Table of Contents

Table of Contents (Continued)

List of Figures

List of Tables

2.0 PRINCIPAL DESIGN CRITERIA

The Universal Storage System is a canister-based spent fuel dry storage cask system that is designed to be compatible with the Universal Transportation System. It is designed to store a variety of intact PWR and BWR fuel assemblies. This chapter presents the design bases, including the principal design criteria, limiting load conditions, and operational parameters of the Universal Storage System. The principal design criteria are summarized in Table 2-1.

Table 2-1 Summary of Universal Storage System Design Criteria

Parameter	Criteria
Design Life	50 years
Design Code - Confinement	ASME Code, Section III, Subsection NB [1] for confinement boundary
Design Code - Nonconfinement	
Basket Vertical Concrete Cask Transfer Cask	ASME Code, Section III, Subsection NG [2] and NUREG/CR-6322 [3] ACI-349 [4], ACI-318 [5] ANSI N14.6 [6] and NUREG-0612 [7]
Maximum Weight: Canister with Design Basis PWR Fuel Assembly (dry, including inserts) (Class 2) Canister with Design Basis BWR Fuel (dry) (Class 5) Vertical Concrete Cask (loaded) (Class 5) Transfer Cask (Class 3)	 72,729 lbs. 75,910 lbs. 312,224 lbs. 120,010 lbs.
Thermal: Maximum Fuel Cladding Temperature: PWR Fuel BWR Fuel Ambient Temperature: Normal (average annual ambient) Off-Normal (extreme cold; extreme hot) Accident Concrete Temperature: Normal Conditions Off-Normal/Accident Conditions	 Variable Based on Fuel Type, Burnup and Cool Time. See Table 4.4.7-5 for Temperature Limits. 1058°F (570°C) Off-Normal/Accident/Transfer [21] Variable Based on Fuel Type, Burnup and Cool Time. See Table 4.4.7-5 for Temperature Limits. 1058°F (570°C) Off-Normal/Accident/Transfer [21] 76°F -40°F; 106°F 133°F ≤ 150°F (bulk); ≤ 200°F (local) [4] ≤ 350°F local/ surface [4]
Cavity Atmosphere	Helium

Table 2-1 Summary of Universal Storage System Design Criteria (Continued)

Radiation Protection/Shielding	**Criteria**
Concrete Cask Side Wall Contact Dose Rate	< 50 mrem/hr. (avg)
Concrete Cask Top Lid Contact Dose Rate	< 50 mrem/hr. (avg)
Concrete Cask Air Inlet/Outlet Dose Rate	< 100 mrem/hr. (max)
Owner Controlled Area Boundary Dose [11]	
Normal/Off-Normal Conditions	25 mrem (Annual Whole Body)
Accident Whole Body Dose	5 rem (Whole Body)

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2.1 Spent Fuel To Be Stored

The Universal Storage System is designed to safely store up to 24 PWR spent fuel assemblies, or up to 56 BWR spent fuel assemblies, contained within a Transportable Storage Canister. Each canister is specifically designed to accommodate one of three classes of PWR fuel assemblies or one of two classes of BWR fuel assemblies. The classification of the fuel assemblies is based primarily on fuel assembly length and cross section. The classes of major fuel assemblies to be stored in the Vertical Concrete Cask and their characteristics are shown in Tables 2.1.1-1 (PWR) and 2.1.2-1 (BWR). The minimum initial enrichment limits are shown in Tables 2.1.1-3 and 2.1.2-3 for PWR and BWR fuel, respectively. The minimum enrichment limits exclude the loading of fuel assemblies enriched to less than 1.9 wt.% ^{235}U, including unenriched fuel assemblies, into the transportable storage canister. However, BWR assemblies with unenriched axial blankets may be loaded into the canister. Any empty fuel rod position must be filled with a solid filler rod fabricated from either Zircaloy or Type 304 stainless steel. The loading of unenriched fuel assemblies into the transportable storage canister is not evaluated and is not authorized. However, BWR assemblies with unenriched axial blankets may be loaded into the canister. Any empty fuel rod position must be filled with a solid filler rod fabricated from either Zircaloy or Type 304 stainless steel. Fuel assembly parameter limits established for the design basis fuel analysis may be exceeded by site specific fuel assemblies. These site specific fuel assembies are separately evaluated, or shown to be bounded by the design basis fuel, as described in Section 2.1.3.

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2.1.1 Bounding Fuel Evaluation: PWR

The classes of PWR fuel assemblies to be stored in the vertical concrete cask and their characteristics are shown in Table 2.1.1-1.

The Westinghouse 17 x 17 fuel assembly, with an initial enrichment of 4.2 wt % ^{235}U, is identified as the most reactive PWR fuel assembly, and it is the design basis PWR fuel for the criticality evaluation. From shielding evaluations, the design basis PWR fuel is the Westinghouse 17 x 17 Standard assembly with a burnup of 40,000 MWD/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time.

The physical parameters for the Westinghouse PWR fuel used in the criticality evaluation are:

• Overall Length (in)	160.1
• Active Length (in)	144.0
• Assembly Width (in)	8.43
• Assembly Weight (lb)	1,602 (including inserts)
• Clad Material	Zircaloy-4

The maximum decay heat load for the Universal Storage System for the storage of all types of PWR fuel is 23 kW. The maximum allowable decay heat load as a function of cool time and burnup are shown in Table 2.1.1-2.

The minimum cool time as a function of assembly class, minimum enrichment and maximum burnup for PWR fuel is shown in Table 2.1.1-3. Minimum cool time is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete cask.

Site specific fuel that does not meet the enrichment and burnup limits of this section and Tables 2.1.1-2 and 2.1.1-3 must be separately evaluated to establish loading limits.

Table 2.1.1-1 PWR Fuel Assembly Characteristics

Fuel Class[1]	Vendor[2]	Array	Max. MTU	No of Fuel Rods	Max. Pitch (in)	Min. Rod Dia. (in)	Min. Clad Thick (in)	Max. Pellet Dia.(in)	Maximum Active Length (in)	Min. Guide Tube Thick (in)
1	CE	14 X 14	0.404	176	0.590	0.438	0.024	0.380	137.0	0.034
1	Ex/ANF	14 X 14	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14 X 14	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14 X 14	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15 X 15	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17 X 17	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17 X 17	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17 X 17	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15 X 15	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17 X 17	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16 X 16	0.442	236	0.506	0.382	0.023	0.3255	150.0	0.035
1	Ex/ANF[3]	14 X 14	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15 X 15	0.432	216	0.550	0.418	0.026	0.358	132.0	----
1	Ex/ANF[3]	15 X 15	0.431	216	0.550	0.417	0.030	0.358	131.8	----
1	CE[3]	16 X 16	0.403	236	0.506	0.382	0.025	0.325	136.7	0.035

1. Maximum Initial Enrichment: 4.2 wt % ^{235}U. All fuel rods are Zircaloy clad.
2. Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.
3. 14x14, 15x15 and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively. These are not generic fuel assemblies provided to multiple reactors.

Table 2.1.1-2 Maximum Allowable Decay Heat for PWR Systems

Cool Time (C) Years[1]	Burnup (B) GWD/MTU		
	$B \leq 35$	**$35 < B \leq 40$**	**$40 < B \leq 45$**
$C = 5$	23 kW	23 kW	23 kW
$5 < C \leq 6$	22.4 kW	22.1 kW	21.9 kW
$6 < C \leq 7$	20.2 kW	20.1 kW	20 kW
$7 < C \leq 10$	19.7 kW	19.6 kW	19.5 kW
$10 < C \leq 15$	19.1 kW	19 kW	18.9 kW

1. Based on 5% Temperature Margin to Allowable.

Table 2.1.1-3 Loading Table for PWR Fuels

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]				30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]			
	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	5	5	5	5	7	7	5	7
2.1 ≤ E < 2.3	5	5	5	5	7	6	5	6
2.3 ≤ E < 2.5	5	5	5	5	6	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	6	6	5	6
2.7 ≤ E < 2.9	5	5	5	5	6	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5	5	5	5
Minimum Initial Enrichment wt % ^{235}U (E)	**35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]**				**40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]**			
	14x14	**15x15**	**16x16**	**17x17**	**14x14**	**15x15**	**16x16**	**17x17**
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	7	9	14	13	10	13
2.3 ≤ E < 2.5	8	8	6	8	12	13	10	12
2.5 ≤ E < 2.7	8	8	6	8	11	13	10	12
2.7 ≤ E < 2.9	7	8	6	8	10	12	9	12
2.9 ≤ E < 3.1	7	8	6	8	9	12	9	11
3.1 ≤ E < 3.3	6	8	6	7	8	12	9	10
3.3 ≤ E < 3.5	6	8	6	7	8	12	9	10
3.5 ≤ E < 3.7	6	8	6	6	8	11	9	10
3.7 ≤ E ≤ 4.2	6	7	6	6	8	10	9	10

2.1.2 Bounding Fuel Evaluation: BWR

The classes of BWR fuel assemblies to be stored in the vertical concrete cask and their characteristics are shown in Table 2.1.2-1.

Criticality evaluations identify the Exxon/ANF 9 x 9 fuel assembly having an initial enrichment of 4.0 wt % ^{235}U, to be the most reactive BWR fuel assembly. The initial enrichment limit represents the maximum initial peak planar-average enrichment allowed for loading into the canister. The initial peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly. From shielding evaluations, the GE 9 x 9 assembly with a burnup of 40,000 MWD/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time is the design basis BWR fuel assembly.

The design basis physical parameters for the BWR fuel criticality evaluation are:

- Overall Length (in) 177.7
- Active Length (in) 150.0
- Assembly Width (in) 5.44
- Assembly Weight (lb) 696.0
- Clad Material Zircaloy-2

The maximum decay heat load for the storage of BWR fuel in the UMS® Storage System is 23.0 kW. The maximum allowable decay heat load for a UMS® Storage Cask, as a function of cool time and burnup, is shown in Table 2.1.2-2. The minimum cooling time as a function of assembly class, minimum enrichment and maximum burnup for BWR fuel is shown in Table 2.1.2-3. Minimum cooling time is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete cask. Based on thermal considerations alone, all of the BWR fuel assembly types meet the decay heat load limits at cooling times less than 15 years. As shown in Table 2.1.2-3, certain higher burnup BWR fuel assemblies require cooling times longer than 15 years, based on dose rate limits established for the UMS® Storage System. The BWR fuel assembly minimum cooling time determination based on dose rate limits is presented in Section 5.5.

