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1.0-19	1A	1.2-26	1A
1.0-20	1A	1.2-27	1A
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1.0-23	1A	1.2-30	1A
1.0-24	1A	1.2-31	1A
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1.0-28	1A	Fig. 1.2.1	1
1.0-29	1A	Fig. 1.2.1A	1
1.0-30	Deleted	Fig. 1.2.2	1
1.0-31	Deleted	Fig. 1.2.3	1
1.0-32	Deleted	Fig. 1.2.4	1
1.0-33	Deleted	Fig. 1.2.5	0
1.1-1	1	Fig. 1.2.6	0
1.1-2	1	Fig. 1.2.7	1
1.1-3	1	Fig. 1.2.8	1
1.1-4	1	Fig. 1.2.8A	1
Fig. 1.1.1	0	Fig. 1.2.9	1
Fig. 1.1.1A	1	Fig. 1.2.10	0
Fig. 1.1.2	0	Fig. 1.2.11	0
Fig. 1.1.3	1	Fig. 1.2.12	0
Fig. 1.1.3A	1	Fig. 1.2.13	Deleted
Fig. 1.1.4	1	Fig. 1.2.14	Deleted
Fig. 1.1.5	1	Fig. 1.2.15	Deleted
1.2-1	1A	Fig. 1.2.16a	0
1.2-2	1A	Fig. 1.2.16b	0
1.2-3	1A	Fig. 1.2.16c	0
1.2-4	1A	Fig. 1.2.16d	0
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Fig 1.A 1	0		
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1.C-2	Deleted		
1.C-3	0		
1.C-4	0		
1.C-5	Deleted		
1.C-6	Deleted		
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2.0-23	1	Fig. 2.1.4	0
2.0-24	1	Fig. 2.1.5	0
2.0-25	1	Fig. 2.1.6	Deleted
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2.1-10	1	2.2-21	1
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3.1-14	1A	3.3-8	1
3.1-15	1A	3.3-9	1
3.1-16	1A	3.3-10	Deleted
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3.1-19	1A	3.4-3	1
3.1-20	1A	3.4-4	1
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3.1-25	1A	3.4-9	1
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3.1-39	1A	3.4-23	1
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3.4-59	1	3.4-111	1
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3.4-61	1	3 4-113	1
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3.4-65	1	3 4-117	1
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3.4-68	1	3 4-120	1
3.4-69	1	3 4-121	1
3 4-70	1	3 4-122	1
3.4-71	1	3 4-123	Deleted
3 4-72	1	3 4-124	Deleted
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3.4-74	1	Fig 3 4.2	1
3.4-75	1	Fig 3.4.3	0
3 4-76	1	Fig 3 4.4	0
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Fig. 3.4.9	0	3.5-10	0
Fig. 3.4.10	1	3.5-11	0
Fig. 3.4.11	0	3.5-12	0
Fig. 3.4.12	0	3.5-13	0
Fig. 3.4.13	0	3.5-14	0
Fig. 3.4.14	0	3.5-15	0
Fig. 3.4.15	0	3.5-16	0
Fig. 3.4.16	0	3.5-17	0
Fig. 3.4.16a	0	3.5-18	0
Fig. 3.4.16b	0	3.5-19	0
Fig. 3.4.17	0	Fig. 3.5.1	0
Fig. 3.4.18	0	Fig. 3.5.2	0
Fig. 3.4.19	0	Fig. 3.5.3	0
Fig. 3.4.20	0	Fig. 3.5.4	0
Fig. 3.4.21	0	Fig. 3.5.5	0
Fig. 3.4.22	0	Fig. 3.5.6	0
Fig. 3.4.23	0	Fig. 3.5.7	0
Fig. 3.4.24	0	Fig. 3.5.8	0
Fig. 3.4.25	0	Fig. 3.5.9	0
Fig. 3.4.26	0	3.6-1	1
Fig. 3.4.27	0	3.6-2	1
Fig. 3.4.28	0	3.6-3	1
Fig. 3.4.29	0	3.6-4	1
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Fig. 3.4.31	1	3.6-6	1
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Fig. 3.4.33	1	3.6-8	1
Fig. 3.4.34	1	3.6-9	1
Fig. 3.4.35	1	3.6-10	Deleted
Fig. 3.4.36	1	3.7-1	1
Fig. 3.4.37	1	3.7-2	1
Fig. 3.4.38	1	3.7-3	1
Fig. 3.4.39	1	3.7-4	1
Fig. 3.4.40	1	3.7-5	1
Fig. 3.4.41	1	3.7-6	1
Fig. 3.4.42	1	3.7-7	1
Fig. 3.4.43	1	3.7-8	1
Fig. 3.4.44	1	3.7-9	1
Fig. 3.4.45	1	3.7-10	1
Fig. 3.4.46	1	3.7-11	1
Fig. 3.4.47	1	3.7-12	1
Fig. 3.4-48	1	3.7-13	Deleted
Fig. 3.4-49	1	3.7-14	Deleted
3.5-1	0	3.8-1	1
3.5-2	0	3.8-2	1
3.5-3	0	3.A-1	1
3.5-4	0	3.A-2	1
3.5-5	0	3.A-3	1
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3.A-10	1	3 B-28	0
3.A-11	1	3.B-29	0
3 A-12	1	3.B-30	0
3.A-13	1	3.B-31	0
3 A-14	1	3.B-32	0
3.A-15	1	3.B-33	0
Fig. 3.A 1	0	3.B-34	0
Fig 3 A.2	0	3.B-35	0
Fig. 3 A 3	0	3 B-36	0
Fig 3.A.4	0	3.B-37	0
Fig 3.A.5	0	3.B-38	0
Fig. 3 A.6	0	3.B-39	0
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Fig. 3.A 8	0	3.B-41	0
Fig 3 A 9	0	3 B-42	0
Fig. 3.A.10	0	3 B-43	0
Fig 3.A.11	0	3 B-44	0
Fig. 3 A.12	0	3.B-45	0
Fig. 3 A 13	0	3.B-46	0
Fig. 3.A.14	0	3 B-47	0
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Fig. 3.A.16	0	3 B-49	0
Fig. 3.A.17	0	3 B-50	0
Fig. 3 A.18	0	3.B-51	0
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3 B-1	0	3.B-53	0
3.B-2	0	3 B-54	0
3.B-3	0	3 B-55	0
3.B-4	0	3 B-56	0
3.B-5	0	3.B-57	0
3 B-6	0	3.B-58	0
3.B-7	0	3.B-59	0
3 B-8	0	3 B-60	0
3.B-9	0	3 B-61	0
3 B-10	0	3 B-62	0
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3 B-12	0	3 C-2	0
3.B-13	0	3.C-3	0
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3.B-17	0	3 C-7	0
3.B-18	0	3 C-8	0
3 B-19	0	Fig. 3.C 1	0
3 B-20	0	Fig 3.C 2	0
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3.D-6	1	3.G-11	1
3.D-7	1	3.G-12	1
3.D-8	1	3.G-13	1
3.D-9	1	Fig. 3.G.1	0
3.D-10	1	Fig. 3.G.2	0
3.D-11	1	Fig. 3.G.3	0
3.D-12	1	Fig. 3.G.4	0
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Fig. 3.D.1	0	3.H-3	1
Fig. 3.D.2a	0	3.H-4	1
Fig. 3.D.2b	0	3.H-5	1
Fig. 3.D.2c	0	3.H-6	1
Fig. 3.D.3	0	3.H-7	1
Fig. 3.D.4a	0	Fig. 3.H.1	0
Fig. 3.D.4b	0	3.I-1	1
Fig. 3.D.4c	0	3.I-2	1
Fig. 3.D.5a	0	3.I-3	1
Fig. 3.D.5b	0	3.I-4	1
Fig. 3.D.5c	0	3.I-5	1
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3.E-2	1	3.I-7	1
3.E-3	1	3.I-8	1
3.E-4	1	3.I-9	1
3.E-5	1	3.I-10	1
3.E-6	1	Fig. 3.I.1	0
3.E-7	1	Appendix 3.J	Deleted
3.E-8	1	3.K-1	1
3.E-9	1	3.K-2	1
3.E-10	1	3.K-3	1
Fig. 3.E.1	0	3.K-4	1
Fig. 3.E.2	0	3.K-5	1
Fig. 3.E.3	0	3.K-6	1
3.F-1	0	3.K-7	1
3.F-2	0	3.L-1	1
3.F-3	0	3.L-2	1
3.F-4	0	3.L-3	1
Fig. 3.F.1	0	3.L-4	1
Fig. 3.F.2	0	3.L-5	1
Fig. 3.F.3	0	3.L-6	1
Fig. 3.F.4	0	3.L-7	1
3.G-1	1	3.L-8	1
3.G-2	1	3.L-9	1
3.G-3	1	3.L-10	1
3.G-4	1	3.L-11	1
3.G-5	1	3.L-12	1
3.G-6	1	3.L-13	1
3.G-7	1	3.M-1	1

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3.M-4	1	3.W-8	1
3 M-5	1	3.W-9	1
3.M-6	1	3.W-10	1
3.M-7	1	Fig. 3.W.1	0
3 M-8	1	3 X-1	1
3.M-9	1	3.X-2	1
3.M-10	1	3 X-3	1
3.M-11	1	3 X-4	1
3.M-12	1	3 X-5	1
3.M-13	1	3 X-6	1
3 M-14	1	3 X-7	1
3.M-15	1	3 X-8	1
3.M-16	1	3 X-9	1
3 M-17	1	3 X-10	1
3.M-18	1	Fig 3.X.1	0
3.M-19	Deleted	Fig. 3.X 2	0
Appendix 3.N	Deleted	Fig. 3.X.3	0
Appendix 3 O	Deleted	Fig. 3.X.4	0
Appendix 3.P	Deleted	Fig. 3.X 5	0
Appendix 3 Q	Deleted	3.Y-1	1
Appendix 3.R	Deleted	3.Y-2	1
Appendix 3.S	Deleted	3 Y-3	1
Appendix 3 T	Deleted	3.Y-4	1
3.U-1	1	3 Y-5	1
3 U-2	1	3 Y-6	1
3.U-3	1	3.Y-7	1
3.U-4	1	3.Y-8	1
3 U-5	1	3 Y-9	1
3.U-6	1	3 Y-10	1
3 U-7	1	3 Y-11	1
3.U-8	1	3 Y-12	1
3.U-9	1	3 Y-13	1
3.U-10	1	3 Y-14	1
Fig. 3.U.1	0	3.Y-15	1
3.V-1	1	3 Y-16	1
3.V-2	1	3 Y-17	1
3 V-3	1	3 Y-18	1
3.V-4	1	3.Y-19	1
3 V-5	1	3 Y-20	1
3 V-6	1	Fig. 3.Y.1	0
3 V-7	1	Fig 3 Y.2	0
3 V-8	1	3.Z-1	0
3 V-9	1	3.Z-2	0
3.V-10	1	3.Z-3	0
Fig 3.V.1	1	3.Z-4	0
3.W-1	1	3.Z-5	0
3.W-2	1	3.Z-6	0
3.W-3	1	3 Z-7	0
3 W-4	1	3.Z-8	0
3.W-5	1	3.Z-9	0

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3.Z-12	0	3.AD-4	1
Fig. 3.Z.1	0	3.AD-5	1
Fig. 3.Z.2	0	3.AD-6	1
Fig. 3.Z.3	0	3.AD-7	1
Fig. 3.Z.4	0	3.AD-8	1
Fig. 3.Z.5	0	3.AD-9	1
Fig. 3.Z.6	0	3.AD-10	1
3.AA-1	1	3.AD-11	1
3.AA-2	1	3.AD-12	1
3.AA-3	1	3.AD-13	1
3.AA-4	1	3.AD-14	1
3.AA-5	1	3.AD-15	1
3.AA-6	1	3.AD-16	1
3.AA-7	1	3.AD-17	1
3.AA-8	1	3.AD-18	1
Fig. 3.AA.1	0	3.AD-19	1
Fig. 3.AA.2	0	Fig. 3.AD.1	0
Fig. 3.AA.3	0	Fig. 3.AD.2	0
Fig. 3.AA.4	0	Fig. 3.AD.3	0
Fig. 3.AA.5	0	3.AE-1	1
Fig. 3.AA.6	0	3.AE-2	1
Fig. 3.AA.7	0	3.AE-3	1
Fig. 3.AA.8	0	3.AE-4	1
3.AB-1	1	3.AE-5	1
3.AB-2	1	3.AE-6	1
3.AB-3	1	3.AE-7	1
3.AB-4	1	Fig. 3.AE.1	1
3.AB-5	1	Fig. 3.AE.1b	Deleted
3.AB-6	1	Fig. 3.AE.1c	Deleted
3.AB-7	1	Fig. 3.AE.2	1
3.AB-8	1	Fig. 3.AE.3	0
3.AB-9	1	Fig. 3.AE.4	Deleted
3.AB-10	1	3.AF-1	1
3.AB-11	1	3.AF-2	1
3.AB-12	1	3.AF-3	1
3.AB-13	1	3.AF-4	1
3.AB-14	1	3.AF-5	1
3.AC-1	1	3.AF-6	1
3.AC-2	1	3.AF-7	1
3.AC-3	1	3.AF-8	1
3.AC-4	1	3.AG-1	1
3.AC-5	1	3.AG-2	1
3.AC-6	1	3.AG-3	1
3.AC-7	1	3.AG-4	1
3.AC-8	1	3.AG-5	1
3.AC-9	1	3.AG-6	1
3.AC-10	1	3.AG-7	1
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3.AG-15	1	3 AJ-9	1
3 AG-16	1	3.AJ-10	1
3 AG-17	1	3 AJ-11	1
3 AG-18	1	3.AJ-12	1
3 AG-19	1	3.AJ-13	1
3 AG-20	1	3.AJ-14	1
3 AG-21	1	3.AJ-15	1
3 AG-22	1	3.AJ-16	1
3 AG-23	1	3 AJ-17	1
3 AG-24	1	3.AJ-18	1
3.AH-1	1	3 AJ-19	1
3 AH-2	1	Fig 3.AJ.1	0
3 AH-3	1	Fig. 3.AJ.2	0
3.AH-4	1	Fig 3 AJ.3	0
3.AH-5	1	3 AK-1	1
3 AH-6	1	3 AK-2	1
3.AH-7	1	3.AK-3	1
3.AH-8	1	3 AK-4	1
3 AH-9	1	3 AK-5	1
3 AI-1	1	3.AK-6	1
3 AI-2	1	3.AK-7	1
3 AI-3	1	3.AK-8	1
3 AI-4	1	3.AK-9	1
3.AI-5	1	3.AK-10	1
3 AI-6	1	3 AK-11	1
3 AI-7	1	3 AK-12	1
3.AI-8	1	3 AK-13	1
3.AI-9	1	3 AK-14	1
3.AI-10	1	3 AK-15	1
3.AI-11	1	3 AK-16	1
3.AI-12	1	3 AK-17	1
3.AI-13	1	3.AK-18	1
3.AI-14	1	3.AL-1	1
3 AI-15	1	3.AL-2	1
3 AI-16	1	3.AL-3	1
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3.AI-18	1	3 AL-5	1
3.AI-19	1	3 AL-6	1
Fig. 3.AI.1	0	3.AL-7	1
Fig. 3.AI 2	0	3 AL-8	1
Fig 3 AI.3	0	3.AL-9	1
Fig. 3.AI.4	0	3 AL-10	1
Fig 3 AI.5	0	3 AM-1	0
Fig. 3 AI.6	0	3 AM-2	0
3.AJ-1	1	3 AM-3	0
3.AJ-2	1	3 AM-4	0
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3 AJ-4	1	3.AM-6	0

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3.AM-8	0	Fig. 3.AN.16	0
3.AM-9	0	Fig. 3.AN.17	0
3.AM-10	0	Fig. 3.AN.18	0
3.AM-11	0	Fig. 3.AN.19	0
3.AM-12	0	Fig. 3.AN.20	0
3.AM-13	0	Fig. 3.AN.21	0
3.AM-14	0	Fig. 3.AN.22	0
3.AM-15	0	Fig. 3.AN.23	0
3.AM-16	0	Fig. 3.AN.24	0
3.AM-17	0	Fig. 3.AN.25	0
3.AM-18	0	Fig. 3.AN.26	0
3.AM-19	0	Fig. 3.AN.27	0
3.AM-20	0	Fig. 3.AN.28	0
3.AM-21	0	Fig. 3.AN.29	0
3.AM-22	0	Fig. 3.AN.30	0
3.AM-23	0	3.AO-1	Not Used
3.AM-24	0	3.AP-1	Not Used
3.AM-25	0	3.AQ-1	1
3.AM-26	0	3.AQ-2	1
3.AM-27	0	3.AQ-3	1
3.AM-28	0	3.AQ-4	1
3.AM-29	0	3.AQ-5	1
3.AM-30	0	3.AQ-6	1
3.AN-1	1A	3.AQ-7	1
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3.AN-5	1A	3.AR-1	1
3.AN-6	1A	3.AR-2	1
3.AN-7	1A	3.AR-3	1
3.AN-8	1A	3.AR-4	1
3.AN-9	1A	3.AR-5	1
3.AN-10	1A	3.AR-6	1
3.AN-11	1A	3.AR-7	1
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3.AN-13	1A	3.AR-9	1
3.AN-14	1A	3.AR-10	1
Fig. 3.AN.1	0	3.AR-11	1
Fig. 3.AN.2	0	3.AS-1	1
Fig. 3.AN.3	0	3.AS-2	1
Fig. 3.AN.4	0	3.AS-3	1
Fig. 3.AN.5	0	3.AS-4	1
Fig. 3.AN.6	0	3.AS-5	1
Fig. 3.AN.7	0	3.AS-6	1
Fig. 3.AN.8	0	3.AS-7	1
Fig. 3.AN.9	0	3.AS-8	1
Fig. 3.AN.10	0	3.AS-9	1
Fig. 3.AN.11	0	3.AS-10	1
Fig. 3.AN.12	0	3.AS-11	1
Fig. 3.AN.13	0	3.AS-12	1
Fig. 3.AN.14	0	3.AS-13	1

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4 1-2	1	4.4-7	1A
4 1-3	1	4.4-8	1A
4.1-4	1	4 4-9	1A
4 1-5	1	4 4-10	1A
4.1-6	1	4 4-11	1A
4.1-7	1	4.4-12	1A
4 2-1	1	4 4-13	1A
4 2-2	1	4 4-14	1A
4.2-3	1	4 4-15	1A
4.2-4	1	4 4-16	1A
4.2-5	1	4.4-17	1A
4.2-6	1	4.4-18	1A
4.2-7	1	4.4-19	1A
4.2-8	1	4 4-20	1A
4.2-9	1	4 4-21	1A
4.2-10	1	4 4-22	1A
4.2-11	1	4 4-23	1A
Fig. 4.2.1	0	4.4-24	1A
Fig. 4.2.2	0	4.4-25	1A
4.3-1	1	4.4-26	1A
4.3-2	1	4.4-27	1A
4.3-3	1	4.4-28	1A
4.3-4	1	4.4-29	1A
4.3-5	1	4.4-30	1A
4.3-6	1	4.4-31	1A
4.3-7	1	4.4-32	1A
4.3-8	1	4.4-33	1A
4.3-9	1	4.4-34	1A
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4.3-12	1	4.4-37	1A
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4.3-14	1	4.4-39	1A
4.3-15	1	4 4-40	1A
4.3-16	1	4 4-41	1A
4.3-17	1	4.4-42	1A
4.3-18	1	4.4-43	1A
4.3-19	1	4.4-44	1A
4.3-20	1	4 4-45	1A
4.3-21	1	4.4-46	1A
4.3-22	1	4.4-47	1A
4.3-23	Deleted	4.4-48	1A
Fig. 4.3.1	0	4.4-49	1A
Fig. 4.3.2	0	4.4-50	1A
Fig. 4.3.3	0	4.4-51	1A
Fig 4.3.4	0	4.4-52	1A
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4.4-60	1A	4.5-18	1A
4.4-61	1A	4.5-19	1A
4.4-62	1A	4.5-20	1A
4.4-63	1A	4.5-21	1A
4.4-64	1A	4.5-22	1A
4.4-65	1A	4.5-23	1A
4.4-66	1A	4.5-24	1A
4.4-67	1A	4.5-25	1A
4.4-68	1A	4.5-26	1A
Fig. 4.4.1	0	Fig. 4.5.1	0
Fig. 4.4.2	0	Fig. 4.5.2	1
Fig. 4.4.3	0	Fig. 4.5.3	Deleted
Fig. 4.4.4	0	4.6-1	0
Fig. 4.4.5	0	4.6-2	0
Fig. 4.4.6	0	4.7-1	1
Fig. 4.4.7	1	4.7-2	1
Fig. 4.4.8	0	4.7-3	1
Fig. 4.4.9	1	4.A-1	1A
Fig. 4.4.10	0	4.A-2	1A
Fig. 4.4.11	0	4.A-3	1A
Fig. 4.4.12	0	4.A-4	1A
Fig. 4.4.13	0	4.A-5	1A
Fig. 4.4.14	Deleted	4.A-6	1A
Fig. 4.4.15	0	4.A-7	1A
Fig. 4.4.16	1	4.A-8	1A
Fig. 4.4.17	1	4.A-9	1A
Fig. 4.4.18	0	4.A-10	1A
Fig. 4.4.19	1	4.A-11	1A
Fig. 4.4.20	1	4.A-12	1A
Fig. 4.4.21	0	4.A-13	1A
Fig. 4.4.22	Deleted	4.A-14	1A
Fig. 4.4.23	Deleted	4.A-15	1A
Fig. 4.4.24	0	4.A-16	1A
Fig. 4.4.25	1	4.A-17	1A
Fig. 4.4.26	1	4.A-18	1A
4.5-1	1A	4.A-19	1A
4.5-2	1A	4.A-20	1A
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4.5-9	1A	Fig. 4.A.4	1
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5.1-3	1	5.2-23	1
5.1-4	1	5.2-24	1
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5.1-6	1	5.2-26	1
5.1-7	1	5.2-27	1
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5.1-9	1	5.2-29	1
5.1-10	1	5.2-30	1
5.1-11	1	5.2-31	1
5.1-12	1	5.2-32	1
5.1-13	1	5.2-33	1
5.1-14	1	5.2-34	1
5.1-15	1	5.2-35	1
5.1-16	1	5.2-36	1
5.1-17	1	5.2-37	1
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Fig. 5.1.4	0	5.2-43	1
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Fig. 5.1.6	0	5.2-45	1
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Fig. 5.1.8	0	5.2-47	1
Fig. 5.1.9	0	5.2-48	1
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Fig. 5.1.12	1	5.2-51	1
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5.2-11	1	5.3-9	1
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Fig 5.3.7	1	5.6-3	1
Fig. 5.3.8	0	5.A-1	0
Fig. 5.3.9	0	5.A-2	0
Fig. 5.3.10	1	5.A-3	0
Fig. 5.3.11	1	5.B-1	0
Fig. 5.3.12	0	5.B-2	0
Fig. 5.3.13	0	5 B-3	0
Fig. 5.3.14	1	5 B-4	0
Fig. 5.3.15	1	5 B-5	0
Fig. 5.3.16	1	5 B-6	0
Fig 5.3.17	1	5 B-7	0
Fig 5.3.18	1	5.C-1	0
Fig. 5.3.19	1	5.C-2	0
Fig. 5.3-20	1	5.C-3	0
Fig. 5.3-21	1	5 C-4	0
5 4-1	1	5 C-5	0
5 4-2	1	5.C-6	0
5.4-3	1	5.C-7	0
5.4-4	1	5 C-8	0
5.4-5	1	5 C-9	0
5.4-6	1	5.C-10	0
5.4-7	1	5.C-11	0
5.4-8	1	5.C-12	0
5 4-9	1	5.C-13	0
5 4-10	1	5 C-14	0
5 4-11	1	5.C-15	0
5.4-12	1	5.C-16	0
5.4-13	1	5.C-17	0
5.4-14	1	5.C-18	0
5.4-15	1	5.C-19	0
5.4-16	1	5.C-20	0
5.4-17	1	5.C-21	0
5.4-18	1	5 C-22	0
5 4-19	1	5.C-23	0
5 4-20	1	5.C-24	0
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5.4-24	1	5.C-28	0
5.4-25	1	5 C-29	0
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5 4-27	1	5 C-31	0
5 4-28	1	5.C-32	0
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5 4-30	1	5.C-34	0
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6.1-13	1	6.2-47	1
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Appendix 13.A	Deleted		
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CHAPTER 1[†]: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) for Holtec International's HI-STORM 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under requirements specified in 10CFR72 [1.0.1]. This FSAR describes the basis for NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L, for the HI-STORM 100 System to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI). This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM 100 System, drawings of the structures, systems, and components important to safety, and the qualifications of the certificate holder. This report is also suitable for incorporation into a site-specific Safety Analysis Report which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility similar in objective and scope. Table 1.0.1 contains a listing of the terminology and notation used in this FSAR.

