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| Fig. 3.A.12 | 0 | | 3.B-34 | Deleted in Rev. 2 |
| Fig. 3.A.13 | 0 | | 3.B-35 | Deleted in Rev. 2 |
| Fig. 3.A.14 | 0 | | 3.B-36 | Deleted in Rev. 2 |
| Fig. 3.A.15 | 0 | | 3.B-37 | Deleted in Rev. 2 |
| Fig. 3.A.16 | 0 | | 3.B-38 | Deleted in Rev. 2 |
| Fig. 3.A.17 | 0 | | 3.B-39 | Deleted in Rev. 2 |
| Fig. 3.A.18 | 0 | | 3.B-40 | Deleted in Rev. 2 |
| Fig. 3.A.19 | Deleted in Rev. 1 | | 3.B-41 | Deleted in Rev. 2 |
| Fig. 3.A.20 | Deleted in Rev. 1 | | 3.B-42 | Deleted in Rev. 2 |
| Fig. 3.A.21 | Deleted in Rev. 1 | | 3.B-43 | Deleted in Rev. 2 |
| Fig. 3.A.22 | Deleted in Rev. 1 | | 3.B-44 | Deleted in Rev. 2 |
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| 3.B-3 | Deleted in Rev. 2 | | 3.B-55 | Deleted in Rev. 2 |
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| 3.B-5 | Deleted in Rev. 2 | | 3.B-57 | Deleted in Rev. 2 |
| 3.B-6 | Deleted in Rev. 2 | | 3.B-58 | Deleted in Rev. 2 |
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| 3.B-10 | Deleted in Rev. 2 | | 3.B-62 | Deleted in Rev. 2 |
| 3.B-11 | Deleted in Rev. 2 | | 3.C-1 | Deleted in Rev. 2 |
| 3.B-12 | Deleted in Rev. 2 | | 3.C-2 | Deleted in Rev. 2 |
| 3.B-13 | Deleted in Rev. 2 | | 3.C-3 | Deleted in Rev. 2 |
| 3.B-14 | Deleted in Rev. 2 | | 3.C-4 | Deleted in Rev. 2 |
| 3.B-15 | Deleted in Rev. 2 | | 3.C-5 | Deleted in Rev. 2 |
| 3.B-16 | Deleted in Rev. 2 | | 3.C-6 | Deleted in Rev. 2 |
| 3.B-17 | Deleted in Rev. 2 | | 3.C-7 | Deleted in Rev. 2 |
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| 3.D-1 | Deleted in Rev. 2 | | 3.G-8 | Deleted in Rev. 2 |
| 3.D-2 | Deleted in Rev. 2 | | 3.G-9 | Deleted in Rev. 2 |
| 3.D-3 | Deleted in Rev. 2 | | 3.G-10 | Deleted in Rev. 2 |
| 3.D-4 | Deleted in Rev. 2 | | 3.G-11 | Deleted in Rev. 2 |
| 3.D-5 | Deleted in Rev. 2 | | 3.G-12 | Deleted in Rev. 2 |
| 3.D-6 | Deleted in Rev. 2 | | 3.G-13 | Deleted in Rev. 2 |
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| 3.E-8 | Deleted in Rev. 2 | | 3.K-2 | Deleted in Rev. 2 |
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| 3.F-4 | Deleted in Rev. 2 | | 3.L-4 | Deleted in Rev. 2 |
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| 3.G-4 | Deleted in Rev. 2 | | 3.M-6 | Deleted in Rev. 2 |
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| 3.M-9 | Deleted in Rev. 2 | | 3.X-2 | Deleted in Rev. 2 |
| 3.M-10 | Deleted in Rev. 2 | | 3.X-3 | Deleted in Rev. 2 |
| 3.M-11 | Deleted in Rev. 2 | | 3.X-4 | Deleted in Rev. 2 |
| 3.M-12 | Deleted in Rev. 2 | | 3.X-5 | Deleted in Rev. 2 |
| 3.M-13 | Deleted in Rev. 2 | | 3.X-6 | Deleted in Rev. 2 |
| 3.M-14 | Deleted in Rev. 2 | | 3.X-7 | Deleted in Rev. 2 |
| 3.M-15 | Deleted in Rev. 2 | | 3.X-8 | Deleted in Rev. 2 |
| 3.M-16 | Deleted in Rev. 2 | | 3.X-9 | Deleted in Rev. 2 |
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| 3.Q-1 | Deleted in Rev. 2 | | 3.Y-2 | Deleted in Rev. 2 |
| 3.R-1 | Deleted in Rev. 2 | | 3.Y-3 | Deleted in Rev. 2 |
| 3.S-1 | Deleted in Rev. 2 | | 3.Y-4 | Deleted in Rev. 2 |
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| 3.U-3 | Deleted in Rev. 2 | | 3.Y-8 | Deleted in Rev. 2 |
| 3.U-4 | Deleted in Rev. 2 | | 3.Y-9 | Deleted in Rev. 2 |
| 3.U-5 | Deleted in Rev. 2 | | 3.Y-10 | Deleted in Rev. 2 |
| 3.U-6 | Deleted in Rev. 2 | | 3.Y-11 | Deleted in Rev. 2 |
| 3.U-7 | Deleted in Rev. 2 | | 3.Y-12 | Deleted in Rev. 2 |
| 3.U-8 | Deleted in Rev. 2 | | 3.Y-13 | Deleted in Rev. 2 |
| 3.U-9 | Deleted in Rev. 2 | | 3.Y-14 | Deleted in Rev. 2 |
| 3.U-10 | Deleted in Rev. 2 | | 3.Y-15 | Deleted in Rev. 2 |
| Fig. 3.U.1 | Deleted in Rev. 2 | | 3.Y-16 | Deleted in Rev. 2 |
| 3.V-1 | Deleted in Rev. 2 | | 3.Y-17 | Deleted in Rev. 2 |
| 3.V-2 | Deleted in Rev. 2 | | 3.Y-18 | Deleted in Rev. 2 |
| 3.V-3 | Deleted in Rev. 2 | | 3.Y-19 | Deleted in Rev. 2 |
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| 3.V-10 | Deleted in Rev. 2 | | 3.Z-3 | Deleted in Rev. 2 |
| 3.W-1 | Deleted in Rev. 2 | | 3.Z-4 | Deleted in Rev. 2 |
| 3.W-2 | Deleted in Rev. 2 | | 3.Z-5 | Deleted in Rev. 2 |
| 3.W-3 | Deleted in Rev. 2 | | 3.Z-6 | Deleted in Rev. 2 |
| 3.W-4 | Deleted in Rev. 2 | | 3.Z-7 | Deleted in Rev. 2 |
| 3.W-5 | Deleted in Rev. 2 | | 3.Z-8 | Deleted in Rev. 2 |
| 3.W-6 | Deleted in Rev. 2 | | 3.Z-10 | Deleted in Rev. 2 |
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| 3.AA-1 | Deleted in Rev. 2 | | 3.AD-11 | Deleted in Rev. 2 |
| 3.AA-2 | Deleted in Rev. 2 | | 3.AD-12 | Deleted in Rev. 2 |
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| 3.AI-4 | Deleted in Rev. 2 | | 3.AK-12 | Deleted in Rev. 2 |
| 3.AI-5 | Deleted in Rev. 2 | | 3.AK-13 | Deleted in Rev. 2 |
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| 3.AJ-6 | Deleted in Rev. 2 | | 3.AM-11 | Deleted in Rev. 2 |
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| 3.AJ-11 | Deleted in Rev. 2 | | 3.AM-16 | Deleted in Rev. 2 |
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| 3.AN-13 | Deleted in Rev. 2 | | 3.AR-9 | Deleted in Rev. 2 |
| 3.AN-14 | Deleted in Rev. 2 | | 3.AR-10 | Deleted in Rev. 2 |
| Fig. 3.AN.1 | Deleted in Rev. 2 | | 3.AR-11 | Deleted in Rev. 2 |
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| Fig. 3.AN.3 | Deleted in Rev. 2 | | 3.AS-2 | Deleted in Rev. 2 |
| Fig. 3.AN.4 | Deleted in Rev. 2 | | 3.AS-3 | Deleted in Rev. 2 |
| Fig. 3.AN.5 | Deleted in Rev. 2 | | 3.AS-4 | Deleted in Rev. 2 |
| Fig. 3.AN.6 | Deleted in Rev. 2 | | 3.AS-5 | Deleted in Rev. 2 |
| Fig. 3.AN.7 | Deleted in Rev. 2 | | 3.AS-6 | Deleted in Rev. 2 |
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| Fig. 3.AN.29 | Deleted in Rev. 2 | | | |
| Fig. 3.AN.20 | Deleted in Rev. 2 | | | |
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| 3.AQ-1 | Deleted in Rev. 2 | | | |
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| 3.AQ-3 | Deleted in Rev. 2 | | | |
| 3.AQ-4 | Deleted in Rev. 2 | | | |
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| 3.AQ-6 | Deleted in Rev. 2 | | | |
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| Fig. 4.3.1 | 0 | | 4.4-48 | 2 |
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| Fig. 4.4.2 | 0 | | 4.5-25 | 2 |
| Fig. 4.4.3 | 0 | | 4.5-26 | 2 |
| Fig. 4.4.4 | 0 | | 4.5-27 | 2 |
| Fig. 4.4.5 | 0 | | Fig. 4.5.1 | 0 |
| Fig. 4.4.6 | 0 | | Fig. 4.5.2 | 1 |
| Fig. 4.4.7 | 0 | | Fig. 4.5.3 | 1 |
| Fig. 4.4.8 | 0 | | 4.6-1 | 1 |
| Fig. 4.4.9 | 0 | | 4.6-2 | 1 |
| Fig. 4.4.10 | 0 | | 4.7-1 | 1 |
| Fig. 4.4.11 | 0 | | 4.7-2 | 1 |
| Fig. 4.4.12 | 0 | | 4.7-3 | 1 |
| Fig. 4.4.13 | 0 | | 4.A-1 | 2 |
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| Fig. 4.4.15 | 0 | | 4.A-3 | 2 |
| Fig. 4.4.16 | 2 | | 4.A-4 | 2 |
| Fig. 4.4.17 | 2 | | 4.A-5 | 2 |
| Fig. 4.4.18 | 0 | | 4.A-6 | 2 |
| Fig. 4.4.19 | 2 | | 4.A-7 | 2 |
| Fig. 4.4.20 | 2 | | 4.A-8 | 2 |
| Fig. 4.4.21 | 0 | | 4.A-9 | 2 |
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| Fig. 4.4.24 | 0 | | 4.A-12 | 2 |
| Fig. 4.4.25 | 1 | | 4.A-13 | 2 |
| Fig. 4.4.26 | 2 | | 4.A-14 | 2 |
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| Fig. 4.A.3 | 1 | | | |
| Fig. 4.A.4 | 1 | | | |
| Fig. 4.A.5 | 1 | | | |
| Fig. 4.A.6 | 2 | | | |
| Fig. 4.A.7 | 1 | | | |
| Fig. 4.A.8 | 1 | | | |
| Fig. 4.A.9 | 1 | | | |
| Fig. 4.A.10 | 1 | | | |
| Fig. 4.A.11 | 1 | | | |
| Fig. 4.A.12 | 1 | | | |
| Fig. 4.A.13 | 1 | | | |
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| 4.B-10 | 2 | | | |
| 4.B-11 | 2 | | | |
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| Fig. 4.B.2 | 1 | | | |
| Fig. 4.B.3 | 1 | | | |
| Fig. 4.B.4 | 1 | | | |
| Fig. 4.B.5 | 1 | | | |
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| 5.1-18 | 2 | | 5.2-38 | 2 |
| 5.1-19 | 2 | | 5.2-39 | 2 |
| Fig. 5.1.1 | 1 | | 5.2-40 | 2 |
| Fig. 5.1.2 | 0 | | 5.2-41 | 2 |
| Fig. 5.1.3 | 2 | | 5.2-42 | 2 |
| Fig. 5.1.4 | 0 | | 5.2-43 | 2 |
| Fig. 5.1.5 | 0 | | 5.2-44 | 2 |
| Fig. 5.1.6 | 0 | | 5.2-45 | 2 |
| Fig. 5.1.7 | 0 | | 5.2-46 | 2 |
| Fig. 5.1.8 | 0 | | 5.2-47 | 2 |
| Fig. 5.1.9 | 0 | | 5.2-48 | 2 |
| Fig. 5.1.10 | 0 | | 5.2-49 | 2 |
| Fig. 5.1.11 | 0 | | 5.2-50 | 2 |
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| 5.2-14 | 2 | | 5.3-11 | 1 |
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| Fig. 5.3.6 | 0 | | 5.A-1 | 0 |
| Fig. 5.3.7 | 0 | | 5.A-2 | 0 |
| Fig. 5.3.8 | 0 | | 5.A-3 | 0 |
| Fig. 5.3.9 | 0 | | 5.B-1 | 0 |
| Fig. 5.3.10 | 1 | | 5.B-2 | 0 |
| Fig. 5.3.11 | 0 | | 5.B-3 | 0 |
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| Fig. 5.3.13 | 0 | | 5.B-5 | 0 |
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| Fig. 5.3.15 | 0 | | 5.B-7 | 0 |
| Fig. 5.3.16 | 0 | | 5.C-1 | 0 |
| Fig. 5.3.17 | 0 | | 5.C-2 | 0 |
| Fig. 5.3.18 | 1 | | 5.C-3 | 0 |
| Fig. 5.3.19 | 1 | | 5.C-4 | 0 |
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| 5.4-33 | 2 | | 5.C-37 | 0 |
| 5.4-34 | 2 | | 5.C-38 | 0 |
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| 6.2-30 | 2 | | Fig. 6.3.1 | 0 |
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| Fig. 6.3.4 | 0 | | Fig. 6.4.13 | 1 |
| Fig. 6.3.4A | 1 | | Fig. 6.4.14 | 1 |
| Fig. 6.3.5 | 1 | | Fig. 6.4.15 | 1 |
| Fig. 6.3.6 | 0 | | Fig. 6.4.16 | 2 |
| Fig. 6.3.7 | 1 | | 6.5-1 | 0 |
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| 6.4-2 | 2 | | 6.7-1 | 0 |
| 6.4-3 | 2 | | 6.7-2 | 0 |
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| 6.4-6 | 2 | | 6.A-3 | 1 |
| 6.4-7 | 2 | | 6.A-4 | 1 |
| 6.4-8 | 2 | | 6.A-5 | 1 |
| 6.4-9 | 2 | | 6.A-6 | 1 |
| 6.4-10 | 2 | | 6.A-7 | 1 |
| 6.4-11 | 2 | | 6.A-8 | 1 |
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| 6.4-16 | 2 | | 6.A-13 | 1 |
| 6.4-17 | 2 | | 6.A-14 | 1 |
| 6.4-18 | 2 | | 6.A-15 | 1 |
| 6.4-19 | 2 | | 6.A-16 | 1 |
| 6.4-20 | 2 | | 6.A-17 | 1 |
| 6.4-21 | 2 | | 6.A-18 | 1 |
| 6.4-22 | 2 | | 6.A-19 | 1 |
| 6.4-23 | 2 | | 6.A-20 | 1 |
| 6.4-24 | 2 | | Fig. 6.A.1 | 0 |
| 6.4-25 | 2 | | Fig. 6.A.2 | 0 |
| 6.4-26 | 2 | | Fig. 6.A.3 | 0 |
| 6.4-27 | 2 | | Fig. 6.A.4 | 0 |
| 6.4-28 | 2 | | Fig. 6.A.5 | 0 |
| 6.4-29 | 2 | | Fig. 6.A.6 | 0 |
| 6.4-30 | 2 | | 6.B-1 | 0 |
| 6.4-31 | 2 | | 6.B-2 | 0 |
| 6.4-32 | 2 | | 6.C-1 | 2 |
| 6.4-33 | 2 | | 6.C-2 | 2 |
| Fig. 6.4.1 | 0 | | 6.C-3 | 2 |
| Fig. 6.4.2 | 1 | | 6.C-4 | 2 |
| Fig. 6.4.3 | 1 | | 6.C-5 | 2 |
| Fig. 6.4.4 | 1 | | 6.C-6 | 2 |
| Fig. 6.4.5 | 1 | | 6.C-7 | 2 |
| Fig. 6.4.6 | 1 | | 6.C-8 | 2 |
| Fig. 6.4.7 | 1 | | 6.C-9 | 2 |
| Fig. 6.4.8 | 1 | | 6.C-10 | 2 |
| Fig. 6.4.9 | 1 | | 6.C-11 | 2 |
| Fig. 6.4.10 | 0 | | 6.C-12 | 2 |
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| 6.C-17 | 2 | | | |
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| 6.D-2 | 0 | | | |
| 6.D-3 | 0 | | | |
| 6.D-4 | 0 | | | |
| 6.D-5 | 0 | | | |
| 6.D-6 | 0 | | | |
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| 6.D-29 | 0 | | | |
| 6.D-30 | 0 | | | |
| 6.D-31 | 0 | | | |
| 6.D-32 | 0 | | | |
| 6.D-33 | 0 | | | |