Table 2.1.2-1 BWR Fuel Assembly Characteristics

Fuel Class[1,5]	Vendor[4]	Array	Max. MTU	No of Fuel Rods	Max. Pitch (in)	Min. Rod Dia. (in)	Min. Clad Thick (in)	Max. Pellet Dia.(in)	Max. Active Length (in)[2]
4[5]	Ex/ANF	7 X 7	0.196	48	0.738	0.570	0.036	0.490	144.0
4	Ex/ANF	8 X 8	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9 X 9	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7 X 7	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7 X 7	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8 X 8	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8 X 8	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8 X 8	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8 X 8	0.180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9 X 9	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[6]	Ex/ANF	9 X 9	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7 X 7	0.198	49	0.738	0.563	0.032	0.487	144.0
5	GE	8 X 8	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8 X 8	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8 X 8	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9 X 9	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9 X 9	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

1. Maximum Initial Peak Planar-Average Enrichment 4.0 wt % ^{235}U. All fuel rods are Zircaloy clad.
2. 150 inch active fuel length assemblies contain 6" natural uranium blankets on top and bottom.
3. Shortened active fuel length in some rods.
4. Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.
5. UMS Class 4 and 5 Class of fuel for BWR/2-3.
6. Assembly width including channel. Unchanneled or channeled assemblies may be loaded based on a maximum channel thickness of 120 mill.
7. Exxon/ANF assembly weights are listed without channel.

Table 2.1.2-2 Maximum Allowable Decay Heat for BWR Systems

Cool Time (C) Years[1]	Burnup (B) GWD/MTU		
	B ≤35	35 < B ≤ 40	40 < B ≤ 45
C = 5	23 kW	23 kW	23 kW
5 < C ≤ 6	23 kW	23 kW	23 kW
6 < C ≤ 7	22.1 kW	21.9 kW	21.8 kW
7 < C ≤ 10	21.6 kW	21.5 kW	21.4 kW
10 < C ≤ 15	21.1 kW	20.9 kW	20.7 kW

1. Based on 5% Temperature Margin to Allowable.

Table 2.1.2-3 Loading Table for BWR Fuels

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]			30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]		
	7x7	8x8	9x9	7x7	8x8	9x9
1.9 ≤ E < 2.1	5	5	5	8	7	7
2.1 ≤ E < 2.3	5	5	5	6	6	6
2.3 ≤ E < 2.5	5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5
3.7 ≤ E ≤ 4.0	5	5	5	5	5	5
Minimum Initial Enrichment wt % ^{235}U (E)	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]			40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]		
	7x7	8x8	9x9	7x7	8x8	9x9
1.9 ≤ E < 2.1	16	14	15	26	24	25
2.1 ≤ E < 2.3	13	12	12	23	21	22
2.3 ≤ E < 2.5	9	8	8	18	16	17
2.5 ≤ E < 2.7	8	7	7	15	14	14
2.7 ≤ E < 2.9	7	6	6	13	11	12
2.9 ≤ E < 3.1	6	6	6	11	10	10
3.1 ≤ E < 3.3	6	5	6	9	8	9
3.3 ≤ E < 3.5	6	5	6	8	7	8
3.5 ≤ E < 3.7	6	5	6	7	7	7
3.7 ≤ E ≤ 4.0	6	5	5	7	6	7

Note: Minimum cooling times greater than 15 years are based on the concrete cask surface dose rate limits established for the UMS® Storage System as described in Section 5.5.

2.1.3 Site Specific Spent Fuel

The UMS® Storage System design basis PWR fuel assemblies are described in Section 2.1.1. Four different assembly arrays: 14x14, 15x15, 16x16 and 17x17, produced by several different fuel vendors, were evaluated in the determination of the PWR design basis fuel.

The design basis BWR fuel assemblies are described in Section 2.1.2. Three different arrays: 7x7, 8x8 and 9x9, produced by several different fuel vendors were evaluated in the determination of the UMS® BWR design basis fuel.

This section describes site specific spent fuel, i.e., fuel assemblies that are configured differently or that have different fuel parameters, such as enrichment or burnup, than the design basis fuel assemblies. The site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations or from the insertion of control components or other items within the fuel assembly.

A summary description of the site specific spent fuels is presented in Section 1.3.2. The site specific spent fuel configurations are either shown to be bounded by the design basis fuel analysis or are separately evaluated. Unless specifically excepted, site specific spent fuel must also meet the conditions specified for the design basis fuel presented in Section 1.3.1.

2.1.3.1 Maine Yankee Site Specific Spent Fuel

The Maine Yankee site specific spent fuel assemblies are categorized as intact (undamaged) or damaged as defined in Table 1-1. All damaged fuel and certain undamaged fuel configurations are placed in a Maine Yankee fuel can for storage in the Transportable Storage Canister.

The configurations of Maine Yankee site specific spent fuel that have been evaluated and found to be acceptable contents are summarized in Section 1.3.2.1, and include those standard fuel assembly configurations, which were modified by the installation or removal of fuel or non-fuel-bearing components.

The three principal types of these modifications are:

- The removal of fuel rods without replacement.
- The replacement of removed fuel rods or burnable poison rods with rods of another material, such as stainless steel, or with fuel rods of a different enrichment.
- The insertion of control elements, or instrument or plug thimbles, in guide tube positions.

Site specific spent fuel also includes fuel assemblies that are uniquely designed to support reactor physics. These fuel assemblies include those that are variably enriched or that are variably enriched with annular axial blankets. Generally, these fuel assemblies (described in Sections 6.6.1.2.2 and 6.6.1.2.3) are bounded by the evaluation of the design basis fuel.

As described in Section 2.1.3.1.6, certain of the site specific spent fuel configurations, including damaged and consolidated fuel, must be preferentially loaded in corner positions of the PWR fuel basket. In addition, certain of the site specific fuel has experienced burnup that exceeds the 45,000 MWD/MTU design basis burnup. The thermal evaluation of these fuel assemblies is presented in Section 4.5.1. The results of that evaluation show that a fuel assembly with a burnup between 45,000 and 50,000 MWD/MTU must be preferentially loaded in peripheral fuel positions in the basket.

2.1.3.1.1 Damaged Fuel Lattices

There are two lattices for damaged fuel rods in the current Maine Yankee fuel inventory, designated CF1 and CA3, that are loaded in Maine Yankee fuel cans. CF1 is a lattice having roughly the same dimensions as a standard fuel assembly. It is a 9x9 array of tubes, some of which contain damaged fuel rods. CA3 is a previously used fuel assembly lattice that has had all of the rods removed, and into which, damaged fuel rods have been inserted. The CF1 and CA3 lattices are placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.2 Maine Yankee Consolidated Fuel

The Maine Yankee fuel inventory includes two consolidated fuel lattices, which house intact fuel rods taken from three fuel assemblies. Each lattice is a 17x17 array formed using stainless steel grids and top and bottom stainless steel end fittings. Four solid stainless steel connector rods connect the end fittings. The top end fitting is designed so that the lattice can be handled by the standard fuel assembly lifting fixture (grapple). These lattices were not used in the reactor and the stainless steel hardware is not activated.

One of these lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the 76 stainless steel dummy rods in the outer periphery of the lattice.

The consolidated fuel is placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.3 Maine Yankee Spent Fuel with Inserted Integral Hardware

Certain Maine Yankee fuel assemblies have either a Control Element Assembly or an Instrument Thimble inserted in the fuel assembly. These components add to the gamma radiation source term of the standard fuel assembly.

A Maine Yankee Control Element Assembly (CEA) consists of five control rods mounted on a Type 304 stainless steel spider assembly. The five control rods are inserted in the fuel assembly guide tubes when the CEA is inserted in the fuel assembly. When fully inserted, the control element spider rests on the fuel assembly upper end fitting. The rods are fabricated from Inconel 625 or stainless steel and encapsulate B_4C as the primary neutron poison material. Fuel assemblies with a control element installed must be loaded into a Class 2 canister because of the additional height that the control element spider adds to the fuel assembly overall length.

Some standard fuel assemblies have an in-core instrument thimble inserted in the center guide tube of the fuel assembly. The detector material and lead wire have been removed from the thimble assembly. The thimble top end and tube are primarily Zircaloy. When installed, the instrument thimble does not add to the overall fuel assembly length. Consequently, fuel assemblies with instrument thimbles are loaded in the Class 1 canister.

2.1.3.1.4 Maine Yankee Spent Fuel with Unique Design

Certain Maine Yankee fuel assemblies were uniquely designed to accommodate reactor physics. These assemblies incorporate variable radial enrichment and axial blankets.

Two batches of fuel used at Maine Yankee contain variably enriched fuel rods. The maximum fuel rod enrichment of one batch is 4.21 wt % ^{235}U with the variably enriched rods enriched to 3.5 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.99 wt % ^{235}U. For the other batch, the maximum fuel rod enrichment is 4.0 wt % ^{235}U, with the variably enriched rods enriched to 3.4 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.92 wt % ^{235}U.

One batch of variably enriched fuel also incorporates axial end blankets with fuel pellets that have a center hole, referred to as annular fuel pellets. Annular fuel pellets are used in the top and bottom 5% of the active fuel length of each fuel rod in this batch.

2.1.3.1.5 Maine Yankee Fuel Can

Fuel assemblies classified as damaged and certain undamaged fuel configurations are loaded in a Maine Yankee fuel can, which is shown in Drawings 412-501 and 412-502. The fuel can may be loaded only in a corner position in the basket of a Class 1 canister. The fuel can analysis assumes the failure of 100 % of the fuel rods held in the fuel can.

