detected, such as during operation at design-basis failed fuel level, the four columns can be arranged to operate in series so that the treated liquid meets recycle and/or release specifications.

Two waste monitor tanks are provided, while one is in the receiving mode, the other waste monitor tank can be standing by, in sampling and analysis, or in transferring release mode.

Two waste holdup tank pumps and two waste monitor tank pumps are provided for processing and transfer operations. Normally only one of each is required for recirculation and processing and transferring.

The detergent waste processing subsystem consists of one detergent drain tank, one detergent drain tank pump, one filter, one detergent drain monitor tank, and one detergent drain monitor tank pump. A process flow diagram for this subsystem is presented in Figure 11.2-1(Sheet 2 of 3).

The detergent drain~~age~~ and monitor tanks and their associated pumps are located at an elevation of -26'-4" in the A/B.

The chemical drainage subsystem consists of a chemical drain tank with pH adjustment, waste analysis features, and a chemical drain tank pump. A process flow diagram for this subsystem is presented in Figure 11.2-1(Sheet 2 of 3). The chemical drain tank and pump are located at an elevation of -26'-4". in the A/B.

Inputs to the liquid waste processing system include the following:

- Equipment drainage (major contributor)
- Floor drainage (major contributor)
- Detergent drainage (minor contributor)
- Chemical drainage (minor contributor)

See Chapter 9, Section 9.3 for a more detailed discussion of the drainage systems.

Table 11.2-2 contains specific inputs for the LWMS. These inputs are taken from ANSI/ANS 55.6, Table 7 (Ref. 11.2-6).

The reactor coolant drainage system consists of the CVDT and two containment vessel reactor coolant drain pumps. The process flow diagram for this subsystem is presented in Figure 11.2-1(Sheet 3 of 3). CVDT and containment vessel reactor coolant drain pumps are located inside the containment. (Figure 11.5-2e and 11.5-2c).

Major inputs to the reactor coolant drainage system are as follows:

- RCPs seal leakage
- Excess letdown water
- Leakage from reactor vessel flanges
- Reactor coolant loop (RCL) drainage
- Leakage from valves inside the containment

- ACC drainage
- Pressurizer relief tank drainage

These liquids drain to the CVDT or to the suction of the containment vessel reactor coolant drain pump which is located inside the containment. A nitrogen cover gas is maintained over the liquid in the tank to preserve the quality of the water and to minimize the potential for the buildup of a flammable mixture. The water entering the tank can be at a relatively high temperature (up to 200 °F), therefore, the tank is equipped with instrumentation to monitor the temperature. Prior to transferring the water to the holdup tank (HT) in CVCS via one of two containment vessel reactor coolant drain pumps, the water temperature is decreased below 200 °F by the addition of PMW. The tank is generally maintained at a near constant level to minimize both the amount of gas sent to the GWMS and the amount of nitrogen cover gas required.

11.2.2.1.2.2 Other Anticipated Operations

In the event that the liquid collected in the CVDT is either oxygenated or above the specified radiation limits, it is sent to the WHTs for processing.

11.2.2.1.2.3 Maintenance/Refueling Operations

During refueling, the containment vessel reactor coolant drain pumps are used to drain water from the reactor cavity and the fuel transfer canal to the refueling water storage auxiliary tank (RWSAT). In this case, typically both pumps are used to speed up the transfer of water from these areas. In this mode, the water is transferred directly to the RWSAT without entering the CVDT. During maintenance or outages, any remaining gas is purged from the system to the GWMS using nitrogen.

Recyclable reactor-grade effluents enter this subsystem from various locations inside the containment and are collected without exposure to air (which would degrade the quality of the water). These liquids are collected in the CVDT which is situated inside the containment. The contents of the tank are maintained in a nitrogen-rich environment to minimize the potential for degradation of the water quality and minimize the potential for formation of a flammable mixture. The tank is a cylindrical stainless steel vessel and oriented horizontally. The tank is vented to the GWMS. The tank is equipped with a relief valve, which vents to the containment sump. The purge water head tank in CVCS shares the same vapor space with the CVDT.

The liquid level and temperature within the tank are monitored. The liquid temperature is maintained below 200 °F by the addition of PMW, as necessary, in order to minimize the potential for damage to downstream equipment, including diaphragm valves. The liquid is transferred via one of two reactor coolant drainage system pumps to the CVCS HT. The liquid can also be routed to the WHT and the RWSAT depending on the quality of the water and plant conditions.

The reactor coolant drainage system pumps are also used for transferring water from the reactor cavity, the emergency core cooling system ACCs, and the fuel transfer canal to the RWSAT after refueling operations.

contain significant quantities of chemical constituents, may be transferred to the floor drainage processing subsystem.

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

11.2.2.2.8 Detergent Drain Subsystem

Detergent waste is collected in the detergent drain tank. This waste stream consists primarily of material from sinks, showers, emergency showers, etc. This waste stream does not typically contain any significant levels of radioactive contaminants. This waste stream is filtered and released through the discharge header.

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

- Detergent drain tank and associated detergent drain tank pump
- Filtration system
- Detergent drain monitor tank and associated detergent drain monitor tank pump
- Sample points
- Cross connection to the other liquid processing system
- Associated utilities

After processing, the waste is held in the monitor tank(s) where a sample is taken, and if discharge standards are met, the waste is discharged off site. Any waste not meeting discharge requirements is transferred to the WHT for further processing. The detergent drain subsystem is shown in Figure11.2-1(Sheet 2 of 3).

11.2.3 Radioactive Effluent Releases

11.2.3.1 Radioactive Effluent Releases and Dose Calculation in Normal Operation

The radioactive constituents of the waste stream are removed by the processing equipment such as filters, ion exchange, etc. Each processing equipment has an associated decontamination factor, which is a measure of the removal efficiency of the particular equipment. The decontamination factors are taken from NUREG-0017 Rev.1(Ref.11.2-13) and are presented in Table 11.2-7. The calculations are made based on the assumption as follows:

Table 11.2-4 Component Data – Pumps

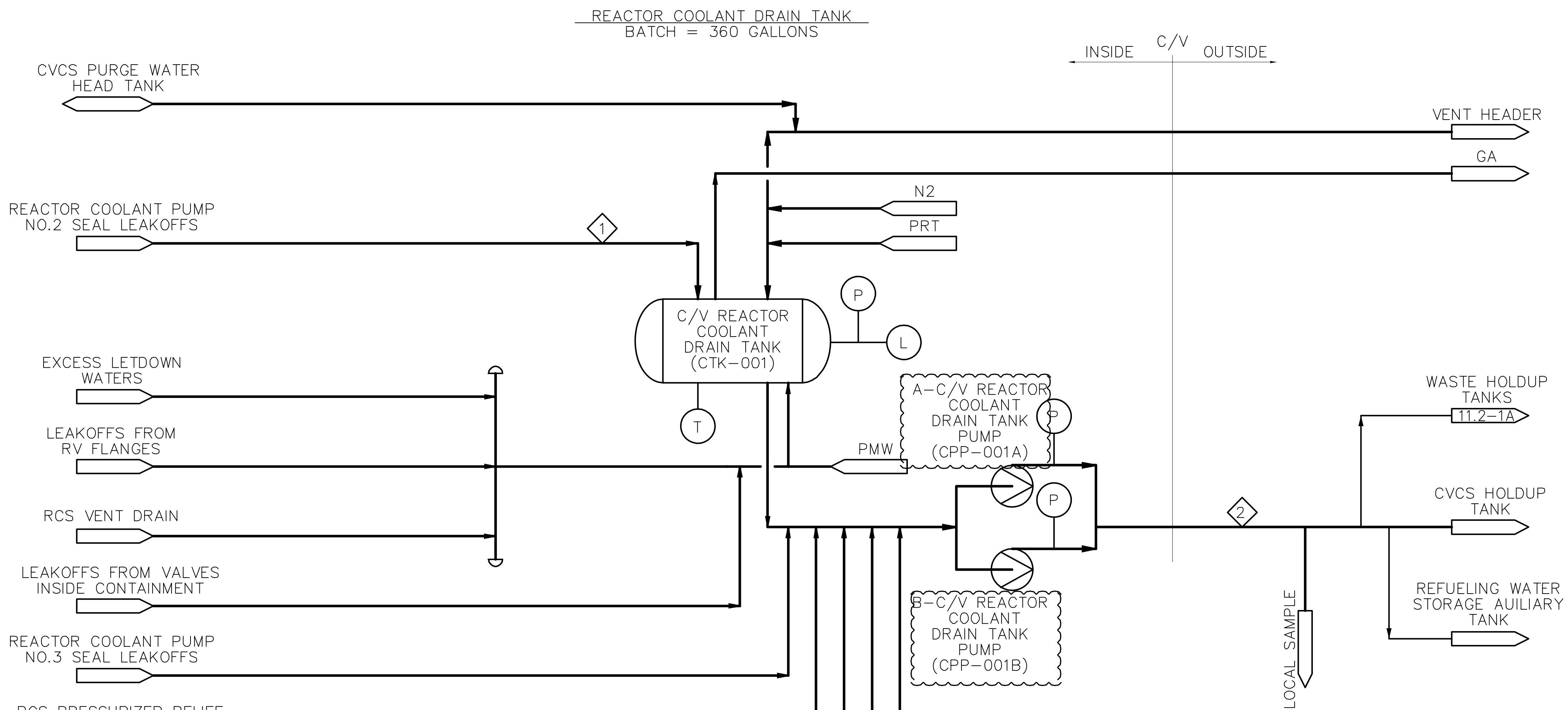
Pump Type	Specifications
Waste Holdup Tank Pumps	
Number of pumps	2
Type	Horizontal, Centrifugal
Design pressure (psig)	200
Design temperature (°F)	175
Design flow (gpm)	200
Material	Stainless Steel
Detergent Drain Tank Pump / Detergent Drain Monitor Tank Pump	
Number of pumps	1 (each)
Type	Horizontal, Centrifugal
Design pressure (psig)	150
Design temperature (°F)	175
Design flow (gpm)	20
Material	Stainless Steel
Chemical Drain Tank Pump	
Number of pumps	1
Type	Horizontal, Centrifugal
Design pressure (psig)	150
Design temperature (°F)	175
Design flow (gpm)	20
Material	Stainless Steel
Waste Monitor Tank Pumps	
Number of pumps	2
Type	Horizontal, Centrifugal
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	200
Material	Stainless Steel
C/V Sump Pumps	
Number of pumps	2
Type	Vertical, Centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gpm)	30
Material	Stainless Steel
C/V Reactor Coolant Drain ~~Tank~~ Pumps	
Number of pumps	2
Type	Horizontal, Centrifugal
Design pressure (psig)	200
Design temperature (°F)	200
Design flow (gpm)	120
Material	Stainless Steel
Sump Tank Pumps	
Number of pumps	6 (2 for each sump tank)
Type	Horizontal, Submerging
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	50
Material	Stainless Steel

Table 11.2-5 Component Data – (Filters)

Filter Type	Specifications
Waste Effluent Inlet Filter	
Number of filters	2
Type	Cartridge
Design pressure (psig)	200
Design temperature (°F)	175
Design flow (gpm)	90
Particle size (micron)	25
Filter housing material	Stainless Steel
Detergent Drain Filter	
Number of filters	1
Type	Cartridge
Design pressure (psig)	150
Design temperature (°F)	175
Design flow (gpm)	~~10~~20
Particle size (micron)	25
Filter housing material	Stainless Steel

Table 11.2-6 Component Data – (Ion Exchangers)

Ion Type	Specifications
Waste Demineralizer	
Number	4
Design pressure (psig)	200
Design temperature (°F)	175
Design flow (gpm)	90
Nominal resin volume (ft^3)	70
Column material	Stainless Steel
Resin type	Mixed cation and anion
Activated Carbon Filter	
Number of filters	1
Media	Activated Carbon
Design pressure (psig)	200
Design temperature (°F)	150
Design flow (gpm)	90
Nominal resin volume (ft^3)	70
Filter housing material	Stainless Steel



	1 (NOTE 1)	2 (NOTE 2)
DESIGN FLOW GPM	25	120–240
DESIGN PRESSURE PSIG	25	200
DESIGN TEMPERATURE °F	200	200

NOTE:
1. THE FLOW IS INTERMITTENT BETWEEN THE SPECIFIED RANGE.
2. CVDT PUMPS CAN OPERATE IN PARALLEL MODE.

Figure 11.2-1 Reactor Coolant Drainage System Process Flow Diagram (Sheet 3 of 3)

11.3.1.6 Mobile or Temporary Equipment

The GWMS is designed with permanently installed equipment. The GWMS does not include the use of mobile or temporary equipment.\

11.3.1.7 Seismic Design

GWMS equipment and piping are classified as non-seismic. However, the A/B is designed to seismic Class II. The SSCs classifications for the GWMS are discussed in Chapter 3, Section 3.2.

11.3.2 System Description

The GWMS consists of two gas compressors, a gas dryer skid, four charcoal delay beds, four gas surge tanks, two hydrogen analyzer units (each with one hydrogen and one oxygen analyzer), and an oxygen analyzer unit containing dual oxygen analyzers. One of the two gas compressors operates continuously to draw gaseous waste from the HTs and the CVDT, and directs the gaseous waste into the gas surge tank for radioactive decay of short-half life isotopes. Upon the completion of decay, or at the operator's discretion, the gaseous waste is processed through the dryer, the charcoal bed adsorbers, and sent to the vent stack for release. When the gas pressure in the VCT reaches the predetermined setpoint, the pressure control valve opens and the gas is released into the gas dryer and charcoal bed adsorbers for treatment and release. A recycle line to the suction side of the gas compressors is provided to direct the gaseous waste from the VCT to go to the gas surge tanks.

The process design of the GWMS is presented in Figures 11.3-1 (Sheet 1 and 2 of 3). The equipment layout is in Figure 11.5-2a through 11.5-~~2j~~2k. Piping and instrumentation diagrams (P&IDs) are to be included in the combined license application.

During normal operation, radioactive isotopes including xenon, krypton, and iodine are generated as fission products. A portion of these nuclides are present in the reactor coolant due to fuel defects. These nuclides are stripped out of the coolant in the VCT and the HTs into the cover gas and form the input to the GWMS.

The main sources of plant radioactive gaseous inputs to the GWMS are the waste gases from the VCT, the CVDT, and the HTs. Since nitrogen is used as a cover gas for the HTs, the gas, after decay in the gas surge tank, is returned back to the HT for reuse. Otherwise, the nitrogen gas is treated and discharged. Hence, the majority of the waste gas entering the GWMS during normal operation is composed of nitrogen cover gas, a small amount of radioactive gaseous isotopes of krypton and xenon, and to a lesser extent, hydrogen and oxygen.

The charcoal bed adsorbers are used to control and minimize the release of radionuclides into the environment by delaying the release of the radioactive noble gases, including krypton and xenon. The charcoal bed adsorbers contain activated charcoal that has been used extensively to remove radioactive iodine and other noble gases before the gaseous waste is routed to the discharge structure. The charcoal bed adsorbers provide up to 45 days of delay time for these gases at the design flow conditions.

Any liquid generated from the operation of the GWMS is collected and routed to the LWMS for further processing. The equipment drainage from the GWMS is routed to the WHTs in the LWMS for further processing.

Some hydrogen and oxygen are generated from the hydrolysis and radiolysis of the coolant-water. At sufficiently high concentrations, these gases can form flammable and explosive mixtures. Hence, streams in the GWMS are monitored for both hydrogen and oxygen content so that a flammable limit will not be reached. The GWMS provides sufficient dilution of nitrogen gas to maintain a hydrogen concentration below 4% by volume and an oxygen concentration below 4% by volume before the gaseous waste is sent to the vent stack. This gas is further diluted with the A/B ventilation flow in the vent stack before it is discharged to the atmosphere.

Initially, the waste gas from the HTs and the CVDT is compressed, cooled, demoisturized, and then routed to be released to the atmosphere via the vent stack. Component cooling water (CCW) is supplied to the gas cooler located downstream from the compressors and is designed to cool the gaseous waste stream to separate the moisture from the gas stream in the moisture separator. The moisture separator has level control and automatically activates the valve to drain the moisture into the LWMS WHTs. The gaseous waste stream is then routed to the molecular sieve tank to remove the remaining moisture with desiccant before the gas is forwarded to the gas surge tank for decay and later on to the charcoal bed adsorbers for removal of the radioactive nuclide gases.

After nuclide removal, the gaseous stream is routed to the vent stack for discharge. The vent stack is located along side of the containment and the discharge point is above the top of the containment. Radiation monitors are provided before the discharge valve so that the release limits are not exceeded. The discharge valves remain open when the radiation setpoint is not exceeded. The gaseous effluent is further diluted with the HVAC ventilation flow in the vent stack. Lack of ventilation flow closes the discharge valve. Two additional normal-range radiation monitors are provided to monitor the releases in the vent stack. Only one of the two monitors is required for normal operation. The other one acts as a standby. Two more radiation monitors, one mid-range and one high-range, are provided to operate during post-accident conditions. All these monitors send data to a processor which provides release data for plant operation and activates valve closure if the setpoints are exceeded. These monitors are used to meet the dose objectives of 10 CFR 50, Appendix I (Ref. 11.3-3) and the dose limits to the public of 10 CFR 20 (Ref. 11.3-4).

The vent stack is the only GWMS release point for both the gaseous system and the HVAC systems associated with the R/B, A/B and the Access Building (AC/B). ~~The vent stack design is site-specific and will be included in the detail design.~~ The COL applicant is responsible to include the site-specific vent stack design in the detail design.

as a backup. The second gas analyzer provides compliance with the redundancy guicance of NUREG-0800 (Ref. 11.3-12), and ANSI/ANS-55.4, Section 4.7 (Ref. 11.3-6). The hydrogen and oxygen analyzers do not operate continuously as does the oxygen analyzer. These instruments are used intermittently, mainly during the discharge mode of operation.

During normal plant operation, hydrogen concentration in the gas stream does not exceed 2% by volume on a moisture-free basis at the defined air flow range, as shown in Table 11.3-1. The 2% limit is set administratively so that the operator has sufficient time to investigate the cause and take corrective actions before the concentration reaches the 4% limit. The 2% limit is discussed in RG 1.189 (Ref. 11.3-8).

During startup or other reduced power operations, hydrogen concentration in the gaseous stream is not likely to come close to 2% due to the lower power operation and reduced radiolysis. However, the administrative limit is still set at 2% to comply with RG 1.189 so that the operator has sufficient time to take corrective actions before the concentration reaches the 4% limit.

In the unlikely case that the oxygen concentration does reach 4%, automatic control features are initiated at this “high-high alarm” setting. When the high-high alarm occurs, the sources of the gas to the charcoal bed are isolated by closing the valves.

11.3.2.1.5 Waste Gas Dryer

The waste gas dryer skid consists of four major components: two gas coolers, one moisture separator tank, three molecular sieve tanks, and three blower fans. The purpose of the waste gas dryer skid is to remove moisture from the effluent and protect the charcoal bed adsorbers. The redundancy provided on this skid allows the operators to cycle between the molecular sieve tanks when one becomes saturated with moisture.

Each gas cooler and molecular sieve tank is sized to handle 100% of the rated load during normal operation, including AOOs. A gas sample tap is provided downstream of the waste gas dryer skid to permit gas sampling. The following provides detailed information on each subcomponent.

11.3.2.1.5.1 Waste Gas Coolers

The waste gas coolers lower the temperature of gases exiting the dryer, allowing entrained moisture to condense before the gases enter the moisture separator tank. Only one of the two waste gas coolers is in service at any given time. The second waste gas cooler serves as a backup. The component cooling water system (CCWS) provides the cooling water required for the waste gas cooler heat exchangers.

11.3.2.1.5.2 Moisture Separator

The moisture separator tank collects the condensed moisture in the gas and routes the collected water to the LWMS. The water seal level in the moisture separator is controlled by a level transmitter and a level control valve. PMW is used during the initial startup and normal operation to maintain the water seal. The level transmitter also activates an alarm in the radwaste control room when the level is too high.

11.3.2.1.5.3 Molecular Sieve Tank

The purpose of the molecular sieve tank is to remove the remaining moisture in the gas stream before it reaches the charcoal beds. There are three molecular sieve tanks on the skid, and each tank is filled with desiccant that captures moisture when the gas is processed through the tank. One molecular sieve tank is required to be in service at any one time to support the system's needs. The second tank serves as a backup while the third is in regeneration and cooling mode. The external electric heater is not used during normal operation, but is used during the regeneration mode to heat the moisture trapped in the tank. A small flow of nitrogen is used as a motive force to remove any moisture trapped in the desiccant (drying agent). During the regeneration mode, the purged gas is routed back to the waste gas cooler to condense the moisture in the gas. Nitrogen gas also is used for purging the gas in these tanks when maintenance is required.

11.3.2.1.5.4 Blower Fans

Each of the three molecular sieve tanks is equipped with a single fan for cooling the exterior of the tank surface when the regeneration mode is complete. Using the fan shortens the cool-down period so that the molecular sieve tank can quickly be returned to service.

11.3.2.1.6 Charcoal Adsorbers

Four charcoal bed adsorbers are provided and arranged in two parallel trains. Each train has two charcoal bed adsorbers in series. During normal operation, both trains are in service and arranged in series. If one adsorber ~~bed~~ train is not available, i.e., saturated with moisture and/or requires maintenance, the train is taken out of service for a short period until the charcoal is replaced and the train is returned back to service. ~~With one train out of service, the system operates at half of its capacity for a short period until the out-of-service train is returned to service.~~ The effluent nuclide activities are also monitored by the radiation instrument at the discharge side of the adsorbers. If the activity exceeds the discharge setpoint, the gas flow is diverted to the gas surge tank and is reprocessed through the charcoal beds. The discharge can be temporarily suspended until the radiation level is below the discharge setpoint or until the out-of-service train is returned to service.

The dynamic adsorption of krypton and xenon isotopes on charcoal delay beds is shown in Table 11.3-1. The volume of charcoal is listed in Table 11.3-2.