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| 7.1-3 | 1 | | 7.A-17 | Deleted |
| 7.1-4 | 1 | | 7.A-18 | Deleted |
| 7.1-5 | 1 | | 7.A-19 | Deleted |
| 7.1-6 | 1 | | 7.A-20 | Deleted |
| 7.1-7 | 1 | | 7.A-21 | Deleted |
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| 7.2-3 | 1 | | 7.A-25 | Deleted |
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| 7.2-7 | 1 | | 7.A-29 | Deleted |
| 7.2-8 | 1 | | 7.A-30 | Deleted |
| 7.2-9 | 1 | | 7.A-31 | Deleted |
| 7.3-1 | 1 | | 7.A-32 | Deleted |
| 7.3-2 | 1 | | 7.A-33 | Deleted |
| 7.3-3 | 1 | | 7.A-34 | Deleted |
| 7.3-4 | 1 | | 7.A-35 | Deleted |
| 7.3-5 | 1 | | 7.A-36 | Deleted |
| 7.3-6 | 1 | | 7.A-37 | Deleted |
| 7.3-7 | 1 | | 7.A-38 | Deleted |
| 7.3-8 | 1 | | 7.A-39 | Deleted |
| 7.3-9 | 1 | | 7.A-40 | Deleted |
| 7.3-10 | 1 | | 7.A-41 | Deleted |
| 7.3-11 | 1 | | 7.A-42 | Deleted |
| 7.3-12 | 1 | | 7.A-43 | Deleted |
| 7.3-13 | 1 | | 7.A-44 | Deleted |
| 7.3-14 | 1 | | 7.A-45 | Deleted |
| 7.3-15 | 1 | | 7.A-46 | Deleted |
| 7.3-16 | 1 | | 7.A-47 | Deleted |
| 7.3-17 | 1 | | 7.A-48 | Deleted |
| 7.3-18 | 1 | | 7.A-49 | Deleted |
| 7.3-19 | 1 | | 7.A-50 | Deleted |
| 7.4-1 | 1 | | 7.A-51 | Deleted |
| 7.4-2 | 1 | | 7.A-52 | Deleted |
| 7.A-1 | Deleted | | 7.A-53 | Deleted |
| 7.A-2 | Deleted | | 7.A-54 | Deleted |
| 7.A-3 | Deleted | | 7.A-55 | Deleted |
| 7.A-4 | Deleted | | 7.A-56 | Deleted |
| 7.A-5 | Deleted | | 7.A-57 | Deleted |
| 7.A-6 | Deleted | | 7.A-58 | Deleted |
| 7.A-7 | Deleted | | 7.A-59 | Deleted |
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| 7.A-9 | Deleted | | 7.A-61 | Deleted |
| 7.A-10 | Deleted | | 7.A-62 | Deleted |
| 7.A-11 | Deleted | | 7.A-63 | Deleted |
| 7.A-12 | Deleted | | 7.A-64 | Deleted |
| 7.A-13 | Deleted | | 7.A-65 | Deleted |

