EA790-FSAR-002

DOCKET No. 72-1015

UMS®

UNIVERSAL MPC SYSTEM®

FINAL SAFETY ANALYSIS REPORT

for the

UMS® Universal Storage System

NOVEMBER 2002 REVISION UMSS-02F

SUBMITTAL

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Chapter 1

Table 1-1 Terminology

Universal Storage System	The storage component of the Universal MPC System (UMS®) designed by NAC for the storage and transportation of spent nuclear fuel.
Universal Transport Cask	The packaging consisting of a Universal Transport Cask body with a closure lid and energy-absorbing impact limiters. The Universal Transport Cask is used to transport a Transportable Storage Canister containing spent fuel. The cask body provides the primary containment boundary during transport.
Confinement System	The components of the Transportable Storage Canister intended to retain the radioactive material during storage.
Contents	Twenty-four PWR fuel assemblies, or fifty-six BWR fuel assemblies. The fuel assemblies may be configured as Site Specific Fuel. The fuel assemblies are contained in a Transportable Storage Canister.
Standard Fuel	Irradiated fuel assemblies with a burnup less than, or equal to, 45,000 MWD/MTU and having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For PWR fuel, a flow mixer, an in-core instrument thimble or a burnable poison rod insert is considered to be a component of standard fuel. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the standard fuel configuration.
Consolidated Fuel	A nonstandard fuel configuration in which the individual intact fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is dimensionally similar to a fuel assembly. Consolidated Fuel is stored in a Maine Yankee Fuel Can.
Intact Fuel (Assembly or Rod) (Undamaged Fuel)	A fuel assembly or fuel rod with no fuel rod cladding defects, or with known or suspected fuel rod cladding defects not greater than pinhole leaks or hairline cracks.

Table 1-1 Terminology (Continued)

Damaged Fuel (Failed Fuel)	A fuel assembly or fuel rod with known or suspected cladding defects greater than pinhole leaks or hairline cracks. Damaged Fuel must be placed in a Maine Yankee Fuel Can.
High Burnup Fuel	A fuel assembly having a burnup between 45,000 and 50,000 MWD/MTU, which must be preferentially loaded in peripheral positions of the basket.

Table 1-1 Terminology (Continued)

- Tie Rod	A stainless steel rod used to align, retain, and support the support disks and the heat transfer disks in the fuel basket structure. The tie rods extend from the top weldment to the bottom weldment.
- Spacer	Installed on the tie rod between the support disks (BWR only) or between the support disks and top and bottom weldments (BWR and PWR) to properly position the disks and provide axial support for the support disks.
- Split Spacer	Spacers installed on the tie rod between the support disks and the heat transfer disks to properly position the disks and provide axial support for the support disks and the heat transfer disks.
Vertical Concrete Cask (Concrete Cask)	A concrete cylinder that contains the Transportable Storage Canister during storage. The Vertical Concrete Cask is formed around a steel inner liner and base and is closed by a shield plug and lid.
- Shield Plug	A thick carbon steel plug, which also contains a neutron shield material, installed in the top end of the Vertical Concrete Cask to reduce skyshine radiation.
- Lid	A thick carbon steel plate that serves as the bolted closure for the Vertical Concrete Cask. The lid precludes access to the canister and provides additional radiation shielding.
- Liner	A thick carbon steel shell that forms the annulus of the concrete cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the air inlets, the concrete cask jacking points and the pedestal that supports the canister inside of the concrete cask.

Table 1-1 Terminology (Continued)