The fuel can is sized to accommodate a fuel assembly and to be inserted in a corner position of the fuel basket. As shown in the drawings, the can is 162.8 inches in length and has an external square dimension of 8.62 inches and an internal square dimension of 8.52 inches. In the top 4.5 inches the external square dimension is 8.82 inches. The fuel can is closed on the bottom end by a 0.63-inch thick plate that is welded to the can shell. The plate has drilled holes in each corner to allow water to drain from the can. A screen covers the holes to preclude the release of gross particulates from the fuel can. A lid having an overall depth dimension of 2.38 inches closes the can. The lid is not secured to the can shell, but is held in place when the shield lid is installed in the canister. The lid also has four drilled and screened holes. The damaged fuel is inserted in the fuel can and the lid is installed. Lifting lugs in the can shell allow the loaded can to be lifted and installed in the basket. Alternately, the fuel can may be inserted in a basket corner position before the damaged fuel assembly is inserted in the fuel can. Since the fuel can lid is held in place by the canister shield lid, the fuel can may be used only in the Class 1 canister.

A Maine Yankee fuel can containing fuel debris with greater than 20 Curies of plutonium, requires double containment for transport conditions in accordance with 10 CFR 71.63 (b).

The Maine Yankee fuel can design and fabrication specification summary is provided in Table 2.1.3.1-2. The major physical design parameters of the Maine Yankee fuel can are provided in Table 2.1.3.1-3. The structural evaluation of the Maine Yankee site specific fuel configurations is provided in Section 3.6.1. As shown in Section 4.5.1, the maximum allowable heat load for the contents of a Maine Yankee fuel can is 0.958 kW.

2.1.3.1.6 Maine Yankee Site Specific Spent Fuel Preferential Loading

The estimated Maine Yankee site specific spent fuel inventory is shown in Table 2.1.3.1-1. (Note that the population of fuel in a given configuration may change based on future spent fuel inspection or survey.) As shown in this table, certain fuel configurations are preferentially loaded to take advantage of the design features of the Transportable Storage Canister and basket to allow the loading of fuel that does not specifically conform to the design basis spent fuel. The designated preferential loading positions are shown in Figure 2.1.3.1-1. The corner positions are designated by the letter "C." These positions are used primarily for the loading of fuel with missing fuel rods, fuel with fuel rods that have been replaced by rods of other material, for consolidated fuel lattices, and for damaged fuel. The requirements for preferential loading schemes using the corner positions result primarily from shielding or criticality evaluations of the designated fuel configurations.

Maine Yankee consolidated fuel is loaded in a Maine Yankee fuel can and is, therefore, designated for a corner position. Preferential loading is also used for spent fuel having a burnup between 45,000 and 50,000 MWD/MTU. This fuel is assigned to peripheral locations designated by the letter "P" in Figure 2.1.3.1-1. The thermal analysis supporting the use of these locations for higher burnup fuel is presented in Section 4.5.1. As described in that section, the interior locations must be loaded with fuel that has lower burnup and/or longer cool times in order to maintain the design basis heat load and component temperature limits. Loading tables, which provide the limits for decay heat on a per assembly basis, are also provided in Section 4.5.1.

High burnup fuel (45,000 – 50,000 MWD/MTU) may be loaded as intact fuel provided that for a given fuel assembly, the cladding oxide layer on no more than 1% of the fuel rods has a peak thickness greater than 80 microns and no more than 3% of the fuel rods have a peak oxide layer thickness greater than 70 microns. The high burnup fuel must be loaded as failed fuel (i.e., in a Maine Yankee fuel can), if the cladding oxide layer criteria are not met, or if the oxide layer is detached or spalled from the cladding. Since the transportable storage canister is tested to be leak tight, no additional confinement analysis is required for the high burnup fuel.

Fuel assemblies with a control element inserted will be loaded in a Class 2 canister and basket for storage and transport due to the increased length of the assembly with the control element installed. However, these assemblies are not restricted as to loading position within the basket.

2.1.3.1.7 Maine Yankee High Burnup Fuel

There are ninety (90) Maine Yankee fuel assemblies that have achieved a burnup between 45,000 and 50,000 MWD/MTU. As described in Section 2.1.3.1.6, these fuel assemblies are preferentially loaded in peripheral locations in the basket. The high burnup assemblies are similar to the other Maine Yankee fuel planned to be placed in dry storage (i.e., those with burnup less than 45,000 MWD/MTU), but have design differences that support the high burnup objective.

The Combustion Engineering 14x14 high burnup fuel assemblies incorporate a lower (fuel rod) internal pressure than the UMS design basis fuel, which results in lower cladding stress throughout their reactor and storage life, and a greater cladding thickness. The greater cladding thickness, together with a larger fuel rod diameter, provide additional margin against regulatory limits. Some of the fuel assemblies have a "low tin" Zircaloy cladding, which results in lower hydrogen pick-up in the cladding and a lower cladding oxide layer thickness.

Publicly available DOE-sponsored research studies on high burnup fuel have measured irradiated Zircaloy material properties. These studies show that even at burnups over 50,000 MWD/MTU, Zircaloy cladding has adequate material strength and ductility to maintain the fuel rod integrity throughout all conditions of storage. The technical details of the DOE sponsored research studies include hot cell examinations of actual irradiated fuel from the Fort Calhoun [22] and Oconee [23] reactors.

The published reports conclude that there is an increase in the yield and ultimate strengths of the high burnup fuel rod Zircaloy cladding, with an oxide layer thickness less than or equal to 80 microns while the material ductility decreases. The Fort Calhoun and Oconee fuel rods examined had maximum burnup up to 55,700 and 54,800 MWD/MTU, respectively. Localized burnup of these fuel rods reached over 60,000 MWD/MTU. The burnups encompass the burnups of the Maine Yankee high burnup fuel. Tables 17, 18 and 19 of the Fort Calhoun report (DOE/ET/34030-11) demonstrate that, at the respective burnups, there is a significant increase in the yield and ultimate strengths of the Zircaloy cladding with a corresponding decrease in the material ductility (plastic strain). This is further confirmed in the Oconee fuel examination report (DOE/ET/34212-50) in Table 20. These studies show that the Zircaloy material property changes occur during the early stages of irradiation and do not change significantly during the higher burnup periods. The Fort Calhoun and Maine Yankee fuels are essentially identical and are fabricated by the same supplier (Combustion Engineering). Therefore, it is concluded that the Maine Yankee high burnup fuel (45,000 < Burnup < 50,000 MWD/MTU) Zircaloy cladding ultimate and yield strengths are greater than those of standard burnup fuel assemblies, while maintaining adequate material ductility to perform its design functions.

The Maine Yankee high burnup fuel assemblies were fabricated according to their respective fuel specifications without any discrepancies or deviations that affected cladding. Review of Plant Operating Data demonstrates that the fuel has not been subjected to any unanalyzed events that could potentially lead to excessive cladding stress.

Review of fuel inspection records and video tapes of the Maine Yankee high burnup fuel assemblies shows that the fuel is essentially identical to fuel that is burned less than 45,000 MWD/MTU, with no evidence of damage or excessive cladding oxidation.

The supporting data and information demonstrates that the physical and mechanical characteristics of the Maine Yankee high burnup fuel assemblies (45,000 < Burnup < 50,000 MWD/MTU) are essentially identical to those of the fuel assemblies with burnup less than 45,000 MWD/MTU.

As noted in Section 2.1.3.1.6, high burnup fuel (45,000 – 50,000 MWD/MTU) may be loaded as intact fuel provided that for a given fuel assembly, the cladding oxide layer on no more than 1% of the fuel rods has a peak thickness greater than 80 microns and no more than 3% of the fuel rods have a peak oxide layer thickness greater than 70 microns. The high burnup fuel must be loaded as failed fuel (i.e., in a Maine Yankee fuel can), if the fuel is not classified as intact, does not meet the oxide layer thickness criteria, or if the oxide layer is detached or spalled from the fuel cladding. Since the transportable storage canister is tested to be leak tight, no additional confinement analysis is required for the high burnup fuel.

Figure 2.1.3.1-1 Preferential Loading Diagram for Maine Yankee Site Specific Spent Fuel

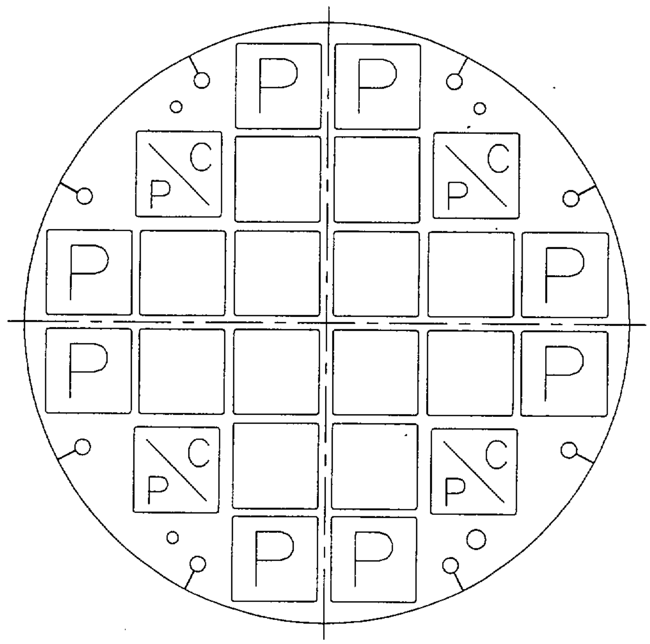

Table 2.1.3.1-1 Maine Yankee Site Specific Fuel Population

Site Specific Spent Fuel Configuration	Est. Number of Assemblies[1,2]	Canister Loading Position[3]
Inserted Control Element Assembly (CEA)[4,5,6]	168	Any
Inserted In-Core Instrument (ICI) Thimble	138	Any
Consolidated Fuel	2	Corner[7,8,9]
Fuel Rod Replaced by Rod Enriched to 1.95 wt %	3	Any
Fuel Rod Replaced by Stainless Steel Rod or Zircaloy Rod	18	Any
Fuel Rods Removed	10	Corner
Variable Enrichment	72	Any
Variable Enrichment and Axial Blanket	68	Any
Burnable Poison Rod Replaced by Hollow Zircaloy Rod	80	Corner
Damaged Fuel[10,11]	12	Corner
Burnup between 45,000 and 50,000 MWD/MTU	90	Periphery[12]
Maine Yankee Fuel Can	As Required	Corner[13]

1. The total number of fuel assemblies in inventory, including these site specific spent fuel configurations and standard spent fuel assemblies, is approximately 1,434.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.
3. Standard fuel assemblies may be loaded in any position.
4. A fuel assembly with an inserted CEA must be loaded in a Class 2 canister.
5. A fuel assembly without an inserted CEA must not be loaded in a Class 2 canister.
6. CEAs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods.
7. Basket corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1. Corner positions are also periphery positions.
8. Only one Consolidated Fuel lattice may be loaded in any Transportable Storage Canister.
9. Consolidated Fuel must be loaded in a Maine Yankee fuel can.
10. All fuel classified as damaged must be placed in a Maine Yankee fuel can, including fuel assemblies with damaged fuel rods or poison rods inserted in guide tubes.
11. All spent fuel, including that held in a Maine Yankee fuel can, must conform to the loading limits presented in Tables 12B2-8 and 12B2-9 for cool time. The fuel maximum planar average enrichment should be used for variable enriched assemblies when determining minimum cool time.
12. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24 in Figure 12B2-1. Periphery positions include the corner positions.
13. The Maine Yankee fuel can may be loaded only in a Class 1 canister.