The piping allows for the isolation and bypass of one train of charcoal bed adsorbers should it become saturated with moisture. The bypass valves are sequenced so that the charcoal bed adsorbers can be bypassed using a one-step control.

The waste gas compressors continuously keep the GWMS piping system pressurized, preventing any airborne contaminants from entering into the system. Therefore, airborne contaminants have no effect on the charcoal adsorbers' life expectancy. The

Table 11.3-3 Equipment Malfunction Analysis (Sheets 2 of 2)

Equipment Item	Malfunction	Result(s)	Alternate Action
Charcoal bed	Charcoal gets wet.	Dew point indication shows high moisture content. The charcoal performance deteriorates gradually as moisture deposits. Holdup times for Krypton and Xenon would decrease, and plant emissions would increase. Provisions made for drying charcoal as required during annual outage.	One train can be isolated and bypassed. Line up the second train. Purge the 1^{st} train with N_2 gas to remove moisture or replace the bed with fresh charcoal bed.
~~Hydrogen gas analyzer~~	~~Hydrogen gas concentration increases above the explosive limit.~~	~~Release of radioactivity if the pressure boundary fails. The radiation monitor for the HVAC system or area radiation monitor will be activated.~~	~~The main process equipment and piping are designed to contain a detonation. The system will be isolated and repaired and the redundant components will be used.~~
Instruments	Instrument failure.	Instrumentation sensors can be isolated and repaired or replaced.	All the essential instruments for the hydrogen/oxygen analyzers and oxygen analyzer are redundant.
Gaseous system	Earthquake damage.	Release of radioactivity.	Dose consequences are within the design guidance of Branch Technical Position 11-5.
Gas surge tank	Leak by or leak out.	The room becomes contaminated. Area rad monitors and HVAC discharge radiation monitor actuate alarms in the control room.	Isolate the damaged vessel and use the backup vessel.

be fully loaded with suspended solids and dissolved solids using the design basis source term, which is also used to determine the thicknesses of the shield walls.

- The SRSTs are cross-connected so that the failure or maintenance of one component does not impair the system or the plant operation. Table 11.4-5 provides typical failure scenarios. The spent resin storage tanks (SRSTs) are housed in individual cubicles, each with a shield wall thickness commensurate with the projected maximum dose rate of its content. The cubicles are ~~lined with steel~~ epoxy coated to ease decontamination. Further, the epoxy coating~~steel liner~~ also serves to minimize the potential for contamination of groundwater in the event that a tank fails or overflows. Other design features addressing release requirements are described in Section 11.2

- Interconnections between the SWMS and other plant systems are designed so that contaminations of non-radioactive systems are precluded and the potential for uncontrolled and unmonitored releases of radiation to the environment from a single failure are minimized.

- Storage is provided to facilitate radioactive decay of spent resin and to provide adequate holding in the case of a delay in processing due to maintenance and/or a delay in transportation for disposal.

- Any liquids and gases generated from the operation of the SWMS are processed by the LWMS (Section 11.2) and plant ventilation system (Chapter 9, Section 9.4). Based on typical PWR experience, a small quantity of sludge and oily waste is expected to be generated. Sludge is stabilized and transported to a disposal facility. Oily waste is collected and sent to appropriately licensed offsite vendors for processing and disposal.

- Collection, processing, packaging, and storage of radioactive wastes is performed to maintain any potential radiation dose to plant personnel ALARA in accordance with RG 8.8 (Ref. 11.4-2) and within the limits of 10 CFR 20 (Ref. 11.4-3). Some of the design features incorporated to maintain exposure levels ALARA include remote system operation and remotely actuated flushing, and an equipment layout that shields personnel from components containing radioactive materials.

- The SWMS is designed to package radioactive wastes in accordance with 10 CFR 61 (Ref. 11.4-4) and the applicable portions of 10 CFR 60 (Ref 11.4-5) and 10 CFR 63 (Ref. 11.4-6). The containers meet the requirement of 49 CFR 171 (Ref. 11.4-7). Solid wastes are processed and packaged for transportation and disposal according to the requirements specified in 49 CFR 173, Subpart I (Ref. 11.4-8).

- Sufficient onsite storage is provided to hold solid waste for at least 30 days in accordance with ANSI 55.1 (Ref. 11.4-9).

- The SWMS is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 61 (Ref. 11.4-10) to assure adequate safety under normal and postulated accident conditions, and GDC 63 so that the SWMS has an ability to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions.

- Spent resin is sampled for analysis and the volume to be transferred into the high-integrity container is predetermined. After the filling operation, the radiation level of the container is monitored prior to offsite shipment assuring that the containers meet regulatory radiation limit and waste acceptance criteria, achieving compliance with 10 CFR 50, Appendix A, GDC 64 (Ref. 11.4-10).

- The SWMS is designed to meet the requirements of 10 CFR 20.1301(e), 10 CFR 20.2006, 10 CFR 20.2007and 10 CFR 20.2108.

- The SWMS is designed to minimize the potential for the release of radioactive effluent to the environment. Any liquids and gases from the operation of the SWMS are routed to the LWMS and GWMS for treatment. Liquid from the de-watering of spent resin and spent carbon, equipment and piping flushes and decontamination, and local area drainage are forwarded to the WHT or liquid radwaste sump tanks located in the A/B for collection and are pumped to the LWMS for further processing. During spent resin transfer and filling operations, the gases from the SRSTs are routed to the HT to be processed by the GWMS.

- The SWMS is designed to handle and package radioactive solid wastes remotely to keep radiation doses ALARA. This design approach results in radiation doses for individuals and the general population that are well within the limits of 10 CFR 20 (Ref. 11.4-3) and 10 CFR 50, Appendix I (Ref. 11.4-12).

- The SRSTs are housed in individual cubicles, each with a shield wall thickness commensurate with the projected maximum radioactive dose rate of its content. The cubicles are ~~steel lined~~ epoxy coated to minimize the potential for contamination of the groundwater system in the event that the tank fails or overflows. Other design features addressing the release requirements are described in Section 11.2.

11.4.1.3 Other Design Considerations

In addition to the listed design criteria, the following considerations are satisfied:

- The SWMS is a non-safety system and the components are non-seismic. The design specifications of the tanks and piping are consistent with those in Table 1 of RG 1.143 (Ref. 11.4-1).
- The SWMS consists of independent subsystems and components designed to handle different types of wastes. The control philosophy for the spent resin transfer operation is based on a manual start and an automatic stop. Interlocking is provided in some functions to provide a fail-safe mode of operation. The control panel for the fillhead operation is located in the A/B close to the area

- The SWMS contains 30-day storage for processed wastes in accordance with the guidance set forth in ANSI/ANS 55.1 (Ref. 11.4-9). Storage facility is designed with adequate shielding to minimize the radiation dose to the operators.

- The SWMS is operated from the radwaste control room and/or via a local control panel. Priority alarm signals such as a SRST level exceeding the setpoint are sent to the radwaste control room in the A/B.

- The SWMS is designed to operate continuously during normal condition and AOOs. For equipment sizing and process capability determination, the SWMS is designed to process the maximum design basis input in one week, assuming 40 hours work week, or processing one tank of SRST in one operating shift, whichever is controlling. When wastes are accumulated in excess of the normal operation, additional solid waste processing can be performed.

- The SWMS, including the modular de-watering subsystem, is connected to non-radioactive systems such as the PMW, nitrogen, and service air systems. The non-radioactive connections (e.g., PMW for flushing, nitrogen gas for sluicing spent resin, and service air to operate valves and pumps) to the SWMS components, including the modular de-watering system, contain double isolation valves and special fittings (e.g., one check valve and one isolation valve) to minimize the potential for cross contamination of the non-radioactive system. These methods provide compliance with 10 CFR 20.1406 (Ref. 11.4-16)

- Each of the SRST cubicles is designed to contain the maximum liquid inventory in the event that the tank ruptures. Tank cubicles are ~~steel-lined~~epoxy coated to minimize the potential for accidental releases to the environment in accordance with BTP 11-3 (Ref. 11.4-14), 10 CFR 20.1302 (Ref. 11.4-15) and 10 CFR 20.1406 (Ref. 11.4-16).

- The SWMS has a temporary equipment connection for sending the waste directly to mobile equipment or to a high-integrity container for processing the solid radwaste. This connection meets RG 1.143 (Ref. 11.4-1) and ANSI 40.37-200 "Mobile Low-level Radioactive Waste Processing System" (Ref. 11.4-17) requirements.

11.4.1.5 Site-Specific Cost-Benefit Analysis

The SWMS is designed to be used for any site or plant. The design is flexible so that site-specific requirements, such as preference of technologies, degree of automated operation, and radioactive waste storage, can be incorporated with minor modifications to the design.

RG1.110 (Ref. 11.4-13) outlines compliance with to 10 CFR 50, Appendix I (Ref. 11.4-12) numerical guidelines for offsite radiation doses as a result of gaseous or liquid radioactive effluents during normal operations, including AOOs. The cost-benefit numerical analysis as required by 10 CFR 50, Appendix I (Ref. 11.4-12), Section II, Paragraph D demonstrates that the addition of items of reasonably demonstrated technology will not provide a favorable cost benefit. The COL applicant will perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.

11.4.1.6 Mobile or Temporary Equipment

The SWMS is designed with permanently installed equipment (i.e., tanks and a crane, etc.), modular equipment (the spent resin de-watering subsystem), and some mobile equipment. The purpose of the modular and mobile design is to provide ease of equipment replacement due to either advances in treatment technologies and/or broken equipment ; such are further discussed in Sections 11.4.2, 11.4.4, and 11.4.4.5. The provision and conformance requirements for the mobile system or temporary equipment for solid radioactive waste processing shall be in accordance with ANSI/ANS-40.37-1993:Mobile Radioactive Waste Processing System. This is the responsibility of the COL applicant.

11.4.2 System Description

The SWMS controls, collects, handles, processes, packages, and temporarily stores dry and wet solid waste generated by the plant prior to offsite shipping and disposal resulting from normal operations, including AOOs. The SWMS processes and packages waste from the LWMS, the CVCS, and the SFPCS. The SWMS also can receive solid waste from the CPS and the SG blowdown when the waste becomes radioactive. Waste from these systems consists of spent resin, spent charcoal, sludge, general contaminated plant debris, and spent filter elements. As these waste types differ in characteristics and contamination levels, the SWMS is divided into five subsystems that handle the following waste types:

- Dry active waste
- Spent filter elements
- Spent resin
- Spent activated carbon
- Oil and sludge

The boundary of the SWMS starts at specific waste generation streams and ends at the waste storage and truck bay of shipment of all solid waste. There is no direct discharge of waste to the environment but the packaged wastes are transferred to licensed offsite waste processing and disposal facilities.

For spent resins and spent activated carbon, the boundary to the SWMS starts downstream of the spent resin isolation valve from each of the demineralizers and activated carbon bed. Spent resin and spent carbon are transferred into the spent resin storage tanks for staging in the SWMS and later on transferred to disposal containers. The containerized wastes are then moved into the dry waste storage area until transportation for offsite processing and/or disposal is arranged. Hence for spent resin and spent carbon, the boundary terminates at the truck bay.

For spent filters, the boundary starts at the filter replacement operation during which the spent filter is picked up by the spent filter cask and transferred by a cart into the spent filter loading area. The spent filter is then dropped into a disposal container in the loading area. When the disposal container is full, it is then moved into the storage area until transportation for offsite processing and/or disposal is arranged. Hence for spent filter, the boundary terminates at the truck bay.

Sludge and mixed wastes are separately collected at the point of generation, i.e., sump and special containers placed at specific location for collection. When the container is full, it is moved into the SWMS staging area inside the low activity storage area. Hence there are no specific process streams that serve as starting point, but that they are placed in the SWMS for staging until transportation for offsite processing and/or disposal is arranged. For these wastes, the boundary terminates at the truck bay.

Similarly dry active wastes are collected at the point of generation during maintenance activities. For low contamination wastes, such as clothing and maintenance parts, they are collected in bags and boxes and brought the SWMS for staging. These wastes are placed into seal-land containers and/or B-25 boxes in the low activity storage area until transportation for offsite processing and/or disposal is arranged. The current design allows for maximum flexibility for the COL Applicant to treat and process these wastes to meet site-specific conditions. For these wastes, the boundary terminates at the truck bay.

11.4.2.1 Dry Solid Waste

Dry solid waste includes dry active waste and spent filter elements.

11.4.2.1.1 Dry active wastes

The dry active waste handling subsystem consists of an onsite storage area equipped with an overhead crane and a truck bay to load packaged waste for offsite transportation and disposal. Dry active waste is normally separated into three categories: non-contaminated wastes (clean), contaminated metal wastes, and the other wastes (i.e., clothing, plastics, HEPA filters, components, etc.). Non-contaminated wastes (clean) are not processed in the SWMS and are handled separately.

Dry active waste consists of contaminated air filters, contaminated equipment and equipment parts, solid laboratory wastes, and general plant waste that cannot be effectively decontaminated. The process control program contains plant-specific actions and procedures to handle and manage these wastes. The radioactivity of much of the dry active waste is low enough to permit contact handling and temporary storage in unshielded areas. Dry active waste is sorted and packaged in suitably sized containers that meet DOT requirements for offsite processing or final disposal. Higher activity dry active waste is separated from the low-activity waste, handled remotely, and transported in shielded containers.

General dry active waste consisting of contaminated clothing, broken and small contaminated tools and parts, contaminated pieces of wood, glass, and other materials is collected at the point of generation, surveyed, and segregated according to contamination types and radioactivity levels before they are transferred to the SWMS for packaging. The dry active waste handling and storage operation is outlined in Figure 11.4-1.

Table 11.4-2 provides an estimate of expected annual dry solid wastes and the anticipated waste classification based on operating experience and industry practices in similar PWR plants. The nuclide contamination (isotopes and activities) for these wastes is consistent with the realistic source term provided in Section 11.2. During some AOOs, such as a refueling condition, the rate of dry active waste generation is higher than the normal operation. For design purposes, a margin of 40% is included in the design of dry active waste generation and storage consideration. The SWMS is designed to operate continuously during normal condition and AOOs. For spent resin transfer equipment sizing and process capability determination, the SWMS is designed to process the maximum design basis volume in one operating shift. During the peak generation rate, additional waste handling and shipping operations can be planned to support operational needs. Maintenance activities may generate large-size dry active waste, such as wood planks and broken equipment. Handling of large-size waste must comply with the radiation protection program established for plant operation. Major equipment items, such as SGs, reactor vessels and vessel heads, are unique and are beyond the scope of the design of the SWMS. The licensee will dispose the spent equipment separately.

The design includes a truck bay located next to the packaged waste storage area. This design provides an enclosed area to bring a shipping container and load packaged waste onto the truck during periods of peak waste generation for offsite burial or processing in offsite facilities.

A permanently installed overhead crane is provided to move the packaged waste into and out of the storage area, and to load the waste onto shipping trucks. The crane has a span of about 55 ft and a lifting capacity of 40 tons. This capacity covers commonly used disposal containers in the industry.

~~The COL applicant is to include descriptions~~ Descriptions of wastes other than normally accumulated non-radioactive wastes such as wasted activated carbon from GWMS charcoal beds, solid wastes coming from component (Steam generator, Reactor vessel etc.) replacement activities, and other unusual cases will be described by the COL applicant in the process control program and will be implemented in accordance with the milestones.

11.4.2.1.2 Spent Filter Element Handling Subsystem

The spent filter element handling subsystem consists of a spent filter transfer cask, a hoist, and a laydown area for spent filter handling. In order to access the spent filter element, the shield plug and the filter-housing flange have to be removed manually before the filter can be accessed remotely. Spent filter elements are handled remotely using a mobile spent filter transfer cask, which provides remote changing of filter cartridges, dripless transport to the storage area, and transfer of the filter cartridges into and out of filter storage, and loading of filter cartridges into disposal containers. Figure 11.4-1 is a process flow diagram of the spent filter handling subsystem.

11.4.2.2 Wet Solid Waste

Wet solid wastes include spent resin, spent charcoal, sludge, and oily waste. Each of these wastes is handled separately as described below. Table 11.4-1 provides an estimate of expected annual wet solid wastes based on operating experience and industry practices in similar PWR plants. During some AOOs, such as a refueling condition, the rate of wet solid waste generation is higher than the normal operation. For design purposes, a margin of 40% is included in the design of the total generation and storage. The design is based on a 40-hour work week. During the peak generation rate, additional waste handling and shipping operation can be planned to support operational needs.

11.4.2.2.1 Spent Resin Handling and De-watering Subsystem

The spent resin handling and dewatering subsystem consists of two SRSTs and a modular de-watering station consisting of a control console, a fillhead, and a de-watering pump.

The SRSTs are located in the A/B and are individually located in shielded cubicles. The de-watering equipment is in a shielded cubicle near the storage area. The SRSTs receive spent resin from various plant sources including the LWMS, CVCS, SFPCS, SG blowdown system, and the condensate polisher ion exchange columns. The SRSTs provide staging for decay and transfer capability into disposal containers for offsite disposal. There are two SRSTs: One for low radioactive resin/carbon such as those from the LWMS, SG blowdown treatment system, and the condensate polisher (Class A waste) and the other for high radioactive resin such as those from the CVCS and SFPCS (potentially Class B or C waste). The two SRSTs are cross-tied to provide redundancy for operational flexibility. Figure 11.4-2 is a process flow diagram of the spent resin handling subsystem. Piping and instrumentation diagrams(P&IDs) are to be included in the combined license application.

Nitrogen gas is used as a motive force to transfer resin from the SRSTs to a high-integrity container via a fillhead. The fillhead is lifted from the stand to the high-integrity container with a hoist and placed into position by aligning the fillhead and the high-integrity container. The fillhead is designed to be mounted manually on top of the high-integrity container and disengaged automatically when it is lifted after the sluicing. This design keeps the dose to the operator ALARA.

Proper controls, including flow elements, interlocks and level and temperature indications, are provided with the fillhead to control slurry flow so that only the required amount of spent resin is transferred into the high-integrity container. The resin transfer automatically stops when high level or high temperature setpoints are reached in the high-integrity container. The operator also can stop this operation manually when the closed circuit~~captioned~~ television (CCTV) camera indicates a high level. The CCTV camera is used in case of level transmitter failure to minimize the potential for an overflow condition. The de-watering pump serves to remove and reduce standing water in the high-integrity container to less than 0.5% by volume to meet the transportation requirements of 49 CFR 173, Subpart I (Ref. 11.4-8). Gaseous exhaust from the fillhead during transfer and dewatering activities is vented via a vent port connection on the fillhead to the A/B ventilation system.

11.4.2.2.2 Spent Charcoal Handling

The spent charcoal handling subsystem shares the use of the spent resin tanks and the de-watering equipment as described above.

Mixing waste is not recommended, therefore, the spent activated carbon from the LWMS is normally sent directly to disposal containers. De-watering of the spent carbon uses the same process as for spent resin. If the SRST is empty, the spent carbon can also be sent to the SRST for temporary storage until further processing is warranted. The process flow diagram of the spent carbon handling subsystem is presented in Figure 11.4-2.

The resin fill tank provides a means of filling demineralizers with fresh resins and charcoal adsorber with activated charcoal. The demineralizers are filled by gravity feed from a stationary resin fill tank (hopper). This tank is equipped with hoses to reach all the demineralizers. This tank is discussed in Section 9.3.

11.4.2.2.3 Oil and Sludge Handling

In areas where rotating equipment requires the use of oil for lubrication and decontamination for maintenance, the area drainage may contain lubricants and other waste solvents. This drainage is collected in the area sump tanks, which are specially designed to provide staging and oil separation by gravity. The separated oils are transferred directly into disposable drums. This waste may contain a low level of radioactive contaminants and is forwarded to an offsite vendor for final treatment and disposal. Operating procedures control all the chemicals that are used in the plant. The sump tank is designed to separate suspended solids. The suspended solids are extracted from the sump tank and transferred into the disposal container as sludge. The process flow diagram of the oily waste and sludge subsystem is presented in Figure 11.4-3.

The following design features identified under "Additional Design Features" in BTP 11.3 (Ref. 11.4-14) are incorporated into the SWMS:

1. Components and piping which contain slurries have flushing connections.

2. Tanks and equipment, which use compressed gases for transport of resin or filter sludge, are vented to the GWMS system or plant ventilation system. The SRST vents are routed to a breakpot to minimize the potential for liquids and solids (resin) from entering the vent header. The breakpot level indication alarms when the water level reaches the setpoint. This isolates the vent valve and opens the drain valve to direct the overflow water to the A/B sump, at the same time the alarm in the radwaste control room informs the operator to take corrective action if required.

11.4.2.2.4 Mixed Waste Handling

The standard SWMS is not designed to process any mixed wastes which contain both hazardous chemical and radioactive waste. The generation of mixed waste is prevented or minimized by the design specifications in that hazardous material shall not be used. Operating procedures are used to be instituted to prevent or minimize introduction of any listed or characteristic hazardous chemicals. In the event that it cannot be avoided, the waste needs to be segregated and disposed of in order to minimize cross-contamination. Potential sources for this category are the laboratory cleaning solutions and lubricants for rotating machinery. Non-hazardous chemicals should be used, however, if hazardous chemicals must be used, the waste must be collected separately in the chemical drain tank and pumped directly into a disposal container for offsite treatment and disposal. To prevent cross contamination, it is not to be mixed with other liquid waste sources.

11.4.2.3 Packaging, Storage, and Shipping

The SWMS is designed to use DOT-approved containers for packaging of radioactive wastes. Specific container types are determined by the facility operating procedures. To estimate the number of containers and the number of potential shipments, typical high-integrity containers with useful volumes of about 100 ft^3 for Class B or C waste, and 174 ft^3 for Class A waste were assumed. However, the design is flexible to allow the use of other DOT-approved containers.