Transfer Cask	A shielded lifting device for handling of the Transportable Storage Canister during loading of spent fuel, canister closure operations, and transfer of the canister into or out of the Vertical Concrete Cask during storage, or into or out of the Universal Transport Cask during transportation. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks. The transfer cask is provided in either the standard or the advanced configuration. The advanced configuration has a higher weight capacity.
- Transfer Cask Lifting Trunnions	Four low alloy steel trunnions used to lift and move the transfer cask in a vertical orientation.
Transfer Adapter	A carbon steel plate assembly that is positioned on to the top of the transport or concrete cask to facilitate installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
NS-4-FR or NS-3	Solid hydrogenous materials with neutron absorption capabilities.
Air Pad Rig Set (Air Pallet)	A device used to lift the Vertical Concrete Cask by using high volume air.
Heavy Haul Trailer	The trailer used to transport the empty or loaded Vertical Concrete Cask.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined, and is defined as the allowable stress or displacement of a material divided by its actual (calculated) value.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Minimum Lifetime	10 CFR Part 72 requires an analysis and evaluation of DCSS thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years.	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits.
2. Spent Fuel Cladding Protection	The spent fuel cladding must be protected against degradation that may lead to gross ruptures.	Tables 4.1-4 and 4.1-5 demonstrate that the fuel cladding temperatures are maintained within allowable limits.
3. Thermal Structures, Systems, and Components	Thermal structures, systems, and components important to safety must be described in sufficient detail to permit evaluation of their effectiveness. Applicable thermal requirements are identified, in part, in 10 CFR 72.24©(3), 72.24(d), 72.122(h)(1), 72.122(l), 72.128(a)(4), 72.236(f), 72.236(g), and 72.236(h).	The discussion of the thermal design features of the system is presented in Section 4.1.
	10 CFR 72.24©(3) Contents of Application: Descriptions of Components Important to Safety 10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Tables 4.1-4 and 4.1-5 demonstrate that the temperatures of SSCs are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
	10 CFR 72.122(h)(1) Overall Requirements: Confinement Barriers and Systems 10 CFR 72.122(l) Overall Requirements: Retrievability	The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system.
	10 CFR 72.128(a)(4) Criteria for Spent Fuel Storage and Handling: Testable Heat Removal Capacity	As specified in Sections 9.1.7 and Appendix 12A, Section 3.1 6, the air temperature at the outlet vents is measured to ensure proper operation of the passive heat removal system.
	10 CFR 72.236(f) Specific Requirements for Spent Fuel Storage Cask Approval: Passive Heat Removal 10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits.
	10 CFR 72.236(h) Specific Requirements for Spent Fuel Storage Cask Approval: Wet/Dry Loading and Unloading Compatibility	The operating procedures for the system, presented in Chapter 8, include procedures for wet and dry loading and unloading operations. A discussion is provided for development of dry loading and unloading procedures for dry cask handling facilities.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Long-term Cladding Temperatures	Fuel cladding (zircaloy) temperature at the beginning of dry cask storage should generally be below the allowable temperature of 400°C (752°F) per ISG-11, Rev. 2.	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below allowable temperature limits for zircaloy clad fuel as determined in accordance with ISG 11, Rev. 2.
2. Short Term Cladding Temperatures	Fuel cladding temperature should generally be maintained below 570°C (1058°F) for short-term, off-normal and accident conditions (PNL 4835). For fuel transfer operations (e.g., vacuum drying of the cask or dry transfer), the temperature should generally be maintained below 400°C (752°F). (ISG-11, Rev 2)	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below 570°C (1058°F) for short-term, off-normal and accident conditions. For transfer operations, the fuel cladding temperatures are maintained below 400°C (752°F).
3. Maximum Internal Pressure	The maximum internal pressure of the cask should remain within its design pressures for normal, off-normal, and accident conditions assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.	The maximum normal condition pressure calculation is presented in Section 4.4.5. The accident condition pressure calculation is presented in Section 11.2.1. The off-normal condition is bounded by the accident condition, which assumes 100% failure of the cladding.
4. Maximum Material Temperatures	Cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions in order to enable components to perform their intended safety functions.	Tables 4.1-4 and 4.1-5 demonstrate that the temperatures are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
5. Fuel Cladding Protection	The spent fuel cladding is the primary structural component that is used to ensure that the spent fuel is contained in a known geometric configuration.	As concluded in ISG-11, Rev. 2, creep under normal conditions of storage will not cause gross rupture of the cladding, and the geometric configuration of the spent fuel will be preserved provided that the maximum cladding temperature does not exceed 400°C (752°F).
6. Long-Term Cladding Damage	Creep is the dominant mechanism for cladding deformation under normal conditions of storage. The relatively high temperatures, differential pressures, and corresponding hoop stress on the cladding will result in permanent creep deformation of the cladding over time	A temperature limit of 400°C (752°F) for normal conditions of storage and for short-term storage operations will limit cladding hoop stresses and creep and limit the amount of soluble hydrogen available to form radial hydrides. (ISG-11, Rev. 2)
7. Passive Cooling	The cask system should be passively cooled. [10 CFR 72.236(f)]	As stated in Sections 1.2 and 4.1, the system is passively cooled.
8. Thermal Operating Limits	The thermal performance of the cask should be within the allowable design criteria specified in SAR Section 2 (e.g , materials, decay heat specifications) and SAR Section 3 (e.g., thermal stress analysis) for normal, off-normal, and accident conditions.	The thermal stress analyses of the canister and Vertical Concrete Cask for normal conditions are provided in Sections 3.4.4.1.1 and 3.4.4.2.3, respectively. The system is evaluated for off-normal thermal loading in Section 11 1 2, and the system is analyzed for accident thermal loading in Sections 11.2 6, 11 2 7 and 11.2 13

24. American Society for Testing and Materials, "Specification for Fly Ash and Raw or Calcined Natural Pozzolan for Use as a Mineral Admixture in Portland Cement Concrete," ASTM C 618-87.

25. American Welding Society, "Structural Welding Code Steel," AWS D1.1-96, 1996.

26. American Society for Testing and Materials, "Standard Practice for Sampling Freshly Mixed Concrete," ASTM C 172-90, June 1990.

27. American Society for Testing and Materials, "Method of Making and Curing Concrete Test Specimens in the Field," ASTM C 31-88.

28. American Society for Testing and Materials, "Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens," ASTM C 39-94, January 1995.

29. Nuclear Regulatory Commission, "Cladding Considerations for the Transport and Storage of Spent Fuel," Interim Staff Guidance-11, Revision 2.

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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	ASME Code, Section III, Subsection NG [2] and NUREG/CR-6322 [3]
Vertical Concrete Cask	ACI-349 [4], ACI-318 [5]
Transfer Cask	ANSI N14.6 [6] and NUREG-0612 [7]
Maximum Weight:	
Canister with Design Basis PWR Fuel Assembly (dry, including inserts) (Class 2)	72,900 lbs.
Canister with Design Basis BWR Fuel (dry) (Class 5)	75,600 lbs.
Vertical Concrete Cask (loaded) (Class 5)	313,900 lbs.
Transfer Cask (Class 3)	121,500 lbs.
Thermal:	
Maximum Fuel Cladding Temperature:	
PWR Fuel	752°F (400°C) for Normal and Transfer [25] 1058°F (570°C) Off-Normal and Accident [21]
BWR Fuel	752°F (400°C) for Normal and Transfer [25] 1058°F (570°C) Off-Normal and Accident [21]
Ambient Temperature:	
Normal (average annual ambient)	76°F
Off-Normal (extreme cold; extreme hot)	-40°F; 106°F
Accident	133°F
Concrete Temperature:	
Normal Conditions	≤ 150°F (bulk); ≤ 300°F (local) [24]
Off-Normal/Accident Conditions	≤ 350°F local/ surface [4]
Cavity Atmosphere	Helium

2.1.1 PWR Fuel Evaluation

The parameters of the PWR fuel assemblies that may be loaded in the transportable storage canister (canister) are shown in Table 2.1.1-1. The maximum initial enrichment limit represents the maximum fuel rod enrichment limit for variably enriched PWR assemblies. Each canister may contain up to 24 intact PWR fuel assemblies.