Table 2.1.3.1-2 Maine Yankee Fuel Can Design and Fabrication Specification Summary

Design

- The Maine Yankee Fuel Can shall be designed in accordance with ASME Code, Section III, Subsection NG except for: 1) the noted exceptions of table 12B3-1 for fuel basket structures; and 2) the Maine Yankee Fuel Can may deform under accident conditions of storage.
- The Maine Yankee Fuel Can will have screened vents in the lid and base plate. Stainless steel meshed screens (250x250) shall cover all openings.
- The Maine Yankee Fuel Can shall limit the release of material from damaged fuel assemblies and fuel debris to the canister cavity.
- The Maine Yankee Fuel Can lifting structure and lifting tool shall be designed with a minimum factor of safety of 3.0 on material yield strength.

Materials

- All material shall be in accordance with the referenced drawings and meet the applicable ASME Code sections.
- All structural materials are ASME SA 240, Type 304 stainless steel.

Welding

- All welds shall be in accordance with the referenced drawings.
- The final surface of all welds shall be liquid penetrant examined in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code, Section NG-5350.

Fabrication

- All cutting, welding, and forming shall be in accordance with ASME Code Section III, NG-4000.

Acceptance Testing

- The Maine Yankee Fuel Can (first unit) and handling tool shall be load tested and visually inspected at the completion of fabrication.

Quality Assurance

- The Maine Yankee Fuel Can shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G. The quality assurance program must be accepted by NAC International and the licensee prior to initiation of the work.
- A Certificate of Conformance (or Compliance) shall be issued by the fabricator stating that the component meets the specifications and drawings.

Table 2.1.3.1-3 Major Physical Design Parameters of the Maine Yankee Fuel Can

Parameter	**Value**
Overall Length (in.)	162.8
Inside Cross Section (in.)	8.52 x 8.52
Outside Cross Section (in.) [(1)]	8.62 x 8.62
Can Wall Thickness	18 Gauge (0.048 in.)
Internal Cavity Length (in.)	160.0
Empty Weight (nominal) (lbs.)	130

Note [(1)]Outside cross section of Maine Yankee Fuel Can upper structure is 8.82 x 8.82 in. at top (4.5 in.) for lid engagement and fuel can lifting. This upper structure is located above the top weldment plate of the fuel basket assembly.

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2.2 Design Criteria for Environmental Conditions and Natural Phenomena

This section presents the design criteria for site environmental conditions and natural phenomena applied in the design basis analysis of the UMS® Universal Storage System. These criteria reflect conditions and phenomena to which the Storage System could be exposed during the period of storage. The system is designed to withstand the loads imposed by these environmental conditions and natural phenomena. Analyses to demonstrate that the design basis system meets the design criteria defined in this section are presented in the appropriate chapters of this Safety Analysis Report.

The use of the UMS® Universal Storage System at a specific site requires that the site either meet the design criteria of this section or be separately evaluated against the site specific conditions to ensure the acceptable performance of the UMS® Universal Storage System. Site specific evaluations are incorporated in designated sections of each chapter of this Safety Analysis Report. Site specific evaluations for environmental conditions and natural phenomena are presented in Section 11.2.15.

2.2.1 Tornado and Wind Loadings

The Vertical Concrete Casks are typically placed outdoors on an unsheltered reinforced concrete storage pad at an ISFSI site. This storage condition exposes the casks to tornado and wind loading.

2.2.1.1 Applicable Design Parameters

The design basis tornado and wind loading is defined based on Regulatory Guide 1.76 [9] Region 1 and NUREG-0800 [10]. The tornado and wind loading criteria are:

Tornado and Wind Condition	**Limit**
Rotational Wind Speed, mph	290
Translational Wind Speed, mph	70
Maximum Wind Speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

2.2.1.2 Determination of Forces on Structures

Tornado wind forces on the Vertical Concrete Cask are calculated by multiplying the dynamic wind pressure by the frontal area of the cask normal to the wind direction. Wind forces are applied to the cask in the wind direction. No streamlining is assumed. The evaluation of wind loading and tornado missile effects on the cask is presented in Section 11.2.11. The total design basis wind loading on the projected area of the cask is determined in Section 11.2.11. The cask is demonstrated to remain stable under design basis tornado wind loading in conjunction with impact from a high energy tornado missile.

2.2.1.3 Tornado Missiles

The design basis tornado missile impacts are defined in Paragraph 4, Subsection III, Section 3.5.1.4 of NUREG-0800 [10]. The design basis tornado is considered to generate three types of missiles that impact the cask at normal incidence:

1. Massive Missile - (Deformable w/high kinetic energy)	Weight = 4,000 lbs Frontal Area = 20 sq.-ft
2. Penetration Missile - (Rigid hardened steel)	Weight = 280 lbs Diameter = 8.0 in
3. Protective Barrier Missile - (Solid steel sphere)	Weight = 0.15 lbs Diameter = 1.0 in

Each missile is assumed to impact the cask at a velocity of 126 miles per hour, horizontal to the ground, which is 35 percent of the maximum wind speed of 360 miles per hour. For missile impacts in the vertical direction, the assumed missile velocity is (0.7)(126) = 88.2 miles per hour.

The detailed analysis of the Vertical Concrete Cask for missile impacts applies the laws of conservation of momentum and conservation of energy to determine the rigid body response of the concrete cask. Each missile impact is evaluated, and all missiles are assumed to impact in a manner that produces the maximum damage to the cask. The tornado and wind driven missile impact evaluation is presented in Section 11.2.11.

2.2.2 Water Level (Flood) Design

The Vertical Concrete Cask may be exposed to a flood during storage on an unsheltered concrete storage pad at an ISFSI site. The source and magnitude of the probable maximum flood depend on specific site characteristics.

2.2.2.1 Flood Elevations

The Vertical Concrete Cask is evaluated in Section 11.2.9 for a maximum flood water depth of 50 feet above the base of the cask. The flood water velocity is assumed to be 15 feet per second. Results of the evaluation show that under design basis flood conditions, the cask does not float, tip, or slide on the storage pad, and that the confinement function is maintained.

2.2.2.2 Phenomena Considered in Design Load Calculations

The occurrence of flooding at an ISFSI site is dependent upon the specific site location and the surrounding geographical features, natural and man-made. Some possible sources of a flood at an ISFSI site are: (1) overflow from a river or stream due to unusually heavy rain, snow-melt runoff, a dam or major water supply line break caused by a seismic event (earthquake); (2) high tides produced by a hurricane; and (3) a tsunami (tidal wave) caused by an underwater earthquake or volcanic eruption.

Flooding at an ISFSI site is highly improbable because of the extensive environmental impact studies that are performed during the selection of a site for a nuclear facility.

2.2.2.3 Flood Force Application

The evaluation of the Universal Storage System for a flood condition determines a maximum allowable flood water current velocity and a maximum allowable flood water depth. The criteria employed in the determination of the maximum allowable values are that a cask sliding or tip-over will not occur, and that the canister material yield strength is not exceeded. The evaluation of the effects of flood conditions on the system is presented in Section 11.2.9.

The force of the flood water current on the cask is calculated as a function of the current velocity by multiplying the dynamic water pressure by the frontal area of the cask that is normal to the current direction. The dynamic water pressure is calculated using Bernoulli's equation relating fluid velocity and pressure. The force of the flood water current is limited such that the overturning moment on the cask will be less than that required to tip the cask over.

2.2.2.4 Flood Protection

The inherent strength of the reinforced concrete cask provides a substantial margin of safety against any permanent deformation of the cask for a credible flood event at an ISFSI site. Therefore, no special flood protection measures for the cask are necessary. The evaluation presented in Section 11.2.9 shows that for the design basis flood, the allowable stresses in the canister are not exceeded.

2.2.3 Seismic Design

An ISFSI site may be subject to seismic events (earthquakes) during its lifetime. The seismic response spectra experienced by the cask depends upon the geographical location of the specific site and the distance from the epicenter of the earthquake. The only significant effect of a seismic event on the Vertical Concrete Cask is a possible tip-over; however, tip-over does not occur during the design basis 0.26g earthquake. Seismic response of the cask is presented in Section 11.2.8.

2.2.3.1 Input Criteria

The Transportable Storage Canister and Vertical Concrete Cask are designed and analyzed for sites east of the Rocky Mountain front by applying a 0.26g seismic acceleration at the top surface of the ISFSI pad.

2.2.3.2 Seismic - System Analyses

The analysis for the earthquake condition applied to nuclear facilities east of the Rocky Mountain front is provided in Section 11.2.8.2. The evaluation shows that the concrete cask does not tip over or slide in the design basis earthquake. Evaluation of the consequences of a hypothetical tip-over event is provided in Section 11.2.12.

2.2.4 Snow and Ice Loadings

The criterion for determining design snow loads is based on ANSI/ASCE 7-93 [12], Section 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7 C_e C_t I p_g$$

where:

p_f = flat roof snow load (psf)
C_e = Exposure factor = 1.0
C_t = Thermal factor = 1.2
I = Importance factor = 1.2
p_g = ground snow load, (psf) = 100

The numerical values of C_e, C_t, I and p_g are obtained from Tables 18, 19, 20 and Figure 7, respectively, of ANSI/ASCE 7-93.