# LIST OF EFFECTIVE PAGES FOR PROPOSED FSAR REVISION 2

| Page | Revision | | Page | Revision |
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| 7.A-70 | Deleted | | | |
| 7.A-71 | Deleted | | | |
| 7.A-72 | Deleted | | | |
| 7.A-73 | Deleted | | | |
| 7.A-74 | Deleted | | | |
| 7.A-75 | Deleted | | | |
| 7.A-76 | Deleted | | | |
| 7.A-77 | Deleted | | | |
| 7.A-78 | Deleted | | | |
| 7.A-79 | Deleted | | | |
| 7.A-80 | Deleted | | | |
| 7.A-81 | Deleted | | | |
| 7.A-82 | Deleted | | | |
| 7.A-83 | Deleted | | | |
| 7.A-84 | Deleted | | | |
| 7.A-85 | Deleted | | | |
| 7.A-86 | Deleted | | | |
| 7.A-87 | Deleted | | | |
| 7.A-88 | Deleted | | | |
| 7.A-89 | Deleted | | | |
| 7.A-90 | Deleted | | | |
| 7.A-91 | Deleted | | | |

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| 8.0-3 | 1 | | Fig. 8.1.2b | 0 |
| 8.0-4 | 1 | | Fig. 8.1.2c | 0 |
| 8.0-5 | 1 | | Fig. 8.1.2d | 0 |
| 8.0-6 | 1 | | Fig. 8.1.2e | 0 |
| 8.1-1 | 1 | | Fig. 8.1.2f | 0 |
| 8.1-2 | 1 | | Fig. 8.1.3 | 0 |
| 8.1-3 | 1 | | Fig. 8.1.4 | 0 |
| 8.1-4 | 1 | | Fig. 8.1.5 | 0 |
| 8.1-5 | 1 | | Fig. 8.1.6 | 0 |
| 8.1-6 | 1 | | Fig. 8.1.7 | 0 |
| 8.1-7 | 1 | | Fig. 8.1.8 | 0 |
| 8.1-8 | 1 | | Fig. 8.1.9 | 0 |
| 8.1-9 | 1 | | Fig. 8.1.10 | 0 |
| 8.1-10 | 1 | | Fig. 8.1.11 | 0 |
| 8.1-11 | 1 | | Fig. 8.1.12 | 0 |
| 8.1-12 | 1 | | Fig. 8.1.13 | 0 |
| 8.1-13 | 1 | | Fig. 8.1.14 | 0 |
| 8.1-14 | 1 | | Fig. 8.1.15 | 0 |
| 8.1-15 | 1 | | Fig. 8.1.16 | 0 |
| 8.1-16 | 1 | | Fig. 8.1.17 | 0 |
| 8.1-17 | 1 | | Fig. 8.1.18 | 0 |
| 8.1-18 | 1 | | Fig. 8.1.19 | 0 |
| 8.1-19 | 1 | | Fig. 8.1.20 | 0 |
| 8.1-20 | 1 | | Fig. 8.1.21 | 0 |
| 8.1-21 | 1 | | Fig. 8.1.22a | 1 |
| 8.1-22 | 1 | | Fig. 8.1.22b | 1 |
| 8.1-23 | 1 | | Fig. 8.1.23 | 0 |
| 8.1-24 | 1 | | Fig. 8.1.24 | 0 |
| 8.1-25 | 1 | | Fig. 8.1.25 | 0 |
| 8.1-26 | 1 | | Fig. 8.1.26 | 0 |
| 8.1-27 | 1 | | Fig. 8.1.27 | 0 |
| 8.1-28 | 1 | | Fig. 8.1.28 | 0 |
| 8.1-29 | 1 | | Fig. 8.1.29a | 0 |
| 8.1-30 | 1 | | Fig. 8.1.29b | 0 |
| 8.1-31 | 1 | | Fig. 8.1.30 | 0 |
| 8.1-32 | 1 | | Fig. 8.1.31 | 0 |
| 8.1-33 | 1 | | Fig. 8.1.32 | 0 |
| 8.1-34 | 1 | | Fig. 8.1.33 | 0 |
| 8.1-35 | 1 | | Fig. 8.1.34a | 0 |
| 8.1-36 | 1 | | Fig. 8.1.34b | 0 |
| 8.1-37 | 1 | | Fig. 8.1.35 | 0 |
| 8.1-38 | 1 | | Fig. 8.1.36 | 0 |
| 8.1-39 | 1 | | Fig. 8.1.37 | 0 |
| 8.1-40 | 1 | | 8.2-1 | 1 |
| 8.1-41 | 1 | | 8.3-1 | 2 |
| 8.1-42 | 1 | | 8.3-2 | 2 |
| 8.1-43 | 1 | | 8.3-3 | 2 |
| 8.1-44 | 1 | | 8.3-4 | 2 |
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# LIST OF EFFECTIVE PAGES FOR PROPOSED FSAR REVISION 2

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| 8.3-6 | 2 | | | |
| 8.3-7 | 2 | | | |
| 8.3-8 | 2 | | | |
| 8.3-9 | 2 | | | |
| 8.3-10 | 2 | | | |
| 8.3-11 | 2 | | | |
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| Fig. 8.3.2a | 0 | | | |
| Fig. 8.3.2b | 0 | | | |
| Fig. 8.3.2c | 0 | | | |
| Fig. 8.3.2d | 0 | | | |
| Fig. 8.3.3 | 0 | | | |
| Fig. 8.3.4 | 0 | | | |
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| 8.4-2 | 1 | | | |
| 8.5-1 | 2 | | | |
| 8.5-2 | 2 | | | |
| 8.5-3 | 2 | | | |
| 8.5-4 | 2 | | | |
| 8.6-1 | 0 | | | |

# LIST OF EFFECTIVE PAGES FOR PROPOSED FSAR REVISION 2

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| 9.1-3 | 1 | | | |
| 9.1-4 | 1 | | | |
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| 9.1-7 | 1 | | | |
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| 9.1-10 | 1 | | | |
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| 9.1-21 | 1 | | | |
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| 9.2-2 | 0 | | | |
| 9.2-3 | 0 | | | |
| 9.2-4 | 0 | | | |
| 9.3-1 | 0 | | | |
| 9.4-1 | 0 | | | |