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable FSAR section that addresses each topic.

The HI-STORM 100 FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all guidance. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.3 herein. Table 1.0.3 also contains the justification for the alternative method for compliance adopted in this FSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

[†] This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

The generic design basis and the corresponding safety analysis of the HI-STORM 100 System contained in this FSAR are intended to bound the SNF characteristics, design, conditions, and interfaces that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This FSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design basis and safety analysis documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM 100 System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM 100 System FSAR identifies a limited number of conditions that are necessarily site-specific and are to be addressed in the licensee's 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage pad (including the embedment for anchored cask users) and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.
- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.
- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.
- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.
- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.
- Performance of pre-operational testing.
- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.
- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

The generic safety analyses contained in the HI-STORM 100 FSAR may be used as input and for guidance by the licensee in performing a 10CFR72.212 evaluation.

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.2.3 is the third figure in Section 1.2 of Chapter 1.

Revisions to this document are made on a section level basis. Complete sections have been replaced if any material in the section changed. The specific changes are noted with revision bars in the right margin. Figures are revised individually. Drawings are controlled separately within the Holtec QA program and have individual revision numbers. Bills-of-Material (BOMs) are considered separate drawings and are not necessarily at the same revision level as the drawing(s) to which they apply. If a drawing or BOM was revised in support of the current FSAR revision, that drawing/BOM is included in Section 1.5 at its latest revision level. Drawings and BOMs appearing in this FSAR may be revised between formal updates to the FSAR. Therefore, the revisions of drawings/BOMs in Section 1.5 may not be current.

1.0.1 Engineering Change Orders

The changes authorized by the following Holtec Engineering Change Orders (ECOs) are reflected in Revision 1 of this FSAR:

MPC-68/68F/68FF: ECOs 1021-1 through 4, 7, 8, 12 through 16, 18 through 23, 27 through 30, 33, 34, 36, 38, 39, 41, 43, and 44; and 71188-43.

MPC-24/24E/24EF: ECOs 1022- 1 through 7, 9, 10, 12 through 26, 28, 31, and 34 through 38.

MPC-32: ECOs 1023-1 and 3 through 10.

HI-STORM overpack: ECOs 1024-1 through 4, 6 through 16, 18 through 21, 24, 25, 27 through 38, 42 through 47, 50, 51, 52, 54 through 58, and 60.

HI-TRAC 125 transfer cask: ECOs 1025-1 through 32, 35, and 36.

HI-TRAC 100 transfer cask: ECOs 1026- 1 through 29.

Ancillary Equipment: ECOs 1027-27, 31, 33, 46, and 53.

General FSAR changes: ECOs 5014-36, 47, 49, 50, 51, 53, 54, 56, 58 through 64, 66, 67, and 68.

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

BoralTM means Boral manufactured by AAR Advanced Structures.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

Cooling Time for a spent fuel assembly is the time between its discharge from the reactor (reactor shutdown) and the time the spent fuel assembly is loaded into the MPC.

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container (or Canister) means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The Damaged Fuel Container/Canister (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this FSAR, if operated and maintained in accordance with this FSAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM 100 System.

Enclosure Vessel means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, and closure ring which provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Fracture Toughness is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

Fuel Basket means a honeycombed structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Debris refers to ruptured fuel rods, severed rods, loose fuel pellets, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

High Burnup Fuel is a spent fuel assembly with an average burnup greater than 45,000 MWD/MTU.

HI-TRAC transfer cask or HI-TRAC means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. HI-TRAC is an acronym for Holtec International Transfer Cask. In this FSAR there are three HI-TRAC transfer casks, the 125 ton standard design HI-TRAC (HI-TRAC-125), the 125-ton dual-purpose lid design (HI-TRAC 125D), and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to all three HI-TRAC transfer cask design, unless the discussion requires distinguishing among the three. The HI-TRAC is equipped with a pair of lifting trunnions and the HI-TRAC 100 and HI-TRAC 125 designs also include pocket trunnions. The trunnions are used to lift and downend/upend the HI-TRAC with a loaded MPC.

TERMINOLOGY AND NOTATION

HI-STORM overpack or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. The term "overpack" as used in this FSAR refers to both the standard design overpack (HI-STORM 100), the alternate design overpack (HI-STORM 100S), and either of these as an overpack designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA), unless otherwise clarified.

HI-STORM 100 System consists of a loaded MPC placed within the HI-STORM 100 overpack.

HoltiteTM is the trade name for all present and future neutron shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-ATM is the first and only shielding material qualified under the Holtite R&D program. As such, the terms Holtite and Holtite-A may be used interchangeably throughout this FSAR.

HoltiteTM -A is a trademarked Holtec International neutron shield material.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

License Life means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

METCON™ is a trade name for the HI-STORM 100 overpack. The trademark is derived from the metal-concrete composition of the HI-STORM 100 overpack.

MGDS is an acronym for Mined Geological Disposal System.

Moderate Burnup Fuel is a spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions.

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), water displacement guide tube plugs, and orifice rod assemblies.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Plain Concrete is concrete that is unreinforced and is of density specified in this FSAR.

Preferential Fuel Loading is a requirement in the CoC applicable to uniform fuel loading whenever fuel assemblies with significantly different post-irradiation cooling times (≥ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling time are loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times are placed toward the center of the basket. Regionalized fuel loading meets the intent of preferential fuel loading. Preferential fuel loading is a requirement in addition to other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers.

TERMINOLOGY AND NOTATION

Post-Core Decay Time (PCDT) is synonymous with cooling time.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high heat emitting fuel assemblies to be stored in fuel storage locations in the center of the fuel basket provided lower heat emitting fuel assemblies are stored in the peripheral fuel storage locations. Users choosing regionalized fuel loading must also consider other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers. Regionalized fuel loading meets the intent of preferential fuel loading.

SAR is an acronym for Safety Analysis Report (10CFR71).

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Single Failure Proof means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

Thermosiphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.

FSAR is an acronym for Final Safety Analysis Report (10CFR72).

Uniform Fuel Loading is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as preferential fuel loading, and those applicable to non-fuel hardware, and damaged fuel containers.

ZPA is an acronym for zero period acceleration.

Table 1.0.2

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
1. General Description			
1.1 Introduction	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.1
1.2 General Description	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2
1.2.1 Cask Characteristics	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.1
1.2.2 Operational Features	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.2
1.2.3 Cask Contents	1.III.3 DCSS Contents	10CFR72.2(a)(1) 10CFR72.236(a)	1.2.3
1.3 Identification of Agents & Contractors	1.III.4 Qualification of the Applicant	10CFR72.24(j) 10CFR72.28(a)	1.3
1.4 Generic Cask Arrays	1.III.1 General Description & Operational Features	10CFR72.24(c)(3)	1.4
1.5 Supplemental Data	1.III.2 Drawings	10CFR72.24(c)(3)	1.5
NA	1.III.6 Consideration of Transport Requirements	10CFR72.230(b) 10CFR72.236(m)	1.1
NA	1.III.5 Quality Assurance	10CFR72.24(n)	1.3
2. Principal Design Criteria			
2.1 Spent Fuel To Be Stored	2.III.2.a Spent Fuel Specifications	10CFR72.2(a)(1) 10CFR72.236(a)	2.1
2.2 Design Criteria for Environmental Conditions and Natural Phenomena	2.III.2.b External Conditions, 2.III.3.b Structural, 2.III.3.c Thermal	10CFR72.122(b)	2.2
		10CFR72.122(c)	2.2.3.3, 2.2.3.10
		10CFR72.122(b)(1)	2.2
		10CFR72.122(b)(2)	2.2.3.11
		10CFR72.122(h)(1)	2.0
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.5

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b) (2)	2.2.3.6
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b) (2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b) (2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a) (4)	2.3.2.2, 4.0
NA	2.III.3f Operating Procedures	10CFR72.24(f) 10CFR72.128(a) (5) 10CFR72.236(h) 10CFR72.24(1)(2) 10CFR72.236(1) 10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0 8.0 1.2.1, 1.2.2 2.3.2.1 10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a) (1)	9.0
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a) (2)	2.3.5.2
		10CFR72.128(a) (3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
10CFR72.236(e)	2.3.2.1		
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h) (4) 10CFR72.122(i) 10CFR72.128(a) (1)	2.3.5
2.3.4 Nuclear Criticality Safety	2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236(h)	2.4
	14.III.1 Design	10CFR72.130	2.4
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(1)

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	14.III.4 License Termination	10CFR72.54	(1)
3. Structural Evaluation			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.24(c)	3.1
3.2 Weights and Centers of Gravity	3.V.1.b.2 Structural Design Features	--	3.2
3.3 Mechanical Properties of Materials	3.V.1.c Structural Materials	10CFR72.24(c)(3)	3.3
	3.V.2.c Structural Materials		
NA	3.III.2 Radiation Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(l)	3.4.4.3 3.4.7.3 3.4.10
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4.3
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.102(f)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.236(g)	3.4.11 3.4.12
3.4 General Standards for Casks	--	--	3.4
3.4.1 Chemical and Galvanic Reactions	3.V.1.b.2 Structural Design Features	--	3.4.1
3.4.2 Positive Closure	--	--	3.4.2
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions --	--	3.4.3, Appendices 3.E, 3.AC, 3.D
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4, Appendices 3.I, 3.U, 3.V, 3.W

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h) (1)	3.5
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a) (4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria	10CFR72.122(h) (1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
5. Shielding Evaluation			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
5.3.1 Description of the Radial and Axial Shielding Configurations	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a) (2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
6. Criticality Evaluation			
6.1 Discussion and Results	--	--	6.1
6.2 Spent Fuel Loading	6.V.2 Fuel Specification	--	6.1, 6.2
6.3 Model Specifications	6.V.3 Model Specification	--	6.3
6.3.1 Description of Calculational Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6 Supplemental Data	6.V.5 Supplemental Info.	--	Appendices 6.B, 6.C, and 6.D
7. Confinement			
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety	10CFR72.24(c)(3) 10CFR72.24(1)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h) (l)	7.1, 7.1.1, 7.2.2
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.236(1)	7.2
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7.2.1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h) (4) 10CFR72.128(a) (l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	7.III.8 Annual Dose	10CFR72.104(a)	7.3.5
7.2.2 Pressurization of Confinement Vessel	--	--	7.2.2
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.3
7.3.1 Fission Gas Products	--	--	7.3.1
7.3.2 Release of Contents	--	--	7.3.3
NA	--	10CFR72.106(b)	7.3
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.1.5, 8.5.2
	8.III.4 Written Procedures	10CFR72.212(b) (9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0 Introduction
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0 Introduction
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	8.1, 8.3
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.3
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.3
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.3.3

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	8.III.4 Written Procedures	10CFR72.212(b) (9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.8 Ready Retrieval	10CFR72.122(1)	8.3
8.3 Preparation of the Cask	--	--	8.3.2
8.4 Supplemental Data	--	--	Tables 8.1.1 to 8.1.10
NA	8.III.9 Design to Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a) (5)	8.1, 8.3
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 8.1.6
9. Acceptance Criteria and Maintenance Program			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	8.1, 9.1
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	9.1
	9.III.1.d Test Program	10CFR72.162	9.1
	9.III.1.e Appropriate Tests	10CFR72.236(1)	9.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1 ⁽²⁾
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a) (1)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b) (8)	9.2

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	(3)
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	(4)
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	9.1.7, 9.1.1.(12)
	9.III.3 Cask Identification		
10. Radiation Protection			
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	10.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
11. Accident Analyses			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1

Table 1.0.2 (continued)

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
12. Operating Controls and Limits			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart E 10CFR72 Subpart F	12.0

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 12.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 12.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SCC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	8.0
NA	12.III.2 Confinement Effectiveness	10CFR72.236(j)	9.0
NA	12.III.2 Evaluation for Confinement	10CFR72.236(l)	7.1, 7.2, 9.0
13. Quality Assurance			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Notes:

- (1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application.
- (2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission's acceptance of the tests defined in Chapter 9.
- (3) Not applicable to HI-STORM 100 System. The functional adequacy of all important to safety components is demonstrated by analyses.
- (4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.
- (5) The stated requirement is not applicable to the HI-STORM 100 System. No monitoring is required for accident conditions.
- “—” There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.
- “NA” There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>2.V.2.(b)(1) "The NRC accepts as the maximum and minimum "normal" temperatures the highest and lowest ambient temperatures recorded in each year, averaged over the years of record."</p>	<p><u>Exception:</u> Section 2.2.1.4 for environmental temperatures utilizes an upper bounding value of 80°F on the annual average ambient temperatures for the United States.</p>	<p>The 80°F temperature set forth in Table 2.2.2 is greater than the annual average ambient temperature at any location in the continental United States. Inasmuch as the primary effect of the environmental temperature is on the computed fuel cladding temperature to establish long-term fuel cladding integrity, the annual average ambient temperature for each ISFSI site should be below 80°F. The large thermal inertia of the HI-STORM 100 System ensures that the daily fluctuations in temperatures do not affect the temperatures of the system. Additionally, the 80°F ambient temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours.</p>
<p>2.V.2.(b)(3)(f) "10CFR Part 72 identifies several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for spent fuel storage."</p>	<p><u>Clarification:</u> A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM 100 System must be performed prior to use if these events are applicable to the site.</p>	<p>In accordance with NUREG-1536, 2.V.(b)(3)(f), if seiche, tsunami, and hurricane are not addressed in the SAR and they prove to be applicable to the site, a safety analysis is required prior to approval for use of the DCSS under either a site specific, or general license.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."</p>	<p><u>Clarification:</u> As stated in NUREG-1536, 3.V.(d), page 3-11, "Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted outside the spent fuel building, or the maximum deceleration that the cask could experience in a drop." The maximum deceleration for a corner drop is specified as 45g's for the HI-STORM overpack. No carry height limit is specified for the corner drop.</p>	<p>In Chapter 3, the MPC and HI-STORM overpack are evaluated under a 45g radial loading. A 45g axial loading on the MPC is bounded by the analysis presented in the HI-STAR FSAR, Docket 72-1008, under a 60g loading, and is not repeated in this FSAR. In Chapter 3, the HI-STORM overpack is evaluated under a 45g axial loading. Therefore, the HI-STORM overpack and MPC are qualified for a 45g loading as a result of a corner drop. Depending on the design of the lifting device, the type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses may be required to demonstrate that the deceleration limit of 45g's is not exceeded.</p>
<p>3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced..."</p> <p>3.V.2.b.i.(2)(b), Page 3-20, Para. 1, "The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359".</p> <p>3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and construction of reinforced concrete structures important to safety but not within the scope of ACI 359 should comply with the</p>	<p><u>Exception:</u> The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used for the material selection and specification, and construction of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack. ACI 318-95 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.</p>	<p>Concrete is provided in the HI-STORM overpack solely for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose.</p> <p>To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. In addition, the temperature limits for normal and off-normal condition from ACI 349 will be imposed.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
requirements of ACI 349".		Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.
3.V.3.b.i.(2), Page 3-29, Para. 1, "The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with Section III of the ASME B&PV Code."	<u>Clarification:</u> The HI-STORM overpack steel structure is designed in accordance with the ASME B&PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.	The overpack structure is a steel weldment consisting of "plate and shell type" members. As such, it is appropriate to design the structure to Section III, Class 3 of Subsection NF. The very same approach has been used in the structural evaluation of the "intermediate shells" in the HI-STAR 100 overpack (Docket Number 72-1008) previously reviewed and approved by the USNRC.
<p>4.IV.5, Page 4-2 "for each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage."</p> <p>4.IV.1, Page 4-3, Para 1 "the staff should verify that cladding temperatures for each fuel type proposed for storage will be below the expected damage thresholds for normal conditions of storage."</p> <p>4 IV.1, Page 4-3, Para. 2 "fuel cladding limits for each fuel type should be defined in the SAR with thermal restrictions in the DCSS technical specifications."</p> <p>4.V.1, Page 4-3, Para. 4 "the applicant should verify that these cladding temperature limits are appropriate for all fuel types proposed for storage,</p>	<u>Clarification:</u> As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak fuel cladding temperature limit.	As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak fuel cladding temperature limit. All major variations in fuel parameters are considered in the determination of the peak fuel cladding temperature limits. Minor variations in fuel parameters within an array type are bounded by the conservative determination of the peak fuel cladding temperature limit.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
and that the fuel cladding temperatures will remain below the limit for facility operations (e.g., fuel transfer) and the worst-case credible accident."		
4.V.4.a, Page 4-6, Para. 3 "applicants seeking NRC approval of specific internal convection models should propose, in the SAR, a comprehensive test program to demonstrate the adequacy of the cask design and validation of the convection models."	<u>Exception:</u> The natural convection model described in Subsection 4.4.1. is based on classical correlations for natural convection in differentially heated cavities which have been validated by many experimental studies. Therefore, no additional test program is proposed.	Many experimental studies of this mechanism have been performed by others and reported in open literature sources. As discussed in Subsection 4.4.1, natural convection has been limited to the relatively large MPC basket to shell peripheral gaps. Subsection 4.4.1 provides sufficient references to experiments which document the validity of the classical correlation used in the analysis.
4.V.4.a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."	<u>Clarification:</u> As discussed in Subsection 4.4.2, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based update of Wooten-Epstein (described in Subsection 4.4.1.1.2)	The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature difference conservatively maximizes the basket wall temperature.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>4.V.4 b, Page 4-7, Para. 2 "if the thermal model is axisymmetric or three-dimensional, the longitudinal thermal conductivity should generally be limited to the conductivity of the cladding (weighted fractional area) within the fuel assembly."</p>	<p><u>Clarification:</u> As described in Subsection 4.4.1.1.4, the axial thermal conductivity of the fuel basket is set equal to the cross-sectional thermal conductivity.</p>	<p>Due to the large number of gaps in the cross-sectional heat transfer paths, use of the fuel basket cross-sectional thermal conductivity for the axial thermal conductivity severely underpredicts the axial thermal conductivity of the fuel basket region. This imposed axial thermal conductivity restriction is even more limiting than that imposed by this requirement of NUREG-1536.</p>
<p>4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."</p>	<p><u>Exception:</u> All calculations of fuel assembly effective thermal conductivities, described in Subsection 4.4.1.1.2, use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.</p>	<p>Within Subsection 4.4.1.1.2, the calculated effective thermal conductivities based on nominal design fuel dimensions are compared with available literature values and are demonstrated to be conservative by a substantial margin.</p>
<p>4.V.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented."</p>	<p><u>Clarification:</u> No additional heat balance is performed or provided.</p>	<p>The FLUENT computational fluid dynamics program used to perform evaluations of the HI-STORM Overpack and HI-TRAC transfer cask, which uses a discretized numerical solution algorithm, enforces an energy balance on all discretized volumes throughout the computational domain. This solution method, therefore, ensures a heat balance at the surface of the cask.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."	<u>Exception:</u> No input or output file listings are provided in Chapter 4.	A complete set of computer program input and output files would be in excess of three hundred pages. All computer files are considered proprietary because they provide details of the design and analysis methods. In order to minimize the amount of proprietary information in the FSAR, computer files are provided in the proprietary calculation packages.
4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures.	<u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) are calculated using maximum weights and minimum densities.	Calculating the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) using maximum weights and minimum densities conservatively overpredicts the volume occupied by the internal components and correspondingly underpredicts the remaining free volume.
7.V.4.c, Page 7-7, Para. 2 and 3 "Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145 is not typically applied." and "Note that for an instantaneous release (and instantaneous exposure), the time that an individual remains at the controlled area boundary is not a factor in the dose calculation."	<u>Exception:</u> As described in Section 7.3, in lieu of an instantaneous release, the assumed leakage rate is set equal to the leakage rate acceptance criteria (5×10^{-6} atm-cm ³ /s) plus 50% for conservatism, which yields 7.5×10^{-6} atm-cm ³ /s. Because the release is assumed to be a leakage rate, the individual is assumed to be at the controlled area boundary for 720 hours. Additionally, the atmospheric dispersion factors of Regulatory Guide 1.145 are applied.	The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g., helium leakage, hydrostatic, and volumetric weld inspection). The NRC letter to Holtec International dated 9/15/97, Subject: Supplemental Request for Additional Information - HI-STAR 100 Dual Purpose Cask System (TAC No. L22019), RAI 7.3 states "use the verified confinement boundary leakage rate in lieu of the assumption that the confinement boundary fails."
9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with	<u>Clarification:</u> Section 9.1.1.1 and the Design Drawings specify that the	In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
NB/NC-5330, "Ultrasonic Acceptance Standards".	ASME Code, Section III, Subsection NB, Article NB-5332 will be used for the acceptance criteria for the volumetric examination of the MPC lid-to-shell weld.	as defined by either the ASME code, or an alternative approach..." The ASME Code, Section III, Subsection NB, Paragraph NB-5332 is appropriate acceptance criteria for pre-service examination.
9.V.1.d, Para. 1 "Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron absorbing material."	<u>Exception:</u> Subsection 9.1.5 describes the control of special processes, such as neutron shield material installation, to be performed in lieu of scanning or probing with neutron sources.	<p>The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation.</p> <p>The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures.</p> <p>The composition of the Holtite-A is confirmed by inspection and tests prior to first use.</p> <p>Following the first loading for the HI-TRAC transfer cask and each HI-STORM overpack, a shield effectiveness test is performed in accordance with written approved procedures, as specified in Section 9.1.</p>
ISG-15, Section X.5.4.2, "No more than 1% of the rods in an assembly have peak cladding oxide thicknesses greater than 80 micrometers and no more than 3% of the rods in an assembly have peak cladding oxide thicknesses greater than 70 micrometers. A high burnup fuel assembly should be treated as potentially damaged fuel if the assembly does not meet both of the above criteria of if the fuel assembly contains fuel rods with oxide that has become detached or spalled from the cladding	The Fuel Cladding Oxide Thickness Evaluation Program in Section 5.0 of Appendix A to the CoC provides an equation to calculate the maximum allowable high burnup fuel cladding oxide thickness, based on fuel assembly type.	FSAR Appendix 4.A, Section 4.A.9 provides the justification for this deviation from NUREG-1536 (ISG-15).