The exposure factor, C_e, accounts for wind effects. The site of the Universal Storage System is assumed to be a location typical for siting Category C, which is defined to be "locations in which snow removal by wind cannot be relied on to reduce roof loads because of terrain, higher structures, or several trees nearby."

The thermal factor, C_t, accounts for the importance of buildings and structures in relation to public health and safety. The Universal Storage System is conservatively classified as Category III.

Ground snow loads for the contiguous United States are given in Figures 5, 6 and 7 of ANSI/ ASCE 7-93. A worst case value of 100 lbs per square ft is assumed.

Based on the above, the design criterion for snow and ice loads is:

Flat Roof Snow Load, p_f = (0.7) (1.0) (1.2) (1.2) (100) = 100.8 psf

This load is bounded by the weight of the loaded transfer cask on the top of the concrete cask shell and by the tornado missile loading on the concrete cask lid. The snow load is considered in the load combinations described in Section 3.4.4.2.2.

2.2.5 Combined Load Criteria

Each normal, off-normal and accident condition has a combination of load cases that defines the total combined loading for that condition. The individual load cases considered include thermal, seismic, external and internal pressure, missile impacts, drops, snow and ice loads, and/or flood water forces.

The load conditions to be evaluated for storage casks are identified in 10 CFR 72[11] and ANSI/ANS-57.9 [13].

2.2.5.1 Load Combinations and Design Strength - Vertical Concrete Cask

The load combinations specified in ANSI/ANS 57.9 for concrete structures are applied to the concrete casks as shown in Table 2.2-1. The live loads are considered to vary from 0 percent to 100 percent to ensure that the worst-case condition is evaluated. In each case, use of 100 percent of the live load produces the maximum load condition. The steel liner of the concrete cask is a stay-in-place form and it provides radiation shielding. The concrete cask is designed to the requirements of ACI 349 [4].

In calculating the design strength of concrete in the Vertical Concrete Cask body, nominal strength values are multiplied by a strength reduction factor in accordance with Section 9.3 of ACI 349.

2.2.5.2 Load Combinations and Design Strength - Canister and Basket

The canister is designed in accordance with the 1995 edition of the ASME Code, Section III, Subsection NB [1] for Class 1 components. The basket structure is designed in accordance with

ASME Code, Section III, Subsection NG [2]. Structural buckling of the basket is evaluated in accordance with NUREG/CR-6322 [3].

The load combinations for all normal, off-normal, and accident conditions and corresponding service levels are shown in Table 2.2-2. The table, therefore, defines the canister design and service loadings. Levels A and D service limits are used for normal and accident conditions, respectively. Levels B and C service limits are used for off-normal conditions. The analysis methods of the ASME Code are employed. Stress intensities caused by pressure, temperature, and mechanical loads are combined before comparing them to ASME code allowables. The Code allowables are listed in Table 2.2-3.

2.2.5.3 Design Strength - Transfer Cask

The transfer cask is a special lifting device. It is designed and fabricated to the requirements of ANSI N14.6 [6] and NUREG 0612 [7] for the lifting trunnions and supports, and ANSI/ANS-57.9 [13] for the remainder of the structure. The criteria are:

1. The combined shear stress or maximum tensile stress during the lift (with 10 percent dynamic load factor) shall be $\leq$ $S_y/6$ and $S_u/10$ for a nonredundant load path, or shall be $\leq S_y/3$ and $S_u/5$ for redundant load paths.

2. The ferritic steel material used for the load bearing members of the transfer cask shall satisfy the material toughness requirements of ANSI N14.6, paragraph 4.2.6.

Load testing of the transfer cask is described in Section 2.3.3.1.

2.2.6 Environmental Temperatures

A normal, long-term annual average design ambient temperature of 76°F is selected to bound most annual average temperatures seen by a cask over its lifetime. This temperature is based on the maximum average annual temperature in the 48 contiguous United States, specifically, Miami, FL., at 75.6°F [14], and is, therefore, used so as to bound existing and potential ISFSI sites.

The 76°F normal temperature is used as the base for thermal evaluations. The evaluation of this environmental condition is discussed along with the thermal analysis models in Chapter 4.0. The thermal stress evaluation for the normal operating conditions is provided in Section 3.4.4. Normal temperature fluctuations are bounded by the severe ambient temperature cases that are evaluated as off-normal and accident conditions.

Off-normal, severe environmental conditions are defined as -40°F with no solar loads and 106°F with solar loads. An extreme environmental condition of 133°F with maximum solar loads is evaluated as an accident case (11.2.7) to show compliance with the maximum heat load case required by ANSI/ANS-57.9. Thermal performance is also evaluated assuming half-blockage of the concrete cask air inlets and the complete blockage of the air inlets and outlets. Thermal analyses for these cases are presented in Sections 11.1.2 and 11.2.13. The evaluation based on ambient temperature conditions is presented in Section 4.4.

The design basis temperatures used in the Universal Storage System analysis are shown below. Solar insolance is as specified in 10 CFR 71.71 [15] and Regulatory Guide 7.8 [16].

Condition	Ambient Temperature	Solar Insolance
Normal	76°F	yes
Off-Normal - Severe Heat	106°F	yes
Off-Normal - Severe Cold	-40°F	no
Accident - Extreme Heat	133°F	yes

Table 2.2-1 Load Combinations for the Vertical Concrete Cask

Load Combination	Condition	Dead	Live	Wind	Thermal	Seismic	Tornado/ Missile	Drop/ Impact	Flood
1	Normal	1.4D	1.7L						
2	Normal	1.05D	1.275L		$1.275T_o$				
3	Normal	1.05D	1.275L	1.275W	$1.275T_o$				
4	Off-Normal and Accident	D	L		T_a				
5	Accident	D	L		T_o	E_{ss}			
6	Accident	D	L		T_o			A	
7	Accident	D	L		T_o				F
8	Accident	D	L		T_o		W_t		

Load Combinations are from ANSI/ANS-57.9 [13] and ACI 349 [4].

D = Dead Load

L = Live Load

W = Wind

T_o = Normal Temperature

F = Flood

T_a = Off- Normal or Accident Temperature

E_{ss} = Design Basis Earthquake

W_t = Tornado/Tornado Missile

A = Drop/Impact

Table 2.2-2 Load Combinations for the Transportable Storage Canister

LOAD		NORMAL			OFF-NORMAL					ACCIDENT					
ASME Service Level		A			B		C			D					
Load Combinations		1	2	3	1	2	3	4	5	1	2	3	4	5	6
Dead Weight	Canister with fuel	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	In Storage Cask 76° F Ambient	X		X				X		X	X	X	X	X	
	In Transfer Cask 76° F Ambient		X		X		X								X
	In Storage Cask -40°F or 106°F Ambient					X			X						
Internal Pressure	Normal	X	X	X			X	X	X	X	X	X	X		
	Off-Normal				X	X									
	Accident													X	X
Handling Load	Normal		X	X	X										
	Off-Normal						X	X	X						
Drop/Impact	Accident									X					
Seismic	Accident										X				
Flood	Accident											X			
Tornado	Accident												X		

Table 2.2-3 Structural Design Criteria for Components Used in the Transportable Storage Canister

	Component	Criteria
1.	Normal Operations: Service Level A Canister: ASME Section III, Subsection NB [1] Basket: ASME Section III, Subsection NG [2] Lifting Devices: ANSI N14.6 [6] and NUREG 0612 [7]	 $P_m \leq S_m$ and $P_L + P_b \leq 1.5\ S_m$ $P_L + P_b + Q \leq 3S_m$ Redundant load path: combined shear or max. tensile stress $\leq S_u/5$ and $S_y/3$
2.	Off-Normal Operations: Service Level B Canister: ASME Section III, Subsection NB	 $P_m < 1.1\ S_m$ and $P_L + P_b < 1.65\ S_m$
3.	Off-Normal Operations: Service Level C Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG Note: Subsection NB allowables for Service Level C are conservatively applied to the basket.	Subsection NB Allowables: $P_m < 1.2\ S_m$ or S_y (whichever is greater) and $P_L + P_b < 1.8\ S_m$ or $1.5\ S_y$ (whichever is less)
4.	Accident Conditions, Service Level D Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG	 $P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (whichever is less) and $P_L + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (whichever is less)
5.	Basket Structural Buckling	NUREG/CR-6322 [3]

Symbols:

S_m = material design stress intensity
S_u = material ultimate strength
S_y = material yield strength
P_L = primary local membrane stress
P_m = primary general membrane stress
P_b = primary bending stress

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2.3 Safety Protection Systems

The Universal Storage System relies upon passive systems to ensure the protection of public health and safety, except in the case of fire or explosion. As discussed in Section 2.3.6, fire and explosion events are effectively precluded by site administrative controls that prevent the introduction of flammable and explosive materials. The use of passive systems provides protection from mechanical or equipment failure.

2.3.1 General

The Universal Storage System is designed for safe, long-term storage of spent nuclear fuel. The system will withstand all of the evaluated normal, off-normal, and postulated accident conditions without release of radioactive material or excessive radiation exposure to workers or the general public. The major design considerations that are incorporated in the Universal Storage System to assure safe, long-term fuel storage are:

1. Continued containment in postulated accidents.
2. Thick concrete and steel biological shield.
3. Passive systems that ensure reliability.
4. Inert helium atmosphere to provide corrosion protection for fuel cladding and enhanced heat transfer for the stored fuel.

Each component of the Universal Storage System is classified with respect to its function and corresponding effect on public safety. In accordance with Regulatory Guide 7.10 [17], each system component is assigned safety classification into Category A, B, or C, as shown in Table 2.3-1. The safety classification is based on review of each component's function and the assessment of the consequences of its failure following the guidelines of NUREG/CR-6407 [18]. The safety classification categories are defined as follows:

- Category A - Components critical to safe operations whose failure or malfunction could directly result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.
- Category B - Components with major impact on safe operations whose failure or malfunction could indirectly result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

- Category C - Components whose failure would not significantly reduce the packaging effectiveness and would not likely result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

As discussed in the following sections, the Universal Storage System design incorporates features addressing the above design considerations to assure safe operation during loading, handling, and storage of spent nuclear fuel.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity that the Universal Storage System must confine originates from the spent fuel assemblies to be stored and residual contamination that may remain inside the canister as a result of contact with water in the fuel pool where the canister loading is conducted. The system is designed to confine this radioactive material.