Packaging of spent resin, spent charcoal, and spent filter is performed remotely and the operation is controlled from the radwaste control room and/or local control console for filter replacement and spent resin dewatering. The filling and dewatering area and the waste staging area are shielded, and ventilation is provided to insure that airborne activity in this area is controlled and not spreading to other areas. This approach keeps radiation doses ALARA. Waste is classified as A, B, C, or greater than Class C in accordance with 10 CFR 61.55 and 61.56 (Ref. 11.4-19, 11.4-20). The expected annual volumes of solid radwaste and its classification to be shipped offsite are estimated in Table 11.4-3. The packaging and shipment of radioactive solid waste for disposal complies with 10 CFR 20 Appendix G (Ref. 11.4-21) and 49 CFR 173 Subpart I (Ref.11.4-8). Waste to be packaged is sampled and analyzed; the radioactivity level of the waste is also monitored during the filling operation to insure meeting disposal requirements for the licensed land disposal facility. Each container of processed waste is classified as Class A, B, or C waste using a site-specific 10 CFR 61 (Ref. 11.4-22) Waste Form, in compliance with the site-specific process control program.

Some of the dry active waste is only slightly contaminated and permits contact handling. The SWMS design does not include compaction equipment or drum dryer equipment but provides the flexibility for the site-specific utilities to add compaction equipment or to adopt contract services from specialized facilities. The COL applicant is required to provide information on the adoption of compaction equipment.

Storage for packaged radioactive wastes is provided in a shielded area. The storage area is conveniently located next to the truck bay at an elevation of 3’ 7” in the A/B. An overhead crane is provided to move the waste from the de-watering area to the storage

The spent resin transfer and dewatering operations are controlled from the radwaste control room in the A/B. The dewatering operation can also be controlled from a local control console located next to the dewatering cubicle. The spent resin filling and de-watering operations have level control setpoints. The ranges, setpoints, and references for these instruments are to be developed in the detailed design phase and are not provided in this DCD.

During the operation of the SWMS, the individual component vent is processed through the GWMS (e.g., SRST vent) or the HVAC system (e.g., high-integrity containers) so that neither dust nor contaminated air is released to the workspaces.

11.4.2.5 Operation and Personnel Doses

The SRSTs are located in individually shielded cubicles in the A/B. The SRST rooms are ~~equipped with a steel liner~~ epoxy coated up to the cubicle wall height equivalent to full tank volume to minimize the potential for cross contamination to the groundwater system in accordance with BTP 11-3 (Ref. 11.4-14) and 10 CFR 20.1406 (Ref. 11.4-16). The ~~steel lining~~epoxy coated approach is also beneficial for ease of decontamination and decommissioning. Normally, these cubicles are not occupied and the entrance is under administrative control and physical control with locked doors. Entrances are provided for inspection for ease of ingress and egress; therefore, minimizing stay time and radiation doses.

The de-watering operation is also performed in a shielded and ~~steel~~epoxy-lined cubicle. The fillhead is handled remotely and has motors to automatically dislodge from the high-integrity containers. This design minimizes contact handling after the high-integrity container is filled. Control of the de-watering operation is performed in a separately shielded cubicle and/or in the radwaste control room. This design approach minimizes personnel radiation doses and complies with the dose limits of RG 8.8 (Ref. 11.4-2) and 10 CFR 20 (Ref. 11.4-3).

Ventilation air is designed to flow from areas of low contamination to areas of high contamination. Ventilation air in the dewatering area is controlled and exhausted from the de-watering area to maintain air quality and keep airborne activity to a minimum.

Except for low activity waste, such as contaminated clothing and decontaminated component parts and/or broken tools, radioactive waste is processed and stored in shielded areas. Access to these areas is under physical control (i.e., locked door) and administrative control to minimize radiation doses. Plant operating procedures are implemented so that radiation doses are ALARA. (see Chapter 12.)

11.4.3 Radioactive Effluent Releases

11.4.3.1 Radioactive Effluent Monitoring

The SWMS does not have radioactive liquid effluent. De-watered effluent and drainage are collected and forwarded to the WHTs for processing in the LWMS. The equipment and storage area vents are processed by the A/B HVAC system for discharge. The vents are blended with other HVAC vents and the effluent gases from the GWMS and

During normal operation and maintenance, a small amount of oily waste and sludge is generated (Table 11.4-1). These wastes are collected into drums and sorbent is added as required to stabilize the liquid for transportation to an approved facility for treatment and disposal.

30 days of on-site storage space is provided for packaged waste. Wastes are packaged in high-integrity containers, shielded filter containers, 55-gallon (200-liter) drums, and boxes. Temporary storage is provided on-site in the A/B next to the truck bay as shown in Figures 11.5-2c. A movable wall is used to shield high-integrity containers located in the on-site storage area. This wall separates the low activity storage from the high-level storage area. There is a notch on the wall to allow high-integrity containers or drums to be moved between the high and low activity storage areas and into the truck bay area.

11.4.4 Component Description

A description of the SWMS components, including the design of the tanks, pumps, and de-watering mobile unit, is shown in Table 11.4-4. The capacities, materials of construction, and applicable codes are included.

11.4.4.1 Tanks

11.4.4.1.1 Spent Resin Storage Tank

Each of the two SRSTs is a cylindrical vertical tank with a hemispherical head and bottom and is sized to collect a volume of spent resin for one fuel cycle of operation (two years). The SRSTs are sized to accommodate normal operation and AOOs including refueling operation.

The tanks are stainless steel pressure vessels for nitrogen gas operation. Each tank is protected by a pressure relief valve and is vented through a breakpot from which the exhaust air is discharged to the vent header in the GWMS.

11.4.4.1.2 Breakpot Tank

~~Both SRST vents are routed to a breakpot located downstream of the SRST relief valve. The function of the stainless steel breakpot is to minimize the potential of water entering the vent header. The breakpot is equipped with a level alarm indication to transfer the water to the A/B sump and notify the operator of an abnormal condition.~~The breakpot tank is a cylindrical vertically mounted stainless steel vessel with hemispherical heads. The tank is constructed per standards and design parameters in Table11.4-4. Both SRST vents are routed to a breakpot tank located downstream of the SRST relief valve to minimize the potential for liquids and solids (resin) from entering the vent header.The breakpot tank is equipped with a level alarm indication to transfer the water to the A/B sump and notify the operator of an abnormal condition. The tank will be sized to minimize the possibility of an overflow condition of the spent resin storage tanks (SRSTs) causing fouling of the Gaseous Waste Management System.

11.4.4.2 Pumps

11.4.4.2.1 ~~Sump~~ Sludge Pump

A ~~sump~~ sludge pump is used to transfer sludge to drums. The pump is manually started and is turned off at the operator's discretion. The fill line is equipped with a level switch to turn the pump off when the drum is full.

11.4.4.2.2 De-watering Pump

A diaphragm pump is used for spent resin and spent carbon de-watering. This pump is connected to the fillhead on top of the high-integrity container and is used to pull vacuum and moisture out of the high-integrity container after it is filled with spent resin. This pump is controlled from a local panel and is turned off manually when the moisture has reached the specified limit.

11.4.4.3 Piping

Piping used for the hydraulic transport of slurries, such as ion exchange resins and sludge, is specifically designed for trouble-free operation. Pipe flow velocities are maintained in a turbulent flow regime appropriate for the slurry being transported (ion exchange resins or tank sludge). Appropriate valves and pipe fittings are used to maintain unhindered flow. An adequate water/solids ratio is maintained throughout the transfer. Slurry piping is provided with washing and flushing capability with sufficient water to flush the pipe after each use (e.g., at least two pipe volumes).

11.4.4.4 Venting and Relief Valve

The SRST is vented to the vent header via a breakpot. The SRST is equipped with a relief valve to protect the SRST from over-pressurization. The relief valve vents to the vent header. When sluicing resin to the high-integrity container, the SRST vent is closed to allow the SRST to be pressurized with nitrogen gas. In addition to the relief valve, the SRST is equipped with a pressure alarm to inform the operator of an abnormal condition.

11.4.4.5 Mobile De-watering System

The mobile de-watering subsystem includes modular skids that are designed to be readily mobile and flexible. This mobile de-watering subsystem is typically comprised of one high-integrity container, a de-watering fillhead station, a pump, a control console, and a CCTV to monitor tank level operation. A level instrument, temperature instrument, and a CCTV are provided as part of the fillhead assembly so that the high-integrity container is not over flowed. A configuration of the mobile de-watering system is depicted in Figure 11.4-2. ~~Adoption of a mobile de-watering system is the responsibility of the COL applicant.~~ Identification of mobile/portable SWMS connections that are considered non-radioactive but later may become radioactive through contact or contamination with radioactive systems and preparation of operating procedures are the responsibility of the COL applicant.

as required to provide operational information and performance assessment. Key system alarms, such as a high level in the SRST, are repeated in the MCR. A list of alarm instruments and location of readouts is presented in Table 11.4-6.

Instruments, including back flushing provisions, are located in low radiation areas when possible for accessibility and fulfillment of the ALARA provisions.

11.4.8 Combined License Information

The COL applicant is to provide the following, which apply on a plant-specific basis:

COL 11.4(1) *The current design meets the waste storage requirements in accordance with ANSI/ANS-55.1. When the COL applicant desires additional storage capability beyond that which is discussed in this Tier 2 document, the COL applicant will identify plant-specific needs for on-site waste storage and provide a discussion of on-site storage of low-level waste.*

COL 11.4(2) *Deleted*

COL 11.4(3) *The COL applicant is to prepare a plan for the process control program describing the process and effluent monitoring and sampling program. The plan should include the proposed implementation milestones.*

COL 11.4(4) *The COL applicant is responsible for the identification of mobile/portable SWMS connections that are considered non-radioactive but later may become radioactive through contact or contamination with radioactive systems (i.e., a non-radioactive system becomes contaminated due to leakage, valving errors, or other operating conditions in the radioactive systems). The COL applicant is to prepare a plan to develop and use operating procedures so that the guidance and information in Inspection and Enforcement (IE) Bulletin 80-10 (Ref. 11.4-29) is followed.*

COL 11.4(5) *The current design provides collection and packaging of ~~dry active wastes~~ potentially contaminated clothing for offsite shipment and/or disposal. Depending on site-specific requirements, the COL applicant can send the wastes ~~for~~to an offsite laundry facility processing and/or bring in a mobile compaction unit for volume reduction. The laundry services, including contracted services and/or a temporary mobile compaction subsystem, are ~~is a~~ COL items.*

COL 11.4(6) *The COL applicant is required to perform a site-specific cost benefit analysis to demonstrate compliance with the regulatory requirements.*

COL 11.4(7) *~~The COL applicant can adopt~~ The SWMS design does no include solid waste processing facility (e.g. de-watering system, compactor for reducing waste volume) ~~depending on site-specific requirements. These facilities are COL item.~~ but provides the flexibility for the site-specific utilities to add compaction equipment or to adopt contract services from specialized facilities. This is the responsibility of the COL applicant.*

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Table 11.4-4 Solid Waste Management System Component Data Summary

Component	Quantity	Standards	Type	capacity	Design Pressure(psi)	Design Temp (°F)	Normal Operating Pressure (psi)	Normal Operating Temp (°F)	Material
Spent Resin Storage Tank	2	ASME BPVC, Div 1 or Div 2	Cylindrical, Vertical	800 (ft^3)	150	150	slightly positive	Ambient	SS
Spent Filter Cask	1	Industrial Standard	Cylindrical	Not Applicable	Not Applicable	Not Applicable	Atm	Ambient	CS
Fillhead	1	Industrial Standard	Circular	Not Applicable	150	150	Atm	Ambient	SS
Breakpot Tank	1	ASME BPVC, Div 1 or Div 2	Cylindrical, Vertical	Approximately 4 (ft^3)	100	200	slightly positive	Ambient	SS
De-watering Vacuum Pump	1	API-610; API-674; API-675; ASME Code Section VIII, Div. 1 or Div. 2	Vacuum	45 (gpm)	Vacuum	150	Vacuum	Ambient	SS
Sludge Pump	1	API-610; API-674; API-675; ASME Code Section VIII, Div. 1 or Div. 2	Centrifugal/ Mechanical Seal	45 (gpm)	150	150	Atm	Ambient	SS

Table 11.4-5 Equipment Malfunction Analysis

Equipment Item	Malfunction	Result(s)	Alternate Action
High integrity container level instrument	Level instrument failure	Loss of automatic filling function. Instrumentation sensors can be isolated and repaired or replaced.	Visual inspection can be performed by the camera located on top of the high-integrity container to view high-integrity container level.
SWMS	Earthquake damage	Spent resin storage tank rupture; resin and fluid leakage.	Both resin and liquid are contained in the room. Dose consequences are within the design guidance of SRP 11.4 appendix 11.4-A.
Spent resin storage tank external valve leak	Fluid leaks out	Local contamination.	Isolate leak and use other SRST. Repair the leak. The floor is ~~steel~~ epoxy lined to contain the leak.
Plant makeup water inlet flow	Flow instrument failure	Instrumentation sensors can be isolated and repaired or replaced.	A controlotron can be installed outside the piping to measure the flow.
Spent resin tank level	Level instrument failure	Instrumentation sensors can be isolated and repaired or replaced.	Tygon tubing can be used to measure tank level or the loop seal to fill up the tank.
Spent resin tank pressure	Pressure instrument failure	Instrumentation sensors can be isolated and repaired or replaced.	A temporary gauge can be installed on the drainage line to measure tank pressure.
High integrity container temperature	Temperature instrument failure	Instrumentation sensors can be isolated and repaired.	Sluicing can be delayed until temperature instrument is fixed.
Crane failure and load drop high-integrity container or drum	Crane failure	Auxiliary building contamination.	The area is roped off temporarily until the area has been decontaminated .
Breakpot level	Level instrument failure	Instrumentation sensors can be isolated and repaired or replaced.	Sluicing can be delayed until level sensor is fixed.

Table 11.4-6 Instrument Indication and Alarm Information Page

Instrumentation	Readout Indication	Indication Location	Alarm	Alarm Location
Spent Resin Tanks				
PMW inlet flow	X	Radwaste Control Room	NA	NA
Spent Resin Tank Level	X	Radwaste Control Room	High/Low	Radwaste Control Room (High/Low) Main Control Room (High)
Spent Resin Tank Pressure	X	Radwaste Control Room	High	Radwaste Control Room
High Integrity Container				
High Integrity Container Level	X	Local Control Panel	High-High/High	Local Control Panel
High Integrity Container Temperature	X	Local Control Panel	High	Local Control Panel
Breakpot				
Breakpot Level	X	Radwaste Control Room	High	Radwaste Control Room
Spent Resin Transfer				
Turbidity Meter	X	Radwaste Control Room	NA	NA

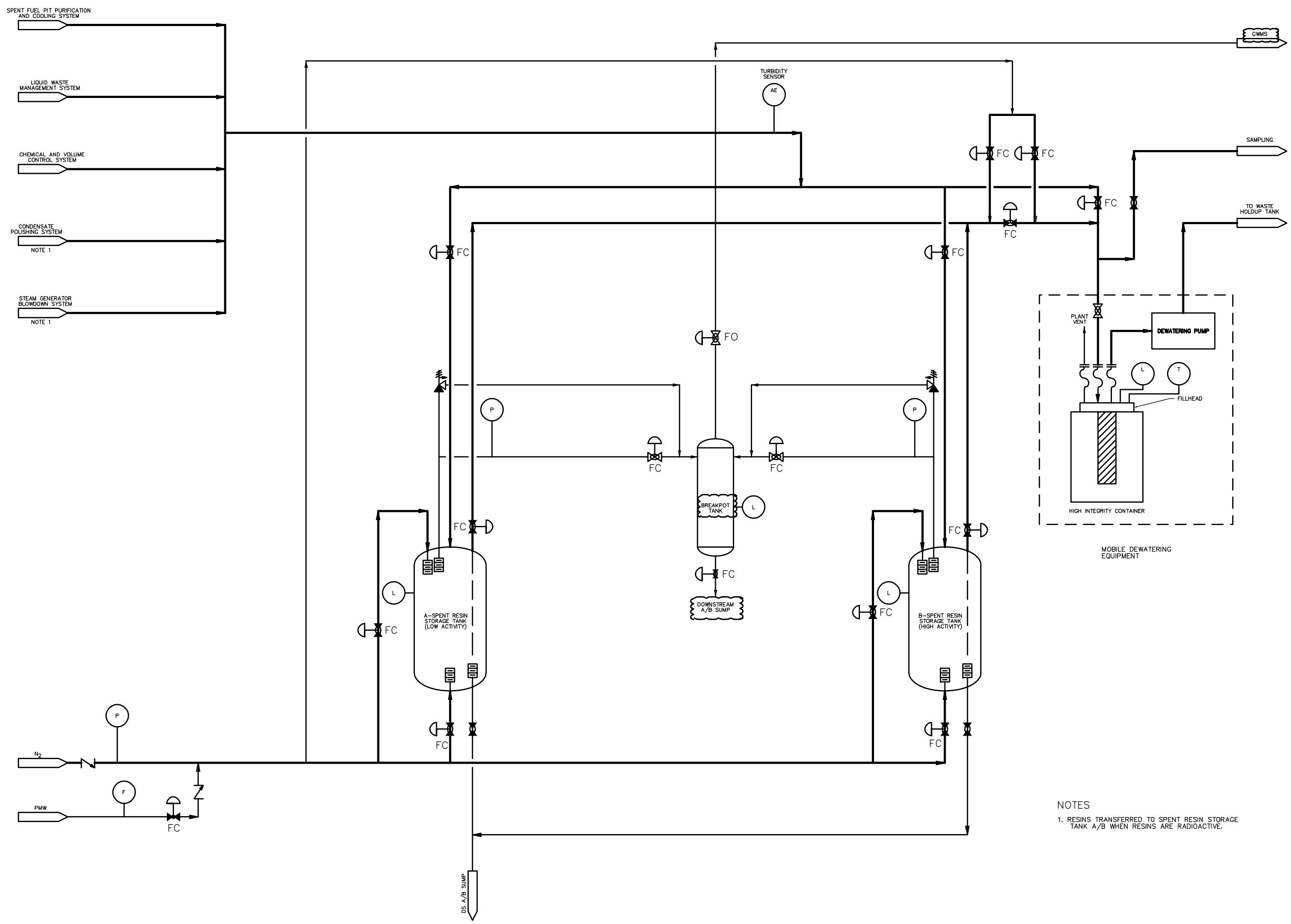

Figure 11.4-2 Process Flow Diagram of SWMS Spent Resin and Charcoal Handling Sub-System

11.5 Process Effluent Radiation Monitoring and Sampling Systems

The process effluent radiation monitoring and sampling system is designed to:

- Sample, measure, control, and record the radioactivity levels of selected process streams within the plant and effluent streams released into the environment
- Activate alarms and control releases of radioactivity
- Provide data to keep doses to workers ALARA
- Provide process data to support plant operation

The process and effluent radiological monitoring and sampling system is used to verify that releases to the environment are within the dose limit of 10 CFR 20.1301 (Ref. 11.5-1), 10 CFR 20.1302 (Ref. 11.5-2), and are within the numerical guidelines of 10 CFR 50, Appendix I (Ref. 11.5-3). The process and effluent radiological monitoring and sampling system is designed to perform its monitoring and recording functions during normal operation, AOOs, and under post-accident conditions, when and where required. Individual instruments of the process and effluent radiological monitoring and sampling system are designed to either monitor or collect samples from the gaseous and liquid streams at key process locations. All potential release points are either monitored or sampled in accordance with 10 CFR 50.34 (f)(2)(xvii) (Ref. 11.5-4). Data from these monitors is used to support the preparation of the radiological release reports required by 10 CFR 50.36a (Ref. 11.5-5).

Grab samples at key locations are taken for chemical and radiological analyses to confirm isotopic compositions and radiation levels. The grab sampling locations, methodology, analysis objectives, and frequencies are described in Chapter 9, Subsection 9.3.2.

Schematics of the monitors, including the sampling features, are presented in Figures 11.5-1a through 11.5-1j. Locations of the monitors are presented in Figures 11.5-2a through 11.5-~~2j~~2k.

The COL applicant is responsible for the additional site-specific aspects of the process and effluent monitoring and sampling system beyond the standard design, in accordance with RGs 1.21, 1.33 and 4.15 (Ref. 11.5-12, 11.5-17, 11.5-14).

11.5.1 Design Bases

11.5.1.1 Design Objective

The design objective of the process and effluent radiological monitoring and sampling system is to provide process data to support plant operation through monitoring, sampling, measuring, controlling, and recording of the radioactivity levels of selected process streams within the plant and effluent streams released into the environment. The system is also used to activate alarms. The process effluent radiation monitoring

- Gaseous process streams that are radioactive or have the potential of becoming radioactive from cross-contamination
- Liquid effluent streams that are radioactive or have the potential of becoming radioactive from cross-contamination
- Gaseous effluent streams that are radioactive or have the potential of becoming radioactive from cross-contamination

The radiation monitors and samplers are summarized in the following tables:

- Table 11.5-1 Process Gas and Particulate Monitors
- Table 11.5-2 Process Liquid Monitors
- Table 11.5-3 Effluent Gas Monitors
- Table 11.5-4 Effluent Liquid Monitors
- Table 11.5-5 Samplers

The process configurations for the radiation monitors are schematically presented in Figures 11.5-1a through 11.5-1j. The locations of the monitors are identified in the general arrangement drawings in Figures 11.5-2a through 11.5-~~2j~~2k. The locations of the sampling points and stations are discussed in Chapter 9, Subsection 9.3.2.

The types of monitors and their ranges are listed in Tables 11.5-1 through 11.5-4. The ranges are based on expected operating ranges for this design and industrial experience. The setpoints for the trips and alarms associated with the monitors are based on specific detail design.

Descriptions of provisions for purging grab sample lines, representative sampling, and frequency of sampling are described in Chapter 9, Subsection 9.3.2. Sampling analytical results are used to calibrate the monitoring instrumentation for normal operation and AOOs. Sampling analytical results are used to assess the plant status during post-accident events.