1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

1.2.1 System Characteristics

The basic HI-STORM 100 System consists of interchangeable MPCs providing a confinement boundary for BWR or PWR spent nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the storage overpack. Figure 1.2.1 provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a storage overpack. Figure 1.2.1A provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a HI-STORM 100S storage overpack. Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. All structures, systems, and components of the HI-STORM 100 System which are identified as Important to Safety are specified in Table 2.2.6. This discussion is supplemented with a full set of drawings in Section 1.5.

The HI-STORM 100 System is comprised of three discrete components:

- i. multi-purpose canister (MPC)
- ii. storage overpack (HI-STORM)
- iii. transfer cask (HI-TRAC)

Necessary auxiliaries required to deploy the HI-STORM 100 System for storage are:

- i. vacuum drying (or other moisture removal) system
- ii. helium (He) backfill system with leakage detector
- iii. lifting and handling systems
- iv. welding equipment
- v. transfer vehicles/trailer

All MPCs have identical exterior dimensions that render them interchangeable. The outer diameter of the MPC is 68-3/8 inches[†] and the overall length is 190-1/2 inches. See Section 1.5 for the detailed design drawings. Due to the differing storage contents of each MPC, the maximum loaded weight differs among MPCs. See Table 3.2.1 for each MPC weight. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons. Tables 1.2.1 and 1.2.2 contain the key parameters for the MPCs.

A single, base HI-STORM overpack design is provided which is capable of storing each type of MPC. The overpack inner cavity is sized to accommodate the MPCs. The inner diameter of the overpack inner shell is 73-1/2 inches and the height of the cavity is 191-1/2 inches. The overpack

[†] Dimensions discussed in this section are considered nominal values.

inner shell is provided with channels distributed around the inner cavity to present an inside diameter of 69-1/2 inches. The channels are intended to offer a flexible medium to absorb some of the impact during a non-mechanistic tip-over, while still allowing the cooling air flow through the ventilated overpack. The outer diameter of the overpack is 132-1/2 inches. The overall height of the HI-STORM 100 is 239-1/2 inches. There are two versions of the HI-STORM 100S overpack, differing only in height and weight. The HI-STORM 100S(232) is 232 inches high, and the HI-STORM 100S(243) is 243 inches high. The HI-STORM 100S(243) is approximately 10,100 lbs heavier, including concrete. Hereafter in the text, these two versions of the HI-STORM 100S overpack will only be referred to as HI-STORM 100S and will be discussed separately only if the design feature being discussed is different between the two overpacks. See Section 1.5 for drawings. The weight of the overpack without an MPC is approximately 135 tons. See Table 3.2.1 for the detailed weights.

Before proceeding to present detailed physical data on the HI-STORM 100 System, it is of contextual importance to summarize the design attributes which enhance the performance and safety of the system. Some of the principal features of the HI-STORM 100 System which enhance its effectiveness as an SNF storage device and a safe SNF confinement structure are:

- the honeycomb design of the MPC fuel basket;
- the effective distribution of neutron and gamma shielding materials within the system;
- the high heat dissipation capability;
- engineered features to promote convective heat transfer;
- the structural robustness of the steel-concrete-steel overpack construction.

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flange plate weldment where all structural elements (i.e., box walls) are arrayed in two orthogonal sets of plates. Consequently, the walls of the cells are either completely co-planar (i.e., no offset) or orthogonal with each other. There is complete edge-to-edge continuity between the contiguous cells.

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass of the basket over the entire length of the basket. Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform basket. In other words, the honeycomb basket is a most effective radiation attenuation device. The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the MPC an effective heat rejection device.

The composite shell construction in the overpack, steel-concrete-steel, allows ease of fabrication and eliminates the need for the sole reliance on the strength of concrete.

A description of each of the components is provided in the following sections, along with information with respect to its fabrication and safety features. This discussion is supplemented with the full set of drawings in Section 1.5.

1.2.1.1 Multi-Purpose Canisters

The MPCs are welded cylindrical structures as shown in cross sectional views of Figures 1.2.2 through 1.2.4. The outer diameter and cylindrical height of each MPC are fixed. Each spent fuel MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring, as depicted in the MPC cross section elevation view, Figure 1.2.5. The number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics.

There are seven MPC models, distinguished by the type and number of fuel assemblies authorized for loading. The MPC-24 is designed to store up to 24 intact PWR fuel assemblies. The MPC-24E is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies. The MPC-24EF is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies or fuel classified as fuel debris. The MPC-68 is designed to store up to 68 total BWR fuel assemblies including up to 68 damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies. Damaged BWR fuel assemblies other than Dresden Unit 1 and Humboldt Bay are limited to 16 fuel storage locations in the MPC-68 with the remainder being intact BWR fuel assemblies, up to a total of 68. The MPC-68F is designed to store up to 68 intact or damaged Dresden Unit 1 and Humboldt Bay BWR fuel assemblies. Up to four of the 68 fuel storage locations in the MPC-68F may be Dresden Unit 1 and Humboldt Bay BWR fuel assemblies classified as fuel debris. The MPC-68FF is designed to store up to 68 total BWR fuel assemblies including up to 16 damaged BWR fuel assemblies. Up to eight (8) of the 16 BWR damaged fuel assembly storage locations may be filled with BWR fuel classified as fuel debris. In addition, all fuel loading combinations permitted in the MPC-68F are also permitted in the MPC-68FF. Drawings for all of the MPCs are provided in Section 1.5.

The MPC provides the confinement boundary for the stored fuel. Figure 1.2.6 provides an elevation view of the MPC confinement boundary. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring. The confinement boundary is a strength-welded enclosure of all stainless steel construction.

The PWR MPC-24, MPC-24E and MPC-24EF differ in construction from the MPC-32 and the MPC-68 (including the MPC-68F and MPC-68FF) in one important aspect: the fuel storage cells are physically separated from one another by a "flux trap", for criticality control. The PWR MPC-32 is designed similar to the MPC-68 (without flux traps) and its design includes credit for soluble boron in the MPC water during wet fuel loading and unloading operations for criticality control.

The MPC fuel baskets of non-flux trap construction (namely, MPC-68, MPC-68F, MPC-68FF, and MPC-32) are formed from an array of plates welded to each other at their intersections. In the flux-trap type fuel baskets (MPC-24, MPC-24E, and MPC-24EF), formed angles are interposed onto the orthogonally configured plate assemblage to create the required flux-trap channels (see MPC-24 and MPC-24E design drawings in Section 1.5). In both configurations, two key attributes of the basket are preserved:

- i. The cross section of the fuel basket simulates a multi-flanged closed section beam, resulting in extremely high bending rigidity.
- ii. The principal structural frame of the basket consists of co-planar plate-type members (i.e., no offset).

This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls that must transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (e.g., non-mechanistic tipover, uncontrolled lowering of a cask during on-site transfer, or off-site transport events, etc.).

The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. Between the periphery of the basket, the MPC shell, and the basket supports, optional heat conduction elements may be installed. These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design that allows a snug fit in the confined spaces and ease of installation. If used, the heat conduction elements are installed along the full length of the MPC basket except at the drain pipe location to create a nonstructural thermal connection that facilitates heat transfer from the basket to shell. In their operating condition, the heat conduction elements contact the MPC shell and basket walls.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit placement of the empty MPC into the HI-TRAC transfer cask. The lifting lugs also serve to axially locate the MPC lid prior to welding. These internal lifting lugs are not used to handle a loaded MPC. Since the MPC lid is installed prior to any handling of a loaded MPC, there is no access to the lifting lugs once the MPC is loaded.

The top end of the MPC incorporates a redundant closure system. Figure 1.2.6 shows the MPC closure details. The MPC lid is a circular plate (fabricated from one piece, or two pieces - split top and bottom) edge-welded to the MPC outer shell. If the two-piece lid design is employed, only the top piece is analyzed as part of the enclosure vessel pressure boundary. The bottom piece acts as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. The lid is equipped with vent and drain ports that are utilized to remove moisture and air from the MPC, and backfill the MPC with a specified amount of inert gas (helium). The vent and drain ports are covered and seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by threaded holes in the MPC lid.

To maintain a constant exterior axial length between the PWR MPCs and the BWR MPCs the thickness of the PWR MPCs' lid is 1/2 inch thinner than the MPC-68's lid to accommodate the longest PWR fuel assembly which is approximately a 1/2 inch longer than the longest BWR fuel assembly. For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5.

The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested values for the upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type. The actual length of fuel spacers will be determined on a site-specific or fuel assembly-specific basis.

The MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and optional aluminum heat conduction elements). No carbon steel parts are permitted in the MPC. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the MPC. All structural components in a MPC shall be made of Alloy X, a designation which warrants further explanation.

Alloy X is a material that is expected to be acceptable as a Mined Geological Disposal System (MGDS) waste package and which meets the thermophysical properties set forth in this document.

At this time, there is considerable uncertainty with respect to the material of construction for an MPC that would be acceptable as a waste package for the MGDS. Candidate materials being considered for acceptability by the DOE include:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The DOE material selection process is primarily driven by corrosion resistance in the potential environment of the MGDS. As the decision regarding a suitable material to meet disposal requirements is not imminent, the MPC design allows the use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this FSAR) may be one of the following materials. Only a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC basket or shell - the basket and shell may be of different alloys in the same MPC.

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable for the entire group for the analysis in question. For example, when

calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, we have defined a material, which is referred to as Alloy X, whose thermophysical properties, from the MPC design perspective, are the least favorable of the candidate materials.

The evaluation of the Alloy X constituents to determine the least favorable properties is provided in Appendix 1.A.

The Alloy X approach is conservative because no matter which material is ultimately utilized in the MPC construction, the Alloy X approach guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

1.2.1.2 Overpacks

1.2.1.2.1 HI-STORM 100 Overpack (Storage)

The HI-STORM 100 and 100S overpacks are rugged, heavy-walled cylindrical vessels. Figures 1.2.7, 1.2.8, and 1.2.8A provide cross sectional views of the HI-STORM 100 System, showing both of the overpack designs, respectively. The HI-STORM 100A is an anchored variant of the same structure and hereinafter is identified by name only when the discussion specifically applies to the anchored overpack. The HI-STORM 100A differs only in the diameter of the overpack baseplate and the presence of bolt holes and associated anchorage hardware (see Figures 1.1.4 and 1.1.5). The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The overpack plain concrete is enclosed by cylindrical steel shells, a thick steel baseplate, and a top plate. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation in the vertical direction.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC. The inner shell of the overpack has channels attached to its inner diameter. The channels provide guidance for MPC insertion and removal and a flexible medium to absorb impact loads during the non-mechanistic tip-over, while still allowing the cooling air flow to circulate through the overpack. Shims may be attached to channels to allow the proper inner diameter dimension to be obtained.

The storage system has air ducts to allow for passive natural convection cooling of the contained MPC. Four air inlets and four air outlets are located at the lower and upper extremities of the storage system, respectively. The location of the air outlets in the HI-STORM 100 and the HI-STORM 100S design differ in that the outlet ducts for the HI-STORM 100 overpack are located in the overpack body and are aligned vertically with the inlet ducts at the bottom of the overpack body. The air outlet ducts in the HI-STORM 100S are integral to the lid assembly and are not in vertical alignment with the inlet ducts. The location of the air inlet ducts is same for both the HI-

STORM 100 and the HI-STORM 100S. The air inlets and outlets are covered by a fine mesh screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, described in Chapter 11 of this FSAR, evaluates the effects of partial and complete blockage of the air ducts.

The four air inlets and four air outlets are penetrations through the thick concrete shielding provided by the HI-STORM 100 overpack. The outlet air ducts for the HI-STORM 100S overpack, integral to the lid, present a similar break in radial shielding. Within the air inlets and outlets, an array of gamma shield cross plates are installed (see Figure 5.3.19 for a pictorial representation of the gamma shield cross plate designs). These gamma shield cross plates are designed to scatter any particles traveling through the ducts. The result of scattering the particles in the ducts is a significant decrease in the local dose rates around the four air inlets and four air outlets. The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized. The shielding analysis conservatively credits only the mandatory version of the gamma shield cross plate design because they provide less shielding than the optional design. Conversely, the thermal analysis conservatively evaluates the optional gamma shield cross plate design because it conservatively provides greater resistance to flow than the mandatory design.

Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM 100) or the inlet air duct horizontal plates (HI-STORM 100S) (see Figure 1.2.7). The four anchor blocks are located on 90° arcs around the circumference of the overpack. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, Hillman rollers, or other design based on site-specific needs and capabilities. Slings or other suitable devices mate with lifting lugs that are inserted into threaded holes in the top surface of the overpack lid to allow lifting of the overpack lid. After the lid is bolted to the storage overpack main body, these lifting bolts shall be removed and replaced with flush plugs.

The plain concrete between the overpack inner and outer steel shells is specified to provide the necessary shielding properties (dry density) and compressive strength. The concrete shall be in accordance with the requirements specified in Appendix 1.D.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, in an implicit manner it helps enhance the performance of the HI-STORM overpack in other respects as well. For example, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. The case of a postulated fire accident at the ISFSI is another example where the high thermal inertia characteristics of the HI-STORM concrete control the temperature of the MPC. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the inter-shell space, such that, while its cracking and crushing under a tip-

over accident is not of significant consequence, its deformation characteristics are germane to the analysis of the structural members.

Density and compressive strength are the key parameters which delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal, and HI-STORM lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.0.5] are used.

To ensure the stability of the concrete at temperature, the concrete composition has been specified in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" [1.0.3]. Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM overpack concrete.

There are two base HI-STORM overpack designs - HI-STORM 100 and HI-STORM 100S. The significant differences between the two are overpack height, MPC pedestal height, location of the air outlet ducts, and the vertical alignment of the inlet and outlet air ducts. The HI-STORM 100 overpack is approximately 240 inches high from the bottom of the baseplate to the top of the lid bolts and 227 inches high without the lid installed. There are two versions of the HI-STORM 100S overpack design, differing only in height and weight. The HI-STORM 100S(232) is approximately 232 inches from the bottom of the baseplate to the top of the lid in its final storage configuration and 211 inches high without the lid installed. The HI-STORM 100S(243) is approximately 243 inches from the bottom of the baseplate to the top of the lid in its final storage configuration and 222 inches high without the lid installed.

The anchored embodiment of the HI-STORM overpack is referred to as HI-STORM 100A. As explained in the foregoing, the HI-STORM overpack is a steel weldment, which makes it a relatively simple matter to extend the overpack baseplate, form lugs, and then anchor the cask to the reinforced concrete structure of the ISFSI. In HI-STORM terminology, these lugs are referred to as "sector lugs." The sector lugs, as shown in Figure 1.1.5 and the drawing in Section 1.5, are formed by extending the HI-STORM overpack baseplate, welding vertical gussets to the baseplate extension and to the overpack outer shell and, finally, welding a horizontal lug support ring in the form of an annular sector to the vertical gussets and to the outer shell. The baseplate is equipped with regularly spaced clearance holes (round or slotted) through which the anchor studs can pass. The sector lugs are bolted to the ISFSI pad using anchor studs that are made of a creep-resistant, high-ductility, environmentally compatible material. The bolts are pre-loaded to a precise axial stress using a "stud tensioner" rather than a torque wrench. Pre-tensioning the anchors using a stud tensioner eliminates any shear stress in the bolt, which is unavoidable if a torquing device is employed (Chapter 3 of the text "Mechanical Design of Heat Exchangers and Pressure Vessel Components", by Arcturus Publishers, 1984, K.P. Singh and A.I. Soler, provides additional information on stud tensioners). The axial stress in the anchors induced by pre-tensioning is kept below 75% of the material yield stress, such that during the seismic event the maximum bolt axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF (for Level D conditions). Figures 1.1.4 and

1.1.5 provide visual depictions of the anchored HI-STORM 100A configuration. This configuration also applies to the HI-STORM 100SA.

The anchor studs pass through liberal clearance holes (circular or slotted) in the sector lugs (0.75" minimum clearance) such that the fastening of the studs to the ISFSI pad can be carried out without mechanical interference from the body of the sector lug. The two clearance hole configurations give the ISFSI pad designer flexibility in the design of the anchor embedment in the ISFSI concrete. The axial force in the anchors produces a compressive load at the overpack/pad interface. This compressive force, F , imparts a lateral load bearing capacity to the cask/pad interface that is equal to μF ($\mu \leq 0.53$ per Table 2.2.8). As is shown in Chapter 3 of this FSAR, the lateral load-bearing capacity of the HI-STORM/pad interface (μF) is many times greater than the horizontal (sliding) force exerted on the cask under the postulated DBE seismic event. Thus, the potential for lateral sliding of the HI-STORM 100A System during a seismic event is precluded, as is the potential for any bending action on the anchor studs.

The seismic loads, however, will produce an overturning moment on the overpack that would cause a redistribution of the compressive contact pressure between the pad and the overpack. To determine the pulsation in the tensile load in the anchor studs and in the interface contact pressure, bounding static analysis of the preloaded configuration has been performed. The results of the static analysis demonstrate that the initial preloading minimizes pulsations in the stud load. A confirmatory non-linear dynamic analysis has also been performed using the time-history methodology described in Chapter 3, wherein the principal nonlinearities in the cask system are incorporated and addressed. The calculated results from the dynamic analysis confirm the static analysis results and that the presence of pre-stress helps minimize the pulsation in the anchor stud stress levels during the seismic event, thus eliminating any concern with regard to fatigue failure under extended and repetitive seismic excitations.