The Transportable Storage Canister is closed by welding. The shield lid weld is pressure tested. All of the field-installed shield lid welds are liquid penetrant examined following the root and final weld passes. The shield lid welds are leak tested. The installation of the canister structural lid, which provides a redundant closure over the shield lid and port covers, is accomplished by multi-pass welding that is either: 1) progressively liquid penetrant examined; or 2) ultrasonically examined in conjunction with a liquid penetrant examination of the final weld surface. The longitudinal and girth welds of the canister shell are full penetration welds that are radiographically examined during fabrication. The weld that joins the bottom plate to the canister shell is ultrasonically and liquid penetrant examined during fabrication.

The canister welds are an impenetrable boundary to the release of fission gas products during the period of storage. There are no evaluated normal, off-normal, or accident conditions that result in the breach of the canister and the subsequent release of fission products. The canister is designed to withstand a postulated drop accident in the UMS Universal Transport Cask without precluding the subsequent removal of the fuel (i.e., the fuel tubes do not deform such that they bind the fuel assemblies).

Personnel radiation exposure during handling and closure of the canister is minimized by the following steps:

1. Placing the shield lid on the canister while the transfer cask and canister are under water in the fuel pool.
2. Decontaminating the exterior of the transfer cask prior to draining the canister to preserve the shielding benefit of the water.
3. Using temporary shielding.
4. Using a retaining ring on the transfer cask to ensure that the canister is not raised out of the shield provided by the transfer cask.
5. Placing a shielding ring over the annular gap between the transfer cask and the canister.

2.3.2.2 Cask Cooling

The loaded Vertical Concrete Cask is passively cooled. Cool (ambient) air enters at the bottom of the concrete cask through four inlet vents. Heated air exits through the four outlets at the top of the cask. Radiant heat transfer also occurs from the canister shell to the concrete cask liner. Consequently, the liner also heats the convective air flow. Conduction does not play a substantial role in heat removal from the canister surface. This natural circulation of air inside the Vertical Concrete Cask, in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the concrete cask component temperatures below their design limits. The cask cooling system is described in detail in Sections 4.1 and 4.4.

2.3.3 Protection by Equipment and Instrumentation Selection

The Universal Storage System is a passive storage system that does not rely on equipment or instruments to preserve public health or safety and to meet its safety functions in long-term storage. The system employs support equipment and instrumentation to facilitate operations. These items, and the actions taken to assure performance, are described below.

2.3.3.1 Equipment

The equipment that is important-to-safety employed in the use and operation of the Universal Storage System is the transfer cask and the lifting yoke used to lift the transfer cask. The transfer cask lifting yoke is designed to meet the requirements of ANSI N14.6 and NUREG-0612 and is designed as a special lifting device for critical loads. The lifting yoke is proof load tested to 300 percent of design load (660,000 lbs) when fabricated. The lifting yoke has no welds in the lifting load path. Following the load test, the bolted connections are disassembled, and the components are inspected for deformation. Permanent deformation of components is not acceptable. The lifting yoke is inspected for visible defects prior to each use and is inspected annually.

The transfer cask is used to move the empty and loaded Transportable Storage Canister in all of the operations that precede the installation of the loaded canister in the Vertical Concrete Cask. The transfer cask is evaluated as a lifting component. The principal design criteria of the transfer cask are presented in Section 2.2.5.3, above. The transfer cask design meets the requirements of ANSI N14.6 and NUREG-0612. The transfer cask has 2 pairs of lifting trunnions. Each pair is designed as a special lifting device for critical loads, but both pairs may be used together in order to provide a redundant load path. Each pair of transfer cask trunnions is load tested to 300 percent of the maximum calculated service load, or 660,000 lbs. The service load includes the transfer cask weight, the loaded canister, and water in the canister. Following the load test, the trunnion welds and other welds in the load path are inspected for indications of cracking or deformation. The principal load bearing welds and the transfer cask lifting trunnions are evaluated in Section 3.4.3.3.

The transfer cask bottom shield doors support the canister from the bottom during handling of the canister. The shield doors are also load tested to 300 percent of the maximum calculated service load, or 265,000 lbs. The service load includes the weight of the loaded canister and water in the canister. Following the load test, the load bearing surface areas of the doors, rails, and attachment welds are examined for evidence of cracking or deformation.

The transfer cask welds are subjected to a liquid penetrant examination, performed in accordance with the ASME Code, Section V, Article 6. Acceptance criteria is in accordance with the ASME Code, Paragraph NF-5350.

Any evidence of permanent deformation, cracking, galling of bearing surfaces, or unacceptable liquid penetrant examination results is cause for rejection. Any identified defects must be repaired and the load test repeated prior to final acceptance.

2.3.3.2 Instrumentation

A remote temperature measuring system is employed to measure the outlet air temperature of the Vertical Concrete Casks in long-term storage. The outlet temperature is recorded daily as a check of the thermal performance of the heat rejection capability of the storage cask (see Section 12.2). The outlet temperature is expected to increase in the unlikely event that one or more inlets or outlets become blocked. Consequently, visual inspection of the inlets and outlets is required when the temperature differential between the ambient air temperature and the outlet air temperature exceeds 102°F for the PWR configuration or 92°F for the BWR configuration during normal operations. In addition, visual inspection of the inlets and outlets is required following any natural phenomena event, such as an earthquake, that could lead to a reduction in efficiency of the cooling system.

The canister shield lid weld is helium leak tested during closure. The leak detector is checked against a known helium source immediately prior to, and after, use to preclude unknown leak detector failure.

2.3.4 Nuclear Criticality Safety

The Universal Storage System design includes features to ensure that nuclear criticality safety is maintained (i. e., the cask remains subcritical) under normal, off-normal, and accident conditions. The design of the canister and fuel basket is such that, under all conditions, the highest neutron multiplication factor (k_{eff}) is less than 0.95. The criticality evaluation for the design basis fuel is presented in Section 6.4.

2.3.4.1 Control Methods for Prevention of Criticality

Criticality control in the PWR basket is achieved using a neutron flux trap configuration. Individual fuel assemblies are surrounded by four BORAL sheets, one on each side of the assembly, that provide absorption of moderated neutrons. The assemblies are separated by a gap that is filled with water during hypothetical accident conditions when the canister is flooded. Fast neutrons escaping one fuel assembly are moderated in the gap between the assemblies and absorbed by the BORAL neutron poison surrounding the assemblies. The minimum loading of the BORAL sheets is 0.025 g $^{10}B/cm^2$. The BORAL sheets are mechanically supported by the fuel tube structure to ensure that the neutron absorber sheets remain in place during the design basis normal, off-normal, and accident events.

Individual fuel assemblies in the BWR basket are separated from adjacent assemblies by a single BORAL sheet between fuel assemblies. Of the total 56 fuel tubes, 42 tubes contain BORAL sheets on two sides of the tubes, 11 tubes contain BORAL sheets on one side, and the remaining 3 tubes contain no BORAL sheets. The arrangement of the fuel tubes ensures that there is at least one BORAL sheet between adjacent fuel assemblies. Although this configuration of water gaps and BORAL neutron absorber sheets does not form a classic neutron flux trap, the design assures that there is sufficient absorption of moderated neutrons by the BORAL to maintain criticality control in the basket ($k_{eff} < 0.95$). The minimum loading of the BORAL sheets in the BWR fuel tubes is 0.011 g $^{10}B/cm^2$. The BORAL sheets are mechanically supported by the fuel tube structure to ensure that the neutron absorber sheets remain in place during the design basis normal, off-normal, and accident events.

The efficiency of the BORAL sheets in preserving nuclear criticality safety is demonstrated by the criticality results presented in Section 6.4.3.

The principal criticality design criterion is that k_{eff} remain below 0.95 under all conditions. Assumptions made in the analyses used to demonstrate conformance to this criterion include:

1. Fuel assembly with maximum ^{235}U loading (95% theoretical density);
2. 75 percent of the nominal ^{10}B loading in the BORAL;
3. Infinite array of casks in the X-Y (horizontal) plane;
4. Infinite fuel length with no inclusion of end leakage effects;
5. No credit is taken for soluble boron in spent fuel pool water;
6. No credit taken for structural material present in the assembly; and,
7. No credit taken for fuel burnup or for the buildup of fission product neutron poisons.

Use of administrative controls of fuel burnup levels, neutron absorption properties of the burned fuel, and the presence of steel shell of the canister provide further criticality controls in the Universal Storage System.

2.3.4.2 Error Contingency Criteria

The calculated values of k_{eff} include error contingencies and calculation and modeling biases. The standards and regulations of criticality safety require that k_{eff}, including uncertainties, k_s, be less than 0.95. The bias and 95/95 uncertainty are applied to the calculation of k_s by using:

$$k_s = k_{nom} + 0.0052 + [(0.0087)^2 + (2\sigma_{MC})^2]^{1/2} \leq 0.95$$

where:

k_{nom} = the nominal k_{eff} for the cask, and
σ_{MC} = the Monte Carlo uncertainty.

The calculation of error contingencies and uncertainties is presented in Section 6.4.

2.3.4.3 Verification Analyses

The CSAS25 criticality analysis sequence is benchmarked through a series of calculations based on 63 critical experiments. These experiments span a range of fuel enrichments, fuel rod pitches, poison sheet characteristics, shielding materials, and geometries that are typical of light water reactor fuel in a cask. To achieve accurate results, three-dimensional models, as close to the actual experiment as possible, are used to evaluate the experiments. The results of the benchmark calculations are provided in Section 6.5.

2.3.5 Radiological Protection

The Universal Storage System, in keeping with the As Low As Is Reasonably Achievable (ALARA) philosophy, is designed to minimize, to the extent practicable, operator radiological exposure.