11.5.2.2 Process Gas and Particulate Monitors Component Description

11.5.2.2.1 Containment Radiation Monitors (RMS-RE-41 and RMS-RE-40)

The two containment radiation monitors are a beta (β)-scintillation (RMS-RE-41) and a gamma (γ) (RMS-RE-40) monitor. The detection ranges and other details are summarized in Table 11.5-1, item numbers 1 and 2, respectively. The process configuration for these monitors is shown in Figure 11.5-1a. These monitors are in the R/B as shown in Figure 11.5-2h.

Three radiation monitors comprise a monitoring set. One monitor is a radioactive gas detector (RMS-RE-84A/RMS-RE-84B), one monitor is an iodine detector (RMS-RE-85A/RMS-RE-85B), and one monitor is a particulate detector (RMS-RE-83A/RMS-RE-83B). There are two sets of monitors (RMS-RE-84A, RMS-RE-85A, and RMS-RE-83A; and RMS-RE-84B, RMS-RE-85B, and RMS-RE-83B). An air sample is continuously drawn into the chamber where the radiation levels are monitored: first the particulate, then the gas. A parallel sample is drawn into the iodine monitor. Both samples are returned back to the stream downstream of the sample point. Detection of radioactivity levels in the stream exceeding the predetermined setpoints automatically activates signals to start the main control room isolation, and activates an alarm in the MCR for operator actions.

Piping taps are provided for the purging and cleaning of the monitors. These monitor sets are safety-related and are connected to the emergency power supply system. The redundant monitor sets have independent backup power units.

The Main Control Room Outside Air Intake Radiation Monitors are part of the ESF Systems described in Section 7.3. The I&C design of the ESF Systems, including the Main Control Room Outside Air Intake Radiation Monitors, conform to the requirements of IEEE Standard 603-1991.

11.5.2.2.7 TSC Outside Air Intake Radiation Monitors (RMS-RE-100, RMS-RE-101 and RMS-RE-102)

The TSC outside air intake radiation monitors are β–scintillation (RMS-RE-101) and γ (RMS-RE-100, and RMS-RE-102). The detection ranges and other details are summarized in Table 11.5-1, item numbers 11, 12, and 13. The process configuration of the monitors is schematically presented in Figure 11.5-1e. The monitors are located in the A/B as shown in Figure 11.5-2g.

Three radiation monitors comprise a monitoring set. One monitor is a radioactive gas detector (RMS-RE-101), one monitor is an iodine detector (RMS-RE-102), and one monitor is a particulate detector (RMS-RE-100). An air sample is continuously drawn into the chamber where the radiation levels are monitored: first the particulate, then the gas. A parallel sample is drawn into the iodine monitor. Both samples are returned back to the stream downstream of the sample point. Detection of radioactivity levels in the stream exceeding the predetermined setpoints automatically activates signals to start technical support center isolation, and activates an alarm in the MCR for operator actions.

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

11.5.2.3.6 SG Blowdown Return Water Radiation Monitor (RMS-RE-36)

The SG blowdown return water radiation monitor is a γ monitor; the detection range and other details are summarized in Table 11.5-2, item number 19. The process configuration of the monitor is schematically presented in Figure 11.5-1d. The monitor is located in the R/B as shown in Figure 11.5-2~~c~~e. The monitor is located downstream of the SG demineralizer outlet.

This monitor measures the radiation level in the SG blowdown water after it is treated and before it is returned into the condensate storage tank. A sample from the SG blowdown mix bed demineralizers is sampled for radiation monitoring. Normally the treated SG blowdown water is slightly radioactive. In the event of significant primary-to-secondary system leakage due to a SG tube leak, it is possible that the SG blowdown liquid may be contaminated with radioactive material. Detection of radiation above a predetermined setpoint activates an alarm in the MCR for operator actions, including automatically closing of the valve to transfer the treated liquid to the condensate storage tank. Plant personnel are required to manually sample the SG blowdown water for analysis. If it is confirmed that the liquid is contaminated, the liquid is routed to the LWMS for processing.

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

11.5.2.4 Effluent Gaseous Monitors Component Description

11.5.2.4.1 Plant Vent Radiation Gas Monitors (RMS-RE-21A, RMS-RE-21B, RMS-RE-80A and RMS-RE-80B)

The plant vent radiation gas monitors are β (RMS-RE-21A, RMS-RE-21B) and β/γ (RMS-RE-80A, and RMS-RE-80B) monitors; the detection range and other details are summarized in Table 11.5-3, item numbers 20, 21, and 22. There are a total of four monitors: two for normal range (RMS-RE-21A/B), one for mid range (RMS-RE-80A), and one for high range (RMS-RE-80B). The process configuration of the monitors is schematically presented in Figure 11.5-1h. The monitors are located in the R/B as shown in Figure 11.5-2h.

The plant vent radiation gas monitor (RMS-RE-21A) measures the concentration of radioactive gases released through the vent stack during normal operation. The monitoring of this gaseous effluent stream assures that the radiological releases are kept ALARA. The normal range monitor (RMS-RE-21B) and the extended range monitors (RMS-RE-80A, and RMS-RE-80B) work together under post-accident conditions. Detection of radiation above a predetermined setpoint activates an alarm in the MCR for operator actions and also activates the closure of the discharge valve to isolate the GWMS discharge stream and containment ventilation isolation.

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

11.5.2.6 Reliability and Quality Assurance

The reliability of the process and effluent radiological monitoring and sampling system is based on providing redundant instruments, where required, and the use of manual sampling and analysis to verify the performance of the installed instruments.

To assure that quality assurance is maintained, only instruments designed and manufactured for the intended services, and instruments with industry-proven performances are used. Lessons learned regarding instrument reliability will be used in the final instrument selection. Safety-related monitoring instruments are qualified as described in Chapter 3, Section 3.11. Monitors that are not safety-related are designed in accordance with ANSI N42.18-2004 (Ref. 11.5-11), and are qualified in accordance with RG 1.143 Section IV (Ref. 11.5-16).

Calibration and inspection procedures are developed and put in place by the COL applicant in accordance with RG 1.33 (Ref. 11.5-17) and RG 4.15 (Ref. 11.5-14). Inspections are conducted daily on the process and effluent monitoring and sampling system through observance of the system channels. Periodically, the system is further checked during the course of reactor operation through the implementation of a check source. The detector response is compared to the instrument's background count rate to determine functionality. Calibration of monitors is conducted through the use of known radionuclide sources as documented by national standards. Maintenance is conducted routinely on the monitoring and sampling system, which is easily accessible, as is the accompanying power supply. Electronic and sampling components undergo a full servicing, periodically, as detailed in the operational instructions in order to maintain consistent operation.

The COL applicant is to develop procedures which are of inspection, decontamination, and replacement related to radiation monitoring instruments. The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and type of sampling media for site-specific matter.

11.5.2.7 Determination of Instrumentation Alarm Setpoints for Effluents

Alarm setpoints for effluent monitors follow site-specific requirements, such as release flow rates and associated conditions, operator response preferences, and detailed design radiation monitor selection. The alarm setpoints are developed during the detailed design stage by the COL applicant in the offsite dose calculation manual. Setpoint determination will need to follow the guidance of NUREG-1301 (Ref. 11.5-21), and NUREG-0133 (Ref. 11.5-18) so that the effluent releases to unrestricted areas do not exceed those given in 10 CFR 20, Appendix B, Table 2 (Ref. 11.5-19). Setpoint determinations will also consider local meteorological conditions.

11.5.2.8 Compliance with Effluent Release Requirements

The radiological monitoring and sampling systems are designed to assure compliance with the requirements of 10 CFR 20.1301 (Ref. 11.5-1), 10 CFR 20.1302 (Ref. 11.5-2), 10 CFR 20, Appendix B (Ref. 11.5-19), and 40 CFR 190 (Ref. 11.5-20), and the numerical design guidelines stated in 10 CFR 50, Appendix I (Ref. 11.5-3). The monitoring and sampling systems are designed to collect and analyze radiation readings to generate the annual radiological effluent release report required by 10 CFR 50.36a (Ref. 11.5-5). Site-specific procedures on equipment inspection, calibration, and maintenance, and by regulated recordkeeping are developed by the COL applicant to meet the requirements of 10 CFR 20.1301 (Ref. 11.5-1) and 10 CFR 20.1302 (Ref. 11.5-2), and meet the numerical guidelines stated in 10 CFR 50, Appendix I (Ref. 11.5-3), but will also be able to comply with 10 CFR 50.34a (Ref. 11.5-6) and keep releases ALARA.

The COL applicant is to develop procedures which are of inspection, decontamination, and replacement related to radiation monitoring instruments. The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and type of sampling media for site-specific matter.

11.5.2.9 Offsite Dose Calculation Manual

An offsite dose calculation manual that contains a description of the methodology and parameters used for calculation of offsite doses for the gaseous and liquid effluents will be prepared by the COL applicant. The manual will also contain the planned effluent discharge flow rates and addresses the numerical guidelines stated in 10 CFR 50, Appendix I (Ref. 11.5-3). The manual will be produced in accordance with the guidance of NUREG-1301 (Ref. 11.5-21), and NUREG-0133 (Ref.11.5-18), and with the guidance of RG 1.109 (Ref. 11.5-22), RG 1.111 (Ref. 11.5-23), or RG 1.113 (Ref. 11.5-24). The manual will include a discussion of how the NUREGs, RGs, or alternative methods are implemented.

The COL applicant is responsible for assuring the fulfillment of the guidelines issued in 10 CFR 50, Appendix I (Ref. 11.5-3) regarding the offsite doses released through gaseous and liquid effluent streams.

11.5.2.10 Radiological Environmental Monitoring Program

The COL applicant is to develop a radiological and environmental monitoring program taking into consideration local land use census data in identifying all potential radiation dose pathways. The program will take into account associated radioactive materials present in liquid and gaseous effluents and direct external radiation from SSCs. The COL applicant is to follow the guidance outlined in NUREG-1301 (Ref. 11.5-21), and NUREG-0133 (Ref. 11.5-18) when developing the radiological environmental monitoring program.

11.5.2.11 Site –Specific Cost-Benefit Analysis

RG1.110 provides compliance with 10 CFR 50, Appendix I (Ref. 11.5-3) numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal operations, including AOOs. The cost-benefit numerical analysis as required by 10 CFR 50, Appendix I, Section II, Paragraph D (Ref. 11.5-3)

11.5-23 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.

11.5-24 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.

11.5-25 Deleted

11.5-26 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. Regulatory Guide 8.8.

11.5-27 Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable. Regulatory Guide 8.10, May 1977.

11.5-28 Occupational dose limits for adults. NRC Regulations Title 10 Code of Federal Regulations, 10 CFR Part 20.1201.

11.5-29 Compliance with requirements for summation of external and internal doses. NRC Regulations Title 10 Code of Federal Regulations, 10 CFR Part 20.1202.

11.5-30 NEI 07-09, "Generic Template Guidance for Offsite Dose Calculation Manual Program Description," Revision 1.

11.5-31 Reactor Coolant Pressure Boundary Leakage Detection Systems. Regulatory Guide 1.45, May 1973.

11.5-32 U.S. Nuclear Regulatory Commission, Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses, NUREG-0718.

11.5-33 Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors. Regulatory Guide 1.110, March 1976.

11.5-34 IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, IEEE Std 603-1991.

Table 11.5-2 Process Liquid Monitors

Item No.	Monitor Number	Service	Type	Range $\mu Ci/cm^3$	Calibration Isotopes	Check Source	Safety-Related	Control Function	Quantity	Schematic Number	GA Drawing Number
14	RMS-RE-56A RMS-RE-56B	CCW radiation *The concentration of radioactive material in the CCW*	γ	1E-6 to 1E-2	Cs-137	Yes	No	Termination	2	11.5 – 1f	11.5 – 2a
15	RMS-RE-57	Auxiliary steam condensate water radiation *The concentration of radioactive material in the auxiliary steam condensate water*	γ	1E-6 to 1E-2	Cs-137	Yes	No	Termination	1	11.5 – 1f	11.5 – 2a
16	RMS-RE-70	Primary coolant radiation *The concentration of radioactive material in the CVCS line*	γ	1E-4 R/h to 10 R/h	Cs-137	Yes	No	No	1	11.5 – 1c	11.5 – 2e
17	RMS-RE-58	Turbine building floor drain radiation *The concentration of radioactive material discharged from the Turbine Building*	γ	1E-7 to 1E-2	Cs-137	Yes	No	Diverse	1	11.5 – 1g	11.5－2c
18	RMS-RE-55	SG blowdown water radiation *The concentration of radioactive material in the SG blowdown water*	γ	1E-7 to 1E-3	Cs-137	Yes	No	Termination	1	11.5 - 1d	11.5 – 2h
19	RMS-RE-36	SG Blowdown return water radiation *The concentration of radioactive material in the SG return blowdown water*	γ	1E-7 to 1E-3	Cs-137	Yes	No	Diverse	1	11.5 - 1d	11.5 - 2~~c~~e

Chapter 12

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Table 12.2-1 Radiation Sources Parameters (Sheet 1 of 46)

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

* The regenerative heat exchanger consists of three shells.

Table 12.2-1 Radiation Sources Parameters
(Sheet 2 of 46)

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

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Table 12.2-1 Radiation Sources Parameters
(Sheet 3of 46)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Auxiliary Building							
Mix bed demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Cation-bed demineralizer*	15.9	65.6	Homogeneous	Water	62.4	ignored	1
Deborating demineralizer*	23.7	68.9	Homogenous	Water	62.4	ignored	2
Holdup tank Liquid Phase Vapor Phase	147.6	410.0 229.7 180.3	 Homogenous Homogenous	 Water Air	 62.4 7.6E-02	ignored	3
B.A. evaporator feed demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	1
Spent fuel pit demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Steam generator blowdown demineralizer*	44.3	63.4	Homogeneous	Water	62.4	ignored	4
Waste holdup tank	128.0	138.6	Homogeneous	Water	62.4	ignored	4
Waste demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	4
Charcoal bed Charcoal Phase Vapor Phase	23.7	126.0 68.8 57.2	 Homogenous Homogenous	 Charcoal Air	 34.4 7.6E-02	ignored	4
Waste gas surge tank	74.8	167.0	Homogeneous	Air	7.6E-02	1.0	4
Spent resin storage tank	59.1	131.2	Homogeneous	Water	62.4	ignored	2
B.A. evaporator	26.9	188.5	Homogeneous	Water	62.4	ignored	1
B.A. evaporator vent condenser	5.0	78.1	Homogeneous	Air	7.6E-02	ignored	1
Boric acid tank	118.1	361.5	Homogeneous	Water	62.4	ignored	2

* Parameters from the US-APWR demineralizers are tabulated in Table 12.2-73.

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Table 12.2-1 Radiation Sources Parameters (Sheet 4~~3~~of ~~4~~6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Plant Yard Area (Outside the Power Block)							
Refueling water storage auxiliary tank	236.2	536.6	Homogeneous	Water	62.4	ignored	1
Primary makeup water tank	183.1	316.9	Homogeneous	Water	62.4	ignored	2

Table 12.2-1 Radiation Sources Parameters
(Sheet ~~4~~ 5 of ~~4~~6)

Components	Assumed Shielding Sources							
	Source Approximate Geometry as rectangular parallelepiped Volume			Source Characteristics				Quantity
	Width (in.)	Depth (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Outside the Containment Vessel (Reactor Building)								
Spent fuel pit heat exchanger *	29.5	47.7	88.6	Homogeneous	Source Water 25.5 wt%+ Cooling water 25.5 wt%+ Steel 49 wt%	109.3	ignored	2

* Spent fuel pit exchanger is a plate heat exchanger.

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

Components	Assumed Shielding Sources								
	Source Approximate Geometry as Annular Cylinder Volume			Source Characteristics					Quantity
	Outer Radius (in.)	Inner Radius (in.)	Height (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	Designed Upper limit dose rate (mrem/h)	
Auxiliary Building									
Reactor coolant Filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	2
Mixed bed demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	3
B.A. evaporator feed demineralizer	3.4	2.7	19.7	Homogeneous	Water	62.4	Ignored	100	1
Boric acid filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	1
Seal water injection filter	1.7	1.6	19.9	Homogeneous	Water	62.4	Ignored	100	2
Waste effluent inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SFP filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SG blowdown demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	2

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Table 12.2-62 Chemical and Volume Control System Radiation Sources Deborating Demineralizer Activity (70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	1.5E-02
Br-83	2.6E-02
Br-84	3.1E-03
I-130	1.4E-01
I-131	3.4E+00
I-132	2.1E+00
I-133	4.3E+00
I-134	7.4E-02
I-135	1.5E+00

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Table 12.2-63 Chemical and Volume Control System Radiation Sources Deborating Demineralizer Source Strength (70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	1.3E+01
0.03	2.6E+02
0.08	2.6E+02
0.1	9.0E-01
0.15	6.7E+01
0.2	3.7E+02
0.3	4.0E+03
0.4	4.4E+04
0.5	8.4E+04
0.6	7.0E+04
0.8	8.1E+04
1.0	4.9E+04
1.5	7.1E+04
2.0	1.9E+04
3.0	1.6E+02
4.0	3.3E+01

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Table 12.2-64 Chemical and Volume Control System Radiation Sources
B.A. Evaporator Feed Demineralizer Activity (70 ft^3 of Resin)

Nuclide	Activity (µCi/cm³)	Nuclide	Activity (µCi/cm³)
Br-82	1.2E-02	Te-129m	2.0E-01
Br-83	3.0E-03	Te-129	1.5E-04
Br-84	8.5E-05	Te-131m	3.7E-02
Rb-86	2.9E+00	Te-131	2.1E-05
Rb-88	1.1E+00	Te-132	1.1E+00
Rb-89	1.3E-03	Te-133m	2.0E-04
Sr-89	9.3E-02	Te-134	2.1E-04
Sr-90	5.9E-03	I-130	1.5E+00
Sr-91	7.9E-04	I-131	1.3E+01
Sr-92	6.7E-05	I-132	1.3E+00
Y-90	1.3E-01	I-133	2.2E+00
Y-91m	5.3E-04	I-134	3.8E-03
Y-91	1.3E-02	I-135	3.4E-01
Y-92	2.8E-04	Cs-132	5.5E-01
Y-93	1.6E-04	Cs-134	5.0E+02
Zr-95	1.5E-02	Cs-135m	1.4E-03
Nb-95	4.0E-02	Cs-136	6.3E+01
Mo-99	2.5E+00	Cs-137	2.9E+02
Mo-101	1.7E-05	Cs-138	9.1E-02
Tc-99m	3.5E+00	Ba-137m	2.5E+02
Ru-103	1.1E-02	Ba-140	5.2E-02
Ru-106	5.0E-03	La-140	1.0E-01
Ag-110m	4.5E-05	Ce-141	1.2E-02
Te-125m	1.7E-02	Ce-143	8.0E-04
Te-127m	7.5E-02	Ce-144	1.2E-02
		Pr-144	1.7E-02
		Pm-147	1.4E-03
		Eu-154	1.3E-04
		Na-24	4.2E-02
		Cr-51	1.2E-01
		Mn-54	1.2E-01
		Mn-56	1.1E-02
		Fe-55	1.2E-01
		Fe-59	1.6E-02
		Co-58	2.5E-01
		Co-60	4.2E-02
		Zn-65	3.3E-02

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Table 12.2-65 Chemical and Volume Control System Radiation Sources B.A. Evaporator Feed Demineralizer Source Strength (70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	2.6E+03
0.02	2.7E+02
0.03	3.3E+04
0.04	9.8E+03
0.05	3.0E+02
0.06	1.8E+04
0.08	1.3E+04
0.1	1.1E+03
0.15	6.3E+04
0.2	7.7E+04
0.3	4.3E+05
0.4	1.6E+05
0.5	2.2E+05
0.6	1.9E+07
0.8	1.6E+07
1.0	2.9E+06
1.5	8.9E+05
2.0	2.5E+04
3.0	9.2E+03
4.0	4.3E+01
5.0	3.0E+02

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Table 12.2-73 Parameters for the US-APWR demineralizers

Component	Parameters				Note
	DF	Flow rate	Term of Service	inlet flow stream activity concentration	
Mixed bed demineralizer	Kr, Xe=1, Br, I=100, Cs, Rb=2 , Others=50	180 gpm	731 days	Table 12.2-13	These values in the left columns are listed in Table 11.1-1.
Cation-bed demineralizer	Kr, Xe=1, Br, I=1, Cs, Rb=10 , Others=10	18 gpm	731 days	Table 12.2-74	
Deborating demineralizer	Anion=100, Cs, Rb=1, Others=1	180 gpm	22 hours	Table 12.2-74	
B.A. evaporator feed demineralizer	Anion=10, Cs, Rb=2, Others=10	30 gpm	780 hours	Table 12.2-75	
Waste demineralizer (Anion-bed)	I=100, Cs, Rb=1, Others=1	90 gpm	280 hours	Table 12.2-37	
Waste demineralizer (Cation-bed)	I=1, Cs, Rb=10, Others=10	90 gpm	280 hours	Table 12.2-76	
Waste demineralizer (Mixed bed) In case of treating HT system	Kr, Xe=1, I=5, Cs, Rb=1 , Others=10	30 gpm	780 hours	Table 12.2-77	Parameters used when treating distilled water in the boron recycle system
Waste demineralizer (Mixed bed) In case of treating WHT system	Kr, Xe=1, I=100, Cs, Rb=2 , Others=100	90 gpm	280 hours	Table 12.2-78	Parameters used when treating waste liquid in the waste liquid storage tank
Spent fuel pit demineralizer	Kr, Xe=1, Br, I=100 Cs, Rb=2 , Others=100	265 gpm	731 days	Table 12.2-34	
SG Blowdown demineralizer	Br, I=100, Cs, Rb=100, Others=1000	1.554E+05 lb/hr	731 days	Table 11.1-5	

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Table 12.2-74 Inlet Flow Stream Activity of Cation-bed demineralizer and Deborating demineralizer