The design of the Universal Storage System is based on certain reference fuel assemblies that maximize the source terms used for the shielding and criticality evaluation, and that maximize the weight used in the structural evaluation. These reference fuel assemblies are described in the chapters appropriate to the condition being evaluated. The principal characteristics and parameters of a reference fuel, such as fuel volume, initial enrichment, cool time and burnup, do not represent limiting or bounding values. Bounding values for a fuel class are established based primarily on how principal parameters are combined and on the loading conditions or restrictions established for a class of fuel based on its parameters.

The maximum decay heat load for the storage of all types of PWR fuel assemblies is 23.0 kW (0.958 kW/assembly), except in cases where preferential loading is employed.

The minimum cool time is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete and transfer casks and is presented in Section 5.5. PWR fuel must be loaded in accordance with Table 2.1.1-2.

Site specific fuel that does not meet the enrichment and burnup limits of this section and Table 2.1.1-1 is separately evaluated in Section 2.1.3 to establish loading limits.

Table 2.1.1-1 PWR Fuel Assembly Characteristics

Fuel Class[1, 2]	14 x 14	14 x 14	15 x 15	15 x 15	15 x 15	16 x 16	17 x 17
Fissile Isotopes	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Max Initial Enrichment (wt % ^{235}U)[3]	5.0	5.0	4.6	4.4	4.2	4.8	4.3
Max Initial Enrichment (wt % ^{235}U)[4]	5.0	5.0	5.0	5.0	5.0	5.0	5.0
Number of Fuel Rods	176	179	204	208	216	236	264
Number of Water Holes	5	17	21	17	9	5	25
Max Assembly Average Burnup (MWD/MTU)	45,000	45,000	45,000	45,000	45,000	45,000	45,000
Min Cool Time (years)	5	5	5	5	5	5	5
Min Average Enrichment (wt % ^{235}U)	1.9	1.9	1.9	1.9	1.9	1.9	1.9
Cladding Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Non-Fuel Hardware[5]	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR
Max Weight (lb) per Storage Location[6]	1,602	1,602	1,602	1,602	1,602	1,602	1,602
Max Decay Heat (Watts) per Storage Location[7]	958.3	958.3	958.3	958.3	958.3	958.3	958.3
Fuel Condition	Intact	Intact	Intact	Intact	Intact	Intact	Intact

General Notes:

1. Fuel, except Maine Yankee fuel, must be loaded in accordance with Table 2.1.1-2.
2. Maine Yankee fuel must be loaded in accordance with Tables 2.1.3.1-4 and 2.1.3.1-5, as appropriate.
3. Maximum initial enrichment without boron credit. Represents the maximum fuel rod enrichment for variably enriched assemblies Assemblies meeting this limit may contain a flow mixer (FM), an ICI thimble (T), or a burnable poison rod insert (BPR).
4. Maximum initial enrichment with taking credit for a minimum soluble boron concentration of 1000 ppm in the spent fuel pool water. Represents the maximum fuel rod enrichment for variably enriched assemblies. Assemblies meeting this limit may contain a flow mixer.
5. Assemblies may not contain control element assemblies, except as permitted for site specific fuel.
6. Weight includes the weight of non-fuel bearing components.
7. Maximum decay heat may be higher for site specific fuel configurations, which control fuel loading position.

Table 2.1.1-2 Loading Table for PWR Fuel

Minimum Initial Enrichment	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]				30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
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 < 3.9	5	5	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5	5	5

Minimum Initial Enrichment	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]				40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	6	9	14	13	9	13
2.3 ≤ E < 2.5	8	8	6	8	12	12	8	12
2.5 ≤ E < 2.7	8	7	6	7	11	11	7	11
2.7 ≤ E < 2.9	7	7	6	7	10	10	7	10
2.9 ≤ E < 3.1	7	6	6	7	9	9	7	9
3.1 ≤ E < 3.3	6	6	6	6	9	8	7	8
3.3 ≤ E < 3.5	6	6	6	6	8	8	7	8
3.5 ≤ E < 3.7	6	6	6	6	7	8	7	7
3.7 ≤ E < 3.9	6	6	6	6	7	8	7	7
3.9 ≤ E < 4.1	6	6	6	6	7	7	7	7
4.1 ≤ E < 4.3	5	6	6	6	6	7	7	7
4.3 ≤ E < 4.5	5	6	6	6	6	7	7	7
4.5 ≤ E < 4.7	5	6	5	6	6	7	6	7
4.7 ≤ E < 4.9	5	6	5	6	6	7	6	7
E ≥ 4.9	5	6	5	6	6	7	6	7

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2.1.2 BWR Fuel Evaluation

The parameters of the BWR fuel assemblies that may be loaded in the transportable storage canister (canister) are shown in Table 2.1.2-1. Each canister may contain up to 56 intact BWR fuel assemblies.