The sector lugs in HI-STORM 100A are made of the same steel material as the baseplate and the shell (SA516- Gr. 70) which helps ensure high quality fillet welds used to join the lugs to the body of the overpack. The material for the anchor studs can be selected from a family of allowable stud materials listed in the ASME Code (Section II). A representative sampling of permitted materials is listed in Table 1.2.7. The menu of materials will enable the ISFSI owner to select a fastener material that is resistant to corrosion in the local ISFSI environment. For example, for ISFSIs located in marine environments (e.g., coastal reactor sites), carbon steel studs would not be recommended without concomitant periodic inspection and coating maintenance programs. Table 1.2.7 provides the chemical composition of several acceptable fastener materials to help the ISFSI owner select the most appropriate material for his site. The two mechanical properties, ultimate strength σ_u and yield strength σ_y , are also listed. For purposes of structural evaluations, the lower bound values of σ_u and σ_y from the menu of materials listed in Table 1.2.7 are used (see Table 3.4.10).

As shown in the drawing, the anchor studs are spaced sufficiently far apart such that a practical reinforced concrete pad with embedded receptacles can be designed to carry the axial pull from the anchor studs without overstressing the enveloping concrete monolith. The design specification and supporting analyses in this FSAR are focused on qualifying the overpack structures, including the sector lugs and the anchor studs. The design of the ISFSI pad, and its anchor receptacle will vary

from site to site, depending on the geology and seismological characteristics of the sub-terrain underlying the ISFSI pad region. The data provided in this FSAR, however, provide the complete set of factored loads to which the ISFSI pad, its sub-grade, and the anchor receptacles must be designed within the purview of ACI-349-97 [1.0.4]. Detailed requirements on the ISFSI pads for anchored casks are provided in Section 2.0.4.

1.2.1.2.2 HI-TRAC (Transfer Cask) - Standard Design

Like the storage overpack, the HI-TRAC transfer cask is a rugged, heavy-walled cylindrical vessel. The main structural function of the transfer cask is provided by carbon steel, and the main neutron and gamma shielding functions are provided by water and lead, respectively. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. Figure 1.2.9 provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. See Section 1.2.1.2.3 for discussion of the optional HI-TRAC 125D design.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC. The top lid of the HI-TRAC 125 has additional neutron shielding to provide neutron attenuation in the vertical direction (from SNF in the MPC below). The MPC access hole through the HI-TRAC top lid is provided to allow the lowering/raising of the MPC between the HI-TRAC transfer cask, and the HI-STORM or HI-STAR overpacks. The standard design HI-TRAC (comprised of HI-TRAC 100 and HI-TRAC 125) is provided with two bottom lids, each used separately. The pool lid is bolted to the bottom flange of the HI-TRAC and is utilized during MPC fuel loading and sealing operations. In addition to providing shielding in the axial direction, the pool lid incorporates a seal that is designed to hold clean demineralized water in the HI-TRAC inner cavity, thereby preventing contamination of the exterior of the MPC by the contaminated fuel pool water. After the MPC has been drained, dried, and sealed, the pool lid is removed and the HI-TRAC transfer lid is attached (standard design only). The transfer lid incorporates two sliding doors that allow the opening of the HI-TRAC bottom for the MPC to be raised/lowered. Figure 1.2.10 provides a cross section of the HI-TRAC with the transfer lid installed.

In the standard design, trunnions are provided for lifting and rotating the transfer cask body between vertical and horizontal positions. The lifting trunnions are located just below the top flange and the pocket trunnions are located above the bottom flange. The two lifting trunnions are provided to lift and vertically handle the HI-TRAC, and the pocket trunnions provide a pivot point for the rotation of the HI-TRAC for downending or upending.

Two standard design HI-TRAC transfer casks of different weights are provided to house the MPCs. The 125 ton HI-TRAC weight does not exceed 125 tons during any loading or transfer operation. The 100 ton HI-TRAC weight does not exceed 100 tons during any loading or transfer operation. The internal cylindrical cavities of the two standard design HI-TRACs are identical. However, the external dimensions are different. The 100ton HI-TRAC has a reduced thickness of lead and water shielding and consequently, the external dimensions are different. The structural steel thickness is identical in the two HI-TRACs. This allows most structural analyses of the 125

ton HI-TRAC to bound the 100 ton HI-TRAC design. Additionally, as the two HI-TRACs are identical except for a reduced thickness of lead and water, the 125 ton HI-TRAC has a larger thermal resistance than the smaller and lighter 100 ton HI-TRAC. Therefore, for normal conditions the 125 ton HI-TRAC thermal analysis bounds that of the 100 ton HI-TRAC. Separate shielding analyses are performed for each HI-TRAC since the shielding thicknesses are different between the two.

1.2.1.2.3 HI-TRAC 125D Transfer Cask

As an option to using either of the standard HI-TRAC transfer cask design, users may choose to use the optional HI-TRAC 125D design. Figure 1.2.9A provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. Like the standard design, the HI-TRAC 125D is designed and constructed in accordance with ASME III, Subsection NF, with certain NRC-approved alternatives, as discussed in Section 2.2.4. Functionally equivalent, the major differences between the HI-TRAC 125D design and the standard design are as follows:

- No pocket trunnions are provided for downending/upending
- The transfer lid is not required
- A new ancillary, the HI-STORM mating device (Figure 1.2.18) is required during MPC transfer operations
- A wider baseplate with attachment points for the mating device is provided
- The baseplate incorporates gussets for added structural strength
- The number of pool lid bolts is reduce

The interface between the MPC and the transfer cask is the same between the standard design and the HI-TRAC 125D design. The optional design is capable of withstanding all loads defined in the design basis for the transfer cask during normal, off-normal, and accident modes of operation with adequate safety margins. In lieu of swapping the pool lid for the transfer lid to facilitate MPC transfer, the pool lid remains on the HI-TRAC 125D until MPC transfer is required. The HI-STORM mating device is located between, and secured with bolting to, the top of the HI-STORM overpack and the HI-TRAC 125D transfer cask. The mating device is used to remove the pool lid to provide a pathway for MPC transfer between the overpack and the transfer cask. Section 1.2.2.2 provides additional detail on the differences between the standard transfer cask design and the HI-TRAC 125D design during operations.

1.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure the radiation and exposure requirements in 10CFR72.104 and 10CFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the SNF in the MPC for ALARA considerations during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and Boral[®] neutron poison panels attached to the fuel storage cell vertical surfaces provide the initial attenuation of gamma and neutron radiation

emitted by the radioactive spent fuel. The MPC shell, baseplate, lid and closure ring provide additional thicknesses of steel to further reduce the gamma flux at the outer canister surfaces.

In the HI-STORM storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the lid, and a thick circular concrete pedestal upon which the MPC rests. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shielding integral to the lid provide additional gamma attenuation in the upward direction, reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness. Gamma shield cross plates, as depicted in Figure 5.3.19, provide attenuation of scattered gamma radiation as it exits the inlet and outlet air ducts.

In the HI-TRAC transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the axial direction, shielding is provided by the top lid, and the pool or transfer lid, as applicable. In the HI-TRAC pool lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC. In the transfer lid, layers of steel-lead-steel provide gamma attenuation. For the HI-TRAC 125 transfer lid, the neutron shield material, Holtite-A, is also provided. The HI-TRAC 125 and HI-TRAC 125D top lids are composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The HI-TRAC 100 top lid is composed of steel only providing gamma attenuation.

1.2.1.3.1 Boral Neutron Absorber

Boral is a thermal neutron poison material composed of boron carbide and aluminum (aluminum powder and plate). Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The Boral cladding is made of alloy aluminum, a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal, and chemical environment of a nuclear reactor, spent fuel pool, or dry cask. See Section 3.4.1 for discussion of the reaction of Boral with spent fuel pool water during fuel loading and unloading operations.

The documented historical applications of Boral, in environments comparable to those in spent fuel pools and fuel storage casks, dates to the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor [1.2.2]). Technical data on the material was first printed in 1949, when the report "Boral: A New Thermal Neutron Shield" was published [1.2.3]. In 1956, the first edition of the Reactor Shielding Design Manual [1.2.4] was published and it contained a section on Boral and its properties.

In the research and test reactors built during the 1950s and 1960s, Boral was frequently the

material of choice for control blades, thermal-column shutters, and other items requiring very good thermal-neutron absorption properties. It is in these reactors that Boral has seen its longest service in environments comparable to today's applications.

Boral found other uses in the 1960s, one of which was a neutron poison material in baskets used in the shipment of irradiated, enriched fuel rods from Canada's Chalk River laboratories to Savannah River. Use of Boral in shipping containers continues, with Boral serving as the poison in current British Nuclear Fuels Limited casks and the Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

Boral has been exclusively used in fuel storage applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and several unique characteristics, such as:

- The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.
- Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- Boral is stable, strong, durable, and corrosion resistant.

Boral absorbs thermal neutrons without physical change or degradation of any sort from the anticipated exposure to gamma radiation and heat. The material does not suffer loss of neutron attenuation capability when exposed to high levels of radiation dose.

Holtec International's QA Program ensures that Boral is manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR72, Subpart G. Holtec International has procured over 200,000 panels of Boral from AAR Advanced Structures in over 30 projects. Boral has always been purchased with a minimum ^{10}B loading requirement. Coupons extracted from production runs were tested using the wet chemistry procedure. The actual ^{10}B loading, out of thousands of coupons tested, has never been found to fall below the design specification. The size of this coupon database is sufficient to provide reasonable assurance that all future Boral procurements will continue to

yield Boral with full compliance with the stipulated minimum loading. Furthermore, the surveillance, coupon testing, and material tracking processes which have so effectively controlled the quality of Boral are expected to continue to yield Boral of similar quality in the future. Nevertheless, to add another layer of insurance, only 75% ^{10}B credit of the fixed neutron absorber is assumed in the criticality analysis consistent with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

1.2.1.3.2 Neutron Shielding

The specification of the HI-STORM overpack and HI-TRAC transfer cask neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation to appropriate levels;
- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;
- Stability of the homogeneous nature of the shielding material matrix;
- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and
- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

Other aspects of a shielding material, such as ease of handling and prior nuclear industry use, are also considered, within the limitations of the main criteria. Final specification of a shield material is a result of optimizing the material properties with respect to the main criteria, along with the design of the shield system, to achieve the desired shielding results.

Neutron attenuation in the HI-STORM overpack is provided by the thick walls of concrete contained in the steel vessel, lid, and pedestal. Concrete is a shielding material with a long proven history in the nuclear industry. The concrete composition has been specified to ensure its continued integrity at the long term temperatures required for SNF storage.

The HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) will be added to reduce the freezing point for low temperature operations (e.g., below 32°F) [1.2.7].

Neutron shielding in the HI-TRAC 125 and 125D transfer casks in the axial direction is provided by Holtite-A within the top lid. HI-TRAC 125 also contains Holtite-A in the transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A

is specified with a nominal B₄C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

In the following, a brief summary of the performance characteristics and properties of Holtite-A is provided.

Density

The specific gravity of Holtite-A is 1.68 g/cm³ as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm³. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm³ to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The weight concentration of hydrogen is 6.0%. However, all shielding analyses conservatively assume 5.9% hydrogen by weight in the calculations.

Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (nominal) weight concentration. Holtite-A may be specified with a B₄C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal B₄C weight percent of 1%.

Design Temperature

The design temperature of Holtite-A is set at 300°F. The maximum spatial temperature of Holtite-A under all normal operating conditions must be demonstrated to be below this design temperature.

Thermal Conductivity

The Holtite-A neutron shielding material is stable below the design temperature for the long term and provides excellent shielding properties for neutrons. A conservative, lower bound conductivity is stipulated for use in the thermal analyses of Chapter 4 (Section 4.2) based on information in the technical literature.

1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete

sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7)[†]. Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the baseplate and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are

installed in the ducts (Figures 1.2.8 and 1.2.8A) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets.

In the HI-TRAC transfer cask, the primary gamma shielding is provided by lead. As in the storage overpack, carbon steel supplements the lead gamma shielding of the HI-TRAC transfer cask.

1.2.1.4 Lifting Devices

Lifting of the HI-STORM 100 System may be accomplished either by attachment at the top of the storage overpack ("top lift"), as would typically be done with a crane, or by attachment at the bottom ("bottom lift"), as would be effected by a number of lifting/handling devices.

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The anchor blocks are integrally welded to the overpack radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM100) or inlet air duct horizontal plates (HI-STORM 100S). The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment.

A bottom lift of the HI-STORM 100 storage overpack is effected by the insertion of four hydraulic jacks underneath the inlet vent horizontal plates (Figure 1.2.1). A slot in the overpack baseplate allows the hydraulic jacks to be placed underneath the inlet vent horizontal plate. The hydraulic jacks lift the loaded overpack to provide clearance for inserting or removing a device for transportation.

The standard design HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket trunnions. The HI-TRAC 125D is equipped with only lifting trunnions. The lifting trunnions are positioned just below the top forging. The two pocket trunnions are located above the bottom forging and attached to the outer shell. The pocket trunnions are designed to allow rotation of the HI-TRAC. All trunnions are built from a high strength alloy with proven corrosion and non-galling characteristics. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The lifting trunnions are installed by threading into tapped

[†] The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated. Those overpacks without the shield shell are required to have a higher concrete density in the overpack body to provide compensatory shielding. See Table 1.D.1.

holes just below the top forging.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered through the HI-TRAC transfer cask using lifting cleats. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

1.2.1.5 Design Life

The design life of the HI-STORM 100 System is 40 years. This is accomplished by using material of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 9, is also implemented to ensure the HI-STORM 100 System will exceed its design life of 40 years. The design considerations that assure the HI-STORM 100 System performs as designed throughout the service life include the following:

HI-STORM Overpack and HI-TRAC Transfer Cask

- Exposure to Environmental Effects
- Material Degradation
- Maintenance and Inspection Provisions

MPC

- Corrosion
- Structural Fatigue Effects
- Maintenance of Helium Atmosphere
- Allowable Fuel Cladding Temperatures
- Neutron Absorber Boron Depletion

The adequacy of the HI-STORM 100 System for its design life is discussed in Sections 3.4.11 and 3.4.12.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The HI-STORM 100 System incorporates some unique design improvements. These design innovations have been developed to facilitate the safe long term storage of SNF. Some of the design originality is discussed in Subsection 1.2.1 and below.

The free volume of the MPCs is inerted with 99.995% pure helium gas during the spent nuclear fuel loading operations. Table 1.2.2 specifies the helium fill requirements for the MPC internal cavity.

The HI-STORM overpack has been designed to synergistically combine the benefits of steel and concrete. The steel-concrete-steel construction of the HI-STORM overpack provides ease of fabrication, increased strength, and an optimal radiation shielding arrangement. The concrete is primarily provided for radiation shielding and the steel is primarily provided for structural functions.

The strength of concrete in tension and shear is conservatively neglected. Only the compressive strength of the concrete is accounted for in the analyses.

The criticality control features of the HI-STORM 100 are designed to maintain the neutron multiplication factor k -effective (including uncertainties and calculational bias) at less than 0.95 under all normal, off-normal, and accident conditions of storage as analyzed in Chapter 6. This level of conservatism and safety margins is maintained, while providing the highest storage capacity.

1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STORM 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the drawings in Section 1.5. A summary of the general actions needed for the loading and unloading operations is provided below. Figures 1.2.16 and 1.2.17 provide a pictorial view of typical loading and unloading operations, respectively.

Loading Operations

At the start of loading operations, the HI-TRAC transfer cask is configured with the pool lid installed. The HI-TRAC water jacket is filled with demineralized water or a 25% ethylene glycol solution depending on the ambient temperature conditions. The lift yoke is used to position HI-TRAC in the designated preparation area or setdown area for HI-TRAC inspection and MPC insertion. The annulus is filled with plant demineralized water (borated if necessary), and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with water. Based on the MPC model and fuel enrichment (as required by the CoC), this may be borated water or plant demineralized water. HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielding lid (the MPC lid) is installed. The lift yoke is remotely engaged to the HI-TRAC lifting trunnions and is used to lift the HI-TRAC close to the spent fuel pool surface. As an ALARA measure, dose rates are measured on the top of the HI-TRAC and MPC prior to removal from the pool to check for activated debris on the top surface. The MPC lift bolts (securing the MPC lid to the lift yoke) are removed. As HI-TRAC is removed from the spent fuel pool, the lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination.

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid and around the mid-height circumference of HI-TRAC to ensure that the dose rates are within expected values. The Automated Welding System baseplate shield (if used) is installed to reduce dose rates around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS) or other approved welding process. Liquid penetrant examinations are performed on the root and final passes. A multi-layer liquid penetrant or volumetric examination is also performed on the MPC lid-to-shell weld. The water level is raised to the top of the MPC and the weld is hydrostatically tested. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid-to-shell) to verify weld integrity and to ensure that leakage rates are within acceptance criteria. The MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line.

For storage of moderate burnup fuel, a Vacuum Drying System (VDS) may be used to remove moisture from the MPC cavity. The VDS is connected to the MPC and is used to remove liquid water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated. This process is continued until the pressure in the MPC meets the technical specification limit and can be held there for the required amount of time.

For storage of high burnup fuel and as an option for storage of moderate burnup fuel, the reduction of residual moisture in the MPC to trace amounts is accomplished using a Forced Helium Dehydration (FHD) system, as described in Appendix 2.B. Relatively warm and dry helium is recirculated through the MPC cavity, which helps maintain the SNF in a cooled condition while moisture is being removed. The warm, dry gas is supplied to the MPC drain port and circulated through the MPC cavity where it absorbs moisture. The humidified gas travels out of the MPC and through appropriate equipment to cool and remove the absorbed water from the gas. The dry gas may be heated prior to its return to the MPC in a closed loop system to accelerate the rate of moisture removal in the MPC. This process is continued until the

temperature of the gas exiting the demoinsturizing module described in Appendix 2.B meets the limit specified in the technical specifications.

Following moisture removal, the VDS or FHD system is disconnected and the Helium Backfill System (HBS) is attached and the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides the means of future leakage rate testing of the MPC confinement boundary welds. Cover plates are installed and seal-welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes. The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.

The MPC closure ring is then placed on the MPC, aligned, tacked in place, and seal welded, providing redundant closure of the MPC lid and cover plates confinement closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in the annulus is drained. The AWS Baseplate shield is removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and HI-TRAC dose rates are measured. The HI-TRAC top lid is installed and the bolts are torqued. The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point of the MPC.

Rigging is installed between the MPC lift cleats and the lift yoke. . The rigging supports the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. For the standard design transfer cask, the HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and rigging support the MPC during the transfer operations.

MPC transfer from the HI-TRAC transfer cask into the overpack may be performed inside or outside the fuel building. Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways. The loaded HI-TRAC may be handled in the vertical or horizontal orientation. The loaded HI-STORM can only be handled vertically.

For MPC transfers inside the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100 overpack, the vent duct shield inserts installed. If using HI-TRAC 125D, the HI-STORM mating device is secured to the top of the empty overpack (Figure 1.2.18). The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM, or the mating device, as applicable. After the HI-TRAC is positioned atop the HI-STORM or secured to the mating device, as applicable, the MPC is raised slightly. With the standard HI-TRAC design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and the HI-TRAC is prepared for removal from on top of HI-STORM (with HI-TRAC 125D, the transfer cask must first be disconnected from the mating device). For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above

the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and/or mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs are installed and torqued.

For MPC transfers outside of the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts installed. For HI-TRAC 125D, the mating device is secured to the top of the overpack. The loaded HI-TRAC is transported to the cask transfer facility in the vertical or horizontal orientation. A number of methods may be utilized as long as the handling limitations prescribed in the technical specifications are not exceeded.

To place the loaded HI-TRAC in a horizontal orientation, a transport frame or "cradle" is utilized. If the cradle is equipped with rotation trunnions they are used to engage the HI-TRAC 100 or 125 pocket trunnions. While the loaded HI-TRAC is lifted by the lifting trunnions, the HI-TRAC is lowered onto the cradle rotation trunnions. Then, the crane lowers and the HI-TRAC pivots around the pocket trunnions and is placed in the horizontal position in the cradle.

The HI-TRAC 125D does not include pocket trunnions in its design. Therefore, the user must downend the transfer cask onto the transport frame using appropriately designed rigging in accordance with the site's heavy load control program.

If the loaded HI-TRAC is transferred to the cask transfer facility in the horizontal orientation, the HI-TRAC transport frame and/or cradle are placed on a transport vehicle. The transport vehicle may be an air pad, railcar, heavy-haul trailer, dolly, etc. If the loaded HI-TRAC is transferred to the cask transfer facility in the vertical orientation, the HI-TRAC may be lifted by the lifting trunnions or seated on the transport vehicle. During the transport of the loaded HI-TRAC, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms.

After the loaded HI-TRAC arrives at the cask transfer facility, the HI-TRAC is upended by a crane if the HI-TRAC is in a horizontal orientation. The loaded HI-TRAC is then placed, using the crane located in the transfer area, on top of HI-STORM, which has been inspected and staged with the lid removed, vent duct shield inserts installed, the alignment device positioned, and the mating device installed, as applicable.

After the HI-TRAC is positioned atop the HI-STORM or the mating device, the MPC is raised slightly. In the standard design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and

HI-TRAC is removed from on top of HI-STORM or disconnected from the mating device, as applicable. For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs and nuts are installed.

After the HI-STORM has been loaded either within the fuel building or at a dedicated cask transfer facility, the HI-STORM is then moved to its designated position on the ISFSI pad. The HI-STORM overpack may be moved using a number of methods as long as the handling limitations listed in the technical specifications are not exceeded. The loaded HI-STORM must be handled in the vertical orientation, and may be lifted from the top by the anchor blocks or from the bottom by the inlet vents. After the loaded HI-STORM is lifted, it may be placed on a transport mechanism or continue to be lifted by the lid studs and transported to the storage location. The transport mechanism may be an air pad, crawler, railcar, heavy-haul trailer, dolly, etc. During the transport of the loaded HI-STORM, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms. Once in position at the storage pad, vent operability testing is performed to ensure that the system is functioning within its design parameters.