Nuclide	Activity (µCi/cm^3)	Nuclide	Activity (µCi/cm^3)
Kr-83m	5.5E-01	Te-129m	1.2E-04
Kr-85m	1.8E+00	Te-129	1.5E-04
Kr-85	9.2E+01	Te-131m	3.1E-04
Kr-87	1.2E+00	Te-131	1.7E-04
Kr-88	3.3E+00	Te-132	3.4E-03
Xe-131m	4.1E+00	Te-133m	3.2E-04
Xe-133m	4.2E+00	Te-134	5.8E-04
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	7.9E+00	I-131	1.6E-02
Xe-135	1.2E+01	I-132	6.5E-02
Xe-138	6.7E-01	I-133	2.7E-02
		I-134	6.1E-03
Br-82	8.6E-05	I-135	1.8E-02
Br-83	7.8E-04	Cs-132	4.1E-04
Br-84	4.2E-04	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	4.5E-03
Rb-88	2.1E+00	Cs-136	1.0E-01
Rb-89	4.9E-02	Cs-137	2.2E-01
Sr-89	3.8E-05	Cs-138	4.9E-01
Sr-90	2.4E-06	Ba-137m	1.1E+03
Sr-91	2.5E-05	Ba-140	4.6E-05
Sr-92	1.4E-05	La-140	3.5E-04
Y-90	4.4E-04	Ce-141	7.1E-06
Y-91m	1.8E-04	Ce-143	6.0E-06
Y-91	6.1E-06	Ce-144	5.3E-06
Y-92	2.2E-05	Pr-144	1.0E-01
Y-93	4.8E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.9E-03	Na-24	7.7E-04
Mo-101	3.9E-04	Cr-51	7.5E-05
Tc-99m	8.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	2.6E-03
Ru-106	2.1E-06	Fe-55	5.0E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.7E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

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Table 12.2-75 Inlet Flow Stream Activity of B.A. evaporator feed demineralizer

Nuclide	Activity (µCi/cm^3)	Nuclide	Activity (µCi/cm^3)
Kr-83m	1.6E-01	Te-129m	1.2E-04
Kr-85m	9.7E-01	Te-129	2.8E-05
Kr-85	9.2E+01	Te-131m	2.8E-04
Kr-87	2.4E-01	Te-131	1.1E-05
Kr-88	1.4E+00	Te-132	3.3E-03
Xe-131m	4.1E+00	Te-133m	4.8E-05
Xe-133m	4.0E+00	Te-134	6.6E-05
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	3.3E-01	I-131	1.6E-02
Xe-135	8.4E+00	I-132	2.5E-02
Xe-138	2.6E-02	I-133	2.4E-02
		I-134	9.3E-04
Br-82	7.9E-05	I-135	1.2E-02
Br-83	2.8E-04	Cs-132	4.1E-04
Br-84	3.6E-05	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	6.4E-04
Rb-88	1.4E+00	Cs-136	9.9E-02
Rb-89	2.0E-03	Cs-137	2.2E-01
Sr-89	4.8E-05	Cs-138	6.9E-02
Sr-90	2.4E-06	Ba-137m	8.1E+00
Sr-91	1.9E-05	Ba-140	4.6E-05
Sr-92	5.6E-06	La-140	3.3E-04
Y-90	4.2E-04	Ce-141	7.1E-06
Y-91m	3.4E-05	Ce-143	5.5E-06
Y-91	6.2E-06	Ce-144	5.3E-06
Y-92	1.4E-05	Pr-144	4.8E-03
Y-93	3.6E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.5E-03	Na-24	6.3E-04
Mo-101	1.6E-05	Cr-51	7.5E-05
Tc-99m	5.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	9.8E-04
Ru-106	2.1E-06	Fe-55	4.9E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.6E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

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Table 12.2-76 Inlet Flow Stream Activity of Waste Demineralizer (Cation Bed)

Nuclide	Activity (µCi/cm^3)	Nuclide	Activity (µCi/cm^3)
Xe-131m	3.1E-03	Te-129m	2.5E-03
Xe-133m	1.2E-02	Te-129	2.0E-03
Xe-133	1.8E-01	Te-131m	6.3E-03
Xe-135m	2.5E+00	Te-131	2.2E-03
Xe-135	4.6E-01	Te-132	7.0E-02
		Te-133m	4.3E-03
Br-82	3.5E-03	Te-134	7.6E-03
Br-83	2.4E-02	I-130	2.7E-04
Br-84	1.1E-02	I-131	6.7E-03
Rb-86	1.1E-02	I-132	2.9E-03
Rb-88	1.4E+00	I-133	1.1E-02
Rb-89	2.5E-02	I-134	1.5E-03
Sr-89	8.3E-04	I-135	6.4E-03
Sr-90	5.4E-05	Cs-132	2.2E-03
Sr-91	4.7E-04	Cs-134	2.0E+00
Sr-92	2.2E-04	Cs-135m	2.4E-03
Y-90	1.8E-04	Cs-136	2.5E-01
Y-91m	2.7E-04	Cs-137	1.2E+00
Y-91	1.3E-04	Cs-138	2.6E-01
Y-92	2.1E-04	Ba-137m	8.0E+00
Y-93	9.0E-05	Ba-140	9.8E-04
Zr-95	1.6E-04	La-140	4.2E-04
Nb-95	1.8E-04	Ce-141	1.5E-04
Mo-99	1.8E-01	Ce-143	1.2E-04
Mo-101	5.0E-03	Ce-144	1.2E-04
Tc-99m	1.1E-01	Pr-144	2.9E-03
Ru-103	1.3E-04	Pm-147	1.3E-05
Ru-106	4.7E-05	Eu-154	1.2E-06
Ag-110m	4.3E-07		
Te-125m	1.9E-04	Na-24	1.5E-02
Te-127m	7.5E-04	Cr-51	1.6E-03
		Mn-54	1.1E-03
		Mn-56	4.0E-02
		Fe-55	1.1E-03
		Fe-59	1.9E-04
		Co-58	2.6E-03
		Co-60	3.9E-04
		Zn-65	3.2E-04

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Table 12.2-77 Inlet Flow Stream Activity of Waste Demineralizer (Mixed bed)*

Nuclide	Activity (µCi/cm^3)	Nuclide	Activity (µCi/cm^3)
Kr-83m	1.6E-04	Te-129m	1.2E-08
Kr-85m	9.7E-04	Te-129	2.9E-09
Kr-85	9.2E-02	Te-131m	2.8E-08
Kr-87	2.4E-04	Te-131	1.2E-09
Kr-88	1.4E-03	Te-132	3.3E-07
Xe-131m	4.1E-03	Te-133m	4.8E-09
Xe-133m	4.0E-03	Te-134	6.6E-09
Xe-133	3.1E-01	I-130	6.2E-07
Xe-135m	3.7E-04	I-131	1.6E-05
Xe-135	8.4E-03	I-132	1.2E-04
Xe-138	2.6E-05	I-133	2.4E-05
		I-134	9.8E-07
Br-82	7.9E-09	I-135	1.2E-05
Br-83	2.8E-08	Cs-132	2.1E-07
Br-84	3.6E-09	Cs-134	1.9E-04
Rb-86	1.9E-06	Cs-135m	3.2E-07
Rb-88	7.2E-04	Cs-136	5.0E-05
Rb-89	1.0E-06	Cs-137	1.1E-04
Sr-89	4.8E-09	Cs-138	3.4E-05
Sr-90	2.4E-10	Ba-137m	1.3E-01
Sr-91	1.9E-09	Ba-140	4.6E-09
Sr-92	5.6E-10	La-140	5.9E-08
Y-90	4.4E-08	Ce-141	7.1E-10
Y-91m	1.4E-08	Ce-143	5.5E-10
Y-91	6.3E-10	Ce-144	5.3E-10
Y-92	1.8E-09	Pr-144	1.4E-06
Y-93	3.6E-10	Pm-147	6.0E-11
Zr-95	7.3E-10	Eu-154	5.6E-12
Nb-95	2.4E-09		
Mo-99	8.5E-07	Na-24	6.3E-08
Mo-101	1.6E-09	Cr-51	7.5E-09
Tc-99m	1.3E-05	Mn-54	5.1E-09
Ru-103	6.0E-10	Mn-56	9.8E-08
Ru-106	2.1E-10	Fe-55	4.9E-09
Ag-110m	1.9E-12	Fe-59	8.7E-10
Te-125m	8.6E-10	Co-58	1.2E-08
Te-127m	3.4E-09	Co-60	1.8E-09
		Zn-65	1.4E-09

*: These activities are used when this demineralizer processes the distilled water from the boron recycle system.

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Table 12.2-78 Inlet Flow Stream Activity of Waste Demineralizer (Mixed bed)*

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	2.8E-02	Te-129m	2.5E-04
Xe-131m	3.1E-03	Te-129	2.0E-04
Xe-133m	1.2E-02	Te-131m	6.3E-04
Xe-133	1.8E-01	Te-131	2.2E-04
Xe-135m	2.5E+00	Te-132	7.0E-03
Xe-135	4.6E-01	Te-133m	4.3E-04
		Te-134	7.6E-04
Br-82	3.5E-04	I-130	2.7E-04
Br-83	2.4E-03	I-131	6.7E-03
Br-84	1.1E-03	I-132	2.0E+00
Rb-86	1.1E-03	I-133	1.1E-02
Rb-88	1.4E-01	I-134	7.0E-03
Rb-89	2.5E-03	I-135	6.4E-03
Sr-89	8.3E-05	Cs-132	2.2E-04
Sr-90	5.4E-06	Cs-134	2.0E-01
Sr-91	4.7E-05	Cs-135m	2.4E-04
Sr-92	2.2E-05	Cs-136	2.5E-02
Y-90	3.3E-05	Cs-137	1.2E-01
Y-91m	3.1E-04	Cs-138	2.6E-02
Y-91	1.3E-05	Ba-137m	4.6E+02
Y-92	3.7E-05	Ba-140	9.8E-05
Y-93	9.0E-06	La-140	3.6E-04
Zr-95	1.6E-05	Ce-141	1.5E-05
Nb-95	2.1E-05	Ce-143	1.2E-05
Mo-99	1.8E-02	Ce-144	1.2E-05
Mo-101	5.0E-04	Pr-144	7.3E-03
Tc-99m	1.6E-01	Pm-147	1.3E-06
Ru-103	1.3E-05	Eu-154	1.2E-07
Ru-106	4.7E-06		
Ag-110m	4.3E-08	Na-24	1.5E-03
Te-125m	1.9E-05	Cr-51	1.6E-04
Te-127m	7.5E-05	Mn-54	1.1E-04
		Mn-56	4.0E-03
		Fe-55	1.1E-04
		Fe-59	1.9E-05
		Co-58	2.6E-04
		Co-60	3.9E-05
		Zn-65	3.2E-05

*: These activities are used when this demineralizer processes the liquid effluent from the waste holdup tank.

Chapter 14

Table 14.2-1 Comprehensive Listing of Tests (Sheet 3 of 5)

Section	Test
14.2.12.1.80	Liquid Waste Management System Preoperational Test
14.2.12.1.81	Gaseous Waste Management System Preoperational Test
14.2.12.1.82	Solid Waste Management System Preoperational Test
14.2.12.1.83	Steam Generator Blowdown System Preoperational Test
14.2.12.1.84	Sampling System Preoperational Test
14.2.12.1.85	Spent Fuel Pit Cooling and Purification System (SFPCS) Preoperational Test
14.2.12.1.86	Fuel Handling System Preoperational Test
14.2.12.1.87	Component Cooling Water System Preoperational Test
14.2.12.1.88	Turbine Component Cooling Water System Preoperational Test
14.2.12.1.89	Secondary Side Chemical Injection System Preoperational Test
14.2.12.1.90	Fire Protection System Preoperational Test
14.2.12.1.91	Instrument Air System Preoperational Test
14.2.12.1.92	Station Service Air System Preoperational Test
14.2.12.1.93	Boron Recycle System Preoperational Test
14.2.12.1.94	Offsite Communication System Preoperational Test
14.2.12.1.95	Inplant Communication System Preoperational Test
14.2.12.1.96	Safeguard Component Area Heating, Ventilation, and Air Conditioning (HVAC) System Preoperational Test
14.2.12.1.97	Emergency Feedwater Pump Area HVAC System Preoperational Test
14.2.12.1.98	Class 1E Electrical Room HVAC System Preoperational Test
14.2.12.1.99	Auxiliary Building HVAC System Preoperational Test
14.2.12.1.100	Main Steam/Feedwater Piping Area HVAC System Preoperational Test
14.2.12.1.101	Main Control Room (MCR) HVAC System Preoperational Test (including MCR Habitability)
14.2.12.1.102	Non-Class 1E Electrical Room HVAC System Preoperational Test
14.2.12.1.103	Technical Support Center HVAC System Preoperational Test
14.2.12.1.104	Non-Essential Chilled Water System Preoperational Test
14.2.12.1.105	Vessel Servicing Preoperational Test
14.2.12.1.106	Safety-Related Component Area HVAC System Preoperational Test
14.2.12.1.107	Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test
14.2.12.1.108	Non-Essential Service Water (non-ESW) System Preoperational Test
14.2.12.1.109	Condensate Storage Facilities System Preoperational Test
14.2.12.1.110	Turbine Building Area Ventilation System (General Mechanical Area) Preoperational Test
14.2.12.1.111	Turbine Building Area Ventilation System (Electric Equipment Area) Preoperational Test
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
14.2.12.1.116	Equipment and Floor Drainage System Preoperational Test
14.2.12.1.117	Compressed Gas System Preoperational Test

2. To demonstrate CS/RHRS pump and RHRS performance during discharge to the reactor coolant cold legs and to the minimum flow line and to verify the head/flow characteristics of each installed pump.

3. To demonstrate the RHRS operation during the RCS heatup with letdown through RHRS.

4. To demonstrate RHRS operation during RCS cooldown and reactor coolant cooling by only two of four subsystems.

5. To demonstrate proper operation of the RHRS during low RCS water level (e.g., mid-nozzle level) and to verify sufficient margins exist to prevent vortexing or air entrainment in the suction lines.

6. To provide the RHRS relief valve in order to protect low temperature overpressure for the RCS.

7. To demonstrate proper operation of the SFP gravity drain injection to the RCS.

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 system are available.

5. Required system flushing/cleaning is completed.

6. Required electrical power supplies and control circuits are energized and operational.

7. The CCW system is available to supply water to the CS/RHRS heat exchangers, pump seal coolers, and CS/RHRS pump motors.

C. Test Method

1. System component control and interlock circuits and alarms are verified, including the operation of the CS/RHRS pumps and RHRS valves.

2. CS/RHRS pump and RHRS performance characteristics are verified during RCS circulation.

3. RHRS operation is verified during RCS heatup and cooldown in conjunction with the hot functional test. This includes operation of reactor coolant cooling with only two of four subsystems.

4. Operation of the RHRS during RCS mid-loop hot leg water level is verified.

5. Operation of the SFP gravity injection to the RCS during mid-loop operation is verified.

D. Acceptance Criteria

1. RHRS components respond properly to normal control and interlock signals (see Subsection 5.4.7).

2. CS/RHRS pump and RHRS performance characteristics are within design specifications.

3. RHRS functions as designed during RCS heatup and cooldown.

4. Reactor coolant temperature can be cooled down with only two of four subsystems.

5. The RHRS functions as designed during RCS mid-loop hot leg water level.

6. The RHRS relief valve operation to provide low temperature overpressure protection for RCS is verified by in-service testing specified in subsection 3.9.6.

7. Indications and alarms operate as described in Subsection 5.4.7.2.5.

8. The SFP water can be injected to the RCS by gravity during mid-loop operation.

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

A. Objectives

1. To demonstrate acceptable closing times of the MSIVs and the MSIV bypass valves.

2. To demonstrate acceptable closing times for the MFIVs.

3. To demonstrate failure position of MSIVs and MFIVs upon loss of valve motive force.

4. To demonstrate that the main steam check valve prevents blowdown by steam backflow for intact steam generator in the event of breaking the upstream of main steam check valve.

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 system are available.

3. The test source gas is routed through the charcoal beds to verify performance.

D. Acceptance Criteria

1. The waste gas compressors, waste gas dryer, and waste gas system components respond to normal control, interlock, and alarm signals.

2. The waste gas compressors, waste gas dryer, and waste gas system are operable and their controls operate as described in Section 11.3.

3. The charcoal beds perform as designed (see Section 11.3).

14.2.12.1.82 Solid Waste Management System Preoperational Test

A. Objective

1. To demonstrate the operation and verify the operating characteristics of the solid waste management system, valves, and spent resin storage tanks.

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.

C. Test Method

1. Verify manual and automatic system controls, interlocks, alarms and indications.

2. Demonstrate the ability of the spent resin storage tanks (SRSTs) to receive spent resin from the LWMS, CVCS, SFPCS, SG blowdown system, and the condensate polisher ion exchange columns.

3. Demonstrate the ability of the SWMS to handle the following waste types, consistent with system operations described in Subsection 11.4.2: dry active waste, spent filter elements, spent resin, spent activated carbon, and oil and sludge.

4. The test source gas is routed through the SWMS to verify performance.

D. Acceptance Criterion

1. The operation of the system meets design specifications (Section 11.4) including the capability to assure that the volume of free liquids in packaged wastes is within acceptable limits.

2. The nitrogen supply gas demonstrates conformance with design flows and process capabilities.

14.2.12.1.83 Steam Generator Blowdown System Preoperational Test

A. Objectives

1. To demonstrate that the SG blowdown system (SGBDS) accepts water from each SG blowdown line, processes the blowdown as required, and delivers the processed water to the condensate system.

2. To demonstrate the capability of the SG blowdown sampling system to collect blowdown liquid sample from each SG and the operation of the ~~SGBDS sample monitoring system~~SG blowdown sampling system including monitors, system valve and control circuits.

3. To demonstrate of the performance of laboratory equipment.

B. Prerequisites

1. Required construction testing is completed.

2. Component testing and instrument calibration is completed.

3. Test instrumentation is available and calibrated.

4. Required support systems are available.

5. CCW is available for cooling the sample stream and hot functional test is in progress.

6. The SGBDS to be sampled is at normal pressure and temperature.

7. The condenser or waste water system (WWS) or LWMS is available to receive discharge from the SG blowdown sampling system.

C. Test Method

1. Verify manual and automatic system controls.

2. Verify flowrates and temperatures.

3. Verify indications (flow, temperature and status), and alarms.

4. Verify system isolation using simulated signals.

5. Verify flowpaths.

6. Verify the flow rate and temperature during SG blowdown liquid samples are taken from each SG.

assembly ~~between the transfer cart and the reactor vessel~~ (Subsection 9.1.4). The spent fuel cask handling crane can lift 125% of rated load and satisfactorily pass an inspection, and can raise the new fuel shipping container from the receipt truck (the only potentially heavy load handling for new fuel receipt described in Subsection 9.1.4).

4. Indications and alarms operate as described in Subsection 9.1.4.5.

5. Refueling machine, new fuel elevator, and fuel handling machine testing demonstrates compliance with test requirements specified by ASME NOG-1 (Reference 14.2-30) and ASME B30.20-2006 (Reference 14.2-31) as applicable.

6. Spent fuel cask handling building crane testing demonstrates compliance with test requirements specified by NUREG-0554 (Reference 14.2-24), ASME NOG-1 (Reference 14.2-30) and NUREG-0612 (Reference 14.2-21) as applicable.

7. Fuel handling tools perform their intended design function as identified in Subsection 9.1.4.2.1.

14.2.12.1.87 Component Cooling Water System Preoperational Test

A. Objectives

1. To verify the operation, interlock and alarm of CCW surge tank.
2. To demonstrate the capability of the CCW system to provide cooling water during normal operation, normal cooldown, and postulated loss-of-coolant accident (LOCA) modes of operation.
3. To verify operation of system valves and control circuitry.
4. To demonstrate the operation and verify the operating characteristics of the CCW pumps.

B. Prerequisites

1. Required construction testing is completed.
2. Component testing and instrument calibration is completed.
3. Test instrumentation is available and calibrated.
4. Required support systems are available.
5. Demineralized water is available for system makeup.
6. The CCW is aligned to cool the CCW motors.
7. The ESWS is available to CCW heat exchangers.

C. Test Method

1. The control circuitry of the CCW pumps, surge tanks, and valves is verified.
2. The CCW system pumps are operated, and performance characteristics verified.
3. System flows are balanced, as required, and then verified in each mode of operation. Testing includes verification of coolant flow to the thermal barrier via cross-tie.
4. The cooling ability of the CCW system is verified during RCS heatup and cooldown in conjunction with the RHRS during the hot functional test.
5. CCW surge tank vent valve closure logic is verified using a simulated high CCW radiation monitor condition.
6. The thermal barrier heat exchanger cooling water return line isolation valve logic is verified using a simulated reactor coolant pump thermal barrier heat exchanger cooling water high flow condition.
7. Demonstrate the ability to provide makeup water and verify flow to each pressurized CCW surge tank using DWS, PMWS and RWS supplies.

D. Acceptance Criteria

1. The tank alarms and interlocks operate as designed.
2. The performance characteristics of the CCW pumps are within design specifications (Subsection 9.2.2)
3. Components that are supplied with CCW receive flows that are within the design specifications in each of the operating modes including the supply of coolant flow to the thermal barrier via cross-tie.
4. The pump control and interlocks operate as designed.
5. CCW system performance characteristics are within design specifications.
6. CCW surge tank vent valve high radiation logic operates as described in Subsection 9.2.2.5.2.
7. The thermal barrier heat exchanger cooling water return line isolation valve logic operates as described in Subsection 9.2.2.5.5.
8. The ability to provide makeup water to each pressurized CCW surge tank using DWS, PMWS and RWS supplies is demonstrated.

2. Demonstrate the head and flow characteristics of the fire protection water supply system pumps and the operation of all auxiliaries.

3. Verify control logic of the fire protection water supply system pumps and auxiliaries.

4. Demonstrate flow paths of the fire protection water supply system.

5. Demonstrate operation of the fire alarm system.

6. Verify installation of fire extinguishers.

7. Verify operation of the gaseous fire protection systems.

D. Acceptance Criterion

1. The fire protection system operates as described in Subsection 9.5.1 and Appendix 9A.

14.2.12.1.91 Instrument Air System Preoperational Test

A. Objectives

1. To demonstrate operation of the instrument air system, including compressors, coolers, reservoirs, and dryers and associated controls.