The design of the Universal Storage System is based on certain reference fuel assemblies that maximize the source terms used for the shielding and criticality evaluation, and that maximize the weight used in the structural evaluation. These reference fuel assemblies are described in the chapters appropriate to the condition being evaluated. The principal characteristics and parameters of a reference fuel, such as fuel volume, initial enrichment, cool time and burnup, do not represent limiting or bounding values. Bounding values for a fuel class are established based primarily on how principal parameters are combined and on the loading conditions or restrictions established for a class of fuel based on its parameters.

The maximum canister decay heat load for the storage of all types of BWR fuel assemblies is 23.0 kW (0.411 kW/assembly).

The minimum cooling time determination is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete and transfer casks and is presented in Section 5.5. BWR fuel must be loaded in accordance with Table 2.1.2-2.

Table 2.1.2-1 BWR Fuel Assembly Characteristics

Fuel Class[1]	**7 × 7**	**7 × 7**	**8 × 8**	**8 × 8**	**8 × 8**	**9 × 9**	**9 × 9**
Fissile Isotopes	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Max Initial Enrichment (wt % ^{235}U)[1]	4.5	4.7	4.5	4.7	4.8	4.5	4.6
Number of Fuel Rods	48	49	60	62	63	74	79
Number of Water Holes	1[4]	0	1/4[5]	2	4	2/7[5]	2
Max Assembly Average Burnup (MWD/MTU)	45,000	45,000	45,000	45,000	45,000	45,000	45,000
Min Cool Time (years)	5	5	5	5	5	5	5
Min Average Enrichment (wt % ^{235}U)	1.9	1.9	1.9	1.9	1.9	1.9	1.9
Cladding Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Nonfuel Hardware[2]	Channel	Channel	Channel	Channel	Channel	Channel	Channel
Max Channel Thickness (mil)	120	120	120	120	120	120	120
Max Weight (lb) per Storage Location[3]	702	702	702	702	702	702	702
Max Decay Heat (Watts) per Storage Location	410.7	410.7	410.7	410.7	410.7	410.7	410.7
Fuel Condition	Intact	Intact	Intact	Intact	Intact	Intact	Intact

General Notes:

1. Fuel must be loaded in accordance with Table 2.1.2-2.
2. Each BWR fuel assembly may have a Zircaloy channel or be unchanneled, but cannot have a stainless steel channel
3. Weight includes the weight of the channel.
4. Solid fill or water rod.
5. Water rods may occupy more than one fuel lattice location.

The fuel can is sized to accommodate a fuel assembly and must be loaded 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. Slots 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

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.

Fuel with missing fuel rods, fuel with fuel rods that have been replaced by rods of other material, consolidated fuel lattices and damaged fuel are preferentially loaded in corner positions of the basket, numbered 3, 6, 19 and 22 in Figure 2.1.3.1-1. The requirements for preferential loading

schemes using the corner positions result primarily from shielding or criticality evaluations of the designated fuel configurations.

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 basket locations, which are the outer 12 fuel loading positions shown in Figure 2.1.3.1-1. Locating the high burnup fuel in the peripheral basket locations reduces the maximum temperatures of these assemblies.

High burnup fuel (45,000 – 50,000 MWD/MTU) may be loaded as intact fuel provided that ISG-11, Rev. 2 temperature limits are met. The 752°F (400°C) ISG-11, Rev. 2 fuel temperature limit is met as shown in Table 4.1-4.

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.

Fuel assemblies with a startup source in the center guide tube position must be loaded in one of the basket corner positions. A fuel assembly may not hold more than one startup source.

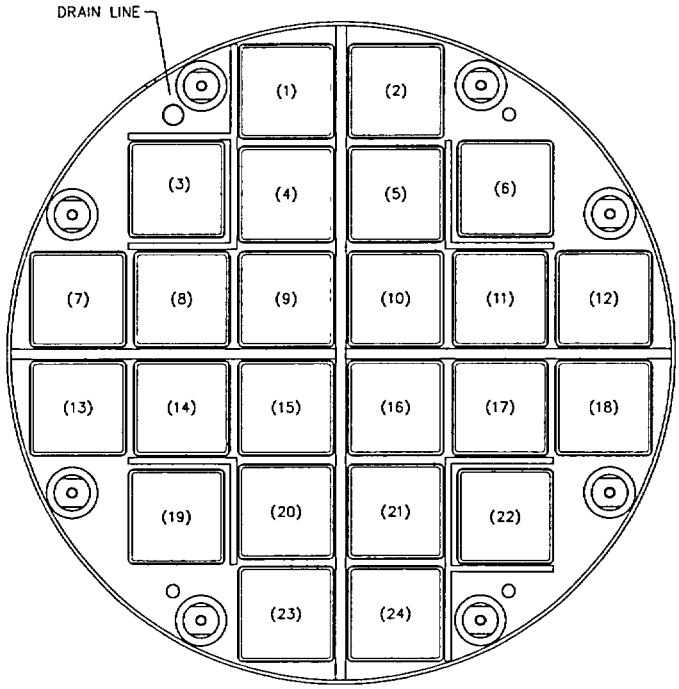
The loading position of fuel assemblies holding the CEA finger tips and/or the ICI segment in a fuel assembly corner guide tube position is not controlled; however, these fuel assemblies must have a CEA flow plug to ensure these items are captured within the guide tube(s).

2.1.3.1.7 Maine Yankee High Burnup Fuel

The storage system has been evaluated to allow the storage of up to 24 fuel rods, in one or more fuel assemblies, classified as damaged (failed) that are not stored in the Maine Yankee fuel can. Subject to the preferential loading controls, this fuel may be stored in any basket periphery fuel position as intact fuel. The 24 fuel rods classified as damaged may be in a single fuel assembly, or be individual fuel rods in up to 12 fuel assemblies (the number of peripheral fuel positions). The damaged fuel classification may arise from the existence of greater than hairline cracks, pinhole leaks, or the cladding oxide layer thickness for 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 the 12 peripheral fuel loading positions 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.