# LIST OF EFFECTIVE PAGES FOR PROPOSED FSAR REVISION 2

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| 10.1-2 | 1 | | 10.4-5 | 2 |
| 10.1-3 | 1 | | 10.5-1 | 0 |
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| 10.1-7 | 1 | | | |
| 10.1-8 | 1 | | | |
| Fig. 10.1.1 | 0 | | | |
| 10.2-1 | 1 | | | |
| 10.3-1 | 1 | | | |
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| 10.3-3 | 1 | | | |
| 10.3-4 | 1 | | | |
| 10.3-5 | 1 | | | |
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| 10.3-9 | 1 | | | |
| 10.3-10 | 1 | | | |
| 10.3-11 | 1 | | | |
| 10.3-12 | 1 | | | |
| 10.3-13 | 1 | | | |
| 10.3-14 | 1 | | | |
| 10.3-15 | 1 | | | |
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| 10.3-34 | 1 | | | |
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| Fig. 10.3.1c | 0 | | | |
| Fig. 10.3.1d | 0 | | | |
| Fig. 10.3.1e | 0 | | | |
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| 10.4-2 | 2 | | | |

LIST OF EFFECTIVE PAGES FOR PROPOSED FSAR REVISION 2

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| 11.1-3 | 2 | | 11.2-39 | 2 |
| 11.1-4 | 2 | | 11.2-40 | 2 |
| 11.1-5 | 2 | | 11.2-41 | 2 |
| 11.1-6 | 2 | | 11.2-42 | 2 |
| 11.1-7 | 2 | | 11.2-43 | 2 |
| 11.1-8 | 2 | | 11.2-44 | 2 |
| 11.1-9 | 2 | | 11.2-45 | 2 |
| 11.1-10 | 2 | | 11.2-46 | 2 |
| 11.1-11 | 2 | | 11.2-47 | 2 |
| 11.1-12 | 2 | | 11.2-48 | 2 |
| 11.1-13 | 2 | | 11.2-49 | 2 |
| 11.1-14 | 2 | | 11.2-50 | 2 |
| 11.1-15 | 2 | | 11.2-51 | 2 |
| 11.1-16 | 2 | | 11.2-52 | 2 |
| 11.2-1 | 2 | | Fig. 11.2.1 | 0 |
| 11.2-2 | 2 | | Fig. 11.2.2 | 0 |
| 11.2-3 | 2 | | Fig. 11.2.3 | 0 |
| 11.2-4 | 2 | | Fig. 11.2.4 | 0 |
| 11.2-5 | 2 | | Fig. 11.2.5 | 0 |
| 11.2-6 | 2 | | Fig. 11.2.6 | 2 |
| 11.2-7 | 2 | | Fig. 11.2.7 | Deleted in Rev. 2 |
| 11.2-8 | 2 | | Fig. 11.2.8 | Deleted in Rev. 1 |
| 11.2-9 | 2 | | 11.3-1 | 0 |
| 11.2-10 | 2 | | | |
| 11.2-11 | 2 | | | |
| 11.2-12 | 2 | | | |
| 11.2-13 | 2 | | | |
| 11.2-14 | 2 | | | |
| 11.2-15 | 2 | | | |
| 11.2-16 | 2 | | | |
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| 11.2-18 | 2 | | | |
| 11.2-19 | 2 | | | |
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| 11.2-22 | 2 | | | |
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| 11.2-26 | 2 | | | |
| 11.2-27 | 2 | | | |
| 11.2-28 | 2 | | | |
| 11.2-29 | 2 | | | |
| 11.2-30 | 2 | | | |
| 11.2-31 | 2 | | | |
| 11.2-32 | 2 | | | |
| 11.2-33 | 2 | | | |
| 11.2-34 | 2 | | | |
| 11.2-35 | 2 | | | |
| 11.2-36 | 2 | | | |

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| 12.1-2 | 2 | | B 3.3.1-3 | 2 |
| 12.1-3 | 2 | | B 3.3.1-4 | 2 |
| 12.2-1 | 2 | | B 3.3.1-5 | 2 |
| 12.2-2 | 2 | | Appendix 12.B Cover | 1 |
| 12.2-3 | 2 | | Comment Resolution Letters | 0 |
| 12.2-4 | 2 | | 21 Pages | |
| 12.2-5 | 2 | | | |
| 12.2-6 | 2 | | | |
| 12.2-7 | 2 | | | |
| 12.2-8 | 2 | | | |
| 12.2-9 | 2 | | | |
| 12.2-10 | 2 | | | |
| 12.3-1 | 0 | | | |
| 12.4-1 | 0 | | | |
| 12.5-1 | 0 | | | |
| Appendix 12.A Cover | 1 | | | |
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| B 3.0-4 | 0 | | | |
| B 3.0-5 | 0 | | | |
| B 3.0-6 | 0 | | | |
| B 3.0-7 | 0 | | | |
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| B 3.1.1-3 | 2 | | | |
| B 3.1.1-4 | 2 | | | |
| B 3.1.1-5 | 2 | | | |
| B 3.1.1-6 | 2 | | | |
| B 3.1.1-7 | 2 | | | |
| B 3.1.2-1 | 2 | | | |
| B 3.1.2-2 | 2 | | | |
| B 3.1.2-3 | 2 | | | |
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| B 3.1.2-5 | 2 | | | |
| B 3.1.2-6 | 2 | | | |
| B 3.1.2-7 | 2 | | | |
| B 3.1.3-1 | 2 | | | |
| B 3.1.3-2 | 2 | | | |
| B 3.1.3-3 | 2 | | | |
| B 3.1.3-4 | 2 | | | |
| B 3.1.3-5 | 2 | | | |
| B 3.1.3-6 | 2 | | | |
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| B 3.2.3-2 | Deleted in Rev. 2 | | | |
| B 3.2.3-3 | Deleted in Rev. 2 | | | |
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| 13.0-2 | 2 | | | |
| 13.1-1 | Deleted in Rev. 2 | | | |
| 13.1-2 | Deleted in Rev. 2 | | | |
| 13.2-1 | Deleted in Rev. 2 | | | |
| 13.3-1 | Deleted in Rev. 2 | | | |
| 13.3-2 | Deleted in Rev. 2 | | | |
| 13.3-3 | Deleted in Rev. 2 | | | |
| 13.3-4 | Deleted in Rev. 2 | | | |
| 13.3-5 | Deleted in Rev. 2 | | | |
| 13.3-6 | Deleted in Rev. 2 | | | |
| 13.3-7 | Deleted in Rev. 2 | | | |
| 13.3-8 | Deleted in Rev. 2 | | | |
| 13.3-9 | Deleted in Rev. 2 | | | |
| 13.3-10 | Deleted in Rev. 2 | | | |
| 13.3-11 | Deleted in Rev. 2 | | | |
| 13.3-12 | Deleted in Rev. 2 | | | |
| 13.3-13 | Deleted in Rev. 2 | | | |
| 13.3-14 | Deleted in Rev. 2 | | | |
| 13.3-15 | Deleted in Rev. 2 | | | |
| 13.3-16 | Deleted in Rev. 2 | | | |
| 13.3-17 | Deleted in Rev. 2 | | | |
| 13.3-18 | Deleted in Rev. 2 | | | |
| 13.4-1 | Deleted in Rev. 2 | | | |
| 13.5-1 | Deleted in Rev. 2 | | | |
| 13.5-2 | Deleted in Rev. 2 | | | |
| 13.6-1 | 2 | | | |
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| 13.B-1 | 2 | | | |
| 13.B-2 | Deleted in Rev. 2 | | | |
| 13.B-3 | Deleted in Rev. 2 | | | |

# 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 application seeks 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 applicant. This report is also suitable for incorporation into a site-specific Safety Analysis Report which may be submitted by an applicant for a 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

† 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).

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.

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 safety 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.

## 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.

**Boral™** 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.

**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. The HI-TRAC is equipped with a pair of lifting trunnions and pocket trunnions to lift and downend/upend the HI-TRAC with a loaded MPC. HI-TRAC is an acronym for **H**oltec **I**nternational **Tra**nsfer **C**ask. In this submittal there are two HI-TRAC transfer casks, the 125 ton HI-TRAC (HI-TRAC-125) and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a

## Table 1.0.1 (continued)

## TERMINOLOGY AND NOTATION

maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to both the 125 ton and 100 ton HI-TRAC.

**HI-STORM 100 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.

**Holtite™** 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-A™ 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.

**Holtite™-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. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, 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 original fuel rod(s).