In the case of HI-STORM 100A, the anchor studs are installed and fastened into the anchor receptacles in the ISFSI pad in accordance with the design requirements.

Unloading Operations

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover HI-TRAC and empty the MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC overpressurization and thermal shock to the stored spent fuel assemblies.

The MPC is recovered from HI-STORM either at the cask transfer facility or the fuel building using any of the methodologies described in Section 8.1. The HI-STORM lid is removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts are installed, and the MPC lift cleats are attached to the MPC. For HI-TRAC 125D, the mating device is installed. Rigging is attached to the MPC lift cleats. For the HI-STORM 100S and the standard HI-TRAC design, the transfer doors may need to be opened to avoid interfering with the MPC lift cleats. For HI-TRAC 125D, the mating device (possibly containing the pool lid) is secured to the top of the overpack. HI-TRAC is raised and positioned on top of HI-STORM or secured to the mating device, as applicable. For HI-TRAC 125D, the pool lid is ensured to be out

of the transfer path for the MPC. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the standard design HI-TRAC transfer lid doors are closed and the locking pins are installed. For HI-TRAC 125D, the pool lid is installed and the transfer cask is unsecured from the mating device. HI-TRAC is removed from on top of HI-STORM.

The HI-TRAC is brought into the fuel building and, for the standard design, manipulated for bottom lid replacement. The transfer lid is replaced with the pool lid. The MPC lift cleats and rigging support the MPC during lid transfer operations.

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the rigging, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water (borated, if necessary). The annulus and HI-TRAC top surfaces are protected from debris that will be produced when removing the MPC lid.

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established around the MPC ports. The RVOAs are attached to the vent and drain port. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding. Following the fuel cool-down, the MPC is flooded with borated or unborated water in accordance with the CoC. The MPC lid-to-MPC shell weld is removed. Then, all weld removal equipment is removed with the MPC lid left in place.

The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris. HI-TRAC and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and HI-TRAC are decontaminated in preparation for re-utilization.

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorbing materials in the fuel basket. The MPC-24, MPC-24E, and 24EF(all with lower enriched fuel) and the MPC-68 do not rely on soluble boron credit during loading or the assurance that water cannot enter the MPC during storage to meet the stipulated criticality limits.

Each MPC model is equipped with Boral neutron absorber plates affixed to the fuel cell walls as shown on the design drawings. The minimum ^{10}B areal density specified for the Boral in each MPC model is shown in Table 1.2.2. These values are chosen to be consistent with the assumptions made in the criticality analyses.

The MPC-24, MPC-24E and 24EF(all with higher enriched fuel) and the MPC-32 take credit for soluble boron in the MPC water for criticality prevention during wet loading and unloading operations. Boron credit is only necessary for these PWR MPCs during loading and unloading operations that take place under water. During storage, with the MPC cavity dry and sealed from the environment, criticality control measures beyond the fixed neutron poisons affixed to the storage cell walls are not necessary because of the low reactivity of the fuel in the dry, helium filled canister and the design features that prevent water from intruding into the canister during storage.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM 100 dry storage system. A detailed evaluation is provided in Section 3.4.

1.2.2.3.3 Operation Shutdown Modes

The HI-STORM 100 System is totally passive and consequently, operation shutdown modes are unnecessary. Guidance is provided in Chapter 8, which outlines the HI-STORM 100 unloading procedures, and Chapter 11, which outlines the corrective course of action in the wake of postulated accidents.

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is seal welded and leak tested. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM overpack exit vents in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 and the Technical Specifications in Appendix A to the CoC for additional details.

1.2.2.3.5 Maintenance Technique

Because of their passive nature, the HI-STORM 100 System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 9 describes the acceptance criteria and maintenance program set forth for the HI-STORM 100.

1.2.3 Cask Contents

The HI-STORM 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key design parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1 and the Approved Contents section of Appendix B to the CoC. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris

in accordance with the definitions of these terms in the CoC. A summary of the types of fuel authorized for storage in each MPC model is provided below. All fuel assemblies must meet the fuel specifications provided in Appendix B to the CoC. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers.

MPC-24

The MPC-24 is designed to accommodate up to twenty-four (24) PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

MPC-24E

The MPC-24E is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

MPC-24EF

The MPC-24EF is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

MPC-32

The MPC-32 is designed to accommodate up to thirty-two (32) PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

MPC-68

The MPC-68 is designed to accommodate up to sixty-eight (68) BWR intact and/or damaged fuel assemblies, with or without channels. For the Dresden Unit 1 or Humboldt Bay plants, the number of damaged fuel assemblies may be up to a total of 68. For damaged fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the number of damaged fuel assemblies is limited to sixteen (16) and must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2).

MPC-68F

The MPC-68F is designed to accommodate up to sixty-eight (68) Dresden Unit 1 or Humboldt Bay BWR fuel assemblies (with or without channels) made up of any combination of fuel assemblies classified as intact fuel assemblies, damaged fuel assemblies, and up to four (4) fuel assemblies classified as fuel debris.

MPC-68FF

The MPC-68FF is designed to accommodate up to sixty-eight (68) BWR fuel assemblies with or without channels. Any number of these fuel assemblies may be Dresden Unit 1 or Humboldt Bay BWR fuel assemblies classified as intact fuel or damaged fuel. Dresden Unit 1 and Humboldt Bay fuel debris is limited to eight (8) DFCs. DFCs containing Dresden Unit 1 or Humboldt Bay fuel debris may be stored in any fuel storage location. For BWR fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the total number of fuel assemblies classified as damaged fuel assemblies or fuel debris is limited to sixteen (16), with up to eight (8) of the 16 fuel assemblies classified as fuel debris. These fuel assemblies must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2). The balance of the fuel storage locations may be filled with intact BWR fuel assemblies, up to a total of 68.

Table 1.2.1

KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	7	4 for PWR 3 for BWR
MPC storage capacity [†] :	MPC-24 MPC-24E MPC-24EF	Up to 24 intact Zircaloy or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to four damaged fuel assemblies may be stored in the MPC-24E and up to four (4) damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24EF OR Up to 32 intact Zircaloy or stainless steel clad PWR fuel assemblies.
	MPC-32	Up to 32 intact Zircaloy or stainless steel clad PWR fuel assemblies.
	MPC-68	Any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68. For damaged fuel other than Dresden Unit 1 and Humboldt Bay, the number of fuel assemblies is limited to 16, with the balance being intact fuel assemblies. OR

[†] See Section 1.2.3 and Appendix B to the CoC for a complete description of cask contents and fuel specifications, respectively.

Table 1.2.2

KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725 [†] /-40 ^{††}	725 [†] /-40 ^{††}
Design internal pressure (psig)		
Normal conditions	100	100
Off-normal conditions	100	100
Accident Conditions	200	200
Total heat load, max. (kW)	27.77 (MPC-24) 28.17 (MPC-24E & MPC-24EF) 28.74 (MPC-32)	28.19 (MPC-68, MPC-68F, & MPC-68FF)
Maximum permissible peak fuel cladding temperature:		
Normal (□F)	See Table 2.2.3	See Table 2.2.3
Short Term & Accident (□F)	1058°	1058°
MPC internal environment	29.3 – 33.3 psig	29.3 – 33.3psig
Helium fill	OR 0.1212 gm-moles/l of free space	OR 0.1218 gm-moles/l of free space
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
Boral ¹⁰ B Areal Density (g/cm ²)	0.0267 (MPC-24) 0.0372 (MPC-24E, MPC-24EF & MPC-32)	0.0372 (MPC-68 & MPC-68FF) 0.01 (MPC-68F)
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

† Maximum normal condition design temperatures for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.2.3.

†† Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2.2 and no fuel decay heat load.

Table 1.2.3

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Table 1.2.4

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Table 1.2.5

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Table 1.2.6

HI-STORM 100 OPERATIONS SEQUENCE

Site-specific handling and operations procedures will be prepared, reviewed, and approved by each owner/user.	
1	HI-TRAC and MPC lowered into the fuel pool without lids
2	Fuel assemblies transferred into the MPC fuel basket
3	MPC lid lowered onto the MPC
4	HI-TRAC/MPC assembly moved to the decon pit and MPC lid welded in place, volumetrically or multi-layer PT examined, hydrostatically tested, and leak tested
5	MPC dewatered, moisture removed, backfilled with helium, and the closure ring welded
6	HI-TRAC annulus drained and external surfaces decontaminated
7	MPC lifting cleats installed and MPC weight supported by rigging
8	HI-TRAC pool lid removed and transfer lid attached (not applicable to HI-TRAC 125D)
9	MPC lowered and seated on HI-TRAC transfer lid (not applicable to HI-TRAC 125D)
9a	HI-STORM mating device secured to top of empty HI-STORM overpack (HI-TRAC 125D only)
10	HI-TRAC/MPC assembly transferred to atop HI-STORM overpack or mating device, as applicable
11	MPC weight supported by rigging and transfer lid doors opened (standard design HI-TRAC) or pool lid removed (HI-TRAC 125D)
12	MPC lowered into HI-STORM overpack, and HI-TRAC removed from atop HI-STORM overpack/mating device
12a	HI-STORM mating device removed (HI-TRAC 125D only)
13	HI-STORM overpack lid installed and bolted in place
14	HI-STORM overpack placed in storage at the ISFSI pad
15	For HI-STORM 100A (or 100SA) users, the overpack is anchored to the ISFSI pad by installation of nuts onto studs and torquing to the minimum required torque.

Table 1.2.7

**REPRESENTATIVE ASME BOLTING AND THREADED ROD MATERIALS
ACCEPTABLE
FOR THE HI-STORM 100A ANCHORAGE SYSTEM**

ASME MATERIALS FOR BOLTING

Composition	I.D.	Type Grade or UNC No.	Ultimate Strength (ksi)	Yield Strength (ksi)	Code Permitted Size Range [†]
C	SA-354	BC K04100	125	109	$t \leq 2.5''$
¼ Cr	SA-574	51B37M	170	135	$t \geq 5/8''$
1 Cr - 1/5 Mo	SA-574	4142	170	135	$t \geq 5/8''$
1 Cr-1/2 Mo-V	SA-540	B21 (K 14073)	165	150	$t \leq 4''$
5 Cr - ½ Mo	SA-193	B7	125	105	$t \leq 2.5''$
2Ni - ¾ Cr - ¼ Mo	SA-540	B23 (H-43400)	135	120	
2Ni - ¾ Cr - 1/3 Mo	SA-540	B-24 (K-24064)	135	120	
17Cr-4Ni-4Cu	SA-564	630 (H-1100)	140	115	
17Cr-4Ni-4Cu	SA-564	630 (H-1075)	145	125	
25Ni-15Cr-2Ti	SA-638	660	130	85	
22CR-13Ni-5Mn	SA-479	XM-19 (S20910)	135	105	

Note: The materials listed in this table are representative of acceptable materials and have been abstracted from the ASME Code, Section II, Part D, Table 3. Other materials listed in the Code are also acceptable as long as they meet the size requirements, the minimum requirements on yield and ultimate strength (see Table 2.0.4), and are suitable for the environment.

[†] Nominal diameter of the bolt (or rod) as listed in the Code tables. Two-inch diameter studs/rods are specified for the HI-STORM 100A.

APPENDIX 1.B: HOLTITE™ MATERIAL DATA

The information provided in this appendix describes the neutron absorber material, Holtite-A for the purpose of confirming its suitability for use as a neutron shield material in spent fuel storage casks. Holtite-A is one of the family of Holtite neutron shield materials denoted by the generic name Holtite™. It is currently the only neutron shield material approved for installation in the HI-STAR 100 cask. It is chemically identical to NS-4-FR which was originally developed by Bisco Inc. and used for many years as a shield material with B₄C or Pb added.

Holtite-A contains aluminum hydroxide (Al(OH)₃) in an epoxy resin binder. Aluminum hydroxide is also known by the industrial trade name of aluminum tri-hydrate or ATH. ATH is often used commercially as a fire-retardant. Holtite-A contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A contains 1% (nominal) by weight B₄C, a chemically inert material added to enhance the neutron absorption property. Pertinent properties of Holtite-A are listed in Table 1.B.1.

The essential properties of Holtite-A are:

1. the hydrogen density (needed to thermalize neutrons),
2. thermal stability of the hydrogen density, and
3. the uniformity in distribution of B₄C needed to absorb the thermalized neutrons.

ATH and the resin binder contain nearly the same hydrogen density so that the hydrogen density of the mixture is not sensitive to the proportion of ATH and resin in the Holtite-A mixture. B₄C is added as a finely divided powder and does not settle out during the resin curing process. Once the resin is cured (polymerized), the ATH and B₄C are physically retained in the hardened resin. Qualification testing for B₄C throughout a column of Holtite-A has confirmed that the B₄C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B₄C is specified in the Holtite-A mixing and pouring procedure as a precaution to assure that the B₄C concentration is always adequate throughout the mixture.

The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity assumed in the shielding analysis includes a 4% reduction to conservatively account for potential weight loss at the design temperature of 300°F or an inability to reach theoretical density. Tests on the stability of Holtite-A were performed by Holtec International. The results of the tests are summarized in Holtec Reports HI-2002396, "Holtite-A Development History and Thermal Performance Data" and HI-2002420, "Results of Pre- and Post-Irradiation Test Measurements." The information provided in these reports demonstrates that Holtite-A™ possesses the necessary thermal and radiation stability characteristics to function as a reliable shielding material in the HI-STAR 100 overpack.

The Holtite-A is encapsulated in the HI-STAR 100 overpack and, therefore, should experience a very small weight reduction during the design life of the HI-STAR 100 System. The data and test results

confirm that Holtite-A remains stable under design thermal and radiation conditions, the material properties meet or exceed that assumed in the shielding analysis, and the B_4C remains uniformly distributed with no evidence of settling or non-uniformity.

Based on the information described above, Holtite-A meets all of the requirements for an acceptable neutron shield material.

Table 1.B.1

REFERENCE PROPERTIES OF HOLTITE-A NEUTRON SHIELD MATERIAL

PHYSICAL PROPERTIES	
% ATH	62 nominal
Specific Gravity	1.68 g/cc nominal
Max. Continuous Operating Temperature	300°F
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5
wt% Hydrogen	6.0
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B ₄ C	1.0

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APPENDIX 1.D: Requirements on HI-STORM 100 Shielding Concrete

1.D.1 Introduction

The HI-STORM 100 overpack utilizes plain concrete for neutron and gamma shielding. While most of the shielding concrete used in the HI-STORM 100 overpack is installed in the annulus between the concentric structural shells, smaller quantities of concrete are also present in the pedestal shield and the overpack lid. Because plain concrete has little ability to withstand tensile stresses, but is competent in withstanding compressive and bearing loads, the design of the HI-STORM 100 overpack places no reliance on the tension-competence of the shielding concrete. ACI 318-95 provides formulas for permissible compressive and bearing stresses in plain concrete which incorporate a penalty over the corresponding permissible values in reinforced concrete. The formulas for permissible compressive and bearing stresses set forth in ACI 318-95 are used in calculations supporting this TSAR in load cases involving compression or bearing loads on the overpack concrete. However, since ACI 318-95 is intended for commercial applications and the overpack concrete is designated as an ITS Category B material, it is necessary to invoke provisions of ACI 349 (85) (which is sanctioned by NUREG-1536) for all requirements except for the allowable stress formulas (which do not exist in ACI 349) and load combinations. This appendix provides a complete set of criteria applicable to the plain concrete in the HI-STORM 100 overpack.

1.D.2 Design Requirements

The primary function of the plain concrete is to provide neutron and gamma shielding. As plain concrete is a competent structural member in compression, the plain concrete's effect on the performance of the HI-STORM overpack is included. The formulas for permissible compressive and bearing stresses set forth in ACI 318-95 are used. However, as plain concrete has very limited capabilities in tension, no tensile strength is allotted to the concrete.

The steel structure of the HI-STORM overpack provides the strength to meet all load combinations specified in Chapters 2 and 3. Credit for the structural strength of the plain concrete is limited to the compressive load carrying capability of the concrete in calculations appropriate to handling and transfer operations, and to demonstrate that the HI-STORM 100 System continues to provide functional performance in a post-accident environment. Therefore, the load combinations provided in ACI 349 and NUREG-1536, Table 3-1 are not applied to the plain concrete.

The shielding performance of the plain concrete is maintained by ensuring that the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions demonstrate that the plain concrete does not exceed the allowable long term temperature limit provided in Table 1.D.1. Under accident conditions, the bulk of the plain concrete in the HI-STORM overpack does not exceed the allowable short term temperature limit provided in Table 1.D.1. Any portion of the plain concrete which exceeds the short term temperature limit under accident conditions is neglected in the post-accident shielding analysis and in any post-accident structural analysis.

1.D.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85) supplemented by the provisions of NUREG 1536 (page 3-21) and standard good practice. Two different minimum concrete densities are specified for the overpack concrete, based on the presence or absence of the steel shield shell.

1.D.4 Construction Requirements

The HI-STORM 100 overpack is composed of a steel structure that houses plain concrete. The steel structure acts as the framework for the pouring of the concrete. The steel structure defines the dimensions of the concrete which ensures that the required thickness of concrete is provided. The fabrication sequence for the HI-STORM 100 overpack as it pertains to the concrete is provided below. All item numbers are taken from the design drawings. All nomenclature is taken from the bills-of-material.

The steel structure of the HI-STORM 100 overpack body is assembled at a qualified steel fabrication facility. However, access remains to the annulus formed by the overpack inner and outer shells (Items 3 and 2, respectively); likewise, the pedestal shell (Item 5) is welded to the baseplate (Item 1) and the pedestal platform (Item 24) to form the pedestal cavity, but penetrations exist in the baseplate to allow placement of concrete. The steel structure of the overpack body is transported to the reactor site or a nearby concrete facility.

Once the steel structure of the body is received, the body will be inspected to ensure the steel structure meets the requirements of Sections 5.1 and 6.1 of ACI 349. The concrete shall be mixed, conveyed, and deposited in accordance with Sections 5.2 through 5.4 of ACI 349. Sufficient rigidity in the steel structure overpack body is provided such that all the concrete may be placed in a single pour into each of the four segments formed by the inner shell (Item 3), outer shell (Item 2), and radial plates (Item 14). If more than one pour is performed, the requirements of Section 6.4 of ACI 349 must be met for construction joints. The pedestal shell may require bracing and support in accordance with Section 6.1 of ACI 349 to maintain the proper position and shape.

Mixing and placing of the concrete shall follow the guidance of Sections 5.6 and 5.7 for cold and hot weather conditions, respectively. Consolidation of the plain concrete shall be performed in accordance with ACI 309-87. As no reinforcement is placed in the concrete, the possibility of voids is greatly diminished. Curing of the concrete shall be in accordance with Section 5.5 of ACI 349. Water curing or accelerated curing using sealing materials methods may be used as described in ACI 308-92, Standard Practice for Curing Concrete. This would include the use of either a plastic film or a curing compound.

Non-shrink grout shall be applied as necessary to account for any deviation of the concrete elevation. To fabricate the overpack lid an identical process is followed.

Table 1.D.1 provides the construction limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

1.D.5 Testing Requirements

Table 1.D.2 provides the testing requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

Table 1.D.1: Requirements for Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density in overpack body (Minimum)	146 lb/ft ³ (HI-STORM 100 up to Serial Number (S/N) 7), 155 lb/ft ³ (HI-STORM 100 S/N 8 and higher, and HI-STORM 100S)
Density in lid and pedestal (Minimum)	146 lb/ft ³
Specified Compressive Strength	3,300 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318-95
Cement Type and Mill Test Report	Type II; Section 3.2 (ASTM C 150 or ASTM C595)
Aggregate Type	Section 3.3 (including ASTM C33(Note 2))
Nominal Maximum Aggregate Size	1 (inch)
Water Quality	Per Section 3.4
Material Testing	Per Section 3.1
Admixtures	Per Section 3.6
Maximum Water to Cement Ratio	0.5 (Table 4.5.2)
Maximum Water Soluble Chloride Ion Cl in Concrete	1.00 percent by weight of cement (Table 4.5.4)
Concrete Quality	Per Chapter 4 of ACI 349
Mixing and Placing	Per Chapter 5 of ACI 349
Consolidation	Per ACI 309-87
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Maximum Local Temperature Limit Under Long Term Conditions	200°F (See Note 3)
Maximum Section Average Temperature Limit Under Short Term Conditions	350°F (Appendix A, Subsection A.4.2)
Aggregate Maximum Value ² of Coefficient of Thermal Expansion (tangent in the range of 70°F to 100°F)	6E-06 inch/inch/°F (NUREG-1536, 3.V.2.b.i.(2)(c)2.b)

Notes:

1. All section and table references are to ACI 349 (85).
2. The coarse aggregate shall meet the requirements of ASTM C33 for class designation 1S from Table 3. However, if the requirements of ASTM C33 cannot be met, concrete that has been shown by special tests or actual service to produce concrete of adequate strength and durability meeting the requirements of Tables 1.D.1 and 1.D.2 is acceptable in accordance with ACI 349 Section 3.3.2.
3. The 200 °F long term temperature limit is specified in accordance with Paragraph A.4.3 of ACI 349 for normal conditions. The 200 °F long term temperature limit is based on (1) the use of Type II cement, specified aggregate criteria, and the specified compressive stress in Table 1.D.1, (2) the relatively small increase in long term temperature limit over the 150°F specified in Paragraph A.4.1, and (3) the very low maximum stresses calculated for normal and off-normal conditions in Section 3.4 of this FSAR.