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



Note: Locations numbered 3, 6, 19 and 22 are corner positions.
Locations numbered 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23 and 24 are periphery positions.
Locations numbered 4, 5, 8, 11, 14, 17, 20 and 21 are intermediate positions.
Locations numbered 9, 10, 15 and 16 are center positions.

Table 2.1.3.1-1 Maine Yankee Site Specific Fuel Population

Site Specific Spent Fuel Configurations[1]	Est. Number of Assemblies[2]
Standard Fuel	1,434
Inserted Control Element Assembly (CEA)	168
Inserted In-Core Instrument (ICI) Thimble	138
Consolidated Fuel	2
Fuel Rod Replaced by Rod Enriched to 1.95 wt %	3
Fuel Rod Replaced by Stainless Steel Rod or Zircaloy Rod	18
Fuel Rods Removed	10
Variable Enrichment	72
Variable Enrichment and Axial Blanket	68
Burnable Poison Rod Replaced by Hollow Zircaloy Rod	80
Damaged Fuel in Maine Yankee Fuel Can	12
Burnup between 45,000 and 50,000 MWD/MTU	90
Maine Yankee Fuel Can	As Required
Inserted Startup Source	5
Inserted CEA Fingertips or ICI String Segment	1

1. The loading of the site specific fuel is controlled by the requirement of Section 12B2 of the Technical Specifications presented in Chapter 12.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.

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 (250×250) 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 × 8.52
Outside Cross Section (in.) (1)	8.62 × 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 × 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.

Table 2.1.3.1-4 Loading Table for Maine Yankee Fuel without Nonfuel Material

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	**30 < Burnup ≤ 35 GWD/MTU – Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2 9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3 3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3 7	5	5	5
3 7 ≤ E ≤ 4.2	5	5	5
Enrichment	**35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	7	7	5
2.1 ≤ E < 2.3	6	6	5
2.3 ≤ E < 2.5	6	6	5
2.5 ≤ E < 2.7	5	6	5
2 7 ≤ E < 2.9	5	6	5
2.9 ≤ E < 3.1	5	6	5
3.1 ≤ E < 3.3	5	6	5
3.3 ≤ E < 3 5	5	6	5
3.5 ≤ E < 3.7	5	6	5
3 7 ≤ E ≤ 4.2	5	6	5

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern: interior basket locations; allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern: periphery basket locations; allowable heat decay = 1 05 kW per assembly

Table 2.1.3.1-4 Loading Table for Maine Yankee Fuel without Nonfuel Material (continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for[1]		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	11	11	6
2.1 ≤ E < 2.3	9	9	6
2.3 ≤ E < 2.5	8	8	6
2.5 ≤ E < 2.7	7	7	6
2.7 ≤ E < 2.9	7	7	6
2.9 ≤ E < 3.1	6	7	6
3.1 ≤ E < 3.3	6	7	5
3.3 ≤ E < 3.5	6	7	5
3.5 ≤ E < 3.7	6	7	5
3.7 ≤ E ≤ 4.2	6	7	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for[1]**		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	Not allowed	Not allowed	7
2.1 ≤ E < 2.3	Not allowed	Not allowed	7
2.3 ≤ E < 2.5	Not allowed	Not allowed	7
2.5 ≤ E < 2.7	Not allowed	Not allowed	7
2.7 ≤ E < 2.9	Not allowed	Not allowed	7
2.9 ≤ E < 3.1	Not allowed	Not allowed	7
3.1 ≤ E < 3.3	Not allowed	Not allowed	7
3.3 ≤ E < 3.5	Not allowed	Not allowed	6
3.5 ≤ E < 3.7	Not allowed	Not allowed	6
3.7 ≤ E ≤ 4.2	Not allowed	Not allowed	6

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern: interior basket locations; allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern: periphery basket locations; allowable heat decay = 1.05 kW per assembly

Table 2.1.3.1-5 Loading Table for Maine Yankee Fuel Containing a CEA

Enrichment	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	6	5	5	5
2.7 ≤ E < 2.9	5	6	5	5	5
2.9 ≤ E < 3.1	5	6	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	7	7	7	7	7
2.7 ≤ E < 2.9	7	7	7	7	7
2.9 ≤ E < 3.1	6	6	6	6	6
3.1 ≤ E < 3.3	6	6	6	6	6
3.3 ≤ E < 3.5	6	6	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6

Note: The No CEA (Class 2) column is provided for comparison. Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

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 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 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 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:

$$p_f = 0.7C_eC_tIp_g$$

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, $p_f = (0.7)\ (1.0)\ (1.2)\ (1.2)\ (100\ ) \quad = 100.8$ 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

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 is provided in the standard and advanced configurations. The 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. Both lifting yokes are proof load tested to 300% of design load when fabricated. The lifting yokes have 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 standard and advanced transfer casks both have two 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% of the maximum calculated service load. 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% of the maximum calculated service load. 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.

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 standard and advanced 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.