## Table 1.0.1 (continued)

## TERMINOLOGY AND NOTATION

**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.

**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.

*METAMIC™ is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs.*

**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 Absorber Material*** *is a generic term used in this FSAR to indicate any neutron absorber material qualified for use in the HI-STORM 100 System MPCs.*

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

## Table 1.0.1 (continued)

## TERMINOLOGY AND NOTATION

**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*, and vibration suppressor inserts.* |

**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 ($\geq$ 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.

**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

**Table 1.0.1 (continued)**

**TERMINOLOGY AND NOTATION**

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 Character-istics | 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)<br>10CFR72.236(a) | 1.2.3 |
| 1.3 Identification of Agents & Contractors | 1.III.4 Qualification of the Applicant | 10CFR72.24(j)<br>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)<br>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)<br>10CFR72.236(a) | 2.1 |
| 2.2 Design Criteria for Environmental Conditions and Natural Phenomena | 2.III.2.b External Conditions,<br>2.III.3.b Structural,<br>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 |

**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 |
|---|---|---|---|
| 2.2.1 Tornado and Wind Loading | 2.III.2.b External Conditions | 10CFR72.122(b)(2) | 2.2.3.5 |
| 2.2.2 Water Level (Flood) | 2.III.2.b External Conditions<br><br>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)<br>10CFR72.122(b)(2) | 2.2.3.7 |
| 2.2.4 Snow and Ice | 2.III.2.b External Conditions<br><br>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)<br>10CFR72.122(b)(2)(ii) | 2.2.7 |
| NA | 2.III.1 Structures, Systems, and Components Important to Safety | 10CFR72.122(a)<br>10CFR72.24(c)(3) | 2.2.4 |
| NA | 2.III.2 Design Criteria for Safety Protection Systems | 10CFR72.236(g)<br>10CFR72.24(c)(1)<br>10CFR72.24(c)(2)<br>10CFR72.24(c)(4)<br>10CFR72.120(a)<br>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)<br>10CFR72.128(a)(5) | 10.0, 8.0 |
| | | 10CFR72.236(h) | 8.0 |
| | | 10CFR72.24(1)(2) | 1.2.1, 1.2.2 |
| | | 10CFR72.236(1) | 2.3.2.1 |
| | | 10CFR72.24(e)<br>10CFR72.104(b) | 10.0, 8.0 |

**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 |
|---|---|---|---|
| | 2.III.3.g Acceptance Tests & Maintenance | 10CFR72.122(1)<br>10CFR72.236(g)<br>10CFR72.122(f)<br>10CFR72.128(a)(1) | 9.0 |
| 2.3 Safety Protection Systems | -- | -- | 2.3 |
| 2.3.1 General | -- | -- | 2.3 |
| 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)<br>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<br><br>2.3.4 Nuclear Criticality Safety | 2.III.3.d Shielding/ Confinement/ Radiation Protection<br><br>2.III.3.e Criticality | 10CFR72.122(h)(4)<br>10CFR72.122(i)<br>10CFR72.128(a)(1)<br><br>10CFR72.124(a)<br>10CFR72.236(c)<br>10CFR72.124(b) | 2.3.5<br><br>2.3.4, 6.0 |
| 2.3.5 Radiological Protection | 2.III.3.d Shielding/ Confinement/ Radiation Protection | 10CFR72.24(d)<br>10CFR72.104(a)<br>10CFR72.236(d) | 10.4.1 |
| | | 10CFR72.24(d)<br>10CFR72.106(b)<br>10CFR72.236(d) | 10.4.2 |

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 |
|---|---|---|---|
| | | 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)<br>10CFR72.130<br>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) |
| | 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)<br>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)<br>10CFR72.124(a)<br>10CFR72.236(c)<br>10CFR72.236(d)<br>10CFR72.236(1) | 3.4.4.3<br>3.4.7.3<br>3.4.10 |

**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 | 3.III.3 Ready Retrieval | 10CFR72.122(f)<br>10CFR72.122(h)<br>10CFR72.122(l) | 3.4.4.3 |
| NA | 3.III.4 Design-Basis Earthquake | 10CFR72.24(c)<br>10CFR72.102(f) | 3.4.7 |
| NA | 3.III.5 20 Year Minimum Design Length | 10CFR72.24(c)<br>10CFR72.236(g) | 3.4.11<br>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<br>-- | -- | 3.4.3, ~~Appendices 3.E, 3.AC, 3.D~~ |
| 3.4.4 Heat | 3.V.1.d Structural Analysis | 10CFR72.24(d)<br>10CFR72.122(b) | 3.4.4, ~~Appendices 3.I, 3.U, 3.V, 3.W~~ |
| 3.4.5 Cold | 3.V.1.d Structural Analysis | 10CFR72.24(d)<br>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)<br>10CFR72.128(a)(4)<br>10CFR72.236(f)<br>10CFR72.236(h) | 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 |
|---|---|---|---|
| 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)<br>10CFR72.236(g) | 4.4, 4.5 |
| NA | 4.IV Acceptance Criteria | 10CFR72.24(d)<br>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)<br>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 |
| 5.3.1 Description of the Radial and Axial Shielding Configura-tions | 5.V.3.a Configuration of the Shielding and Source | 10CFR72.24(c)(3) | 5.3.1 |

**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 |
|---|---|---|---|
| 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)<br>10CFR72.104(a)<br>10CFR72.106(b)<br>10CFR72.128(a)(2)<br>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 | --<br>10CFR72.124(b)<br>10CFR72.24(c)(3) | 6.3.1 |
| 6.3.2 Cask Regional Densities | 6.V.3.b Material Properties | 10CFR72.24(c)(3)<br>10CFR72.124(b)<br>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<br>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)<br>10CFR72.124<br>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)<br>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)<br>10CFR72.236(1) | 7.2 |
| 7.2.1 Release of Radioactive | 7.III.6 Release of Nuclides to the Environment | 10CFR72.24(1)(1) | 7.2.1 |

## 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 |
|---|---|---|---|
| Material | 7.III.4 Monitoring of Confinement System | 10CFR72.122(h)(4)<br>10CFR72.128(a)(1) | 7.1.4 |
| | 7.III.5 Instrumentation | 10CFR72.24(l)<br>10CFR72.122(i) | 7.1.4 |
| | 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)<br>10CFR72.122(b)<br>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)<br>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 |

## 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.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)<br>10CFR72.104(b) | 8.3 |
| | 8.III.3 Radioactive Effluent Control | 10CFR72.24(1)(2) | 8.3.3 |
| | 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)<br>10CFR72.128(a)(5) | 8.1, 8.3 |
| | 8.III.10 SSCs Permit Inspection, Maintenance, and Testing | 10CFR72.122(f) | Table 8.1.6 |

## 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 |
|---|---|---|---|
| **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 |
| 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) |

**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 |
|---|---|---|---|
| | 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<br>10CFR72.24(e)<br>10CFR72.104(b)<br>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<br>10CFR20.1207<br>10CFR20.1208<br>10CFR20.1301 | 10.3 |
| N/A | 10.III.3 Public Exposure | 10CFR72.104<br>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)<br>10CFR72.104(a)<br>10CFR72.236(d) | 11.1 |
| | 11.III.4 Maintain Subcritical Condition | 10CFR72.124(a)<br>10CFR72.236(c) | 11.1 |
| | 11.III.7 Instrumentation and Control for Off-Normal Condition | 10CFR72.122(i) | 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 |
|---|---|---|---|
| 11.2 Accidents | 11.III.1 SSCs Important to Safety Designed for Accidents | 10CFR72.24(d)(2)<br>10CFR72.122b(2)<br>10CFR72.122b(3)<br>10CFR72.122(d)<br>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)<br>10CFR72.236(c) | 11.2, 6.0 |
| | 11.III.3 Meet Dose Limits for Accidents | 10CFR72.24(d)(2)<br>10CFR72.24(m)<br>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)<br>10CFR72.26<br>10CFR72.44(c)<br>10CFR72 Subpart E<br>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 |

**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 |
|---|---|---|---|
| 12.2.4 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 |
| NA | 12.III.2 Shielding and Confinement | 10CFR20<br>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 |

## 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 |
|---|---|---|---|
| | **13. Quality Assurance** | | |
| 13.1 Quality Assurance | 13.III Regulatory Requirements | 10CFR72.24(n) *10CFR72.140(d)* | 13.0 |
| | 13.IV Acceptance Criteria | 10CFR72, Subpart G | |

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 |
|---|---|---|
| 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." | Exception: 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. | 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. |
| 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." | Clarification: 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. | 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. |

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 |
|---|---|---|
| 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..." | Clarification: 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. | 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. |
| 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..."<br><br>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".<br><br>3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and | Exception: 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. | 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.<br><br>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 |