1 Intentionally deleted.

2 The following aggregate types are a priori acceptable: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite. The thermal expansion coefficient limit does not apply when these aggregates are used. Careful consideration shall be given to the potential of long-term degradation of concrete due to chemical reactions between the aggregate and cement selected for HI-STORM 100 overpack concrete.

Table 1.D.2: Testing Requirements for Plain Concrete

TEST	SPECIFICATION
Compression Test	ASTM C31, ASTM C39, ASTM C192
Unit Weight (Density)	ASTM C138
Maximum Water Soluble Chloride Ion Concentration	Federal Highway Administration Report FHWA-RD-77-85, "Sampling and Testing for Chloride Ion in Concrete"

3.1 STRUCTURAL DESIGN

3.1.1 Discussion

The HI-STORM 100 System consists of three principal components: the Multi-Purpose Canister (MPC), the storage overpack, and the transfer cask. The MPC is a hermetically sealed, welded structure of cylindrical profile with flat ends and a honeycomb fuel basket. A complete description is provided in Subsection 1.2.1.1 wherein the anatomy of the MPC and its fabrication details are presented with the aid of figures. The MPCs utilized in the HI-STORM 100 System are identical to those for the HI-STAR 100 System submitted under Dockets 72-1008 and 71-9261. The evaluation of the MPCs presented herein draws upon the work described in those earlier submittals. In this section, the discussion is confined to characterizing and establishing the structural features of the MPC, the storage overpack, and the HI-TRAC transfer cask. Since a detailed discussion of the HI-STORM 100 Overpack and HI-TRAC transfer cask geometries is presented in Section 1.2, attention is focused here on structural capabilities and their inherent margins of safety for housing the MPC. Detailed design drawings for the HI-STORM 100 System are provided in Section 1.5.

The design of the MPC seeks to attain three objectives that are central to its functional adequacy, namely;

- **Ability to Dissipate Heat:** The thermal energy produced by the stored spent fuel must be transported to the outside surface of the MPC such that the prescribed temperature limits for the fuel cladding and for the fuel basket metal walls are not exceeded.
- **Ability to Withstand Large Impact Loads:** The MPC, with its payload of nuclear fuel, must be sufficiently robust to withstand large impact loads associated with the postulated handling accident events. Furthermore, the strength of the MPC must be sufficiently isotropic to meet structural requirements under a variety of handling and tip-over accidents.
- **Restraint of Free End Expansion:** The membrane and bending stresses produced by restraint of free-end expansion of the fuel basket are categorized as primary stresses. In view of the concentration of heat generation in the fuel basket, it is necessary to ensure that structural constraints to its external expansion do not exist.

Where the first two criteria call for extensive inter-cell connections, the last criterion requires the opposite. The design of the MPC seeks to realize all of the above three criteria in an optimal manner.

From the description presented in Chapter 1, the MPC enclosure vessel is the confinement vessel designed to meet ASME Code, Section III, Subsection NB stress limits. The enveloping canister shell, the baseplate, and the lid system form a complete confinement boundary for the stored fuel that is referred to as the "enclosure vessel". Within this cylindrical shell confinement vessel is an integrally welded assemblage of cells of square cross sectional openings for fuel storage, referred to herein as the fuel basket. The fuel basket is analyzed under the provisions of Subsection NG of Section III of the ASME Code. All multi-purpose canisters designed for deployment in the HI-STORM 100 and HI-STAR 100 systems are exactly alike in their external dimensions. The essential

difference between the MPCs lies in the fuel baskets. Each fuel storage MPC is designed to house fuel assemblies with different characteristics. Although all fuel baskets are configured to maximize structural ruggedness through extensive inter-cell connectivity, they are sufficiently dissimilar in structural details to warrant separate evaluations. Therefore, analyses for each of the MPC types were carried out to ensure structural compliance. Inasmuch as no new MPC designs are introduced in this application, and all MPC designs were previously reviewed by the USNRC under Docket 72-1008, the MPC analyses submitted under Docket Numbers 72-1008 and 71-9261 for the HI-STAR 100 System are not reproduced herein unless they need to be modified by HI-STORM 100 conditions or geometry differences. Analyses provided in the HI-STAR 100 System safety analysis reports that are applicable to the HI-STORM 100 System are referenced in this FSAR by docket number and subsection or appendix.

Components of the HI-STORM 100 System that are important to safety and their applicable design codes are defined in Chapter 2.

Some of the key structural functions of the MPC in the storage mode are:

1. To position the fuel in a subcritical configuration, and
2. To provide a confinement boundary.

Some of the key structural functions of the overpack in the storage mode are:

1. To serve as a missile barrier for the MPC,
2. To provide flow paths for natural convection,
3. To ensure stability of the HI-STORM 100 System, and
4. To maintain the position of the radiation shielding.
5. To allow movement of the overpack with a loaded MPC inside.

Some structural features of the MPCs that allow the system to perform these functions are summarized below:

- There are no gasketed ports or openings in the MPC. The MPC does not rely on any sealing arrangement except welding. The absence of any gasketed or flanged joints makes the MPC structure immune from joint leaks. The confinement boundary contains no valves or other pressure relief devices.

- The closure system for the MPCs consists of two components, namely, the MPC lid and the closure ring. The MPC lid can be either a single thick circular plate continuously welded to the MPC shell along its circumference or two dual lids welded around their common periphery. The MPC closure system is shown in the Design Drawings in Section 1.5. The MPC lid is equipped with vent and drain ports which are utilized for evacuating moisture and air from the MPC following fuel loading, and subsequent backfilling with an inert gas (helium) at a specified mass. The vent and drain ports are covered by a cover plate and welded before the closure ring is installed. The closure ring is a circular annular plate edge-welded to the MPC lid and shell. The two closure members are interconnected by welding around the inner diameter of the ring. Lift points for the MPC are provided in the MPC lid.
- The MPC fuel baskets consist of an array of interconnecting plates. The number of storage cells formed by this interconnection process varies depending on the type of fuel being stored. Basket designs containing cell configurations for PWR and BWR fuel have been designed and are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell. Welding of the basket plates along their edges essentially renders the fuel basket into a multiflange beam. Figure 3.1.1 provides an isometric illustration of a fuel basket for the MPC-68 design.
- The MPC basket is separated from its supports by a gap. The gap size decreases as a result of thermal expansion (depending on the magnitude of internal heat generation from the stored spent fuel). The provision of a small gap between the basket and the basket support structure is consistent with the natural thermal characteristics of the MPC. The planar temperature distribution across the basket, as shown in Section 4.4, approximates a shallow parabolic profile. This profile will create high thermal stresses unless structural constraints at the interface between the basket and the basket support structure are removed.
- The MPCs will be loaded with fuel with widely varying heat generation rates. The basket/basket support structure gap tends to be reduced for higher heat generation rates due to increased thermal expansion rates. Gaps between the fuel basket and the basket support structure are specified to be sufficiently large such that a gap exists around the periphery after any thermal expansion.
- A small number of flexible thermal conduction elements (thin aluminum tubes) are interposed between the basket and the MPC shell. The elements are designed to be resilient. They do not provide structural support for the basket, and thus their resistance to thermal growth is negligible.

It is quite evident from the geometry of the MPC that a critical loading event pertains to the drop condition when the MPC is postulated to undergo a handling side drop (the longitudinal axis of the MPC is horizontal) or tip-over. Under the side drop or tip-over condition the flat panels of the fuel basket are subject to an equivalent pressure loading that simulates the deceleration-magnified inertia load from the stored fuel and the MPC's own metal mass.

The MPC fuel basket maintains the spent nuclear fuel in a subcritical arrangement. Its safe operation is assured by maintaining the physical configuration of the storage cell cavities intact in the aftermath of a drop event. This requirement is considered to be satisfied if the MPC fuel basket meets the stress intensity criteria set forth in the ASME Code, Section III, Subsection NG. Therefore, the demonstration that the fuel basket meets Subsection NG limits ensures that there is no impairment of ready retrievability (as required by NUREG-1536), and that there is no unacceptable effect on the subcritical arrangement.

The MPC confinement boundary contains no valves or other pressure relief devices. The MPC enclosure vessel is shown to meet the stress intensity criteria of the ASME Code, Section III, Subsection NB for all service conditions. Therefore, the demonstration that the enclosure vessel meets Subsection NB limits ensures that there is no unacceptable release of radioactive materials.

The HI-STORM 100 storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of unreinforced (plain) concrete. Additional regions of fully confined (by enveloping steel structure) unreinforced concrete are attached to the lid and to the baseplate. The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a cushion for the MPC in the event of a tip-over accident. The storage overpack is not a pressure vessel since it contains cooling vents that do not allow for a differential pressure to develop across the overpack wall. The structural steel components of the HI-STORM 100 Overpack are designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3. A short version of the HI-STORM 100 overpack, designated as the HI-STORM 100S, is introduced in this revision. To accommodate nuclear plants with limited height access, the HI-STORM 100S has a re-configured lid and a lower overall height. There are minor weight redistributions but the overall bounding weight of the system is unchanged. Therefore, structural analyses are revisited if and only if the modified configuration cannot be demonstrated to be bounded by the original calculation. New or modified calculations focused on the HI-STORM 100 are clearly identified within the text of this chapter. Unless otherwise designated, general statements using the terminology "HI-STORM 100" also apply to the HI-STORM 100S. The HI-STORM 100S can carry all MPC's and transfer casks that can be carried in the HI-STORM 100.

As discussed in Chapters 1 and 2, and Section 3.0, the principal shielding material utilized in the HI-STORM 100 Overpack is plain concrete. Plain concrete was selected for the HI-STORM 100 Overpack in lieu of reinforced concrete, because there is no structural imperative for incorporating tensile load bearing strength into the contained concrete. From a purely practical standpoint, the absence of rebars facilitate pouring and curing of concrete with minimal voids, which is an important consideration in light of its shielding function in the HI-STORM 100 Overpack. Plain concrete, however, acts essentially identical to reinforced concrete under compressive and bearing loads, even

though ACI standards apply a penalty factor on the compressive and bearing strength of concrete in the absence of rebars (vide ACI 318.1).

Accordingly, the plain concrete in the HI-STORM 100 is considered as a structural material only to the extent that it may participate in supporting direct compressive loads. The allowable compression/bearing resistance is defined and quantified in the ACI 318.1(92) Building Code for Structural Plain Concrete.

In general, strength analysis of the HI-STORM 100 Overpack and its confined concrete is carried out only to demonstrate that the concrete is able to perform its radiation protection function and that retrievability of the MPC subsequent to any postulated accident condition of storage or handling is maintained.

A discrete ITS component in the HI-STORM 100 System is the HI-TRAC transfer cask. The HI-TRAC serves to provide a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM 100 Overpack. The HI-TRAC body is a double-walled steel cylinder that constitutes its structural system. Contained between the two steel shells is an intermediate lead cylinder. Attached to the exterior of the HI-TRAC body outer shell is a water jacket that acts as a radiation barrier. The HI-TRAC is not a pressure vessel since it contains a penetration in the HI-TRAC top lid that does not allow for a differential pressure to develop across the HI-TRAC wall. Nevertheless, in the interest of conservatism, structural steel components of the HI-TRAC are subject to the stress limits of the ASME Code, Section III, Subsection NF, Class 3.

Since both the HI-STORM 100 and HI-TRAC may serve as an MPC carrier, their lifting attachments are designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and ANSIN14.6-1993 [3.1.2] for single-failure-proof lifting equipment.

Table 2.2.6 provides a listing of the applicable design codes for all structures, systems, and components which are designated as ITS.

3.1.2 Design Criteria

Principal design criteria for normal, off-normal, and accident/environmental events are discussed in Section 2.2. In this section, the loads, load combinations, and allowable stresses used in the structural evaluation of the HI-STORM 100 System are presented in more detail.

Consistent with the provisions of NUREG-1536, the central objective of the structural analysis presented in this chapter is to ensure that the HI-STORM 100 System possesses sufficient structural capability to withstand normal and off-normal loads and the worst case loads under natural phenomenon or accident events. Withstanding such loadings enables the HI-STORM 100 System to successfully preclude the following negative consequences:

- unacceptable risk of criticality
- unacceptable release of radioactive materials
- unacceptable radiation levels
- impairment of ready retrievability of the SNF

The above design objectives for the HI-STORM 100 System can be particularized for individual components as follows:

- The objectives of the structural analysis of the MPC are to demonstrate that:
 1. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.
 2. The MPC basket does not deform under credible loading conditions such that the subcriticality or retrievability of the SNF is jeopardized.
- The objectives of the structural analysis of the storage overpack are to demonstrate that:
 1. Tornado-generated missiles do not compromise the integrity of the MPC confinement boundary.
 2. The overpack can safely provide for on-site transfer of the loaded MPC and ensure adequate support to the HI-TRAC transfer cask during loading and unloading of the MPC.
 3. The radiation shielding remains properly positioned in the case of any normal, off-normal, or natural phenomenon or accident event.
 4. The flow path for the cooling air flow shall remain available under normal and off-normal conditions of storage and after a natural phenomenon or accident event.
 5. The loads arising from normal, off-normal, and accident level conditions exerted on the contained MPC do not exceed the structural design criteria of the MPC.
 6. No geometry changes occur under any normal, off-normal, and accident level conditions of storage that may preclude ready retrievability of the contained MPC.

7. A freestanding storage overpack can safely withstand a non-mechanistic tip-over event with a loaded MPC within the overpack. The HI-STORM 100A is specifically engineered to be permanently attached to the ISFSI pad. The ISFSI pad engineered for the anchored cask is designated as "Important to Safety". Therefore, the non-mechanistic tipover is not applicable to the HI-STORM 100A.
 8. The inter-cask transfer of a loaded MPC can be carried out without exceeding the structural capacity of the HI-STORM 100 Overpack, provided all required auxiliary equipment and components specific to an ISFSI site comply with their Design Criteria set forth in this FSAR and the handling operations are in full compliance with operational limits and controls prescribed in this FSAR.
- The objective of the structural analysis of the HI-TRAC transfer cask is to demonstrate that:
 1. Tornado generated missiles do not compromise the integrity of the MPC confinement boundary while the MPC is contained within HI-TRAC.
 2. No geometry changes occur under any postulated handling or storage conditions that may preclude ready retrievability of the contained MPC.
 3. The structural components perform their intended function during lifting and handling with the loaded MPC.
 4. The radiation shielding remains properly positioned under all applicable handling service conditions for HI-TRAC.
 5. The lead shielding, top lid, and transfer lid doors remain properly positioned during postulated handling accidents.

The aforementioned objectives are deemed to be satisfied for the MPC, the overpack, and the HI-TRAC, if stresses (or stress intensities, as applicable) calculated by the appropriate structural analyses are less than the allowables defined in Subsection 3.1.2.2, and if the diametral change in the storage overpack (or HI-TRAC), if any, after any event of structural consequence to the overpack (or transfer cask), does not preclude ready retrievability of the contained MPC.

Stresses arise in the components of the HI-STORM 100 System due to various loads that originate under normal, off-normal, or accident conditions. These individual loads are combined to form load combinations. Stresses and stress intensities resulting from the load combinations are compared to their respective allowable stresses and stress intensities. The following subsections present loads, load combinations, and the allowable limits germane to them for use in the structural analyses of the MPC, the overpack, and the HI-TRAC transfer cask.

3.1.2.1 Loads and Load Combinations

The individual loads applicable to the HI-STORM 100 System and the HI-TRAC cask are defined in Section 2.2 of this report (Table 2.2.13). Load combinations are developed by assembling the individual loads that may act concurrently, and possibly, synergistically (Table 2.2.14). In this subsection, the individual loads are further clarified as appropriate and the required load combinations are identified. Table 3.1.1 contains the load combinations for the storage overpack where kinematic stability is of primary importance. The load combinations where stress or load level is of primary importance are set forth in Table 3.1.3 for the MPC fuel basket, in Table 3.1.4 for the MPC confinement boundary, and in Table 3.1.5 for the storage overpack and the HI-TRAC transfer cask. Load combinations are applied to the mathematical models of the MPCs, the overpack, and the HI-TRAC. Results of the analyses carried out under bounding load combinations are compared with their respective allowable stresses (or stress intensities, as applicable). The analysis results from the bounding load combinations are also assessed, where warranted, to ensure satisfaction of the functional performance criteria discussed in the preceding subsection.

3.1.2.1.1 Individual Load Cases

The individual loads that address each design criterion applicable to the structural design of the HI-STORM 100 System are catalogued in Table 2.2.13. Each load is given a symbol for subsequent use in the load combination listed in Table 2.2.14.

Accident condition and natural phenomena-induced events, collectively referred to as the "Level D" condition in Section III of the ASME Boiler & Pressure Vessel Codes, in general, do not have a universally prescribed limit. For example, the impact load from a tornado-borne missile, or the overturning load under flood or tsunami, cannot be prescribed as design basis values with absolute certainty that all ISFSI sites will be covered. Therefore, as applicable, allowable magnitudes of such loadings are postulated for the HI-STORM 100 System. The allowable values are drawn from regulatory and industry documents (such as for tornado missiles and wind) or from an intrinsic limitation in the system (such as the permissible "drop height" under a postulated handling accident). In the following, the essential characteristic of each "Level D" type loading is explained.

3.1.2.1.1.1 Tip-Over

It is required to demonstrate that the free-standing HI-STORM 100 storage overpack, containing a loaded MPC, will not tip over as a result of a postulated natural phenomenon event, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tip-over scenario per NUREG-1536 is analyzed. Since the HI-STORM 100S has an overall length that is less than the regular HI-STORM 100, the maximum impact velocity of the overpack will be reduced. Therefore, the results of the tipover analysis for the HI-STORM 100 (reported in Appendix 3.A) are bounding for the HI-STORM 100S. The potential of the HI-STORM 100 Overpack tipping over during the lowering (or raising) of the loaded MPC into (or out of) it with the HI-TRAC cask mounted on it is ruled out because of the safeguards and devices mandated by this FSAR for such operations

(Subsection 2.3.3.1 and Technical Specification 4.9). The physical and procedural barriers under the MPC handling operations have been set down in the FSAR to preclude overturning of the HI-STORM/HI-TRAC assemblage with an extremely high level of certainty. Much of the ancillary equipment needed for the MPC transfer operations must be custom engineered to best accord with the structural and architectural exigencies of the ISFSI site. Therefore, with the exception of the HI-TRAC cask, their design cannot be prescribed, a priori, in this FSAR. However, carefully drafted Design Criteria and conditions of use set forth in this FSAR eliminate the potential of weakening of the safety measures contemplated herein to preclude an overturning event during MPC transfer operations. Subsection 2.3.3.1 contains a comprehensive set of design criteria for the ancillary equipment and components required for MPC transfer operations to ensure that the design objective of precluding a kinematic instability event during MPC transfer operations is met. Further information on the steps taken to preclude system overturning during MPC transfer operations may be found in Chapter 8, Section 8.0.

In the HI-STORM 100A configuration, wherein the overpack is physically anchored to the ISFSI pad, the potential for a tip-over is a priori precluded. Therefore, the ISFSI pad need not be engineered to be sufficiently compliant to limit the peak MPC deceleration to Table 2.2.8 values. The stiffness of the pad, however, may be controlled by the ISFSI structural design and, therefore, may result in a reduced "carry height" from that specified for a free-standing cask. If a non-single failure proof lifting device is employed to carry the cask over the pad, determination of maximum carry height must be performed by the ISFSI owner once the ISFSI pad design is formalized.

3.1.2.1.1.2 Handling Accident

A handling accident during transport of a loaded HI-STORM 100 storage overpack is assumed to result in a vertical drop. The HI-STORM 100 storage overpack will not be handled in a horizontal position while containing a loaded MPC. Therefore, a side drop is not considered a credible event.

HI-TRAC can be carried in a horizontal orientation while housing a loaded MPC. Therefore, a handling accident during transport of a loaded HI-TRAC in a horizontal orientation is considered to be a credible accident event.

As discussed in the foregoing, the vertical drop of the HI-TRAC and the tip-over of the assemblage of a loaded HI-TRAC on the top of the HI-STORM 100 storage overpack during MPC transfer operations do not need to be considered.

3.1.2.1.1.3 Flood

The postulated flood event results into two discrete scenarios which must be considered; namely,

1. stability of the HI-STORM 100 System due to flood water velocity, and
2. structural effects of hydrostatic pressure and water velocity induced lateral pressure.

The maximum hydrostatic pressure on the cask in a flood where the water level is conservatively set at 125 feet is calculated as follows:

Using

p = the maximum hydrostatic pressure on the system (psi),
 γ = weight density of water = 62.4 lb/ft³
 h = the height of the water level = 125 ft;

The maximum hydrostatic pressure is

$$p = \gamma h = (62.4 \text{ lb/ft}^3)(125 \text{ ft})(1 \text{ ft}^2/144 \text{ in}^2) = 54.2 \text{ psi}$$

The accident condition design external pressure for the MPC (Table 2.2.1) bounds the maximum hydrostatic pressure exerted by the flood.

3.1.2.1.1.4 Explosion

Explosion, by definition, is a transient event. Explosive materials (except for the short duration when a limited quantity of motive fuel for placing the loaded MPC on the ISFSI pad is present in the tow vehicle) are prohibited in the controlled area by specific stipulation in the HI-STORM 100 Technical Specification. However, pressure waves emanating from explosions in areas outside the ISFSI are credible.