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-570	BWR Fuel Basket	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
790-571	Bottom Weldment, BWR Fuel Basket	3	Support	Structural	A
		2	Pad	Structural	A
		1	Plate	Structural	A
		3-5	Support	Structural	A
		2	Ring	Structural	A
		1	Plate	Structural	A
790-572	Top Weldment, BWR Fuel Basket	6	Baffle	Structural	A

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-573	Support Disk and BWR Basket Details	8	Split Spacer	Structural	A
		7	Top Spacer	Structural	A
		5, 6	Tie Rod	Structural	A
		4	Top Nut	Structural	A
		3	Spacer	Structural	A
		1	Support Disk	Structural	A
790-574	Heat Transfer Disk, BWR		Heat Transfer Disk	Thermal	A
790-575	BWR Fuel Tube	7	Flange	Structural	A
		5-6	Cladding	Criticality Control	A
		3-4	Neutron Absorber	Criticality Control	A
		1-2	Tubing	Structural	A
790-581	PWR Fuel Tube	10	Flange	Structural	A
		7-9	Cladding	Criticality Control	A
		4-6	Neutron Absorber	Criticality Control	A
		1-3	Tubing	Structural	A

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-582	Canister, Shell	7	Location Lug	Operations	C
		6	Bottom	Structural/Confinement	A
		1-5	Shell	Structural/Confinement	A
790-583	Drain Tube Assembly	7	Metal Boss Seal	Operations	C
		2-6	Tube	Operations	C
		1	Nipple	Operations	C
790-584	Canister Details	8	Key	Operations	C
		7	Spacer Ring	Structural	C
		6	Lid Support Ring	Structural	B
		5	Cover	Confinement/Operations	B
		4	Structural Lid	Structural	A
		3	Metal Boss Seal	Operations	C
		2	Nipple	Operations	C
		1	Shield Lid	Shielding/Confinement	B
790-585	Transportable Storage Canister	24	Dowel Pin	Operations	NQ
		23	Structural Lid Plug	Operations	NQ
		22	Shield Lid Plug	Operations	NQ
		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

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-587	Spacer Shim, Canister	1-6	Spacer Shims #1 - #6	Operations	C
790-590	Loaded Vertical Concrete Cask	19	Tab	Operations	NQ
		18	Seal Wire	Operations	C
		17	Security Seal	Operations	C
		16	Seal Tape	Operations	NQ
		15	Cover	Operations	C
		14	Washer (Lid Bolt)	Operations	NQ
		13	Lid Bolt	Operations	B
		12	Cask Lid	Operations	B
		11	Shield Plug	Shielding	B
790-591	Bottom Weldment, PWR Basket	5, 6	Support	Structural	A
		4	Pad	Structural	A
		2, 3	Support	Structural	A
		1	Bottom Disk	Structural	A
790-592	Top Weldment, PWR Basket	7	Baffle	Structural	A
		3-6	Support	Structural	A
		2	Ring	Structural	A
		1	Top Disk	Structural	A
790-593	Support Disk and Details, PWR	8	Top Spacer	Structural	A
		5-7	Tie Rod	Structural	A
		4, 9, 10	Top Nut	Structural	A
		3	Spacer	Structural	A
		2	Split Spacer	Structural	A
		1	Support Disk	Structural	A

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-594	Heat Transfer Disk, PWR	1	Heat Transfer Disk	Thermal	A
790-595	PWR Fuel Basket	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
		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	BWR Fuel Tube, Over-Sized	7	Flange	Structural	A
		5-6	Cladding	Criticality Control	A
		3-4	Neutron Absorber	Criticality Control	A
		1-2	Tubing	Structural	A

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

Drawing No.	Description	Item No.	Component	Function	Safety Class
790-613	Supplemental Shielding, VCC Inlets	4	Shims	Operations	NQ
		3	Paint	Operations	NQ
		2	Pipe	Shielding	B
		1	Side Plate	Shielding	B
790-617	Door Stop	6	Attachment Screw	Operations	NQ
		5	Lock Pin	Operations	NQ
		4	Handle	Operations	NQ
		3	Back Plate	Operations	NQ
		2	Top Plate	Operations	NQ
		1	Bottom Plate	Operations	NQ
412-502	Maine Yankee (MY) Fuel Can Details, NAC-UMS®	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

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Chapter 3

Figure 3.4.3-3 (Deleted)

3.4.3.1 Vertical Concrete Cask Lift Evaluation

The vertical concrete cask may be lifted and moved using an air pad system under the base of the cask or four lifting lugs provided at the top of the cask.

Lifting jacks installed at jacking points in the air inlet channels are used to raise the cask so that the air pads can be inserted under the cask. The lifting jacks use a synchronous lifting system to equally distribute the hydraulic pressure among four hydraulic jack cylinders. The calculated weight of the heaviest, loaded concrete cask to be lifted by the jacking system, the BWR Class 5 configuration, is 313,900 pounds with loaded canister and lids (center of gravity is measured from the bottom of the concrete cask). A bounding weight of 320,000 pounds is used for the evaluation in this section.

The lifting lugs are analyzed in accordance with ANSI N14.6 and ACI-349.

3.4.3.1.1 Bottom Lift By Hydraulic Jack

To ensure that the concrete bearing stress at the jack locations due to lifting the cask does not exceed the allowable stress, the area of the surface needed to adequately spread the load is determined in this section. The allowable bearing capacity of the concrete at each jack location is:

$$U_b = \phi f_c' A = \frac{(0.7)(4{,}000)\pi d^2}{4} = 2{,}199.1\ d^2,$$

where:

ϕ = 0.7 strength reduction factor for bearing,
f_c' = 4,000 psi concrete compressive strength,
$A = \frac{\pi d^2}{4}$, concrete bearing area (d = bearing area diameter).

The concrete bearing strength must be greater than the cask weight multiplied by a load reduction factor, $L_f = 1.4$.