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 |
|---|---|---|
| construction of reinforced concrete structures important to safety but not within the scope of ACI 359 should comply with the requirements of ACI 349". | | limits for normal and off-normal condition from ACI 349 will be imposed.<br><br>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." | Clarification: 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. |
| 4.V.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."<br><br>4.V.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."<br><br>4.V.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." | Clarification: 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 |
|---|---|---|
| 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, 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 |
|---|---|---|
| 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." | Clarification: ~~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.~~*An equivalent isotropic thermal conductivity approach is adopted.* | ~~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.~~*The computer code FLUENT used in the thermal analyses is not capable of modeling anisotropy in the thermal conductivity of a porous medium, such as the fuel basket region. An equivalent isotropic conductivity is used to conservatively represent the heat dissipation characteristics of the fuel basket.* |
| 4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity." | Exception: 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. | 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. |

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.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented." | Clarification: No additional heat balance is performed or provided. | 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. |
| 4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations." | Exception: 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. | Exception: 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. |

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 |
|---|---|---|
| 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." | Exception: 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 ($5 \times 10^{-6}$ atm-cm$^3$/s) plus 50% for conservatism, which yields $7.5 \times 10^{-6}$ 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).<br><br>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 NB/NC-5330, "Ultrasonic Acceptance Standards"." | Clarification: Section 9.1.1.1 and the Design Drawings specify that the 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. | In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria 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. |

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 |
|---|---|---|
| 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." | Exception: 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. | The dimensional compliance of all shielding cavities is verified by inspection to Design Drawing requirements prior to shield installation.<br><br>The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures.<br><br>The composition of the Holtite-A is confirmed by inspection and tests prior to first use.<br><br>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. |
| *13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."* | *Exception: Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.* | *The NRC has approved Revision 13 of the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec intends to apply this QA program to dry storage cask activities. Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation.* |

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 |
|---|---|---|
| ISG-15, Section X.5.4.2, "No more than 1% of the rods in an assembly have peak cladding oxide thicknesses greater than 8o 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 form NUREG-1536 (ISG-15). |
| *ISG-5, Revision 1 provides review guidance for dry storage cask confinement analyses. Sections 3 and 4 discuss nuclides with potential for release and confinement analysis, respectively. The ISG is silent with regard to credit for gravitational settling of certain isotopes inside the MPC cavity.* | *The HI-STORM confinement analysis described in Chapter 7 takes a conservative amount of credit for gravitational settling of certain isotopes inside the MPC in calculating the amount of radioactive material that is available for airborne release.* | *NRC Report SMSAB-00-03, "Best-Estimate Offsite Dose from Dry Storage Cask Leakage" provides a methodology for taking credit for gravitational settling of isotopes in the Holtec MPC. Holtec is using this methodology to more accurately estimate the confinement doses, notwithstanding the fact that the MPC confinement boundary is designed to maintain integrity under all normal, off-normal, and accident conditions.* |

## 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 100S System with an MPC inserted into a 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 detailed design 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~~*MPC* 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 and the HI-STORM 100S is 239-1/2 inches and 232 inches, respectively. See Section 1.5 for the detailed design 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 Design Drawings and Bills-of-Material 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.A. 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~~ *eight* MPC models, distinguished by the type and number of fuel assemblies authorized for loading. *Section 1.2.3 and Table 1.2.1 describe the allowable contents for each MPC model.* ~~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. Design~~ 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 *(including the MPC-32F)* and the MPC-68 (including the MPC-68F and MPC-68FF) in one important aspect: the fuel storage cells *in the MPC-24 series* are physically separated from one another by a "flux trap", for criticality control. The PWR MPC-32 *and -32F are* ~~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, *and MPC-32F*) 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 (described in U.S. Patent No. 5,897,747, assigned to Holtec International) 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 *aluminum* heat conduction elements *(AHCEs)* may *have been* ~~be~~ installed *in the early vintage MPCs fabricated and certified under the original version or Amendment 1 of the HI-STORM 100 System CoC. The presence of these aluminum heat conduction elements is acceptable for MPCs fabricated under the original CoC or Amendment 1, since the governing thermal analysis for Amendment 1 conservatively modeled the AHCEs as restrictions to convective flow in the basket, but took no credit for heat transfer through them. The heats loads authorized under Amendment 1 bounds those for the original CoC, with the same MPC design. For MPCs fabricated under Amendment 2 or a later version of the HI-STORM 100 CoC, the aluminum heat conduction elements shall not be installed since they were removed from the thermal model in Amendment 2. MPCs both with and without aluminum heat conduction elements installed are compatible with all HI-STORM overpacks. If used,* ~~T~~*t*hese heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design which 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 which 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 edge-welded to the MPC outer shell. This plate 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 ½ inch thinner than the MPC-68s' lid to accommodate the longest PWR fuel assembly which is approximately a ½ 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 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, this application requests approval for use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this application) 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 which 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.

Other alloy materials which are identified to be more suitable by the DOE for the MGDS in the future and which are also bounded by the Alloy X properties set forth in Appendix 1.A can be used in the MPC after an amendment to this FSAR is approved.

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 1.2.8.A provide cross sectional views of the HI-STORM 100 System, including both of the overpack designs. 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. Stainless steel shims are attached to channels to allow the proper inner diameter dimension to be obtained and to provide a guiding surface for MPC insertion and removal.

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 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 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. 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. The HI-STORM 100S is approximately 232 inches from the bottom of the baseplate to the top of the lid bolts in its final storage configuration and 211 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 the HI-STORM terminology, these lugs are referred too as "sector lugs". The sector lugs, as shown in Figure 1.1.5 and the design 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, imputes a lateral load bearing capacity to the cask/pad interface that is equal to $\mu F$ ($\mu \leq 0.53$ 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 ($\mu F$) 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 preecluded, 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 $\sigma_u$ and yield strength $\sigma_y$ are also listed. For purposes of structural evaluations, the lower bound values of $\sigma_u$ and $\sigma_y$ from the menu of materials listed in Table 1.2.7 are used (see Table 3.4.10).

For convenience in referencing, the representative menu of fastener materials listed in Table 1.2.7 (and other additional acceptable materials from the ASME Code) will be referred to as "Alloy Z."

As shown in the design 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)

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 a HI-TRAC with the pool lid installed.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC. The top lid 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 HI-TRAC 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 which 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. The transfer lid incorporates two sliding doors which 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.

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 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 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.

### 1.2.1.3 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~~*absorber* 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 100 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. 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 125 ton HI-TRAC transfer lid, the neutron shield material, Holtite-A, is also provided. The 125 ton HI-TRAC top lid is composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The 100 ton HI-TRAC top lid is composed of steel only providing gamma attenuation.

1.2.1.3.1 ~~Boral~~ *Fixed* Neutron Absorbers

*(i)Boral™*

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.

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 recently licensed Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

As indicated in Tables 1.2.3-1.2.5, 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 in compliance with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

*(ii) METAMIC™*

*METAMIC ™ is a neutron absorber material developed by the Reynolds Aluminum Company in the mid-1990s for spent fuel reactivity control in dry and wet storage applications. Metallurgically, METAMIC is a metal matrix composite (MMC) consisting of a high purity 6061 Aluminum matrix reinforced with Type 1 ASTM C-750, isotopically graded boron carbide. METAMIC is characterized by an extremely fine aluminum spherical powder (325 mesh or better) and boron carbide powder (average particle size under 10 microns). As described in the U.S. patents held by METAMIC, Inc.*†, the high performance reliability of METAMIC derives from the particle size distribution of its constituents, namely, high purity Aluminum 6061 alloy powder and isotopically*

* *U.S. Patent No. 5,965,829, "Radiation Absorbing Refractory Composition".*
† *U.S. Patent No. 6,042,779, "Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super, Hypereutectic Al/Si."*

*graded $B_4C$ particulate, rendered into an isotropic metal matrix composite state by the powder metallurgy process which yields excellent homogeneity, and which prevents $B_4C$ from clustering in the final product.*

*The powders are carefully blended together without binders, chelating agents, or other additives that could potentially become retained in the final product and deleteriously influence performance. The approximate maximum percentage of $B_4C$ that can be dispersed in the Aluminum alloy 6061 matrix is 40%. The pure blend of powders is cold isostatically compacted into a green billet and vacuum sintered to high theoretical density. According to the manufacturer, billets of any size can be produced using this technology; however, a typical production billet is 8 to 9 inches in diameter by approximately 32 inches long, weighing approximately 210 pounds. This billet is subsequently extruded into one of a number of product forms, ranging from sheet and plate to angle, channel, round and square tube, and other profiles. A typical lot of METAMIC (defined as the quantity processed in one load of the vacuum furnace) is in the range of 1,200 to 2,400 lbs.*

*METAMIC has been subjected to an extensive array of tests sponsored by the Electric Power Research Institute (EPRI) that evaluated the functional performance of the material at elevated temperatures (up to 900ºF) and radiation levels (1E+11 rads gamma). The results of the tests documented in an EPRI report* indicate that METAMIC maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state. The main conclusions provided in the above-referenced EPRI report are summarized below:*

- *The isotropic metal matrix configuration produced by the powder metallurgy process with a complete absence of interconnected internal porosity in METAMIC ensures that its density is essentially equal to the maximum theoretical density.*

- *Measurements of boron carbide particle distribution show extremely small particle-to-particle distance[†] and near-perfect homogeneity.*

- *The physical and neutronic properties of METAMIC are essentially unaltered under exposure to elevated temperatures (750° F - 900° F).*

- *No detectable change in the neutron attenuation characteristics under accelerated test conditions has been observed.*

*An evaluation of the manufacturing technology underlying METAMIC as disclosed in the above-referenced patents and of the extensive third-party tests carried out under the auspices of EPRI had led Holtec International to designate METAMIC an acceptable neutron absorber material for use in the company's MPCs. Holtec's technical position on METAMIC is also supported by the evaluation*

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* "Qualification of *METAMIC* for Spent Fuel Storage Application", EPRI, 1003137, Final Report, October 2001.