Pressure waves from an explosive blast in a property near the ISFSI site result in an impulsive aerodynamic loading on the stored HI-STORM 100 Overpacks. Depending on the rapidity of the pressure build-up, the inside and outside pressures on the HI-STORM METCON™ shell may not equalize, leading to a net lateral loading on the upright overpack as the pressure wave traverses the overpack. The magnitude of the dynamic pressure wave is conservatively set to a value below the magnitude of the pressure differential that would cause a tip-over of the cask if the pulse duration was set at one second. With the maximum design basis pressure pulse established (by setting the design basis pressure differential sufficiently low that cask tip-over is not credible due to the travelling pressure wave), the stress state under this condition requires analysis. The lateral pressure difference, applied over the overpack full height, causes axial and circumferential stresses and strains to develop. Level D stress limits must not be exceeded under this state of stress. It must also be demonstrated that no permanent ovalization of the cross section occurs that leads to loss of clearance to remove the MPC after the explosion.

Once the pressure wave traverses the cask body, then an elastic stability evaluation is warranted. An all-enveloping pressure from the explosion may threaten safety by buckling the overpack outer shell.

In contrast to the overpack, the MPC is a closed pressure vessel. Because of the enveloping overpack around it, the explosive pressure wave would manifest as an external pressure on the external surface of the MPC.

The maximum overpressure on the MPC resulting from an explosion is limited by the HI-STORM Technical Specification to be equal to or less than the accident condition design external pressure or external pressure differential specified in Table 2.2.1. The design external pressure differential is applied as a component of the load combinations.

3.1.2.1.1.5 Tornado

The three components of a tornado load are:

1. pressure changes,
2. wind loads, and
3. tornado-generated missiles.

Wind speeds and tornado-induced pressure drop are specified in Table 2.2.4. Tornado missiles are listed in Table 2.2.5. A central functional objective of a storage overpack is to maintain the integrity of the “confinement boundary”, namely, the multi-purpose canister stored inside it. This operational imperative requires that the mechanical loadings associated with a tornado at the ISFSI do not jeopardize the physical integrity of the loaded MPC. Potential consequences of a tornado on the cask system are:

- Instability (tip-over) due to tornado missile impact plus either steady wind or impulse from the pressure drop (only applicable for free-standing cask).
- Stress in the overpack induced by the lateral force caused by the steady wind or missile impact.
- Loadings applied on the MPC transmitted to the inside of the overpack through its openings or as a secondary effect of loading on the enveloping overpack structure.
- Excessive storage overpack permanent deformation that may prevent ready retrievability of the MPC.
- Excessive storage overpack permanent deformation that may significantly reduce the shielding effectiveness of the storage overpack.

Analyses must be performed to ensure that, due to the tornado-induced loadings:

- The loaded overpack does not become kinematically unstable (only applicable for free-standing cask).

- The overpack does not deform plastically such that the retrievability of the stored MPC is threatened.
- The MPC does not sustain an impact from an incident missile.
- The MPC is not subjected to inertia loads (acceleration or deceleration) in excess of its design basis limit set forth in Chapter 2 herein.
- The overpack does not deform sufficiently due to tornado-borne missiles such that the shielding effectiveness of the overpack is significantly affected.

The results obtained for the HI-STORM 100 bound the corresponding results for HI-STORM 100S because of the reduced height. In the anchored configuration (HI-STORM 100A), the kinematic stability requirement stated above is replaced with the requirement that the stresses in the anchor studs do not exceed level D stress limits for ASME Section III, Class 3, Subsection NF components.

3.1.2.1.1.6 Earthquake

Subsections 2.2.3.7 and 3.4.7 contain the detailed specification of the seismic inputs applied to the HI-STORM 100 System. The design basis earthquake is assumed to be at the top of the ISFSI pad. Potential consequences of a seismic event are sliding/overturning of a free-standing cask, overstress of the sector lugs and anchor studs for the anchored HI-STORM 100A, and lateral force on the overpack causing excessive stress and deformation of the storage overpack.

In the anchored configuration (HI-STORM 100A), a seismic event results in a fluctuation in the state of stress in the anchor bolts and a local bending action on the sector lugs.

Analyses must be performed to ensure that:

- The maximum axial stress in the anchor bolts remains below the Level D stress limits for Section III Class 3 Subsection NF components.
- The maximum primary membrane plus bending stress intensity in the sector lugs during the DBE event satisfies Level D stress limits of the ASME Code, Subsection NF.
- The anchor bolts will not sustain fatigue failure due to pulsation in their axial stress during the DBE event.
- The stress in the weld line joining the sector lugs to the HI-STORM 100 weldment is within Subsection NF limits for Level D condition.

3.1.2.1.1.7 Lightning

The HI-STORM 100 Overpack contains over 25,000 lb of highly conductive carbon steel with over 700 square feet of external surface area. Such a large surface area and metal mass is adequate to dissipate any lightning that may strike the HI-STORM 100 System. There are no combustible materials on the HI-STORM 100 surface. Therefore, lightning will not impair the structural performance of components of the HI-STORM 100 System that are important to safety.

3.1.2.1.1.8 Fire

The potential structural consequences of a fire are: the possibility of an interference developing between the storage overpack and the loaded MPC due to free thermal expansion; and, the degradation of material properties to the extent that their structural performance is affected during a subsequent recovery action. The fire condition is addressed to the extent necessary to demonstrate that these adverse structural consequences do not materialize.

3.1.2.1.1.9 100% Fuel Rod Rupture

The effect on structural performance by 100% fuel rod rupture is felt as an increase in internal pressure. The accident internal pressure limit set in Chapter 2 bounds the pressure from 100% fuel rod rupture. Therefore, no new load condition has been identified.

3.1.2.1.2 Load Combinations

Load combinations are created by summing the effects of several individual loads. The load combinations are selected for the normal, off-normal, and accident conditions. The loadings appropriate for HI-STORM 100 under the various conditions are presented in Table 2.2.14. These loadings are combined into meaningful combinations for the various HI-STORM 100 System components in Tables 3.1.1, and 3.1.3-3.1.5. Table 3.1.1 lists the load combinations that address overpack stability. Tables 3.1.3 through 3.1.5 list the applicable load combinations for the fuel basket, the enclosure vessel, and the overpack and HI-TRAC, respectively.

As discussed in Subsection 2.2.7, the number of discrete load combinations for each situational condition (i.e., normal, off-normal, etc.) is consolidated by defining bounding loads for certain groups of loadings. Thus, the accident condition pressure P_o^* bounds the surface loadings arising from accident and extreme natural phenomenon events, namely, tornado wind W' , flood F , and explosion E^* .

As noted previously, certain loads, namely earthquake E , flowing water under flood condition F , force from an explosion pressure pulse F^* , and tornado missile M , act to destabilize a cask. Additionally, these loads act on the overpack and produce essentially localized stresses at the HI-STORM 100 System to ISFSI interface. Table 3.1.1 provides the load combinations that are relevant to the stability analyses of free-standing casks. The site ISFSI DBE zero period acceleration (ZPA) must be bounded by the design basis seismic ZPA defined by the Load Combination C of Table 3.1.1 to demonstrate that the margin against tip-over during a seismic event is maintained.

The major constituents in the HI-STORM 100 System are: (i) the fuel basket, (ii) the enclosure vessel, (iii) the HI-STORM 100 (or HI-STORM 100S) Overpack, and (iv) the HI-TRAC transfer cask. The fuel basket and the enclosure vessel (EV) together constitute the multi-purpose canister. The multi-purpose canister (MPC) is common to HI-STORM 100 and HI-STAR 100, and as such, has been extensively analyzed in the storage FSAR and transport SAR (Dockets 72-1008 and 71-9261) for HI-STAR 100. Many of the loadings on the MPC (fuel basket and enclosure vessel) are equal to or bounded by loadings already considered in the HI-STAR 100 SAR documents. Where such analyses have been performed, their location in the HI-STAR 100 SAR documents is indicated in this HI-STORM 100 SAR for continuity in narration. A complete account of analyses and results for all load combinations for all four constituents parts is provided in Section 3.4 as required by Regulatory Guide 3.61.

In the following, the loadings listed as applicable for each situational condition in Table 2.2.14 are addressed in meaningful load combinations for the fuel basket, enclosure vessel, and the overpack. Each component is considered separately.

Fuel Basket

Table 3.1.3 summarizes all loading cases (derived from Table 2.2.14) that are germane to demonstrating compliance of the fuel baskets to Subsection NG when these baskets are housed within HI-STORM 100 or HI-TRAC.

The fuel basket is not a pressure vessel; therefore, the pressure loadings are not meaningful loads for the basket. Further, the basket is structurally decoupled from the enclosure vessel. The gap between the basket and the enclosure vessel is sized to ensure that no constraint of free-end thermal expansion of the basket occurs. The demonstration of the adequacy of the basket-to-the-enclosure vessel (EV) gap to ensure absence of interference is a physical problem that must be analyzed.

The normal handling loads on the fuel basket in an MPC within the HI-STORM 100 System or the HI-TRAC transfer cask are identical to or bounded by the normal handling loads analyzed in the HI-STAR 100 FSAR Docket Number 72-1008.

Three accident condition scenarios must be considered: (i) drop with the storage overpack axis vertical; (ii) drop with the HI-TRAC axis horizontal; and (iii) storage overpack tipover. The vertical drop scenario is considered in the HI-STAR 100 SAR.

The horizontal drop and tip-over must consider multiple orientation of the fuel basket, as the fuel basket is not radially symmetric. Therefore, two horizontal drop orientations are considered which are referred to as the 0 degree drop and 45 degree drop, respectively. In the 0 degree drop, the basket drops with its panels oriented parallel and normal to the vertical (see Figure 3.1.2). The 45-degree drop implies that the basket's honeycomb section is rotated meridionally by 45 degrees (Figure 3.1.3).

Enclosure Vessel

Table 3.1.4 summarizes all load cases that are applicable to structural analysis of the enclosure vessel to ensure integrity of the confinement boundary.

The enclosure vessel is a pressure vessel consisting of a cylindrical shell, a thick circular baseplate at the bottom, and a thick circular lid at the top. This pressure vessel must be shown to meet the primary stress intensity limits for ASME Section III Class 1 at the design temperature and primary plus secondary stress intensity limits under the combined action of pressure plus thermal loads.

Normal handling of the enclosure vessel is considered in Docket 72-1008; the handling loads are independent of whether the enclosure vessel is within HI-STAR 100, HI-STORM 100, or HI-TRAC.

The off-normal condition handling loads are identical to the normal condition and, therefore, a separate analysis is not required.

Analyses presented in this chapter are intended to demonstrate that the maximum decelerations in drop and tip-over accident events are limited by the bounding values in Table 3.1.2. The vertical drop event is considered in the HI-STAR 100 SAR Docket 72-1008.

The deceleration loadings developed in the enclosure vessel during a horizontal drop event are combined with those due to P_i (internal pressure) acting alone. The accident condition pressure is bounded by P_i^* . The design basis deceleration for the MPC in the HI-STAR 100 System is 60g's, whereas the design basis deceleration for the MPC in the HI-STORM 100 System is 45g's. The design pressures are identical. The fire event (T^* loading) is considered for ensuring absence of interference between the enclosure vessel and the fuel basket and between the enclosure vessel and the overpack.

It is noted that the MPC basket-enclosure vessel thermal expansion and stress analyses are reconsidered in this submittal to reflect the different MPC-to-overpack gaps that exist in the HI-STORM 100 Overpack versus the HI-STAR 100 overpack, coupled with the different design basis decelerations.

Storage Overpack

Table 3.1.5 identifies the load cases to be considered for the overpack. These are in addition to the kinematic criteria listed in Table 3.1.1. Within these load cases and kinematic criteria, the following items must be addressed:

Normal Conditions

- The dead load of the HI-TRAC with the heaviest loaded MPC (dry) on top of the HI-STORM 100 Overpack must be shown to be able to be supported by the metal-concrete (METCON™) structure consisting of the two concentric steel shells and the steel rib plates, and by the concrete columns away from the vent regions.
- The dead load of the HI-STORM 100 Overpack itself must be supportable by the steel structure with no credit for concrete strength other than self-support in compression.
- Normal handling loads must be accommodated without taking any strength credit from the contained concrete other than self-support in compression.

Accident Conditions

- Maximum flood water velocity for the overpack with an empty MPC must be specified to ensure that no sliding or tip-over occurs.
- Tornado missile plus wind on an overpack with an empty MPC must be specified to demonstrate that no cask tip-over occurs.
- Tornado missile penetration analysis must demonstrate that the postulated large and penetrant missiles cannot contact the MPC. The small missile must be shown not to penetrate the MPC pressure vessel boundary, since it can enter the overpack cavity through the vent ducts.
- Under seismic conditions, a fully loaded, free-standing HI-STORM 100 overpack must be demonstrated to not tip over under the maximum ZPA event. The maximum sliding of the overpack must demonstrate that casks will not impact each other.
- Under a non-mechanistic postulated tip-over of a fully loaded, free-standing HI-STORM 100 overpack, the overpack lid must not dislodge.
- Accident condition stress levels must not be exceeded in the steel and compressive stress levels in the concrete must remain within allowable limits.
- Accident condition induced gross general deformations of the storage overpack must be limited to values that do not preclude ready retrievability of the MPC.

As noted earlier, analyses performed using the HI-STORM 100 generally provide results that are identical to or bound results for the shorter HI-STORM 100S; therefore, analyses are not repeated specifically for the HI-STORM 100S unless the specific geometry changes significantly influence the safety factors.

HI-TRAC Transfer Cask

Table 3.1.5 identifies load cases applicable to the HI-TRAC transfer cask.

The HI-TRAC transfer cask must provide radiation protection, must act as a handling cask when carrying a loaded MPC, and in the event of a postulated accident must not suffer permanent deformation to the extent that ready retrievability of the MPC is compromised. This submittal includes three types of transfer casks: a 125-ton HI-TRAC (referred to as the HI-TRAC 125), a modified version of the HI-TRAC 125 called the HI-TRAC 125D, and a 100-ton HI-TRAC. The details of these three transfer casks are provided in the design drawings in Section 1.5. The same steel structures (i.e., shell thicknesses, lid thicknesses, etc.) are maintained with the only major differences being in the amount of lead shielding, the water jacket configuration, the bottom flange, and the lower dead weight loading. Therefore, all structural analyses performed for the HI-TRAC 125 are repeated for the HI-TRAC 125D and the HI-TRAC 100 only if it cannot be clearly demonstrated that the HI-TRAC 125 calculation is bounding.

3.1.2.2 Allowables

The important to safety components of the HI-STORM 100 System are listed in Table 2.2.6. Allowable stresses, as appropriate, are tabulated for these components for all service conditions.

In Subsection 2.2.5, the applicable service level from the ASME Code for determination of allowables is listed. Table 2.2.14 provides a tabulation of normal, off-normal, and accident conditions and the service levels defined in the ASME Code, along with the applicable loadings for each service condition.

Allowable stresses and stress intensities are calculated using the data provided in the ASME Code and Tables 2.2.10 through 2.2.12. Tables 3.1.6 through 3.1.16 contain numerical values of the stresses/stress intensities for all MPC, overpack, and HI-TRAC load bearing materials as a function of temperature.

In all tables the terms S , S_m , S_y , and S_u , respectively, denote the design stress, design stress intensity, minimum yield strength, and the ultimate strength. Property values at intermediate temperatures that are not reported in the ASME Code are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the Code in any structural calculation.

Additional terms relevant to the analyses are extracted from the ASME Code (Figure NB-3222-1, for example) as follows:

Symbol	Description	Notes
P_m	Average primary stress across a solid section	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.
P_L	Average stress across any solid section	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
P_b	Primary bending stress	Component of primary stress proportional to the distance from the centroid of a solid section. Excludes the effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
P_e	Secondary expansion stress	Stresses that result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration. (Not applicable to vessels.)
Q	Secondary membrane plus bending stress	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion.
F	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. This value is not used in the tables.

It is shown that there is no interference between component parts due to free thermal expansion. Therefore, P_e does not develop within any HI-STORM 100 component.

It is recognized that the planar temperature distribution in the fuel basket and the overpack under the maximum heat load condition is the highest at the cask center and drops monotonically, reaching its lowest value at the outside surface. Strictly speaking, the allowable stresses/stress intensities at any location in the basket, the enclosure vessel, or the overpack should be based on the coincident metal temperature under the specific operating condition. However, in the interest of conservatism, reference temperatures are established for each component that are upper bounds on the metal temperature for each situational condition. Table 3.1.17 provides the reference temperatures for the fuel basket and the MPC canister utilizing Tables 3.1.6 through 3.1.16, and provides conservative numerical limits for the stresses and stress intensities for all loading cases. Reference temperatures for the MPC baseplate and the MPC lid are 400 degrees F and 550 degrees F, respectively, as specified in Table 2.2.3.

Finally, the lift devices in the HI-STORM 100 Overpack and HI-TRAC casks and the multi-purpose canisters, collectively referred to as "trunnions", are subject to specific limits set forth by NUREG-0612: the primary stresses in a trunnion must be less than the smaller of 1/10 of the material ultimate strength and 1/6 of the material yield strength under a normal handling condition (Load Case 01 in Table 3.1.5). The load combination D+H in Table 3.1.5 is equivalent to 1.15D. This is further explained in Subsection 3.4.3.

The region around the trunnions is part of the NF structure in HI-STORM 100 and HI-TRAC and NB pressure boundary in the MPC, and as such, must satisfy the applicable stress (or stress intensity) limits for the load combination. In addition to meeting the applicable Code limits, it is further required that the primary stress required to maintain equilibrium at the defined trunnion/mother structure interface must not exceed the material yield stress at three times the handling condition load (1.15D). This criterion, mandated by Regulatory Guide 3.61, Section 3.4.3, insures that a large safety factor exists on non-local section yielding at the trunnion/mother structure interface that would lead to unacceptable section displacement and rotation.

3.1.2.3 Brittle Fracture

The MPC canister and basket are constructed from a series of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum temperature range of the HI-STORM 100 System. Therefore, brittle fracture is not a concern for the MPC components. Such an assertion can not be made a priori for the HI-STORM storage overpack and HI-TRAC transfer cask that contain ferritic steel parts. In normal storage mode, the lowest service temperature (LST) of the HI-STORM storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). During the HI-STORM handling operations, the applicable lowest service temperature is 0°F (which is the threshold ambient temperature below which lifting and handling of the HI-STORM 100 Overpack or the HI-TRAC cask is not permitted by the Technical Specification). Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM 100 Overpack and the HI-TRAC cask; namely,

LST = 0°F for parts used to lift the overpack or transfer cask (see Table 2.2.2 and Chapter 12). This includes the anchor block in the HI-STORM 100 Overpack, and pocket trunnions, lifting trunnions and the lifting trunnion block in HI-TRAC. Such items will henceforth be referred to as "significant-to-handling" (STH) parts. The applicable code for these elements of the structure is ANSI N14.6.

LST = -40°F for all HI-STORM "NF" components and 0°F for all HI-TRAC "NF" components. This includes all "NF" items not identified as an STH part.

It is important to ensure that all materials designated as "NF" or "STH" parts possess sufficient fracture toughness to preclude brittle fracture. For the STH parts, the necessary level of protection against brittle fracture is deemed to exist if the NDT (nil ductility transition) temperature of the part is at least 40° below the LST. Therefore, the required NDT temperature for all STH parts is -40°F.

It is well known that the NDT temperature of steel is a strong function of its composition, manufacturing process (viz., fine grain vs. coarse grain practice), thickness, and heat treatment. For example, according to Burgreen [3.1.3], increasing the carbon content in carbon steels from 0.1% to 0.8% leads to the change in NDT from -50°F to approximately 120°F. Likewise, lowering of the normalizing temperature in the ferritic steels from 1200°C to 900°C lowers the NDT from 10°C to -50°C [3.1.3]. It, therefore, follows that the fracture toughness of steels can be varied significantly within the confines of the ASME Code material specification set forth in Section II of the Code. For example, SA516 Gr. 70 (which is a principal "NF" material in the HI-STORM 100 Overpack), can have a maximum carbon content of up to 0.3% in plates up to four inches thick. Section II further permits normalizing or quenching followed by tempering to enhance fracture toughness. Manufacturing processes which have a profound effect on fracture toughness, but little effect on tensile or yield strength of the material, are also not specified with the degree of specificity in the ASME Code to guarantee a well defined fracture toughness. In fact, the Code relies on actual coupon testing of the part to ensure the desired level of protection against brittle fracture. For Section III, Subsection NF Class 3 parts, the desired level of protection is considered to exist if the lowest service temperature is equal to or greater than the NDT temperature (per NF 2311(b)(10)). Accordingly, the required NDT temperature for all load bearing metal parts in the HI-STORM 100 Overpack ("NF" and "STH") is -40°F. Likewise, the NDT temperature for all "NF" parts in HI-TRAC (except for "STH" parts) is set equal to 0°F.

From the standpoint of protection against brittle fracture, it should be recognized that setting the LST equal to the NDT temperature ensures that the fracture strength of the material containing small flaws is equal to its yield strength. In fact, as the stress calculations in this chapter (and associated appendices) would attest, the maximum primary tensile stress in the HI-STORM 100 Overpack is below 6,000 psi in all normal conditions of storage operating modes. Even in extreme environmental phenomena events, tensile stresses are below 6,000 psi, except for localized regions under postulated missile impacts or non-mechanistic tip-over. For ferritic steels (please see NF-2311(b)(7)), 6,000 psi is the threshold stress, at or below which crack propagation will not take place, no matter how low the metal temperature [3.1.3, p. 13]. (The threshold stress is the horizontal extension of the crack arrest temperature (CAT) curve in the fracture mechanics literature.)

The generally low value of tensile stress in the HI-STORM 100 storage overpack and in the HI-TRAC cask parts suggest that an NDT temperature requirement is not essential to ensure safety from crack growth. However, the aforementioned NDT temperature requirement of -40°F has been imposed to incorporate an additional layer of conservatism in the design.