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4.0 THERMAL EVALUATION

This section presents the thermal design and analyses of the Universal Storage System for normal conditions of storage of spent nuclear fuel. The analyses include consideration of design basis PWR and BWR fuel. Results of the analyses demonstrate that with the design basis contents, the Universal Storage System meets the thermal performance requirements of 10 CFR 72 [1].

4.1 Discussion

The Universal Storage System consists of a Transportable Storage Canister, Vertical Concrete Cask, and a transfer cask. In long-term storage, the canister is installed in the concrete cask, which provides passive radiation shielding and natural convection cooling. The fuel is loaded in a basket structure positioned within the canister. The transfer cask is used for the handling of the canister. The thermal performance of the concrete cask containing the design basis fuel (during storage) and the performance of the transfer cask containing design basis fuel (during handling) are evaluated herein.

The significant thermal design feature of the Vertical Concrete Cask is the passive convective air flow up along the side of the canister. 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 occurs from the canister shell to the concrete cask liner, which also transmits heat to the adjoining air flow. Conduction does not play a substantial role in heat removal from the canister surface. 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 UMS® Storage System design basis heat load is 23.0 kW for up to 24 PWR (0.958 kW per assembly) or up to 56 BWR (0.411 kW per assembly) fuel assemblies, except in cases where preferential loading patterns are employed.

The thermal evaluation considers normal, off-normal, and accident conditions of storage. Each of these conditions can be described in terms of the environmental temperature, use of solar insolation, and the condition of the air inlets and outlets, as shown in Table 4.1-1. The design conditions for transfer are defined in Table 4.1-2. The transfer conditions consider the transient effect for PWR and BWR fuel, starting from the removal of the transfer cask/canister from the spent fuel pool. The canister is considered under normal operation to be inside the transfer cask and initially filled with water. The canister is vacuum dried, back-filled with helium and then transferred into the Vertical Concrete Cask. As shown in Section 4.4.3, the time duration of the spent fuel in the water and vacuum conditions is administratively controlled to prevent general boiling of the water and to ensure that the allowable temperatures of the limiting components (fuel cladding, structural disks and heat transfer disks) are not exceeded.

This evaluation applies different component temperature limits and different material stress limits for long-term conditions and short-term conditions. Normal storage is considered to be a long-term condition. Off-normal and accident events, as well as the transfer condition that temporarily occurs during the preparation of the canister while it is in the transfer cask, are considered as short-term conditions. Thermal evaluations are performed for the design basis PWR and BWR fuels for all design conditions. The maximum allowable material temperatures for long-term and short-term conditions are provided in Table 4.1-3.

During normal conditions of storage and hypothetical accident conditions, the concrete cask must reject the fuel decay heat to the environment without exceeding the operational temperature ranges of the components important to safety. In addition, to maintain fuel rod integrity for normal conditions of storage the fuel must be maintained at a sufficiently low temperature in an inert atmosphere to preclude thermally induced fuel rod cladding deterioration. To preclude fuel degradation, the maximum allowable cladding temperature under normal conditions of storage and transfer for PWR fuel and BWR fuel is 752°F (400°C) in accordance with ISG-11, Rev 2. For either of these fuel types, the maximum cladding temperature under off-normal and accident conditions must remain below 1,058°F (570°C). Finally, for the structural components of the storage system, the thermally induced stresses, in combination with pressure and mechanical load stresses, must be below material allowable stress levels.

Thermal evaluations for normal conditions of storage and transfer (canister handling) condition operations are presented in Section 4.4. The finite element method is used to calculate the temperatures for the various components of the concrete cask, canister, basket, fuel cladding and transfer cask. Thermal models used in evaluation of normal and transfer conditions are described in Section 4.4.1.

A summary of the thermal evaluation results for the Universal Storage System are provided in Tables 4.1-4 and 4.1-5 for the PWR and BWR cases, respectively. Evaluation results for accident conditions of "All air inlets and outlets blocked" and "Fire" are presented in Chapter 11. The results demonstrate that the calculated temperatures are below the allowable component temperatures for all normal (long-term) storage conditions and for short-term events. The thermally induced stresses, combined with pressure and mechanical load stresses, are also within the allowable levels, as demonstrated in Chapter 3.

Table 4.1-1 Summary of Thermal Design Conditions for Storage

Condition[1]		Environmental Temperature (°F)	Solar Insolation[2]	Condition of Concrete Cask Inlets and Outlets
Normal		76	Yes	All inlets and outlets open
Off-Normal - Half Air Inlets Blocked		76	Yes	Half inlets blocked and all outlets open
Off-Normal - Severe Heat		106	Yes	All inlets and outlets open
Off-Normal - Severe Cold		-40	No	All inlets and outlets open
Accident - Extreme Heat		133	Yes	All inlets and outlets open
Accident - All Air Inlets and Outlets Blocked[3]		76	Yes	All inlets and outlets blocked
Accident - Fire[4]	During Fire	1475	Yes	All inlets and outlets open
	Before and After Fire	76	Yes	All inlets and outlets open

1. Off-normal and accident condition analyses are presented in Chapter 11.
2. Solar Insolation per 10 CFR 71:
   Curved Surface: 400 g cal/cm^2 (1475 Btu/ft^2) for a 12-hour period.
   Flat Horizontal Surface: 800 g cal/cm^2 (2950 Btu/ft^2) for a 12-hour period.
3. This condition bounds the case in which all inlets are blocked, with all outlets open.
4. The evaluated fire accident is the described in Section 11.2.6.