[†] Medium measured neighbor-to-neighbor distance is 10.08 microns according to the article, "METAMIC Neutron Shielding", by K. Anderson, T. Haynes, and R. Kazmier, EPRI Boraflex Conference, November 19-20, 1998.

*carried out by other organizations (see, for example, USNRC's SER on NUHOMS-61BT, Docket No. 72-1004).*

*Consistent with its role in reactivity control, all* METAMIC *material procured for use in the Holtec MPCs will be qualified as important-to-safety (ITS) Category A item. ITS category A manufactured items, as required by Holtec's NRC-approved Quality Assurance program, must be produced to essentially preclude the potential of an error in the procurement of constituent materials and the manufacturing processes. Accordingly, material and manufacturing control processes must be established to eliminate the incidence of errors, and inspection steps must be implemented to serve as an independent set of barriers to ensure that all critical characteristics defined for the material by the cask designer are met in the manufactured product.*

*All manufacturing and in-process steps in the production of* METAMIC *shall be carried out using written procedures that have been reviewed and found to be acceptable by Holtec's QA organization. As required by the company's quality program, the material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances as set forth in the applicable Holtec QA procedures to ensure that all* METAMIC *panels procured meet with the requirements appropriate for the quality genre of the MPCs. Confirmatory tests, prior to the use of the* METAMIC *in Holtec's MPCs, are summarized in Subsection 9.1.5.3 of this FSAR.*

*Because of the absence of interconnected porosities, the time required to dehydrate a* METAMIC-*equipped MPC is expected to be less compared to an MPC containing the rolled cermet class of neutron absorbers such as Boral.*

*Although* METAMIC *exhibits near-theoretical neutron attenuation characteristics (due to the very small* $B_4C$ *particle size distributed in the homogeneous metal matrix), only 75% of the minimum B-10 areal density (as in the case of Boral) is recognized in the criticality analysis.*

*(iii) Structural Integrity of Fixed Neutron Absorbers*

*Both Boral and* METAMIC *neutron absorber panels are completely enclosed in Alloy X (stainless steel) sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld (see the drawings in Section 1.5 for details of this design configuration). Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient evolutions. Finally, the pocket joint detail, borrowed from Holtec's spent fuel rack design (with tens of thousands of successful deployments) ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.*

### 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 125 ton HI-TRAC transfer cask in the axial direction is provided by Holtite-A within the top lid and transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal $B_4C$ 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/cm$^3$ 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/cm$^3$. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm$^3$ 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 $B_4C$ content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal $B_4C$ 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 (Figure 1.2.8) 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). Studs are threaded into the anchor blocks to secure the lid and provide for lifting. These four studs provide for direct attachment of lifting devices which, along with a specially-designed lift rig to ensure a vertical lift, allow lifting by a crane or similar equipment. The lift rig shall be designed to lift a fully-loaded storage overpack with margins of safety specified in ANSI N14.6 [1.2.9].

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 a sufficient height to allow air pads to be placed or removed from under the overpack baseplate.

The HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket 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 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 Design 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 demoisturizing 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. Two cleats provide redundant support of the MPC when it is lifted or supported.

Two or four stays (depending on the site crane hook configuration) are installed between the MPC lift cleats and the lift yoke main pins. The stays secure the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. The HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and stays 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 positioned in the truck bay with the lid removed and, for the HI-STORM 100 overpack, the vent duct shield inserts installed. The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. 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 locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent shield inserts. For the HI-STORM 100S, the 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 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 positioned in the cask transfer facility with the lid removed and, for the HI-STORM 100, the vent duct shield inserts installed. 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. The cradle is equipped with rotation trunnions which engage the HI-TRAC 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.

If the loaded HI-TRAC is transferred to the cask transfer facility in the horizontal orientation, the HI-TRAC and 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. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. 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 locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent duct shield inserts. For the HI-STORM 100S, the 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 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 and torqued.

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. However, the loaded overpack may be lifted from the top through the lid studs 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 and, for the HI-STORM 100, the vent duct shield inserts are installed. The MPC lift cleats are attached to the MPC and the MPC lift slings are attached to the MPC lift cleats. For the HI-STORM 100S, the transfer doors may need to be opened to avoid interfering with the MPC lift cleats. HI-TRAC is raised and positioned on top of HI-STORM. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the HI-TRAC transfer lid doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM.

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

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC stays, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water(borated, if necessary). The annulus shield is installed and pressurized to protect the annulus from debris produced from the lid removal process. Similarly, HI-TRAC top surfaces are covered with a protective fire-retarding blanket.

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~~ *neutron absorber* 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 *and MPC-32F* 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. The quantity of damaged fuel containers with fuel debris is limited to meet the off-site transportation requirements of 10CFR71, specifically, 10CFR71.63(b).

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 1.2.4A).

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 1.2.4A).

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. *Up to eight (8) of these 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 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).*

*MPC-32F*

*The MPC-32F is designed to store up to thirty two (32) PWR fuel assemblies with or without non-fuel hardware. Up to eight (8) of these 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 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).*

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~~8 | ~~4~~5 for PWR<br>3 for BWR |
| MPC storage capacity†: | MPC-24<br>MPC-24E<br>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.<br><br>OR |
| | MPC-32<br>*MPC-32F* | Up to 32 intact zircaloy or stainless steel clad PWR fuel assemblies *with or without non-fuel hardware. Up to 8 damaged fuel assemblies may be stored in the MPC-32 and up to 8 damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-32F.* |
| | 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.<br><br>OR |

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

Table 1.2.1 (continued)
KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

| ITEM | QUANTITY | NOTES |
|---|---|---|
| MPC storage capacity: | MPC-68F | Up to 4 damaged fuel containers with zircaloy clad Dresden Unit 1 (D-1) or Humboldt Bay (HB) BWR fuel debris and the complement damaged zircaloy clad Dresden Unit 1 or Humboldt Bay BWR fuel assemblies in damaged fuel containers or intact Dresden Unit 1 or Humboldt Bay BWR intact fuel assemblies.<br><br>OR |
| | MPC-68FF | Up to 68 Dresden Unit 1 or Humboldt Bay intact fuel or damaged fuel and up to 8 damaged fuel containers containing D-1 or HB fuel debris. For other BWR plants, up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris with the complement intact fuel assemblies, up to a total of 68. The number of damaged fuel containers containing BWR fuel debris is limited to eight (8) for all BWR plants. |

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)<br>Normal conditions<br>Off-normal conditions<br>Accident Conditions | <br>100<br>100<br>200 | <br>100<br>100<br>200 |
| Total heat load, max. (kW) | ~~27.77~~ *37.80* (MPC-24)<br>~~28.17~~ *37.79*(MPC-24E & MPC-24EF)<br><br>~~28.74~~ *38.90* (MPC-32 *& MPC-32F*) | ~~28.19~~ *41.22*<br>(MPC-68, MPC-68F, & MPC-68FF) |
| Maximum permissible peak fuel cladding temperature:<br><br>Normal (°F)<br><br>Short Term & Accident (°F) | <br><br>See Table 2.2.3<br><br>1058° | <br><br>See Table 2.2.3<br><br>1058° |
| MPC internal environment<br><br>Helium fill | ~~29.3 – 33.3 psig~~<br><br>~~OR~~<br><br>~~0.1212 gm-moles/l of free space~~<br><br>*Varies with heat load. See CoC Appendix A, Table 3-1* | ~~29.3 – 33.3psig~~<br><br>~~OR~~<br><br>~~0.1218 gm-moles/l of free space~~<br><br>*Varies with heat load. See CoC Appendix A, Table 3-1* |
| Maximum permissible multiplication factor ($k_{eff}$) including all uncertainties and biases | < 0.95 | < 0.95 |
| ~~Boral~~ *Neutron Absorber* $^{10}$B Areal Density (g/cm$^2$) | 0.0267 (MPC-24)<br><br>0.0372 (MPC-24E, MPC-24EF MPC-32, *& MPC-32F*) | 0.0372<br>(MPC-68 & MPC-68FF)<br><br>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