The STH components (bolt anchor block (HI-STORM), lifting trunnion (HI-TRAC), lifting trunnion block (HI-TRAC), and pocket trunnion (HI-TRAC) have thicknesses greater than 2". SA350-LF3 has been selected as the material for these items (except for the lifting trunnions) due to its capability to maintain acceptable fracture toughness at low temperatures (see Table 5 in SA350 of ASME Section

IIA). Additionally, material for the HI-TRAC top flange, pool lid (100 ton) and pool lid outer ring (125 ton) has been defined as SA350-LF3, SA350-LF2, or SA203E (see Table A1.15 of ASME Section IIA) in order to achieve low temperature fracture toughness. The HI-TRAC lifting trunnion is fabricated from SB-637 Grade N07718, a high strength nickel alloy material. This material has a high resistance to fracture at low temperatures. All other steel structural materials in the HI-STORM 100 overpack and HI-TRAC cask are made of SA516-70 or SA515-70 (with some components having an option for SA203E or SA350-LF3 depending on material availability).

Table 3.1.18 provides a summary of impact testing requirements to satisfy the requirements for prevention of brittle fracture.

3.1.2.4 Fatigue

In storage, the HI-STORM 100 System is not subject to significant cyclic loads. Failure due to fatigue is not a concern for the HI-STORM 100 System.

In an anchored installation, however, the anchor studs sustain a pulsation in the axial load during the seismic event. The amplitude of axial stress variation under the DBE event is computed in this chapter and a significant margin of safety against fatigue failure during the DBE event is demonstrated.

The system is subject to cyclic temperature fluctuations. These fluctuations result in small changes of thermal expansions and pressures in the MPC. The loads resulting from these changes are small and do not significantly contribute to the "usage factor" of the cask.

Inspection of the HI-TRAC trunnions specified in Chapter 9 will preclude use of a trunnion that exhibits visual damage.

3.1.2.5 Buckling

Certain load combinations subject structural sections with relatively large slenderness ratios (such as the enclosure vessel shell) to compressive stresses that may actuate buckling instability before the allowable stress is reached. Tables 3.1.4 and 3.1.5 list load combinations for the enclosure vessel and the HI-STORM 100/HI-TRAC structures; the cases which warrant stability (buckling) check are listed therein (note that a potential buckling load has already been identified as a consequence of a postulated explosion).

TABLE 3.1.1

**LOAD COMBINATIONS SIGNIFICANT TO HI-STORM 100 OVERPACK
KINEMATIC STABILITY ANALYSIS**

Loading Case	Combinations [†]	Comment	Analysis of this Load Case Presented in:
A	D + F	This case establishes flood water flow velocity with a minimum safety factor of 1.1 against overturning and sliding.	Subsection 3.4.6
B	D + M + W'	Demonstrate that the HI-STORM 100 Overpack with minimum SNF stored (minimum D) will not tip over.	Appendix 3.C
C	D + E	Establish the value of ZPA ^{††} that will not cause the overpack to tip over.	Subsection 3.4.7

[†] Loading symbols are defined in Table 2.2.13

^{††} ZPA is zero period acceleration

TABLE 3.1.2

DESIGN BASIS DECELERATIONS FOR THE DROP EVENTS

Case	Value [†] (in multiples of acceleration due to gravity)
Vertical axis drop (HI-STORM 100 Overpack only)	45
Horizontal axis (side) drop (HI-TRAC only)	45

† The design basis value is set from the requirements of the HI-STORM 100 System, as its components are operated as a storage system. The MPC is designed to higher loadings (60g's vertical and horizontal) when in a HI-STAR 100 overpack. Analysis of the MPC in a HI-STAR 100 overpack under a 60g loading is provided in HI-STAR 100 Docket Numbers 71-9261 and 72-1008.

TABLE 3.1.3

LOADING CASES FOR THE FUEL BASKET

Load Case I.D.	Loading [†]	Notes	Location Where this Case is Evaluated
F1	T, T'	Demonstrate that the most adverse of the temperature distributions in the basket will not cause fuel basket to expand and contact the enclosure vessel wall. Compute the secondary stress intensity and show that it is small.	Appendices 3.I, 3.J, 3.U, 3.V, 3.W; Subsection 3.4.4.2
F2 (Note 1)	D + H	Conservatively add the stresses in the basket due to vertical and horizontal orientation handling to form a bounding stress intensity.	Appendix 3.AA of Docket 72-1008
F3 F3.a (Note 2)	D + H'	Vertical axis drop event	Docket Number 72-1008, Subsection 3.4.4.3.1.6
F3.b (Note 3)	D + H'	Side Drop, 0 degree orientation (Figure 3.1.2)	Table 3.4.6
F3.c (Note 3)	D + H'	Side Drop, 45 degree orientation (Figure 3.1.3)	Table 3.4.6

Notes:

1. Load Case F2 for the HI-STORM 100 System is identical to Load Case F2 for the HI-STAR 100 System in Docket Number 72-1008, Table 3.1.3.
2. Load Case F3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008, Subsection 3.4.4.3.1.6. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases F3.b and F3.c are analyzed here for a 45g deceleration, while the MPC is housed within a HI-STORM 100 Overpack or a HI-TRAC transfer cask. The initial clearance at the interface between the MPC shell and the HI-STORM 100 Overpack or HI-TRAC transfer cask is greater than or equal to the initial clearance between the MPC shell and the HI-STAR 100 overpack. This difference in clearance directly affects the stress field. The side drop analysis for the MPC in the HI-STAR 100 overpack under 60g's bounds the corresponding analysis of the MPC in HI-TRAC for 45 g's.

[†] The symbols used for the loadings are defined in Table 2.2.13.

TABLE 3.1.4

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination [†]	Notes	Comments and Location Where this Case is Analyzed
E1 (Note 1)			
E1.a	Design internal pressure, P_i	Primary stress intensity limits in the shell, baseplate, and closure ring	E1.a Lid Docket 72-1008 3.E.8.1.1 Baseplate Docket 72-1008 3.I.8.1 Shell 3.4.4.3.1.2 SupportsN/A
E1.b	Design external pressure, P_o	Primary stress intensity limits, buckling stability	E1.b Lid P_i bounds Baseplate P_i bounds Shell Docket 72-1008 3.H (Case 4) Supports N/A
E1.c	Design internal pressure, P_i , Plus Temperature, T	Primary plus secondary stress intensity under Level A condition	E1.c Lid, Baseplate, and Shell Section 3.4.4.3.1.2
E2	D + H + (P_i, P_o) ^{††}	Vertical lift, internal operating pressure conservatively assumed to be equal to the normal design pressure. Principal area of concern is the lid assembly.	Lid Docket 72-1008 3.E.8.1.2 Baseplate Docket 72-1008 3.I.8.2 Shell Docket 72-1008 3.AA (stress) Docket 72-1008 3.H (Case 4) (buckling) SupportsDocket 72-1008 3.AA

[†] The symbols used for the loadings are defined in Table 2.2.13.

^{††} The notation (P_i, P_o) means that both cases are checked with either P_o or P_i applied.

TABLE 3.1.4 (CONTINUED)

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination [†]	Notes	Comments and Location Where this Case is Analyzed
E3			
E3.a (Note 2)	$D + H' + (P_o, P_i)$	Vertical axis drop event	E3.a Lid Docket 72-10083.E.8.2.1-2 Baseplate Docket 72-10083.I.8.3 Shell Docket 72-10083.H (Case 5) (Buckling) Supports N/A
E3.b (Note 3)	$D + H' + (P_i, P_o)$	Side drop, 0 degree orientation (Figure 3.1.2)	E3.b Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6, 3.Y
E3.c (Note 3)	$D + H' + (P_i, P_o)$	Side drop, 45 degree orientation (Figure 3.1.3)	E3.c Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6, 3.Y
E4	T	Demonstrate that interference with the overpack will not develop for T.	Section 3.4.4.2

[†] The symbols used for the loadings are defined in Table 2.2.13.

TABLE 3.1.4 (CONTINUED)

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination [†]	Notes	Comments and Location Where this Case is Analyzed	
E5 (Note 1)	$P_i^* \text{ or } P_o^* + D + T^*$	Demonstrate compliance with level D stress limits – buckling stability.	Lid Baseplate Shell SupportsN/A	Docket 72-1008 3.E.8.2.1.3 Docket 72-1008 3.I.8.4 Docket 72-1008 3.H (Case 6) (buckling) Docket 72-1008 3.4.4.3.1.5 (thermal stress)

Notes:

1. Load Cases E1.a, E1.b, E2, and E5, are identical to the load cases presented in Docket Number 72-1008, Table 3.1.4. Design pressures and MPC weights are identical.
2. Load Case E3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008, Appendix 3.AA. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases E3.b and E3.c are analyzed in this HI-STORM 100 SAR for a 45g deceleration, while the MPC is housed within the HI-STORM 100 storage overpack. The interface between the MPC shell and storage overpack is not identical to the MPC shell and HI-STAR 100 overpack. The analysis for an MPC housed in HI-TRAC is not performed since results are bounded by those reported in the HI-STAR 100 TSAR for a 60g deceleration.

[†] The symbols used for the loadings are defined in Table 2.2.13.

TABLE 3.1.5

LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK

Load Case I.D.	Loading [†]	Notes	Location in FSAR Where this Case is Analyzed
01	D + H + T + (P _o ,P _i)	Vertical load handling of HI-STORM 100 Overpack/HI-TRAC.	Overpack 3.D HI-TRAC Shell 3.AE Pool lid 3.AB Transfer lid 3.AD
02			
02.a	D + H' + (P _o ,P _i)	Storage Overpack: End drop; primary stress intensities must meet level D stress limits.	Overpack 3.M
02.b	D + H' + (P _o ,P _i)	HI-TRAC: Horizontal (side) drop; meet level D stress limits for NF Class 3 components away from the impacted zone; show lids stay in-place. Show primary and secondary impact decelerations are within design basis. (This case is only applicable to HI-TRAC.)	HI-TRAC Shell 3.Z Transfer Lid 3.AD Slapdown 3.AN
02.c	D + H'	Storage Overpack: Tip-over; any permanent deformations must not preclude ready retrieval of the MPC.	Overpack 3.A,3.B

[†] The symbols used for the loadings are defined in Table 2.2.13

TABLE 3.1.5 (CONTINUED)

LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK

Load Case I.D.	Loading [†]	Notes	Location in FSAR Where this Case is Analyzed
03	D (water jacket)	Satisfy primary membrane plus bending stress limits for water jacket (This case is only applicable to HI-TRAC).	3.AG
04	M (penetrant missiles)	Demonstrate that no thru-wall breach of the HI-STORM overpack or HI-TRAC transfer cask occurs, and the primary stress levels are not exceeded. Small and intermediate missiles are examined for HI-STORM and HI-TRAC. Large missile penetration is also examined for HI-TRAC.	Overpack 3.G HI-TRAC 3.AN, 3.H
05	P _o	Explosion: must not produce buckling or exceed primary stress levels in the overpack structure.	3.B, 3.AK

Notes:

1. Under each of these load cases, different regions of the structure are analyzed to demonstrate compliance.

[†] The symbols used for the loadings are defined in Table 2.2.13

TABLE 3.1.6

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
Material: SA203-E
Service Conditions: Design, Levels A and B
Item: Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)					
	S_m	P_m^\dagger	P_L^\dagger	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	23.3	23.3	35.0	35.0	69.9	69.9
300	23.3	23.3	35.0	35.0	69.9	69.9
400	22.9	22.9	34.4	34.4	68.7	68.7
500	21.6	21.6	32.4	32.4	64.8	64.8

Definitions:

- S_m = Stress intensity values per ASME Code
- P_m = Primary membrane stress intensity
- P_L = Local membrane stress intensity
- P_b = Primary bending stress intensity
- P_e = Expansion stress
- Q = Secondary stress
- $P_L + P_b$ = Either primary or local membrane plus primary bending

Definitions for Table 3.1.6 apply to all following tables unless modified.

Notes:

1. Limits on values are presented in Table 2.2.10.

† Evaluation required for Design condition only.
 †† Evaluation required for Levels A and B only. P_e not applicable to vessels.

TABLE 3.1.7

LEVEL D: STRESS INTENSITY

Code: ASME NB
 Material: SA203-E
 Service Condition: Level D
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	P_m	P_L	$P_L + P_b$
-20 to 100	49.0	70.0	70.0
200	49.0	70.0	70.0
300	49.0	70.0	70.0
400	48.2	68.8	68.8
500	45.4	64.9	64.9

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed $0.42 S_u$.
3. Limits on values are presented in Table 2.2.10.
4. P_m , P_L , and P_b are defined in Table 3.1.6.

TABLE 3.1.8

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
Material: SA350-LF3
Service Conditions: Design, Levels A and B
Item: Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)					
	S_m	P_m^\dagger	P_L^\dagger	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	22.8	22.8	34.2	34.2	68.4	68.4
300	22.2	22.2	33.3	33.3	66.6	66.6
400	21.5	21.5	32.3	32.3	64.5	64.5
500	20.2	20.2	30.3	30.3	60.6	60.6
600	18.5	18.5	27.75	27.75	55.5	55.5
700	16.8	16.8	25.2	25.2	50.4	50.4

Notes:

1. Source for S_m is ASME Code
2. Limits on values are presented in Table 2.2.10.
3. S_m , P_m , P_L , P_b , Q , and P_e are defined in Table 3.1.6.

\dagger Evaluation required for Design condition only.

$\dagger\dagger$ Evaluation required for Levels A and B conditions only. P_e not applicable to vessels.

TABLE 3.1.9

LEVEL D, STRESS INTENSITY

Code: ASME NB
 Material: SA350-LF3
 Service Conditions: Level D
 Item: Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)		
	P _m	P _L	P _L + P _b
-20 to 100	49.0	70.0	70.0
200	48.0	68.5	68.5
300	46.7	66.7	66.7
400	45.2	64.6	64.6
500	42.5	60.7	60.7
600	38.9	58.4	58.4
700	35.3	53.1	53.1

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed 0.42 S_u.
3. Limits on values are presented in Table 2.2.10.
4. P_m, P_L, and P_b are defined in Table 3.1.6.

TABLE 3.1.10

DESIGN AND LEVEL A: STRESS

Code: ASME NF
Material: SA516, Grade 70, SA350-LF3, SA203-E
Service Conditions: Design and Level A
Item: Stress

Temp. (Deg.F)	Classification and Value (ksi)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 650	17.5	17.5	26.3
700	16.6	16.6	24.9

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Limits on values are presented in Table 2.2.12.

TABLE 3.1.11

LEVEL B: STRESS

Code: ASME NF
Material: SA516, Grade 70, SA350-LF3, and SA203-E
Service Conditions: Level B
Item: Stress

Temp. (Deg.F)	Classification and Value (ksi)	
	Membrane Stress	Membrane plus Bending Stress
-20 to 650	23.3	34.9
700	22.1	33.1

Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.10.

TABLE 3.1.12

LEVEL D: STRESS INTENSITY

Code: ASME NF
Material: SA516, Grade 70
Service Conditions: Level D
Item: Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)		
	S_m	P_m	$P_m + P_b$
-20 to 100	23.3	45.6	68.4
200	23.1	41.5	62.3
300	22.5	40.4	60.6
400	21.7	39.1	58.7
500	20.5	36.8	55.3
600	18.7	33.7	50.6
650	18.4	33.1	49.7
700	18.3	32.9	49.3

Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. Limits on values are presented in Table 2.2.12.
4. P_m and P_b are defined in Table 3.1.6.

TABLE 3.1.13

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
 Material: Alloy X
 Service Conditions: Design, Levels A and B
 Item: Stress Intensity

Temp. (Deg.F)	Classification and Numerical Value					
	S_m	P_m^\dagger	P_L^\dagger	$P_L + P_b^\dagger$	$\frac{P_L + P_b^\dagger}{P_b + Q}^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	20.0	20.0	30.0	30.0	60.0	60.0
200	20.0	20.0	30.0	30.0	60.0	60.0
300	20.0	20.0	30.0	30.0	60.0	60.0
400	18.7	18.7	28.1	28.1	56.1	56.1
500	17.5	17.5	26.3	26.3	52.5	52.5
600	16.4	16.4	24.6	24.6	49.2	49.2
650	16.0	16.0	24.0	24.0	48.0	48.0
700	15.6	15.6	23.4	23.4	46.8	46.8
750	15.2	15.2	22.8	22.8	45.6	45.6
800	14.9	14.9	22.4	22.4	44.7	44.7

Notes:

1. S_m = Stress intensity values per Table 2A of ASME II, Part D.
2. Alloy X S_m values are the lowest values for each of the candidate materials at temperature.
3. Stress classification per NB-3220.
4. Limits on values are presented in Table 2.2.10.
5. P_m , P_L , P_b , Q , and P_e are defined in Table 3.1.6.

† Evaluation required for Design condition only.

†† Evaluation required for Levels A, B conditions only. P_e not applicable to vessels.

TABLE 3.1.14

LEVEL D: STRESS INTENSITY

Code: ASME NB
 Material: Alloy X
 Service Conditions: Level D
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	P_m	P_L	$P_L + P_b$
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

Notes:

1. Level D stress intensities per ASME NB-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S_u .
3. Limits on values are presented in Table 2.2.10.
4. P_m , P_L , and P_b are defined in Table 3.1.6.

TABLE 3.1.15

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NG
 Material: Alloy X
 Service Conditions: Design, Levels A and B
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)				
	S_m	P_m	P_m+P_b	P_m+P_b+Q	P_e
-20 to 100	20.0	20.0	30.0	60.0	60.0
200	20.0	20.0	30.0	60.0	60.0
300	20.0	20.0	30.0	60.0	60.0
400	18.7	18.7	28.1	56.1	56.1
500	17.5	17.5	26.3	52.5	52.5
600	16.4	16.4	24.6	49.2	49.2
650	16.0	16.0	24.0	48.0	48.0
700	15.6	15.6	23.4	46.8	46.8
750	15.2	15.2	22.8	45.6	45.6
800	14.9	14.9	22.4	44.7	44.7

Notes:

1. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.
2. Alloy X S_m values are the lowest values for each of the candidate materials at temperature.
3. Classifications per NG-3220.
4. Limits on values are presented in Table 2.2.11.
5. P_m , P_b , Q , and P_e are defined in Table 3.1.6.

TABLE 3.1.16

LEVEL D: STRESS INTENSITY

Code: ASME NG
Material: Alloy X
Service Conditions: Level D
Item: Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)		
	P_m	P_L	P_L + P_b
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

Notes:

1. Level D stress intensities per ASME NG-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S_u.
3. Limits on values are presented in Table 2.2.11.
4. P_m, P_L, and P_b are defined in Table 3.1.6.

TABLE 3.1.17

REFERENCE TEMPERATURES AND STRESS LIMITS
FOR THE VARIOUS LOAD CASES

Load Case I.D.	Material	Reference Temperature [†] , ° F	Stress Intensity Allowables, ksi		
			P _m	P _L + P _b	P _L + P _b + Q
F1	Alloy X	725	15.4	23.1	46.2
F2	Alloy X	725	15.4	23.1	46.2
F3	Alloy X	725	36.9	55.4	NL
E1	Alloy X	450	18.1	27.2	54.3
E2	Alloy X	450	18.1	27.2	54.3
E3	Alloy X	450	43.4	65.2	NL ^{††}
E4	Alloy X	450	18.1	27.2	54.3
E5	Alloy X	775	36.15	54.25	NL

Note:

1. Q, P_m, P_L, and P_b are defined in Table 3.1.6.

[†] Values for reference temperatures are taken as the design temperatures (Table 2.2.3)

^{††} NL: No specified limit in the Code

TABLE 3.1.17 (CONTINUED)

REFERENCE TEMPERATURES AND STRESS LIMITS FOR THE VARIOUS LOAD CASES

Load Case I.D.	Material	Reference Temperature, ^{†,††} ° F	Stress Intensity Allowables, ksi		
			P _m	P _L + P _b	P _L + P _b + Q
O1	SA203-E	400	17.5	26.3	NL ^{†††}
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O2	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL
O3	SA203-E	400	17.5	26.3	NL
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O4	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL

Note:

1. P_m, P_L, P_b, and Q are defined in Table 3.1.6.
2. Load Cases 01 and 03 are for Normal Conditions; therefore the values listed refer to allowable stress, not allowable stress intensity

† Values for reference temperatures are taken as the design temperatures (Table 2.2.3).

†† For storage fire analysis, temperatures are defined by thermal solution

††† NL: No specified limit in the Code

TABLE 3.1.18[†]

FRACTURE TOUGHNESS TEST REQUIREMENTS

Material	Test Requirement	Test Temperature	Acceptance Criterion
Bolting (A193 B7)	Not required (per NF-2311(b)(13) and Note (c) to Figure NF-2311(b)-1)	-	-
Ferritic steel with nominal section thickness of 5/8" or less	Not required per NF-2311(b)(1)	-	-
SA516 Gr. 70, SA515 Gr. 70 (normalized) (thickness less than or equal to 0.75 inch)	Not required per NF-2311(b)(13) and curve D in Figure NF-2311(b)-1	-	-
SA203, SA516 Gr. 70, SA350-LF2, SA350-LF3 (greater than 0.75" thick)	Per NF-2331	See Note 1. (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2 (Also must meet ASME Section IIA requirements)
Weld material	Test per NF-2430 for welds when base metal impact testing is required.	-40 deg.F (HI-STORM) 0 deg.F (HI-TRAC) ("NF" parts) -40 deg.F (HI-TRAC) ("STH" parts)	Per NF-2330

Note:

1. Required NDT temperature = -40 deg.F for all parts in the HI-STORM 100 Overpack, -40 deg.F for HI-TRAC "STH" parts, and 0 deg.F for HI-TRAC "NF" parts.