2.3.5.1 Access Control

Access to a Universal Storage System ISFSI site is controlled by a peripheral fence to meet the requirements of 10 CFR 72 and 10 CFR 20 [19]. Access to the storage area, and its designation as to the level of radiation protection required, are established by site procedure. The storage area is surrounded by a fence, having lockable truck and personnel access gates. The fence has intrusion-detection features as determined by the site procedure.

2.3.5.2 Shielding

The Universal Storage System is designed to limit the dose rates as follows:

- external surface dose (gamma and neutron) to less than 50 mrem/hr (average) on the Vertical Concrete Cask sides.
- external surface dose to less than 50 mrem/hr (average) on the Vertical Concrete Cask top.
- a maximum of 100 mrem/hr at the Vertical Concrete Cask air inlets and outlets.
- less than 300 mrem/hr (average) on the transfer cask side wall.
- the design maximum dose rate at the top of the canister structural lid, with supplemental shielding to less than 300 mrem/hr to limit personnel exposure during canister closure operations.

Sections 72.104 and 72.106 of 10 CFR 72 set whole body dose limits for an individual located beyond the controlled area at 25 millirems per year (whole body) during normal operations and 5 rems (5,000 millirems) from any design basis accident. The analyses showing the actual Universal Storage System doses, and dose rates, are included in Chapters 5.0, 10.0 and 11.0.

2.3.5.3 Ventilation Off-Gas

The Universal Storage System is passively cooled by radiation and natural convection heat transfer at the outer surface of the concrete cask and in the canister-concrete cask annulus. The bottom of the cask is conservatively assumed to be an adiabatic surface. In the canister-concrete cask annulus, air enters the air inlets, flows up between the canister and concrete cask liner in the annulus, and exits the air outlets. The air flow in the annulus is due to the buoyancy effect created by the heating of the air by the canister and concrete cask liner walls. The details of the passive ventilation system design are provided in Chapter 4.0.

The surface of the canister is exposed to cooling air when the canister is placed in the concrete cask. If the surface is contaminated, the possibility exists that contamination could be carried aloft by the cooling air stream. Therefore, during fuel loading, the spent fuel pool water is excluded from the canister exterior by filling the transfer cask/canister annular gap with clean water as the transfer cask is being lowered into the fuel pool. Clean water is injected into the gap during the entire time the transfer cask is submerged. These steps minimize the potential for the intrusion of contaminated water into the canister annular gap.

Once the transfer cask is removed from the pool, a smear survey is taken of the exterior surface of the canister near the top. While no contamination is expected to be found, it is possible that the surface could be contaminated. The allowable upper limit for surface contamination of the canister and transfer cask is provided in LCO 3.2.1 in Appendix 12A of Chapter 12. As described in LCO 3.2.1, if this limit is exceeded, steps to decontaminate the canister surface must be taken and continued until the contamination is less than the allowable limit.

To facilitate decontamination, the canister is fabricated so that its exterior surface is smooth. There are no corners or pockets that could trap and hold contamination.

There are no radioactive releases during normal operations. Also, there are no credible accidents that cause significant releases of radioactivity from the Universal Storage System and, hence, there are no off-gas system requirements for the system during normal storage operation. The only time an off-gas system is required is during the canister drying phase. During this operation, the reactor off-gas system or a HEPA filter system is used.

2.3.5.4 Radiological Alarm Systems

No radiological alarms are required on the Universal Storage System. Justification for this is provided in Chapter 5.0 (Shielding), 10.0 (Radiological Protection), and 11.0 (Accident Analysis).

Typically, total radiation exposure due to the ISFSI installation is determined by the use of Thermo-Luminescent Detectors (TLDs) mounted at convenient locations on the ISFSI fence. The TLDs are read quarterly to provide a record of boundary dose.

2.3.6 Fire and Explosion Protection

Fire and explosion protection of the Universal Storage System is provided primarily by administrative controls applied at the site, which preclude the introduction of any explosive and any excessive flammable materials into the ISFSI area.

2.3.6.1 Fire Protection

A major ISFSI fire is not considered credible, since there is very little material near the casks that could contribute to a fire. The concrete cask is largely impervious to incidental thermal events. Administrative controls are put in place to ensure that the presence of combustibles is minimized. A hypothetical fire event is evaluated as an accident condition in Section 11.2.6. The fire event evaluated is a 1475°F fire of 8 minutes duration. This condition is considered to be highly conservative.

2.3.6.2 Explosion Protection

The Universal Storage System is analyzed to ensure its proper function under an over-pressure condition. As described in Section 11.2.5, in the evaluated 22 psig over-pressure condition, stresses in the canister remain below allowable limits and there is no loss of confinement. These results are conservative, as the canister is protected from direct over-pressure conditions by the concrete cask.

For the same reasons as for the fire condition, a severe explosion on an ISFSI site is not considered credible. The evaluated over-pressure is considered to bound any explosive over-pressure resulting from an industrial explosion at the boundary of the owner-controlled area.

2.3.7 Ancillary Structures

The loading, transfer and transport of the UMS® System requires the use of auxiliary equipment as described in Section 2.3.3 and may require the use of an ancillary structure, referred to as a "Canister Handling Facility." The Canister Handling Facility is an especially designed and engineered structure separate from the 10 CFR 50 facilities at the site. The Canister Handling Facility, if required, would provide a housing for a lifting crane, service air and water, a radiation control area, auxiliary equipment storage and support services and work areas related to canister handling and transfer. Transfer operations could include temporary holding of a loaded canister in the transfer cask to allow repair of a concrete cask, transfer of a canister from one concrete cask to another, or transfer from a concrete cask to a transport cask.

The design of the Canister Handling Facility would meet the requirements of the UMS® Storage System described in Approved Contents and Design Features presented in Appendix 12B of Chapter 12, in addition to those requirements established by the site.

The design, analysis, fabrication, operation and maintenance of the Canister Handling Facility would be performed in accordance with the quality assurance program requirements of the site general licensee, or the site-specific licensee of the ISFSI. The Canister Handling Facility would be classified as Important to Safety or Not Important to Safety in accordance with the guidelines of NUREG-6407.

Table 2.3-1 Safety Classification of Universal Storage System Components

Drawing No.	Sheet No.	Description	Rev No.	Item No.	Component	Function	Safety Class
790-561	1	Weldment, Structure, Vertical Concrete Cask	5	27-30	Shell	Shielding/Structural	B
				26	Screen Table	Structural	C
				25	Baffle	Heat Transfer	B
				21-24	Outlet (4)	Heat Transfer	B
				20	Shield Plate	Shielding	B
				17	Nelson Stud	Structural	B
				16	Base Plate	Structural	B
				15	Stand	Structural	B
				13-14	Inlet (4)	Heat Transfer	B
				12	Bottom	Structural	B
				11	Shield Ring	Shielding	B
				10	Cover	Operations	B
				4-8	Jack (Leveling)	Operations	C
				3	Support Ring	Structural	C
				2	Top Flange	Structural	B
				1	Shell	Structural	B
790-562	1	Reinforcing Bar And Concrete Placement	5	31-33	Lift Anchor	Structural	B
				28-29	Concrete Anchor/Lag Screw	Operations	C
				25	Outlet Screen	Operations	C
				24	Inlet Screen	Operations	C
				20-23	Structure Weldment	Shielding/ Structural	B
				17,19	Screen/Strip/Screw	Operations	C
				15	Concrete Shell	Shielding/ Structural	B
				13	Structure Weldment	Shielding/ Structural	B

Table 2.3-1 Safety Classification of Universal Storage System Components (Continued)

Drawing No.	Sheet No.	Description	Rev No.	Item No.	Component	Function	Safety Class
790-562	1	(Continued)		12	Base Weldment	Shielding/ Structural	B
				1-11	Reinforcing Bar	Structural	B
790-570	1	BWR Fuel Basket	3	24	Heat Transfer Disk	Heat Transfer	A
				23	Flat Washer	Structural	C
				22	Split Spacer	Structural	A
				21	Top Spacer	Structural	A
				13-20	Tube	Structural	A
				11-12	Tie Rod	Structural	A
				10	Top Nut	Structural	A
				8-9	Tube (1-Sided)	Structural	A
				7	Spacer	Structural	A
				5-6	Tube (2-Sided)	Structural	A
				4	Drain Tube Sleeve	Operations	C
				3	Support Disk	Structural	A
				2	Top Weldment	Structural	A
				1	Bottom Weldment	Structural	A
790-575	1	BWR Fuel Tube	4	7	Flange	Structural	A
				5-6	Cladding	Criticality Control	A
				3-4	BORAL	Criticality Control	A
				1-2	Tubing	Structural	A
790-581	1	PWR Fuel Tube	5	10	Flange	Structural	A
				7-9	Cladding	Criticality Control	A
				4-6	BORAL	Criticality Control	A
				1-3	Tubing	Structural	A

Table 2.3-1 Safety Classification of Universal Storage System Components (Continued)

Drawing No.	Sheet No.	Description	Rev No.	Item No.	Component	Function	Safety Class
790-582	1	Canister, Shell	6	7	Location Lug	Operations	C
				6	Bottom	Structural/ Confinement	A
790-584	1	Canister Details	9	2	Nipple	Operations	C
790-585	1	Transportable Storage Canister	7	21	Key	Operations	C
				20	Backing Ring	Structural	C
				19	Structural Lid	Structural	A
				18	Cover	Confinement/ Operations	B
				17	Shield Lid Assembly	Shielding	B
				16	Lid Support Ring	Structural	B
				11-15	Drain Tube Assembly	Operations	C
790-590	1	Loaded Vertical Concrete Cask	1	15	Cover	Operations	B
				14	Washer (Lid Bolt)	Operations	NQ
				13	Lid Bolt	Operations	B
				12	Cask Lid	Operations	B
				11	Shield Plug	Shielding	B
790-595	1	PWR Fuel Basket	4	19-20	Top Nut	Structural	A
				17-18	Top Weldment	Structural	A
				16	Top Spacer	Structural	A
				14-15	Tube	Structural	A
				12-13	Tie Rod	Structural	A
				11	Heat Transfer Disk	Heat Transfer	A
				10	Tie Rod	Structural	A

Table 2.3-1 Safety Classification of Universal Storage System Components (Continued)