BORAL EXPERIENCE LIST
DOMESTIC PRESSURIZED WATER REACTORS

| Plant | Utility |
|---|---|
| Donald C. Cook | American Electric Power |
| Indian Point 3 | New York Power Authority |
| Maine Yankee | Maine Yankee Atomic Power |
| Salem 1,2 | Public Service Electric and Gas |
| Sequoyah 1,2 | Tennessee Valley Authority |
| Yankee Rowe | Yankee Atomic Power |
| Zion 1,2 | Commonwealth Edison Company |
| Byron 1,2 | Commonwealth Edison Company |
| Braidwood 1,2 | Commonwealth Edison Company |
| Three Mile Island I | GPU Nuclear |
| Sequoyah (rerack) | Tennessee Valley Authority |
| D.C. Cook (rerack) | American Electric Power |
| Maine Yankee | Maine Yankee Atomic Power Company |
| Connecticut Yankee | Northeast Utilities Service Company |
| Salem Units 1 & 2 (rerack) | Public Service Electric & Gas Company |

| Table 1.2.4<br><br>BORAL EXPERIENCE LIST<br>DOMESTIC BOILING WATER REACTORS | |
|---|---|
| Browns Ferry 1,2,3 | Tennessee Valley Authority |
| Brunswick 1,2 | Carolina Power & Light |
| Clinton | Illinois Power |
| Dresden 2,3 | Commonwealth Edison Company |
| Duane Arnold Energy Center | Iowa Electric Light and Power |
| J.A. FitzPatrick | New York Power Authority |
| E.I. Hatch 1,2 | Georgia Power Company |
| Hope Creek | Public Service Electric and Gas |
| Humboldt Bay | Pacific Gas and Electric Company |
| LaCrosse | Dairyland Power |
| Limerick 1,2 | Philadelphia Electric Company |
| Monticello | Northern States Power |
| Peachbottom 2,3 | Philadelphia Electric Company |
| Perry 1,2 | Cleveland Electric Illuminating |
| Pilgrim | Boston Edison Company |
| Susquehanna 1,2 | Pennsylvania Power & Light |
| Vermont Yankee | Vermont Yankee Atomic Power |
| Hope Creek | Public Service Electric and Gas Company |

Table 1.2.4 (continued)

BORAL EXPERIENCE LIST
DOMESTIC BOILING WATER REACTORS

| | |
|---|---|
| Shearon Harris Pool B | Carolina Power & Light Company |
| Duane Arnold | Iowa Electric Light and Power |
| Pilgrim | Boston Edison Company |
| LaSalle Unit 1 | Commonwealth Edison Company |
| Millstone Point Unit One | Northeast Utilities Service Company |

Table 1.2.5

BORAL EXPERIENCE LIST
FOREIGN PLANTS

INTERNATIONAL INSTALLATIONS USING BORAL

| COUNTRY | PLANT(S) |
|---|---|
| France | 12 PWR Plants |
| South Africa | Koeberg 1,2 |
| Switzerland | Beznau 1,2<br>Gosgen |
| Taiwan | Chin-Shan 1,2<br>Kuosheng 1,2 |
| Mexico | Laguna Verde Units 1,2 |
| Korea | Ulchin Units 1, 2 |
| Brazil | Angra 1 |
| United Kingdom | Sizewell B |

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 |
| 9 | MPC lowered and seated on HI-TRAC transfer lid |
| 10 | HI-TRAC/MPC assembly transferred to atop HI-STORM overpack |
| 11 | MPC weight supported by rigging and transfer lid doors opened |
| 12 | MPC lowered into HI-STORM overpack and HI-TRAC removed from atop HI-STORM overpack |
| 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<br>K04100 | 125 | 109 | t ≤ 2.5" |
| ¾ Cr | SA-574 | 51B37M | 170 | 135 | t ≥ 5/8" |
| 1 Cr – 1/5 Mo | SA-574 | 4142 | 170 | 135 | t ≥ 5/8" |
| 1 Cr-1/2 Mo-V | SA-540 | B21<br>(K 14073) | 165 | 150 | t ≤ 4" |
| 5 Cr – ½ Mo | SA-193 | B7 | 125 | 105 | t ≤ 2.5" |
| $2N_i$ – ¾ Cr – ¼ Mo | SA-540 | B23<br>(H-43400) | 135 | 120 | |
| $2N_i$ – ¾ Cr – 1/3 Mo | SA-540 | B-24<br>(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. The family of acceptable materials is denoted as "Alloy Z."

† 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.

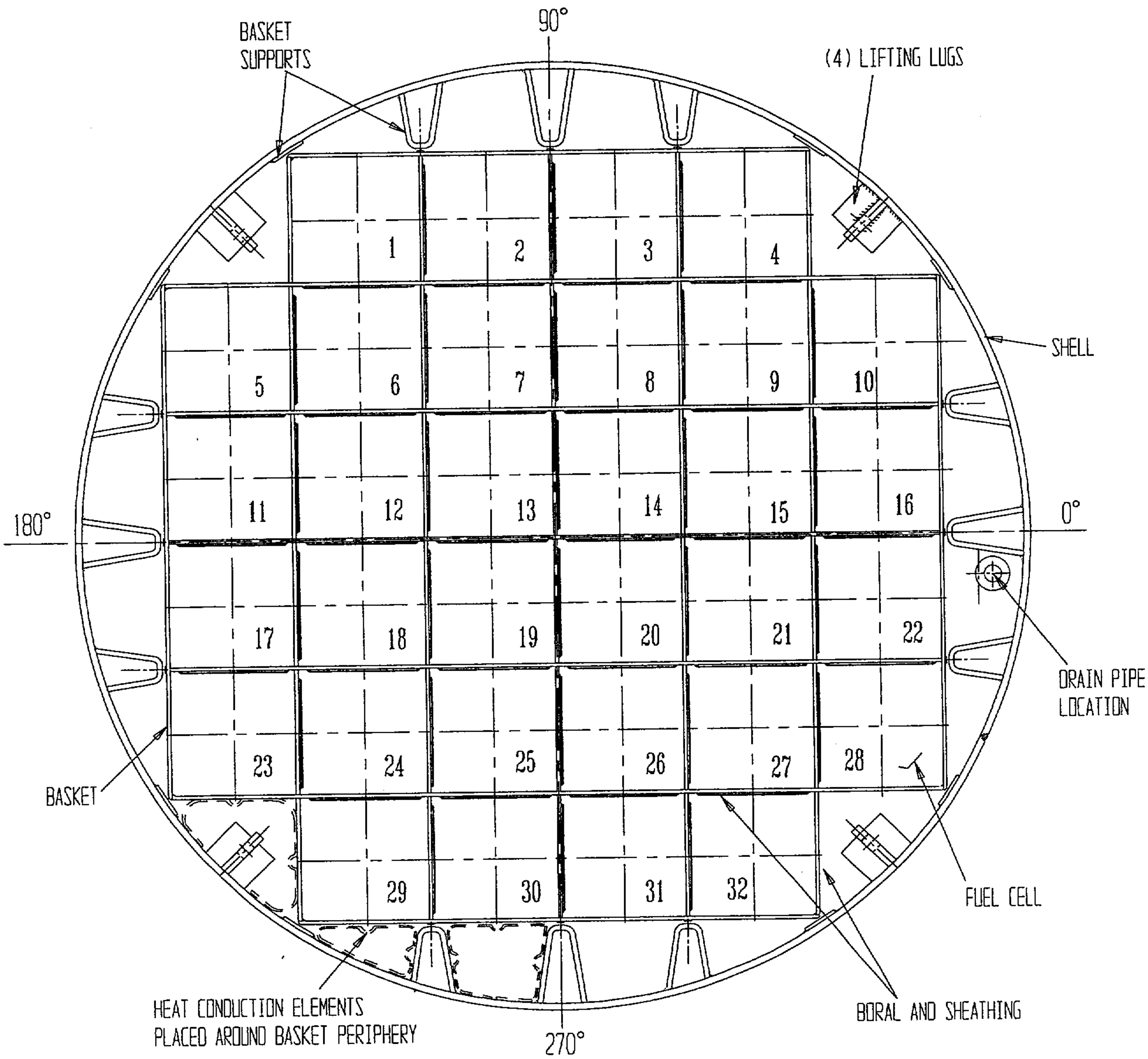


FIGURE 1.2.3; MPC-32/32F CROSS SECTION