2. To assure that the air supply equipment is able to maintain the quality of air supplied within design requirements.

3. To verify that the system responds appropriately to both normal operation of the plant and upset, faulted, or emergency conditions including increases in pressure due to component malfunction or failure, and to verify appropriate response of air-operated valves and other components during and following such upset, faulted or emergency conditions (e.g., fail open, fail closed, fail-as-is).

4. To demonstrate that operation of components requiring large quantities of air does not cause excessive instrument air system pressure transients.

~~Verification of safety-related containment isolation valve position on loss of pressure is described in Subsection 14.2.12.1.62.~~Tests are in accordance with RG 1.68.3 except for C.7.

B. Prerequisites

1. Required construction testing is completed.

2. Component testing and instrument calibration is completed.

3. Test instrumentation is available and calibrated.

4. Required support systems are available.

5. Required ac and dc power sources are available.

C. Test Method

1. Simulate pressure signals to verify alarms.

2. Operate instrument air system to verify operation while recording flow, pressure, and temperature.

   a. Air dryer units are tested for proper functioning, and the units are operated through at least one regeneration cycle. Air dryer testing includes verification of acceptable operation at maximum flow rates.

   b. The appropriate differential pressures and the proper operation of pressure switches, high and low-pressure alarms, safety and relief valves, bypass valves, and alarms and resets are verified.

   c. The operation of compressor unloaders, automatic and manual start and stop circuits of standby compressors, high, and low pressure alarms, pressure indications, and temperature indications are checked. Relief valve settings are verified.

   d. Compressors, aftercoolers, oil separator units, air receivers, and pressure-reducing stations are tested to verify proper operation according to system design.

3. Sample and analyze the air at the end of each feeder line using continuous flow techniques or by analyzing a discrete sample.

4. Simultaneously operate large users of instrument air and monitor the instrument air system pressures.

5. Loads that are a part of (or support the operation of) portions of the facility important to safety, which are identified as susceptible to changes in state or loss of operability upon increases in pressure due to component malfunction or failure are evaluated and tested as determined appropriate, without exceeding allowable component pressure ratings.

6. Test is performed to verify the fail-safe position of safety-related air-operated components for sudden loss of instrument air or gradual loss of pressure as described in Table 9.3.1-1.

D. Acceptance Criteria

1. The instrument air system performs as described in Subsection 9.3.1.

2. The Instrument air systems meets system design specifications relating to flow, pressure, and temperature of the product air. The total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design.

3. Air quality meets the requirements of American National Standards Institute (ANSI) / Instrumentation, Systems, and Automation Society (ISA) S73-1975, “Quality Standard for Instrument Air,” (Reference 14.2-22) with respect to oil, water, and particulate matter contained in the product air.

4. Loads that are a part of (or support the operation of).portions of the facility important to safety respond to pressure transients in accordance with design.

5 Plant equipment designated by design to be supplied by the instrument air system is not being supplied by other compressed air supplies (such as station air) that may have less restrictive air quality requirements without meeting the air quality requirements of ANSI/ISA S7.3-1975 (Reference 14.2-22).

6. Operation of supplied loads is continued in response to credible failures that result in an increase in the supply system pressure.

7. The fail-safe positions of safety-related air-operated components are same as shown in Table 9.3.1-1 for sudden loss of instrument air or gradual loss of pressure.

14.2.12.1.92 Station Service Air System Preoperational Test

A. Objective

1. To demonstrate operation of air compressors, air receivers and air dryers and associated controls.

2. To assure that the air supply equipment is able to maintain the quality of air supplied within design requirements.

3. To verify that the system responds appropriately to both normal operation of the plant and upset, faulted, or emergency conditions including increases in pressure due to component malfunction or failure; and to verify appropriate response of various loads that are important to safety during and following such upset, faulted or emergency conditions (e.g., fail open, fail closed, or fail-as-is).

4. To demonstrate that operation of components requiring large quantities of air does not cause excessive station service air system pressure transients.

Tests are in accordance with RG 1.68.3 except for C.7.

B. Prerequisites

1. Required construction testing is completed.

2. Component testing and instrument calibration is completed.

3. Test instrumentation is available and calibrated.

4. Required support systems are available.

5. Turbine component cooling water system is available.

6. Appropriate ac and dc power sources are available.

C. Test Method

1. Simulate temperature and pressure signals to verify alarms.

2. ~~Operate compressors to verify operation.~~ Operate station service air system to verify operation while recording flow, pressure, and temperature.

   a. Air dryer units are tested for proper functioning. Air dryer testing includes verification of acceptable operation at maximum flow rates.

   b. The automatic and manual start and stop circuits of standby compressors are checked. Relief valve settings are verified.

   c. Proper operation of inlet/air filter/silencer, compressors, intercoolers, aftercoolers and moisture separators are verified according to system design.

3. Sample and analyze the air at the end of each feeder line using continuous flow techniques or by analyzing a discrete sample.

4. Simultaneously operate large users of station service air and monitor the station service air system pressures.

5. Loads that are a part of (or support the operation of) portions of various loads that are important to safety, which are identified as susceptible to changes in state or loss of operability upon increases in pressure due to component malfunction or failure are evaluated and tested, as determined appropriate, without exceeding allowable component pressure ratings.

D. Acceptance Criterion

1. Compressors and air dryers perform as described in Subsection 9.3.1.

2. The station service air systems meets system design specifications relating to flow, pressure, and temperature of the product air.

3. Air quality meets the design specification of the station service air system.

4. Loads that are a part of (or support the operation of) portions of various loads that are important to safety respond to pressure transients in accordance with design.

5. Operation of supplied loads is continued in response to credible failures that result in an increase in the supply system pressure.

14.2.12.1.93 Boron Recycle System Preoperational Test

A. Objectives

9. The stability index of radial power distribution is evaluated periodically after restoration to normal rod position.

D. Acceptance Criteria

1. Measured power distribution and peaking factors are consistent with prediction.

2. The sensitivity of the incore and excore instrumentation signals to the RCCA misalignment is demonstrated by the results of power distribution measurements.

3. The stability index of radial power distribution is negative.

14.2.12.2.4.6 Remote Shutdown Test

A. Objectives

1. To demonstrate the capability of performing a controlled reactor shutdown to the hot standby condition.

2. To maintain the plant in a hot standby condition from outside the control room.

3. To demonstrate the potential for safely cooling down the plant from hot standby to cold shutdown conditions from outside the control room~~by placing the residual heat removal system into service and reducing the reactor coolant temperature~~.

Note: Testing is conducted in accordance with RG 1.68.2.

B. Prerequisites

1. Reactor power is greater than or equal to 10%.

2. The controls and instrumentation associated with the remote shutdown console are available.

3. Plant systems are in the normal operating mode with the turbine generator in operation.

4. Approved operating procedures for performing a remote shutdown are available.

5. Preoperational testing of plant instrumentation, controls, and systems to be used at remote shutdown locations is completed. This preoperational testing includes verification that all systems to be used during shutdown operation from outside the control room are operable in the manner in which they would be used during the operation (i.e., control from remote stations, manual operation, use of available power supplies, etc.) and that communication is established and maintained among the personnel who is performing the shutdown operation.

6. The authority and responsibility of the control room observers are established and documented in the test procedure. Provisions are made for the following actions:

(a) Assumption of control of the plant if an emergency or unsafe condition develops during the testing that cannot be managed by the shutdown crew.

(b) Performance of non safety-related activities that would not be required during an actual remote shutdown. These could include protection of non safety-related equipment from mechanical damage during the transient and the placement of equipment into standby status when no longer required. Such activities have been previously defined and evaluated to ensure that, if they were not performed during an actual remote shutdown, safe shutdown of the plant can still be achieved.

C. Test Method

1. Transfer control from the control room to the remote shutdown console.

2. Perform a controlled reactor shutdown to the hot standby condition from the remote shutdown console.

3. Demonstrate the capability to achieve and maintain the plant in a hot standby condition from the remote shutdown console for a minimum of 30 minutes.

4. During the demonstration, use only the equipment for which credit is taken to perform an actual remote shutdown.

5. Perform the test with the minimum of personnel required to be at the reactor unit at any one time (minimum shift crew).

6. Obtain the data at locations outside the control room.

7. ~~Using the residual heat removal system in steam condensing mode~~Following the hot standby demonstration, starting from approximately 350°F, reduce the reactor coolant temperature by at least 50 °F from outside the control room using the RHRS.~~reduce the reactor coolant temperature from approximately 350°F to approximately 300°F at a rate that does not exceed Technical Specification limits.~~

D. Acceptance Criteria

1. Transfer of control from the control room to the remote shutdown console is achieved.

2. The ability to perform a controlled reactor shutdown to the hot standby condition and to maintain hot standby conditions from the remote shutdown console is demonstrated.

3. The potential ability to cool down from hot standby to ~~hot~~cold shutdown conditions from outside the control room is demonstrated by reducing the reactor coolant temperature by at least 50 °F using the RHRS from outside the control room.

18.11 Design Implementation

18.11.1 Objectives and Scope

The objective of the design implementation is to demonstrate that the design that is implemented (i.e., the "as-built" design) accurately reflects the verified and validated design.

The scope of design implementation includes the effect on personnel performance resulting from design changes and provides the necessary support to ensure safe operations and that the as-built design conforms to the verified and validated design that resulted from the HFE process.

In this section, the referenced changes after V&V apply to the changes made to the US-APWR design following V&V.

18.11.2 Methodology

The detailed HSI ~~HFE~~ design implementation process is performed and documented as described below.

The design implementation methodology includes the following criteria:

- Aspects of the design that were not addressed in the design V&V are evaluated using an appropriate V&V method. Aspects of the design addressed by this criterion may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator, such as control room lighting and noise

- The potential impact on HAs is assessed and a risk significance level is assigned in accordance with the criteria in Reference 18.11-1

- All HFE-related issues documented in the issue tracking system are verified to be adequately addressed

18.11.3 Results

Facility design changes are documented and analyzed for their potential impact on HSIs. Those design implementation issues that negatively impact human performance are identified as HEDs and are tracked and dispositioned. HFE design modifications are documented in a periodic status report.

Chapter 17

DCD_17.04-39

17.3 Quality Assurance Program

The General Manager of Nuclear Energy Systems Headquarters (NESH) is responsible for the Design Certification Activities of US-APWR. The major design activities are performed by the Nuclear Energy Systems Engineering Center engineers. QA Program controls governing the activities are specified in QAPD (PQD-HD-19005 Rev.~~1~~2) (Ref 17.4-2, Ref 17.5.5-4).

Subcontractors of the Nuclear Energy Systems Engineering Center performing design activities in support of the US-APWR are also required to follow QAPD (PQD-HD-19005 Rev.~~1~~2).

For the quality assurance program description during the design certification phase, see Section 17.5.

The COL applicant is responsible for development a Quality Assurance Program Description during design other than the Design Certification, construction and operation phase.

The list of risk-significant SSCs for the D-RAP and its key assumptions shall be maintained by the risk and reliability organization. The list and changes thereof shall be approved by the EP and be provided to design engineering and QA staff working on the US-APWR project.

The risk and reliability organization shall ensure that the design engineers are provided the list of risk-significant SSCs for the D-RAP and its key assumption. The design engineers shall take into account the list of the risk-significant SSCs for the D-RAP and its key assumptions in their design activities and give some feedback to the risk and reliability organization in order to ensure that the key assumptions are realistic and achievable, if necessary.

c. Procedures and Instructions

General Manager, US-APWR project or his designated representative has prepared the procedures and instructions used in implementation of the D-RAP. General Manager, US-APWR project is responsible for development and verification of implementation of the D-RAP, and for assuring all affected MHI organizations are aware of the D-RAP.

d. Records

Records related to the D-RAP which are required to be maintained include the following:

- List of Risk-Significant SSCs
- EP meeting minutes/summaries
- Other quality assurance program records in accordance with the US-APWR QAPD (Ref. 17.4-2) for design certification.

e. Corrective action

Deficiencies identified where design documents address SSC reliability assumptions which are not compatible with the reliability assumptions of the PRA, or are not achievable or are unrealistic shall be entered into the corrective action program (CAP) system and addressed appropriately. The CAP utilized to support the QAPD can be used to implement the corrective actions related to the RAP.

f. Audit

Audit plans shall include for consideration, sampling the effectiveness of implementation of RAP implementation procedure. Audits shall consider several key aspects of the RAP including the identification of risk-significant SSCs, whether design and procurement information is consistent with the risk insights from the PRA, and whether assumed equipment reliability is determined to be practicable or achievable.

17.4.5 Integration into Existing Operational Programs

The US-APWR D-RAP is a source to other administrative and operational programs. Certain risk-significant SSCs identified in the D-RAP are included in existing operational programs such as the technical specifications surveillance requirements and provide assurance that the reliability values assumed in the PRA will be maintained throughout the plant life. The O-RAP implements the measures that yield the significant improvements in the PRA through the plant's existing programs for maintenance or QA. Implementation of the Maintenance Rule requirements contained in 10CFR50.65 (Ref. 17.4-~~2~~3) is an example of how the plant could address the enhanced treatment of certain SSCs in the O-RAP. Per SECY 95-132, the COL Applicant may meet most of the objectives of the O-RAP via existing programs such as maintenance rule, in-service testing, and QA. The COL Applicant must address non-safety risk significant SSCs.

17.4.6 Operating Experience

Consideration and use of operating experience is vital to the overall objective of the D-RAP. Operating experience is considered along with various PRA analytical and importance measures when developing a comprehensive risk analysis. The EP considers component operating history and industry operating experience when it can be applied to assessing risk significance. For example, operating experience indicates that motor driven and turbine driven pumps may have different reliability.

The review of operating experience investigates situations where previous failures of components in similar design applications have led to functional failures of SSCs. The review of operating experiences is not limited to hardware failure but also extends to situations where human performance led to functional failures of SSCs of a similar system design. As an example, the US-APWR design improves reliability and eliminates required operator actions to switch over from injection to recirculation typical in conventional PWRs.

17.4.7 D-RAP

As discussed in Section 17.4.2, Phase I of the D-RAP includes the initial identification of SSCs to be included in the program, implementation of the aspects applicable to design efforts, and definition of the scope, requirements, and implementation options to be included in the later phases.

17.4.7.1 SSCs Identification

During the US-APWR design phase, risk significant SSCs are identified for inclusion in the scope of the D-RAP. A list of risk significant SSCs is developed and controlled as a design input for consideration during the design phase. The list of risk significant SSCs is initially based on the results of the PRA and the EP. For further discussion on PRA, refer to Chapter 19, Section 19.1, of this DCD. The PRA is used to identify risk significant SSCs based on risk achievement worth (RAW) and Fussell-Vesely Worth (FVW). For further information, see Chapter 19, Section 19.1.7.4 of this DCD. The list of risk significant SSCs identified during the design phase is updated when the plant-specific PRA is developed. In addition to the PRA input, information from operating experience of Japanese design plants, as well as US industry experience is considered for identification of risk significant SSCs. A third source in the D-RAP process for identifying risk significant SSCs is the use of an EP consisting of representatives from Design Engineering, PRA, as well as other highly qualified individuals with operations, and maintenance experience who are independent of the PRA Section. The EP also

RAI 17.04-23 (This will be revised incorporating the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 6 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
3	Component cooling water system (CCWS)		
1	CCW pump discharge line check valves [VLV-016A (B,C,D))	RAW/CCF/LPSD	The component cooling water system (CCWS) transfer heat from plant safety-related components to the essential service water system (ESWS). This system supports various safety and non-safety mitigation systems. Accordingly, reliability of CCWS emergency feedwater system (EFWS) has significant impact on risk. CCWS has four trains, each having a component cooling water pump and a component cooling water heat exchanger. Two trains compose a subsystem, which shares a supply / return header and a surge tank. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains. - SSCs that have potential to cause large external leak are risk significant. Since the two trains that compose a subsystem are not physically isolated, large external leak from SSCs that result in loss of inventory is assumed to result in degradation or failure of two trains.
2	A~D-Component cooling water pumps [NCS-RPP-001A (B,C,D)]	FV/RAW/CCF /LPSD	
3	A~D-Component cooling water heat exchangers [NCS-RHX-001A (B,C,D)]	RAW/CCF/LPSD	
4	CCW pump discharge cross tie-line motor operated valves [MOV-020A (B,C,D)]	RAW/CCF/LPSD	
5	CCW pump suction line cross tie-line motor operated valves [MOV-007A (B,C,D)]	RAW/CCF/LPSD	
6	SSCs that compose CCW boundary	RAW/EJ/LPSD	
7	Safety Injection pump motor outlet manual valve [NCS-VLV-114A(B,C,D)]	RAW(L2)	
8	Safety Injection oil cooler outlet manual valve [NCS-VLV-115A(B,C,D)]	RAW(L2)	
9	Safety Injection pump outlet manual valve [NCS -VLV- 116A(B,C,D)]	RAW(L2)	

For Item 10 RAI 17.04-23

For Item11-19 RAI 17.04-24 (This will be revised incorporating the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 7 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
10	Safety Injection pump outlet manual valve [NCS -VLV- 119A(B,C,D)]	RAW(L2)	These valves are used (opened) to provide alternative CCW from the fire suppression system or the non-essential chilled water system to the charging pump cooling line under loss of CCW events. These are important SSCs at loss of CCW events to prevent RCP seal LOCA.
~~7~~ 11	CS/RHR heat exchanger discharge line motor operated valves [MOV-145A (B,C,D)]	FV/RAW/CCF /LPSD	
12	CS/RHR pump outlet manual valve [NCS-VLV-131A(B,C,D)]	RAW(L2)/LPSD	
13	CS/RHR pump motor outlet manual valve [NCS-VLV-128A(B,C,D)]	RAW(L2)/LPSD	
14	CS/RHR pump outlet orifice [NCS-FE1246A(B,C,D)]	RAW(L2)/LPSD	
15	CS/RHR pump motor outlet orifice [NCS-FE1250A(B,C,D)]	RAW(L2)/LPSD	
16	CS/RHR pump inlet manual valve [NCS-VLV-125A(B,C,D)]	RAW(L2)/LPSD	
17	CS/RHR heat exchanger inlet manual valve [NCS-VLV-141A(B,C,D)]	RAW(L2)/LPSD	
18	CS/RHR heat exchanger outlet orifice [NCS-FE1242A(B,C,D)]	RAW(L2)/LPSD	
19	CS/RHR CS/RHR heat exchanger outlet manual valve [NCS-VLV-144A(B,C,D)]	RAW(L2)/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 8 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
20	CS/RHR heat exchanger outlet valve [NCS-MOV-145A(B,C,D)]	RAW/FV,RAW(L2)/ LPSD/ FV,RAW(FLOOD) /RAW(FIRE)	These valves are used (opened) to provide alternative CCW from the fire suppression system or the non-essential chilled water system to the charging pump cooling line under loss of CCW events. These are important SSCs at loss of CCW events to prevent RCP seal LOCA.
~~8~~ 21	Charging injection Pump Cooling Line Check Valves [TBD]	RAW/CCF/LPSD	
~~9~~ 22	Charging injection pump cooling discharge line motor operated valves [TBD]	RAW/CCF/LPSD	
23	CHI seal water heat exchanger inlet manual valve -A(B) [NCS-VLV-311A(B)]	LPSD	
24	CHI oil cooler inlet manual valve-A(B) [NCS-VLV-312A(B)]	LPSD	
25	CHI pump motor inlet manual valve -A(B) [NCS-VLV-301A(B)]	LPSD	
~~10~~ 26	CCWS - fire suppression system boundary motor operated valves [TBD]	RAW/CCF/LPSD	
~~11~~ 27	CCWS - RWSP line boundary check valves [VLV-065A (B)]	RAW/LPSD	
~~12~~ 28	CCWS - RWSP line boundary manual valves [VLV-066A (B)]	RAW/LPSD	

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(This will be revised incorporating the discussion of expert panel.)

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Note: This will be revised incorporating the discussion of expert panel.

Table 17.4-1 Risk significant SSCs (sheet 13 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
7	Emergency power source (EPS)		
1	480V AC motor control center (MCC) buses [~~TBD~~ EPS-4ESBA(B,C,D)]	RAW/LPSD	The EPS consists of four separate trains. Each safety train consists of one 6.9kV AC medium voltage bus and 480V AC low voltage buses (Load Centers, Motor Control Centers). Each AC medium voltage bus connects to class 1E gas turbine generator. This system supports various safety mitigation systems and therefore, reliability of the EPS system has significant impact on risk. Since the EPS consists of four separate trains, single failure in trains not significantly impact risk. However, failure of multiple trains is have significant impact on risk. Accordingly, SSCs that have potential to cause common cause failures among multiple trains are risk significant
2	480V AC load center buses [~~TBD~~ EPS-4LCA(B,C,D)]		
3	6.9kV buses [~~TBD~~ EPS-6ESBA(B,C,D)]	RAW/EJ/LPSD	
4	125V DC buses train A and D [~~TBD~~ EPS- EPS-DCA(D)]	RAW/LPSD	
5	125V DC buses train B and C [~~TBD~~ EPS- EPS-DCB(C)]	RAW(L2)	
6	120V buses train A-D [~~TBD~~ EPS-VITA(B,C,D)]	RAW(L2/ FIRE)	
7	Swing MCC incomer circuit breakers [~~TBD~~ EPS-4SB(D)1]	RAW/CCF/LPSD	
8	Batteries [~~TBD~~ EPS-BA1A(B,C,D)]	RAW/CCF/LPSD	
9	6.9kV AC bus incomer circuit breakers [~~TBD~~ EPS-6HA(B,C,D)]	FV/RAW/CCF/LPSD	
10	Gas turbine discharge circuit breakers [~~TBD~~ EPS- GTBA(B,C,D)]	RAW/CCF/LPSD FV/CCF(FIRE)	

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Note: This will be revised incorporating the discussion of expert panel.