Drawing No.	Sheet No.	Description	Rev No.	Item No.	Component	Function	Safety Class
790-595	1	(Continued)		9	Top Nut	Structural	A
				8	Flat Washer	Structural	C
				7	Split Spacer	Structural	A
				6	Spacer	Structural	A
				4-5	Drain Tube Sleeve/Tube	Operations	C
				3	Support Disk	Structural	A
				2	Top Weldment	Structural	A
				1	Bottom Weldment	Structural	A
790-605	1	BWR Fuel Tube, Over-Sized	5	7	Flange	Structural	A
				5-6	Cladding	Criticality Control	A
				3-4	Neutron Poison	Criticality Control	A
				1-2	Tubing	Structural	A
790-560	1	Assembly, Transfer Cask	7	43	Shielding Ring	Shielding	B
				42	Transfer Adapter SHCS	Shielding	B
				41	Transfer Cask Extension	Shielding	B
				39	Connector	Operations	C
				38	Retaining Ring Bolt	Operations	B
				37	Scuff Plate	Operations	NQ
				36	Gamma Shield Brick	Shielding	B
				35	Fill/ Drain Line	Operations	C
				33-34	NS Cover Plate	Operations	B
				28-32	NS Boundary	Structural	B
				26-27	Bottom Plate	Structural	B
				21-25	Heat Transfer Angle	Thermal	B
				20	Retaining Ring	Operations	B

Table 2.3-1 Safety Classification of Universal Storage System Components (Continued)

Drawing No.	Sheet No.	Description	Rev No.	Item No.	Component	Function	Safety Class
790-560	1	(Continued)	7	19	Door Lock Bolt	Operations	C
				17-18	Shield Door	Structural/Shielding	B
				16	Door Rail	Operations	B
				15	Top Plate	Structural	B
				14	Neutron Shield	Shielding	B
				13	Trunnion Cap	Operations	C
				12	Trunnion	Structural	B
				7-11	Outer Shell	Structural	B
				2-6	Inner Shell	Structural	B
				1	Bottom Plate	Structural	B
412-502	1	Maine Yankee (MY) Fuel Can Details, NAC-UMS®	2	13	Support Ring	Structural/Operations	A
				12	Lift Tee	Structural/Operations	B
				10	Tube Body	Structural/Criticality	A
				9	Side Plate	Structural/Criticality	A
				8	Bottom Plate	Structural/Criticality	A
				7	Backing Screen	Operations	C
				6	Filter Screen	Confinement	B
				4	Wiper	Operations	C
				3	Lid Guide	Operations	C
				2	Lid Plate	Structural/Criticality	A
				1	Lid Collar	Confinement	A

2.4 Decommissioning Considerations

The principal elements of the Universal Storage System are the Vertical Concrete Cask and the Transportable Storage Canister.

The concrete cask provides biological shielding and physical protection for the contents of the canister during long-term storage. The concrete that provides biological shielding is not expected to become contaminated during the period of use, as it does not come into contact with other contaminated objects or surfaces. The concrete cask is not expected to become surface contaminated during use, except through incidental contact with other contaminated surfaces. Incidental contact could occur at the interior surface (liner) of the concrete cask, the top surface that supports the transfer cask during loading and unloading operations, and the base plate of the concrete cask that supports the canister. All of these surfaces are made of carbon steel, and it is anticipated that these surfaces could be decontaminated as necessary for decommissioning.

Activation of the carbon steel liner, concrete, support plates, and reinforcing bar could occur due to neutron flux from the stored fuel. Since the neutron flux rate is low, only minimal activation of carbon steel in the concrete cask is expected to occur. The activity concentrations from activation of storage cask components are listed in Tables 2.4-1 through 2.4-4. Tables 2.4-1 and 2.4-2 provide the activation summaries of the concrete cask and canister for the design-basis PWR fuel, while Tables 2.4-3 and 2.4-4 provide the summaries for the design-basis BWR fuel. These tables include the radiologically significant isotopes, together with a total concentration of all activated nuclides in the respective component. The total concentrations listed include activities of radionuclides, which do not have any substantial contribution to radiation dose and are not specifically identified by 10 CFR 61 waste classification. In particular, the isotope contributing the majority of the carbon steel total curie activity is ^{55}Fe, which decays by electron capture and is not of radiological concern.

Decommissioning of the concrete cask will involve the removal of the canister, and the subsequent disassembly of the concrete cask. It is expected that the concrete will be broken up, and steel components segmented, to reduce volume. Any contaminated or activated items are expected to qualify for near-surface disposal as low specific activity material. The activity concentrations from activation of concrete cask components resulting from the design basis PWR and BWR fuel assemblies are listed in Tables 2.4-1 and 2.4-3, respectively.

The Transportable Storage Canister is designed and fabricated to be suitable for use as part of the waste package for permanent disposal in a deep Mined Geological Disposal System (i.e., it meets the requirements of the DOE MPC Design Procurement Specification [20]. The canister is fabricated from materials having high long-term corrosion resistance, and it contains no paints or coatings that could adversely affect its permanent disposal. Consequently, decommissioning of the canister will occur only if the fuel contained in the canister had to be removed, or if current requirements for disposal were to change. Decommissioning of the canister will require that the closure welds at the canister structural lid, shield lid, and shield lid port covers be cut, so that the spent fuel can be removed. Removal of the contents of the canister will require that the canister be returned to a spent fuel pool or dry unloading facility, such as a hot cell. Closure welds can be cut either manually or with automated equipment, with the procedure being essentially the reverse of that used to initially close the canister.

Following removal of its contents, the canister interior is expected to have significant contamination, and the bottom of the canister may contain "crud" or other residual material. Some effort may be required to remove the surface contamination prior to disposal; however, in practice, it will not be absolutely necessary to decontaminate the canister internals. Since the canister internal contamination will consist only of by-product materials, any contaminated canister and internal components are expected to qualify for near-surface disposal as low specific activity waste without internal contamination. Any required internal decontamination is facilitated, should it become necessary, by the smooth surfaces of the canister and the basket, and by the design that precludes the presence of crud traps. Since the neutron flux rate from the stored fuel is low, only minimal activation of the canister is expected to occur. The activity concentrations from activation of canister components resulting from the design basis PWR and BWR fuel assemblies are listed in Tables 2.4-2 and 2.4-4, respectively.

The unloaded canister can also qualify as a strong, tight container for other waste. In this case, the canister can be filled, within weight limits, with other qualified waste, closed, and transported to a near-surface disposal site. Use of the canister for this purpose can reduce decommissioning costs by avoiding decontamination, segmenting, and repackaging.

The storage pad, fence, and supporting utility fixtures are not expected to require decontamination as a result of use of the Universal Storage System. The design of the cask and canister precludes the release of contamination from the contents over the period of use of the system. Consequently, these items may be reused or disposed of as locally generated clean waste.

Table 2.4-1 Activity Concentration Summary for the Concrete Cask - PWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Concrete Shell	Shell Liner	Shield Plug	Lid	Cover Plate	Bottom	Base Plate
^{14}C	--	--	2.35E-08	--	--	--	--
^{45}Ca	4.62E-06	--	--	--	--	--	--
^{54}Mn	5.13E-08	6.97E-02	1.34E-03	1.63E-04	3.17E-06	5.56E-02	1.88E-02
^{55}Fe	2.30E-05	1.22E+00	2.12E-01	5.49E-02	3.85E-05	7.15E-01	2.27E-01
^{60}Co	1.95E-06	3.43E-04	7.22E-05	1.38E-05	1.54E-05	2.71E-04	8.58E-05
^{63}Ni	--	--	--	--	2.02E-02	--	--
Total	3.09E-05	1.30E+00	2.15E-01	5.54E-02	2.06E-02	7.77E-01	2.48E-01

1. 40-year activation, 1-week cooling.

Table 2.4-2 Activity Concentration Summary for the Canister – PWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Wall	Shield Lid	Structural Lid	Bottom
^{54}Mn	9.94E-05	3.32E-04	4.42E-06	1.00E-04
^{55}Fe	7.94E-04	8.26E-04	3.67E-04	1.05E-03
^{60}Co	3.15E-04	3.31E-04	1.47E-04	4.22E-04
^{59}Ni	3.54E-07	3.67E-07	1.64E-07	4.66E-07
^{63}Ni	4.17E-01	4.33E-01	1.93E-01	5.49E-01
Total	4.27E-01	4.43E-01[2]	1.97E-01[2]	5.63E-01[2]

1. 40-year activation, 1 -week cooling.
2. ^{32}P accounts for most of the unlisted total activity.

Table 2.4-3 Activity Concentration Summary for the Concrete Cask – BWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Concrete Shell	Shell Liner	Shield Plug	Lid	Cover Plate	Bottom	Base Plate
^{14}C	--	--	3.57E-08	--	--	--	--
^{45}Ca	7.91E-06	--	--	--	--	--	--
^{54}Mn	7.74E-08	1.07E-01	1.97E-03	2.39E-04	1.37E-06	7.06E-02	2.40E-02
^{55}Fe	3.93E-05	2.10E00	3.23E-01	8.29E-02	2.08E-05	1.13E-04	3.52E-01
^{60}Co	3.33E-06	5.93E-04	1.10E-04	2.08E-05	8.35E-06	4.26E-04	1.33E-04
^{63}Ni	--	--	--	--	1.09E-02	--	--
Total	5.28E-05	2.22E00	3.27E-01	8.37E-02	1.12E-02	1.21E00	3.79E-01

1. 40-year activation, 1-week cooling.

Table 2.4-4 Activity Concentration Summary for the Canister – BWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Wall	Shield Lid	Structural Lid	Bottom
^{54}Mn	1.53E-04	4.89E-05	6.51E-06	1.26E-04
^{55}Fe	1.39E-03	1.26E-06	5.57E-04	1.68E-03
^{60}Co	5.52E-04	5.04E-04	2.22E-04	6.73E-04
^{59}Ni	6.21E-07	5.60E-07	2.48E-07	7.46E-07
^{63}Ni	7.31E-01	6.60E-01	2.92E-01	8.79E-01
Total	7.49E-01	6.76E-01	2.99E-01	9.00E-01

1. 40-year activation, 1-week cooling.

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