Table 17.4-1 Risk significant SSCs (sheet 14 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
11	Circuit breakers between 6.9kV bus and 6.9kV/480V safety power transformers [~~TBD~~ EPS- 4IA(B,C,D)]	RAW/CCF/LPSD	The “Insights and Assumptions” for these SSCs are described on the previous page.
12	MCC bus incomer circuit breakers [~~TBD~~ EPS- 4JA(B,C,D)]	RAW/CCF/LPSD	
13	Circuit breakers between 125V DC bus and Inverter [~~TBD~~ EPS- VIT4A (B,C,D)]	RAW/CCF/LPSD	
14	Class 1E gas turbine generators [~~TBD~~ EPS- GTA(B,C,D)]	FV/RAW/CCF /LPSD)	
15	Gas turbines generator sequencers [TBD]	RAW/CCF/LPSD FV(FIRE)	
16	Inverters [~~TBD~~ EPS- INVA(B,C,D)]	RAW/CCF/LPSD	
17	Main transformers [~~TBD~~ EPS- MTF]	RAW(L2)	
18	6.9kV/480V safety power transformers [~~TBD~~ EPS- 4PTA(B,C,D)]	RAW/LPSD	

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Note: This will be revised incorporating the discussion of expert panel.

Table 17.4-1 Risk significant SSCs (sheet 15 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
8	Alternative AC power sources (Permanent bus)		
1	Non-class 1E gas turbine generators [~~TBD~~ EPS- GTP1(2)]	FV/RAW/CCF /LPSD	Two non-safety buses called “Permanent bus”, which is connected to Alternative AC (AAC), which consists of non-class 1E gas turbine generators respectively. Each non-class 1E gas turbine generators is manually connected to two safety medium voltage buses via selector circuit under the occurrence of loss of safety AC power. The AAC is a countermeasure against station blackout events. SSCs that have potential to cause failures that degrade the availability to supply AAC power to safety medium voltage are risk significant. Systems for the mitigation of core damage accident are connected to permanent bus.
2	480V permanent buses [TBD]	RAW(L2)	
3	6.9kV permanent buses [TBD]	RAW(L2)	
4	Circuit breakers between 6.9kV bus and 6.9kV/480V power transformer [~~TBD~~ EPS- 4IP1(2)]	RAW(L2)	
5	Batteries [TBD]	RAW/CFF/LPSD	
6	Gas turbine generator discharge circuit breakers [~~TBD~~ EPS- GTBP1(2)]	RAW/CCF/LPSD	
7	AAC selector circuit breakers [~~TBD~~ EPS- 4AA(B,C,D)]	RAW/CCF/LPSD	
8	Circuit breakers between 125V DC bus and Inverter [~~TBD~~ EPS- VIT4P1(2)]	RAW/CCF/LPSD	
9	Inverters [~~TBD~~ EPS- INVP1(2)]	RAW/CCF/LPSD	
10	Gas turbine generator sequencers	RAW/CCF/LPSD	
11	6.9kV/480V power transformers [~~TBD~~ EPS-4PTP1(2)]	RAW/LPSD	

Attachment 2

Table 17.4-1 Risk significant SSCs (sheet 22 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
15	Instrumentation and control (I&C) system		
1	Permanent bus low voltage signal software	RAW/CCF	This software provides start signal to non-class 1E gas turbine generator. Under SBO, This software must operate in order to backup of the safety bus by AAC power source.
2	Component cooling water system train isolation signal software	RAW/CCF	SSCs that have potential to cause common cause failure of signals are risk significant since such failure may result in loss of total system function. EFW T/D pump start signals are risk significant since such failure results in loss of one of two available EFW pumps under, SBO and loss of EFW room cooling conditions. Reliability of signals other than "S signal" is assumed to have same reliability with "P signal".
3	SG isolation signal software	RAW/CCF	
4	Engineered safety features actuation signal software (P,S)	RAW/CCF	
~~5~~	~~SG(EFW) isolation signals~~	~~RAW/CCF~~	
~~6~~5	Main steam line isolation signal software	RAW/CCF	
~~7~~6	Black out signal software	RAW/CCF	
~~8~~	~~CCW start signals~~	~~RAW(L2,FLOOD)~~	
~~9~~7	Containment pressure sensors [TBD]	RAW(L2)/CCF(L2)	
~~10~~	~~A~D Emergency feed water pump start signals~~	~~RAW~~	
8	Steam generator water level sensors [TBD]	EJ	
9	CCW pump breaker position sensing device	EJ	
10	Reactor Protection System	EJ	
11	Engineered Safety Features Actuation System	EJ	
12	Safety Logic System	EJ	
~~11~~13	EFW pump start signal software	RAW/CCF	
~~12~~14	Diverse actuation system	EJ	The unreliability of this system is assumed to be 0.01.

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Note: This will be revised incorporating the discussion of expert panel.

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(This will be revised incorporating the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 26 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
21	Containment spray / residual heat removal (CS/RHR) system		
1	Heat exchanger bypass valves [FCV-604] [FCV-636]	RAW/LPSD	The Containment Spray / Residual Heat Removal (CS/RHR) System consists of four independent trains. The CS/RHR System has the following four functions. a. Containment Spray b. Alternative Core Cooling c. RHR Operation during operating modes 4 , 5 and 6.. Since CS/RHR system consists of four independent trains, failure of one train does not have significant impact on risk. However, failures of SSCs that impact multiple trains are risk significant. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains. - SSCs that have potential to cause loss of RWSP inventory out side the containment due to large external leaks. Loss of RWSP inventory impacts not only all four trains of CS/RHR system but also other systems that use RWSP as water source.
2	RHR line heat exchanger discharge air operated valves [FCV-603] [FCV-633]	RAW/LPSD	
3	Pump suction line check valves [VLV-004A (B,C,D)]	RAW/CCF/LPSD	
4	RHR line containment isolation check valves [VLV-022A (B,C,D)]	RAW/CCF/LPSD	
5	RHR line containment isolation motor operated valves [MOV-021A (B,C,D)]	RAW/CCF/LPSD	
6	A~D-Containment spray/residual heat removal pumps [RHS-RPP-001A (B,C,D)]	RAW/CCF/LPSD FV(FLOOD)	
7	A~D-Containment spray/residual heat removal heat exchangers [RHS-RHX-001A (B,C,D)]	RAW/CCF/LPSD	
8	RHR line boundary check valves [VLV-028A (B,C,D)]	RAW/LPSD	
9	RWSP discharge line isolation valves [~~TBD~~ CSS-MOV-001A(B,C,D)]	RAW	

Table 17.4-1 Risk significant SSCs (sheet 31 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
23	Reactor ~~protection~~trip system (~~RPS~~RTS)		
1	Reactor trip breakers [TBD]	RAW/CCF	These systems are necessary to provide negative reactivity for plan t trip.
2	Control rod (rod injection) [TBD]	FV/RAW/CCF	
24	Chilled water system (VWS)		
1	Chiller units train B and C [~~TBD~~ VWS-PEQ-001B(C)]	FV/RAW/CCF/LPSD	The safety related water system supplies chilled water to safety related HVAC systems. SSCs that have potential to cause common cause failures among trains B and C are risk significant since such failures results in loss room cooling in M/D EWF pump area. SSCs that compose train A and D are not risk significant because the PRA assumes only the M/D EFW pumps to be dependent on room cooling during the mission time.
2	Pumps train B and C [~~TBD~~ VWS-PPP-001B(C)]	RAW/CCF/LPSD	

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For #23 DCD_17.04-38

Note: This will be revised incorporating the discussion of expert panel.

17.4.8 ITAAC for the D-RAP

Tier 1 ITAAC are proposed to verify that the D-RAP provides reasonable assurance that the plant is designed and constructed in a manner that is consistent with the key assumptions and risk insights for risk-significant SSCs~~design of SSCs within the scope of the RAP is consistent with their assumed design reliability~~. The list of risk-significant SSCs for ITAAC will be prepared by introducing the plant's site-specific information to the list shown in Table 17.4-1 in the Phase II of the D-RAP. The ITAAC acceptance criteria are established to ensure that the following three (3) major elements are taken into consideration:~~estimated reliability of each as-built SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.~~

- Identification of all as-built SSCs in the scope of the D-RAP
- Description of the methodology used to identify the as-built SSCs in scope of the D-RAP
- For the as-built SSCs in scope of D-RAP, identify and describe the reliability assurance activities that are accomplished prior to the initial fuel load, which provide reasonable assurance that the plant is designed and constructed in a manner that is consistent with the key assumptions (including reliability and availability assumptions in PRA, when applicable) and risk insights for the risk-significant SSCs.

17.4.9 Combined License Information

COL 17.4(1) *The COL Applicant shall be responsible for the development and implementation of the Phases II and III of the D-RAP, including QA requirements. In the Phase II, the plant's site-specific information should be introduced to the D-RAP process and the site-specific risk-significant SSCs should be combined with the US-APWR design risk-significant SSCs into a list for the specific plant. Phase II is performed during the COL application phase and updated/maintained during the COL license holder phase. In the Phase III, procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP should ensure that significant assumptions, such as equipment reliability, are realistic and achievable. The QA requirements should be implemented during the procurement, fabrication, construction, and pre-operation testing of the SSCs within the scope of the RAP. Phase III is performed during the COL license holder phase and prior to initial fuel loading. The COL applicant will propose a method by which it will incorporate the objectives of the reliability assurance program into other programs for design or operational errors that degrade nonsafety-related, risk-significant SSCs.*

COL 17.4(2) *The COL Applicant shall be responsible for the development and implementation of the O-RAP, in which the RAP activities should be integrated into the existing operational program (i.e., Maintenance Rule, surveillance testing, in-service inspection, in-service testing, and QA). The O-RAP should also include the process for providing corrective actions for design and operational errors that degrade non-safety-related SSCs within the scope of the RAP. A description of the proposed method for developing/integrating the operational RAP into operating plant programs (e.g., maintenance rule, quality assurance) is performed during the COL application phase. The development/integration of the operational RAP is performed during the COL license holder phase and prior to initial fuel loading. All SSCs identified as risk-significant within the scope of the D-RAP should be categorized as high-safety-significant (HSS) within the scope of initial Maintenance Rule.*

17.4.10 References

17.4-1 “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Design,” SECY 95-132, U.S. Nuclear Regulatory Commission, Washington, DC, May 1995.

17.4.2 “Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.2)”

17.4-~~2~~3 ‘Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,’ “Domestic Licensing of Production and Utilization Facilities,” Energy. Title 10, Code of Federal Regulations, Part 50.65, U.S. Nuclear Regulatory Commission, Washington, DC.

17.5 Quality Assurance Program Description

For the Design Certification phase, the MHI-NESH US-APWR Project Quality Assurance Program (QAP) is the top-level policy document that establishes the quality assurance policy and assigns major functional responsibilities for plants designed by MHI-NESH. The QAP describes the methods and establishes QAP and administrative control requirements, described in “Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~1~~2)” (Ref 17.5.5-4), that meet 10 CFR Part 50, Appendix B and 10 CFR Part 52. The QAP is based on the requirements of ASME NQA–1-1994, “Quality Assurance Requirements for Nuclear Facility Applications,” Parts I and II, as specified in Ref.17.5.5-4.

The MHI QAPD for the Design Certification Phase has been prepared on the basis of the NRC approved QAP template (NEI, 06-14A Rev.4 and earlier revisions) (Ref 17.5.5-3) prepared by the Nuclear Energy Institute and has been evaluated against the SRP. The MHI QAPD provides the QAP controls implemented. MHI performed the comparison of SRP (Mar. 2007) (Ref 17.5.5-2) and draft SRP (Sept. 2006) (Ref 17.5.5-1) which was used as a reference for the MHI QAPD and determined that there is no impact to the MHI QAPD.

Business policies of MHI-NESH establish high level responsibilities and authority for carrying out administrative functions which are outside the scope of the QAP.

Procedures establish practices for certain activities which are common to all MHI-NESH organizations performing those activities such that the activity is controlled and carried out in a manner that meets QAP requirements. Organization specific procedures establish detailed implementation requirements and methods, and may be used to implement the business policies of MHI-NESH or be unique to particular functions or work activities.

The COL applicant is responsible for development a Quality Assurance Program Description during design other than the Design Certification, construction and operation.

17.5.1 Combined License Information

COL 17.5(1) *The COL applicant shall develop and implement the design other than the Design Certification, construction and operational QAP that also covers the activities described in Section 17.5.*

17.5.2 References

17.5.5-1 “Draft Standard Review Plan (SRP) 17.5 dated September 22, 2006”

17.5.5-2 “Standard Review Plan (SRP) 17.5 March 2007”

17.5.5-3 “Quality Assurance Program Description (NEI 06-14A Rev.4 and earlier versions)”

17.5.5-4 “Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~1~~2)”

Chapter 18

US-APWR HFE design in the site-specific as-built plant. The site specific HFE processes and procedures will be used for HSI design changes after the certified US-APWR design responsibility is officially turned over to the site specific HFE Team.

18.1.1.1 Assumptions and Constraints Identification

The assumptions and constraints of the design, such as a specific staffing plan or the use of specific HSI technology inherent in are inputs to the HFE program rather than the result of HFE analyses and evaluations. The design assumptions and constraints of the Basic HSI System are clearly identified in Section 5.1.1.2 of Reference 18.1-1. The regulatory requirements applicable to the US-APWR HFE program are listed in Reference 18.1-1, Section 3.0, "Applicable Codes, Standards and Regulatory Guidance".~~The assumptions and constraints of the design, such as a specific staffing plan or the use of specific HSI technology inherent in are inputs to the HFE program rather than the result of HFE analyses and evaluations. The design assumptions and constraints are clearly identified. The regulatory requirements applicable to the US-APWR HFE program are listed in Reference 18.1-1, Section 3.0, "Applicable Codes, Standards and Regulatory Guidance".~~

A fundamental US-APWR HFE design assumption is that it is possible to operate the plant with just one reactor operator (RO) and one senior reactor operator (SRO) in the main control room (MCR) during postulated plant operating modes (Reference 18.1-1, Section 4.1.f, Design Basis, MCR Staff). This MCR staffing meets the regulatory requirements of 10 CFR 50.54(m)(2)(iii) (Reference 18.1-2). The normal MCR staff is supplemented by one additional SRO and one additional RO that is to be at the plant to accommodate unexpected design conditions, including conditions where the human-system interface system (HSIS) is degraded. This overall plant staffing meets the regulatory requirements of 10 CFR 50.54(m)(2)(i) (Reference 18.1-2). While the HSIS is designed to accommodate the minimum MCR and plant staffing described above, the space and layout of the MCR are designed to accommodate the foreseen maximum number of operating and temporary staff.

Reference 18.1-1 describes the US-APWR HSIS design and the HFE design process. The HSIS has been developed and tested for application in both new and existing operating plants in Japan. The functional requirement specification for the Japanese Advanced Pressurized-Water Reactor (APWR) HSIS design serves as the initial source of input to the HSIS design effort. The US-APWR HSIS design is a direct evolution from the predecessor standard Japanese PWR. However, due to differences between existing Japanese nuclear plants and the US-APWR, and the potential for cross-cultural HFE issues, specific changes in the design are addressed in the US-APWR design.

The development of the integrated US-APWR HSIS, as described in Sections 18.7, 18.8, and 18.9 ("Human-System Interface Design," "Procedure Development," and "Training Program Development"), are conducted in an HFE development facility. In addition to HSIS development and testing (Reference 18.1-1, Subsection 5.7.3.3, "HSI Tests and Evaluations"), the verification and validation (V&V) process described in Section 18.10 are conducted in this facility (Reference 18.1-1, Subsection 5.10.2.2.4.b, "Integrated System Validation", "Validation Test Facility"). This facility provides the updated proof-of-

18.11 Design Implementation

18.11.1 Objectives and Scope

The objective of the design implementation is to demonstrate that the design that is implemented (i.e., the “as-built” design) accurately reflects the verified and validated design.

The scope of design implementation includes the effect on personnel performance resulting from design changes and provides the necessary support to ensure safe operations and that the as-built design conforms to the verified and validated design that resulted from the HFE process.

In this section, the referenced changes after V&V apply to the changes made to the US-APWR design following V&V.

18.11.2 Methodology

The detailed HSI ~~HFE~~ design implementation process is performed and documented as described below.

The design implementation methodology includes the following criteria:

- Aspects of the design that were not addressed in the design V&V are evaluated using an appropriate V&V method. Aspects of the design addressed by this criterion may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator, such as control room lighting and noise
- The potential impact on HAs is assessed and a risk significance level is assigned in accordance with the criteria in Reference 18.11-1
- All HFE-related issues documented in the issue tracking system are verified to be adequately addressed

18.11.3 Results

Facility design changes are documented and analyzed for their potential impact on HSIs. Those design implementation issues that negatively impact human performance are identified as HEDs and are tracked and dispositioned. HFE design modifications are documented in a periodic status report.

Tier 1

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Table 2.2-4 Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of ~~3~~4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The structural configurations of the R/B and each~~the~~ PS/B are as described~~shown~~ in ~~Figures 2.2-1 through 2.2-13 and~~ Table 2.2-2.	1. Inspections of the as-built structural configurations of the R/B and each~~the~~ PS/B will be performed.	1. The ~~as-build~~as-built structural ~~design~~ configurations of the R/B and ~~the~~each PS/B are reconciled with descriptions in ~~Figures 2.2-1 through 2.2-13 and~~ Table 2.2-2.
2. The ASME Code Section III components and piping retain their pressure boundary integrity at their design pressure.	2. A hydrostatic test and preoperational NDE will be performed in conjunction with Section III of the ASME Code.	2. The results of the hydrostatic test and preoperational NDE of the as-built components and piping conform to the requirements of the ASME Code, Section III.
3. The PCCV retains structural integrity ~~under~~ with design pressures of 68 psig.	3. A structural integrity test (SIT) will be performed in accordance with the ASME code, Section III.	3. The result of the structural integrity test (SIT) of the as-built PCCV conforms to the requirements in the ASME Code, Section III.
4.a The integrated containment system barrier prevents release of fission products to the atmosphere.	4.a A containment integrated leak rate test will be performed in accordance with 10 CFR 50, Appendix J, Type A testing.	4.a The containment integrated leak rate test verifies that the leak rate is less than the allowable leakage rate specified in 10 CFR 50, Appendix J, Type A testing.
4.b The containment system barrier primary reactor containment penetrations prevent release of fission products to the atmosphere.	4.b A leak rate test will be performed for all Type B containment penetrations in accordance with 10 CFR 50, Appendix J Type B tests.	4.b The containment penetration leak rate tests verifies that the leak rate is less than the allowable leakage rate specified in 10 CFR 50, Appendix J for Type B tests.
5. The PCCV is designed based on the structural design-basis loads.	5. An analysis will be performed to verify that the as-built PCCV structural design-basis loads are reconciled.	5. ASME design report exists for the as-built PCCV, and concludes the PCCV is designed based on the structural design-basis loads.
6. The safety-related standard plant buildings other than the PCCV are designed based on the structural design-basis loads.	6. An analysis will be performed to verify that the as-built safety-related standard plant structures, other than the PCCV, structural design-basis loads are reconciled.	6. Design reports exist for the as-built safety-related standard plant buildings other than the PCCV, and conclude the safety-related standard plant buildings are designed in accordance with structural design-basis loads.
7. The ASME Code, Section III, Class 1 piping systems and components are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design-basis loads.	7. Refer to Table 2.3-2 ITAAC #1.~~Section 2.3 ITAAC #2~~	7. Refer to Table 2.3-2 ITAAC #1.~~Section 2.3 ITAAC #2~~

Table 2.2-4 Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of ~~3~~4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The ASME Code, Section III, Class 2 or 3 piping systems and components are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design-basis loads.	8. Refer to Table 2.3-2 ITAAC #3~~Section 2.3 ITAAC #5~~.	8. Refer to Table 2.3-2 ITAAC #3~~Section 2.3 ITAAC #5~~.
9.~~a~~ Divisional flood barriers are provided in the R/B and each~~the~~ PS/B to protect against the internal and external flooding.	9.~~a~~ An inspection will be performed to verify that the as-built divisional flood barriers exist in the R/B and each~~the~~ PS/B.	9.~~a~~ The as-built divisional flood barriers exist at the appropriate locations in the R/B and each~~the~~ PS/B against the internal and external flooding.
10.~~9.b~~ Water-tight doors are provided in the R/B to protect against the internal and external flooding.	10.~~9.b~~ An inspection of the as-built water- tight doors will be performed.	10.~~9.b~~ The as-built water-tight doors exist at the appropriate locations in the R/B against the internal and external flooding.
11~~10~~. Penetrations in the divisional walls of the R/B and each~~the~~ PS/B, except for water-tight doors, are provided appropriately against the internal and external flooding.	11~~0~~. An inspection of the as-built penetrations will be performed.	11~~0~~. The as-built penetrations in the divisional walls of the R/B and each~~the~~ PS/B are installed at an acceptable level above the floor, and are sealed up to the internal and external design flood levels.
12~~1~~. Safety-related electrical, instrumentation, and control equipment are located in the R/B and each PS/B to protect ~~against~~ them from the design flood~~ level~~.	12~~1~~. An inspection of the as-built safety-related electrical, instrumentation, and control equipment in the R/B and each PS/B will be performed.	12~~1~~. The as-built safety-related electrical, instrumentation, and control equipment in the R/B and each PS/B are located at sufficient height above the floor surface to protect them against the design flood~~ level~~.
1~~2~~3. For the R/B and each~~the~~ PS/B, external wall thicknesses below flood level are a minimum of two feet thick~~provided~~ to protect against water seepage.	13~~2~~. An inspection of the as-built external wall thickness for the R/B and each~~the~~ PS/B will be performed.	13~~2~~. For the R/B and each~~the~~ PS/B, the as-built external walls below flood level are ~~provided with adequate thickness~~ a minimum of two feet thick to protect against water seepage.
14~~3a~~. Flood barriers of the R/B and each~~the~~ PS/B are installed up to the finished plant grade level to protect against water seepage.	14~~3a~~. Inspections of the as-built flood barriers will be performed.	14~~3a~~. The as-built flood barriers are installed up to the finished plant grade level for the R/B and each~~the~~ PS/B to protect against water seepage.

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Table 2.2-4 Structural and Systems Engineerin , Analyses, and Acceptance Criteria (Sheet 3 of ~~3~~4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15~~3b~~. Flood doors and flood barrier penetrations of the R/B and each~~the~~ PS/B are provided with flood protection features.	15~~3b~~. Inspections of the as-built flood doors and flood barrier penetrations will be performed.	15~~3b~~. For the R/B and each PS/B, the as-built flood doors and flood barrier penetrations are provided with flood protection features to protect against water seepage.
16~~4~~. Penetrations in the external walls~~, including those up to the subgrade level if necessary,~~ of the R/B and each PS/B are provided with flood protection features below flood level.	16~~4~~. An inspection will be performed to verify that the flood protection features of the as-built penetrations in the external walls of the R/B and each~~the~~ PS/B exist below flood level.	16~~4~~. The as-built penetrations in the external walls of the R/B and ~~the~~each PS/B are provided with flood protection features below flood level.
17~~5~~. Redundant safe shutdown components and associated electrical divisions outside the containment and the control room complex are separated by 3-hour rated fire barriers to preserve the capability to safely shutdown the plant following a fire. The 3-hour rated fire barriers are placed as required by the FHA.	17~~5~~. An inspection of the as-built fire barriers will be performed.	17~~5~~. The 3-hour rated as-built fire barriers are placed as required by the FHA.
18~~6~~. All penetrations and openings through the fire barriers are protected against fire.	18~~6~~. An inspection will be performed to verify that the as-built components are provided to protect the penetrations and openings through fire barriers.	18~~6~~. All as-built penetrations and openings are protected with rated components (i.e. fire doors in door openings, fire dampers in ventilation duct openings, and penetration seals) consistent with the fire resistance rating of the associated barrier.
19~~7~~. Safety-related SSCs are designed to withstand the dynamic effects of pipe breaks.	19~~7~~. Refer to Table 2.3-2 ITAAC #4.~~Section 2.3 ITAAC #6~~	19~~7~~. Refer to Table 2.3-2 ITAAC #4.~~Section 2.3 ITAAC #6~~
20~~18~~. The key dimensions of the RV conform with the licensed design and are documented in an as-built report.	20~~18~~. Refer to Section 2.4.1 ITAAC #5	20~~18~~. Refer to Section 2.4.1 ITAAC #5

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Table 2.2-4 Structural and Systems Engineering Inspecti
and Acceptance Criteria (Sheet 4 of 4)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
21~~19~~. Safety-related SSCs are protected from any credible internal missile sources inside and outside the containment.	21~~19~~. An inspection will be performed to verify as-built locations of safety-related SSCs are protected from potential impact by credible internal missiles.	21~~19~~. Primary missile protection is provided by locating missile sources behind concrete walls and floors, and/or locating safety-related SSCs outside the zones of credible missile strikes.
22. Special modular construction techniques adequately address the fabrication, shipping, handling, and installation of the SC modules and reconcile the as-built configuration of the plant with the structural design basis of the licensed facility.	22. An inspection will be performed to verify special modular construction techniques adequately address the fabrication, shipping, handling, and installation of the SC modules and reconcile the as-built configuration of the plant with the structural design basis of the licensed facility.	22. Fabrication, shipping, handling, and installation of the SC modules are in accordance with governing programs, codes and specifications.
23. Failure of non-seismic and seismic Category II structures will not impair the ability of near-by safety-related SSCs to perform their safety-related functions (II/I interactions).	23. An inspection will be performed to verify failure of non-seismic and seismic Category II structures will not impair the ability of near-by safety-related SSCs to perform their safety-related functions (II/I interactions).	23. The inspection of design criteria and as-built plant configuration for non-seismic and seismic Category II structures verify the acceptability of II/I interactions.
24. Relief panels exist on the first floor of the T/B.	24. An inspection will be performed of the as-built T/B.	24. The relief panels exist on the first floor of the as-built T/B.
25. The electrical room in the T/B is waterproof.	25. An inspection will be performed of the as-built T/B.	25. The as-built electrical room in the T/B is waterproof.

~~The SWMS is non-seismic category and is not designed to ASME code specifications.~~The portions of the A/B that house the principal SWMS equipment are designed to seismic Category II. The SWMS is a non-safety system and the components are non seismic.

System Operation

The spent resin storage tanks receive spent resin from various plant sources and provide staging for decay and transfer capability into disposal containers for off-site disposal. The spent charcoal handling subsystem shares the use of the spent resin storage tanks and the resin dewatering equipment. Spent resin, spent charcoal, and spent filter packaging operations are controlled remotely and/or from a local control console for filter replacement and spent resin dewatering. Lubricants and waste solvents drainage is collected in the area sump tanks which are specially designed to provide staging and gravitational oil separation. The separated oils are transferred directly into disposable drums.

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 SWMS.

Interlocks

There are no interlocks needed for direct safety functions related to the SWMS.

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.

2.7.4.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.4.3-1 describes the ITAAC for the SWMS.

2.9.1.5 Implementation and Operation

2.9.1.5.1 Design Implementation

The objective of the HSI design implementation is to demonstrate that the HSI design that is implemented (i.e., the “as-built” design) accurately reflects the verified and validated design.

The scope of HSI design implementation includes the effect on personnel performance resulting from design changes and provides the necessary support to ensure safe operations and that the as-built design conforms to the verified and validated design that resulted from the HFE process.

The referenced changes after V&V apply to the changes made to the US-APWR design following V&V.

Facility design changes are documented and analyzed for their potential impact on HSIs. Those design implementation issues that negatively impact human performance are identified as HEDs and are tracked and dispositioned. HFE design modifications are documented in a periodic status report.

2.9.1.5.2 Human Performance Monitoring

Human performance monitoring applies after the plant is in operation. Human performance monitoring within the scope of this program specifically applies to the following:

- Time critical operator actions
- Correct diagnosis of abnormal plant events
- Accuracy of procedure execution

Monitoring of human performance in other areas is within the scope of other plant programs (such as, “Fitness for Duty”).

Human Performance issues are identified as HEDs and are tracked and dispositioned in accordance with the site specific QA program. HED disposition is documented in a periodic status report.

2.9.2 Inspection, Tests, Analyses, and Acceptance Criteria

Table 2.9-1 describes the ITAAC for HFE.

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. ~~HFE program is implemented by a qualified HFE design team.~~Deleted.	1. ~~An analysis will be performed of the experience and training records of HFE design team.~~Deleted	1. ~~HFE program is implemented by a qualified HFE design team.~~Deleted
2. ~~Operating experience review (OER) implements the following process:~~ ~~Extracting and screening HFE-related issues to identify those relevant to HSI System.~~ ~~Evaluating relevant issues.~~ ~~Conducting HFE issues resolution process~~Deleted.	2. ~~An analysis of the OER process will be performed.~~Deleted.	2. ~~The OER evaluation is performed, and associated HFE issues and resolutions have been entered into the HFE Issues tracking system.~~Deleted.
3. Human reliability analysis (HRA) is conducted as an integrated activity to support both the HFE design process and Probabilistic Risk Assessment (PRA) activities.	3. The HRA will be performed.	3. An HRA report exists which contains the following: –Critical Human Actions (HAs) extracted from the PRA results –An evaluation of these HAs which concludes one of the following: ・The assumptions in the PRA regarding the HSI design and the operating procedures are correct and therefore form a sound basis for human error probabilities. ・The assumptions in the PRA are not correct. The HFE issue tracking system manages these issues as further evaluation items (see ITAAC #9). In addition the HRA report provides requirements for subsequent HFE program elements (e.g., detailed HSI design, validation testing, human performance monitoring) to ensure these HAs are properly considered throughout the HSI design life cycle.~~The HRA provides inputs for other HFE design process elements and the optimization of the HSI design.~~

Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. A functional allocation and functional requirements analysis (FA/FRA) is performed to ensure that safety functions are assigned properly as human actions (HAs) or to automated systems.	4. The FA/FRA will be performed.	4. A FA/FRA report exists in which the safety function allocations are evaluated according to human factor perspective using past experience and/or engineering analysis to conclude one of the following: –The safety function is properly assigned as HAs or to automated systems. –The safety function is not properly assigned. The HFE issue tracking system manages these issues as further evaluation items (see ITAAC #9).~~The safety functions is properly assigned as the HAs or to automated systems.~~

Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Task analysis is performed in accordance with the task analysis implementation plan, and includes the following functions: – selected representative and important tasks that affect plant safety from the areas of operations, maintenance, test, inspection, and surveillance – full range of plant operating modes, including startup, normal operations, abnormal and emergency operations, transient conditions, and low-power and shutdown conditions – risk-important human actions that have been found to affect plant risk by means of HRA and PRA importance and sensitivity analyses – internal and external initiating events and actions affecting the PRA Level I and II analyses – human tasks including monitoring of the automated system and execution of backup actions if the system fails	5. The task analysis will be performed.	5. The function-based task analyses are conducted in conformance with the task analysis implementation plan and include the following functions: – selected representative and important tasks that affect plant safety from the areas of operations, maintenance, test, inspection, and surveillance – full range of plant operating modes, including startup, normal operations, abnormal and emergency operations, transient conditions, and low-power and shutdown conditions – risk-important human actions that have been found to affect plant risk by means of HRA and PRA importance and sensitivity analyses – internal and external initiating events and actions affecting the PRA Level I and II analyses – human tasks including monitoring of the automated system and execution of backup actions if the system fails
6. A staffing and qualifications analysis is performed to ensure that personnel are acceptable to permit realistic response to normal and emergency plant conditions.	6. The staffing and qualifications analysis will be performed.	6. A staffing report exists which concludes from a human factors point of view that the staffing and qualifications of plant personnel, are acceptable to perform safety significant tasks for normal and emergency operations.~~The staffing and qualifications of plant personnel are acceptable for normal and emergency operations.~~

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The scope of HSI design, procedures and training, which are developed and/or evaluated by the HFE program, includes operations, accident management, maintenance, tests, inspections and surveillances that are important to safety.	7. An analysis will be performed of the HSI design, procedures, and training for operations, accident management, maintenance, tests, inspections and surveillances.	7. The HSI design, procedures, and training for operations, accident management, maintenance, tests, inspections and surveillances that are important to safety have been developed and/or evaluated by the HFE program.
7a. HSI panels and associated instrumentation, within the scope of the HFE program, comply with quality standards and records.	7a. An analysis will be performed of the panels and associated instrumentation within the scope of the HFE program.	7a. The design documentation exists to verify that panels and associated instrumentation, within the scope of the HFE program, comply with General Design Criteria 1 in Appendix A to ~~10 CFR 70~~10CFR 50 for quality standards and records.
7b. The MCR includes a non safety reactor operator workstation, a non safety supervisor workstation, and a workstation for safety-related displays and controls.	7b. An inspection of the as-built MCR workstations will be performed.	7b. The as-built MCR includes a non safety reactor operator workstation, a non safety supervisor workstation, and a workstation for safety-related displays and controls.
7c. A MCR exists to provide the safety-related and non safety related HSI.	7c. An inspection will be performed of the as-built plant building configuration.	7c. The as-built MCR exists to provide the safety-related and non safety related HSI.
7d. HSI resources available in the MCR include checking the standby condition of equipment before operation, monitoring the plant parameters and identifying plant behavior during operation.	7d. An inspection of the as-built HSI resources available in the as-built MCR will be performed.	7d. The as-built HSI resources in the as-built MCR include the HSI that is needed to check the standby condition of equipment before operation, monitor the plant parameters, and identify plant behavior during operation.
7e. Means are provided in the MCR for manual initiation of protective functions at the system level.	7e. An inspection of the as-built manual initiation functions in the as-built MCR will be performed.	7e. The capability for the as-built manual initiation of protective functions at the system level exists in the as-built MCR.

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 5 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7f. Spatially dedicated continuously visible (SDCV) HSI is provided in the MCR for: – Bypassed or inoperable status indication – Type A and B PAM variables – Safety parameter displays including status of critical safety functions and performance of credited safety systems and preferred non safety systems – Prompting alarms for credited manual operator actions and risk important HAs identified in the HRA – Conventional switches for system level actuation of safety functions	7f. An inspection of the as-built SDCV HSI in the as-built MCR will be performed.	7f. The following minimum inventory of SDCV displays, visual alerts and controls exists for the as-built MCR : –Bypassed or inoperable status indicators on the Large Display Panel for each safety system or function. –Numeric indicators for each Type A and B PAM variable on the Safety VDUs –Status indicators for each critical safety function, and numeric indicators for key parameters which represent the performance of credited safety system and performance of preferred non safety systems on the Large Display Panel –Prompting alarms for credited manual operator actions and risk important HAs identified in the HRA on the Large Display Panel. –Conventional switches for system level actuation of safety functions on Operator Console.~~The minimum inventory of the as-built SDCV displays, visual alerts and controls exists for the as-built MCR that supports the design criteria.~~
7g. Class 1E HSI is provided in the MCR for control of all safety related components and monitoring of all safety-related plant instrumentation.	7g. An inspection of the as-built Class 1E HSI in the as-built MCR will be performed.	7g. The as-built MCR includes the Class 1E HSI for control of all safety related components and monitoring of all safety-related plant instrumentation.
7h. The MCR includes HSI for degraded HSI conditions, including: – Loss of non safety HSI – Loss of safety and non safety HSI due to CCF – Single HSI failures	7h. An inspection of the as-built HSI redundancy and diversity in the as-built MCR will be performed.	7h. The as-built MCR includes alternate HSI for the following degraded HSI conditions: – Loss of non safety HSI – Loss of safety and non safety HSI due to CCF – Single HSI failures

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 6 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7i. A remote shutdown console (RSC) is provided to achieve safe shutdown in the event of evacuation of the MCR. The RSC includes operator workstation(s) from which operators could perform remote shutdown operations.	7i. An inspection of the as-built RSC will be performed.	7i. To achieve safe shutdown in the event of MCR evacuation, the as-built RSC has Operator workstation(s) from which operators could perform shutdown operations. These workstations have the same functions as the MCR operator console for conducting safe shutdown.~~The as-built RSC provides the capability for the operator to achieve safe shutdown.~~
7j. Manual control and monitoring capability is installed at the LCSs (only manned on demand) for the following functions: – On-line testing, radiological protection activities, and required chemical monitoring supporting technical specifications – Maintenance required by technical specifications – Emergency and abnormal response	7j. An inspection of the as-built local control and monitoring functional capability required for the as-built LCSs will be performed.	7j. The as-built LCSs exist at selected locations throughout the plant for the following required functions; –On-line testing, radiological protection activities, and required chemical monitoring supporting technical specifications where HSI is not provided in the MCR. –Maintenance required by technical specifications where HSI is not provided in the MCR. –Emergency and abnormal response for events where MCR HSI cannot be credited.
7k. A TSC exists where effective direction can be given and effective command control can be performed during an emergency.	7k. An inspection of the as-built TSC will be performed.	7k. An as-built TSC exists from which effective direction can be given and effective command control can be exercised during an emergency.
7l. Provisions exist for communications among the MCR, TSC, and EOF; and between the plant, the state and local emergency operations centers, and the field assessment teams; and the appropriate NRC Regional Office Operations Center.	7l. An inspection of the as-built communications functions will be performed.	7l. The as-built functions are made for communications among the MCR, TSC, and EOF; and between the plant and the state and local emergency operations centers, and the field assessment teams; and the appropriate NRC Regional Office Operations Center.
7m. The procedures development process ensures that procedures guide and support human interactions with plant systems and control plant-related events and activities.	7m. An inspection of the as-built procedures development process will be performed.	7m. The as-built procedures exist to support functions important to ensuring plant safety during normal and abnormal operating conditions. These procedures conform to the Procedure Writer's Guide.

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 7 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7n. The training development process ensures that training provided to operations and maintenance personnel is acceptable to maintain plant safety and respond to abnormal plant conditions.	7n. An inspection of the as-built training development process will be performed.	7n. The as-built training program includes plant operations and maintenance activities which are important to maintain plant safety and respond to abnormal plant conditions. The training material conforms the Training Developer's Guide. ~~exists to support functions important to ensuring plant safety during normal and abnormal operating conditions.~~
8. The HFE verification and validation (V&V) program ensures the following: 1) HSI task analysis encompasses a representative range of risk important operational scenarios, events, transients and accidents 2) The inventory and characteristics of the alarms, information, and controls support the tasks generated by the function-based task analyses and the operational sequence analyses, and the HSI design is consistent with the HSI design style guide. 3) The integrated HSI system supports the safe operation of the plant.	8. An inspection of the ~~as-built~~ HFE V&V activities will be performed.	8. The ~~as-built~~HFE V&V program includes the following activities: 1) HSI task support verification includes plant operations and maintenance activities which are important to maintain plant safety and respond to abnormal plant conditions. The training material conforms the Training Developer's Guide. ~~for risk important operational scenarios, events, transients and accidents.~~ 2) HSI design verification demonstrates that the alarms, information, and controls match the display and control requirements generated by the function-based task analyses and the operational sequence analyses, and the HSI design is consistent with the HSI design style guide. 3) Integrated system validation demonstrates that the HSI system supports the safe operation of the plant.
8a. HED resolution during V&V is performed iteratively throughout all V&V activities.	8a. An inspection of the ~~as-built implementation of~~ HED resolution during the HFE~~as-built~~ V&V process will be performed.	8a. HEDs are identified and addressed iteratively throughout all V&V activities and there are no safety significant unresolved HEDs in the final design.~~The as-built HEDs is identified and addressed in the final design.~~

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Table 2.9-1 Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 8 of ~~7~~8)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8b. HSI in the MCR permits execution of tasks by operators to establish operations, accident management, maintenance, test, inspection and surveillances for those systems that are important to safety.	8b. Tests will be performed on the execution of representative tasks by the actual MCR operators.	8b. Test results demonstrate that the as-built MCR HSI can establish operations, accident management, maintenance, test, inspection and surveillances for those systems that are important to safety.
8c. HSI at the RSC permits execution of tasks by operators to establish and maintain cold shutdown.	8c. Tests will be performed on the execution of tasks for the as-built RSC.	8c. Test results demonstrate that actual operators can establish and maintain cold shutdown from the as-built RSC.
9. The HSI design that is implemented (i.e., the “as-built” design) accurately reflects the verified and validated HSI design.	9. An inspection of the as-built ~~HFE~~ HSI design ~~implementation process~~ will be performed.	9. The as-built HSI design reflects the verified and validated design. For any changes from that design, an HSI ~~A HFE~~ design implementation process is performed and documented as described below. ~~The as-built design implementation methodology includes the following criteria:~~ • Aspects of the HSI design that were not addressed in the HSI design V&V are evaluated using an appropriate V&V method. Aspects of the design addressed by this criterion may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator, such as control room lighting and noise • The potential impact on HAs is assessed and a risk significance level is assigned according to the potential impact on plant safety functions~~in accordance with the criteria in Reference 18.11-1~~ • In addition, ~~A~~all ~~HFE-related~~ issues documented in the HFE issue tracking system are verified to be adequately addressed.

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 ~~between as-built the MCR and as-built TSC~~.	2. Walking between the as-built ~~2 areas~~TSC and MCR takes no more than 2 minutes.
3. The TSC provides a habitable workspace environment.	3. See ~~Subsection 2.7.5.4~~ Table 2.7.5.4-2, ITAAC Item 5.	3. See ~~Subsection 2.7.5.4~~ Table 2.7.5.4-2, ITAAC Item 5.
4. Adequate emergency communications systems are in place.~~The means exists for communications from the control room, TSC, and EOF to the NRC headquarters and regional office emergency operations centers (including establishment of the emergency response data system (i.e. ERDS) between the onsite computer system and the NRC Operations Center.)~~	4. See ~~Subsection 2.7.6.10~~ Table 2.7.6.10-1 and Table 2.9-1, ITAAC Items 7.k and 7.l.	4. See ~~Subsection 2.7.6.10~~. Table 2.7.6.10-1 and Table 2.9-1, ITAAC Items 7.k and 7.l.

2.13 DESIGN RELIABILITY ASSURANCE PROGRAM

2.13.1 Design Description

The purposes of the US-APWR design reliability assurance program (D-RAP) are to provide reasonable assurance that:

- The US-APWR is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for the risk-significant structure, system, and components (SSCs).
- The risk-significant SSCs do not degrade to an unacceptable level during plant operations.
- The frequency of transients that challenge risk-significant SSCs is minimized.
- The risk-significant SSCs function reliably when challenged.

The risk-significant SSCs including both safety-related and non safety-related SSCs are identified through utilizing the result of a probabilistic risk assessment (PRA), expert panel, deterministic or other methods for inclusion in the program.

2.13.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.13-1 describes the ITAAC for the D-RAP.

Table 2.13-1 Design Reliability Assurance Program Inspections, Tests, analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The D-RAP provides reasonable assurance that the plant is designed and constructed in a manner that is consistent with the key assumptions and risk insights for risk-significant SSCs.~~1. The D-RAP provides reasonable assurance that the design of the risk-significant SSCs is consistent with the assumptions used in the risk analysis.~~	1. An analysis will confirm the adequacy of the D-RAP.~~1. Inspection will be performed for the existence of a report that establishes the estimated reliability of as-built risk significant SSCs.~~	1. A report exists that includes the following three (3) major elements: · Identification of all as-built SSCs in the scope of the D-RAP · Description of the methodology used to identify the as-built SSCs in scope of the D-RAP · For the as-built SSCs in scope of D-RAP, identify and describe the reliability assurance activities that are accomplished prior to the initial fuel load, which provide reasonable assurance that the plant is designed and constructed in a manner that is consistent with the key assumptions (including reliability and availability assumptions in PRA, when applicable) and risk insights for the risk-significant SSCs.~~1. A report exists and concludes that the estimated reliability of the each as-built SSCs equals or exceeds the assumed reliability and that industry experience with similar SSCs (including operations, maintenance, and monitoring activities) was taken into account in estimating the reliability of the SSCs.~~