Table 4.1-2 Summary of Thermal Design Conditions for Transfer

Condition[1,2]	Maximum Duration (Hours)[3]	
	PWR	BWR
Canister Filled with Water[4]	17	17
Vacuum Drying	27	25
Canister Filled with Helium	20	16

(1) The canister is inside the transfer cask, with an ambient temperature of 76°F.

(2) See Section 8.4 for description of limiting conditions.

(3) Maximum durations based on 23 kW heat load.

(4) The initial water temperature is considered to be 100°F.

Table 4.1-3 Maximum Allowable Material Temperatures

Material	Temperature Limits (°F)		Reference
	Long Term	Short Term	
Concrete	150(B)/300(L)(1)	350	ACI-349 [4]
Fuel Clad PWR Fuel (5-year cooled) BWR Fuel (5-year cooled)	 752 752	 752/1,058(2) 752/1,058(2)	ISG-11 [38] and PNL-4835 [2]
Aluminum 6061-T651	650	750	MIL-HDBK-5G [7]
NS-4-FR	300	300	GESC [8]
Chemical Copper Lead	600	600	Baumeister [9]
SA693 17-4PH Type 630 Stainless Steel	650	800	ASME Code [13] ARMCO [11]
SA240 Type 304 Stainless Steel	800	800	ASME Code [13]
SA240 Type 304L Stainless Steel	800	800	ASME Code [13]
ASTM A533 Type B Carbon Steel	700	700	ASME Code [13]
ASME SA588 Carbon Steel	700	700	ASME Code Case N-71-17 [12]
ASTM A36 Carbon Steel	700	700	ASME Code Case N-71-17 [12]

(1) B and L refer to bulk temperatures and local temperatures, respectively. The local temperature allowable applies to a restricted region where the bulk temperature allowable may be exceeded.

(2) The temperature limit of the fuel cladding is 400°C (752°F) for storage (long-term) and transfer (short-term) conditions. The temperature limit of the fuel cladding is 570°C (1,058°F) for off-normal and accident (short-term) conditions.

Table 4.1-4 Summary of Thermal Evaluation Results for the Universal Storage System: PWR Fuel

Long-Term Condition:						
	Maximum Temperatures (°F)					
Design Condition	Concrete Bulk	Concrete Local	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Normal (76°F Ambient)	135	186	599	601	351	648
Allowable	150	300	650	650	800	752

Short-Term Condition:					
	Maximum Temperatures (°F)				
Design Condition	Concrete	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Off-Normal - Half Inlets Blocked (76°F Ambient)	191	600	603	350	649
Off-Normal - Severe Heat (106°F Ambient)	228	626	628	381	672
Off-Normal - Severe Cold (-40°F Ambient	17	502	505	226	561
Accident - Extreme Heat (133°F Ambient)	262	648	650	408	693
Accident - Fire	244	639	641	391	688
Allowable	350	750	800	800	1058
	Maximum Temperatures (°F)				
Transfer - Vacuum Drying	N/A	641	644	304	732
Transfer - Backfilled with Helium	N/A	680	683	455	732
Allowable	350	750	800	800	752

1. SA 693, 17-4PH Type 630 SS.
2. SA240, Type 304L SS (including canister shell, lid and bottom plate).

Table 4.1-5 Summary of Thermal Evaluation Results for the Universal Storage System: BWR Fuel

Long-Term Condition:						
	Maximum Temperatures (°F)					
Design Condition	Concrete Bulk	Concrete Local	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Normal (76°F Ambient)	136	192	612	614	376	642
Allowable	150	300	650	700	800	752

Short-Term Condition:					
	Maximum Temperatures (°F)				
Design Condition	Concrete	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Off-Normal - Half Inlets Blocked (76°F Ambient)	195	612	614	373	642
Off-Normal - Severe Heat (106°F Ambient)	231	638	640	405	667
Off-Normal - Severe Cold (-40°F Ambient)	20	504	505	252	540
Accident - Extreme Heat (133°F Ambient)	266	662	664	432	690
Accident - Fire	244	652	654	416	682
Allowable	350	750	700	800	1058
	Maximum Temperatures (°F)				
Transfer - Vacuum Drying	N/A	653	659	267	733
Transfer - Backfilled with Helium	N/A	683	686	462	733
Allowable	350	750	700	800	752

1. SA 533, Type B, CS.
2. SA240, Type 304L SS (including canister shell, lid and bottom plate).

Table 4.4.1.2-4 Effective Thermal Conductivities for BWR Fuel Tubes

	Conductivity	Temperature (°F)			
Tubes with Neutron Absorber	**(Btu/hr-in-°F)**	**200**	**400**	**600**	**800**
In CS disk region					
	Kxx	0.017	0.022	0.027	0.032
	Kyy	1.665	1.759	1.815	1.830
	Kzz	1.665	1.759	1.815	1.830
In AL disk region					
	Kxx	0.017	0.022	0.027	0.033
	Kyy	1.665	1.759	1.815	1.830
	Kzz	1.665	1.759	1.815	1.830
Tubes without Neutron Absorber		**200**	**400**	**600**	**800**
In CS disk region					
	Kxx	0.012	0.015	0.018	0.021
	Kyy	0.191	0.202	0.218	0.236
	Kzz	0.191	0.202	0.218	0.236
In AL disk region					
	Kxx	0.012	0.015	0.019	0.023
	Kyy	0.191	0.202	0.218	0.236
	Kzz	0.191	0.202	0.218	0.236

Note: Kxx is in the direction across the thickness of fuel tube wall.
Kyy is in the direction parallel to fuel tube wall.
Kzz is in the canister axial direction.

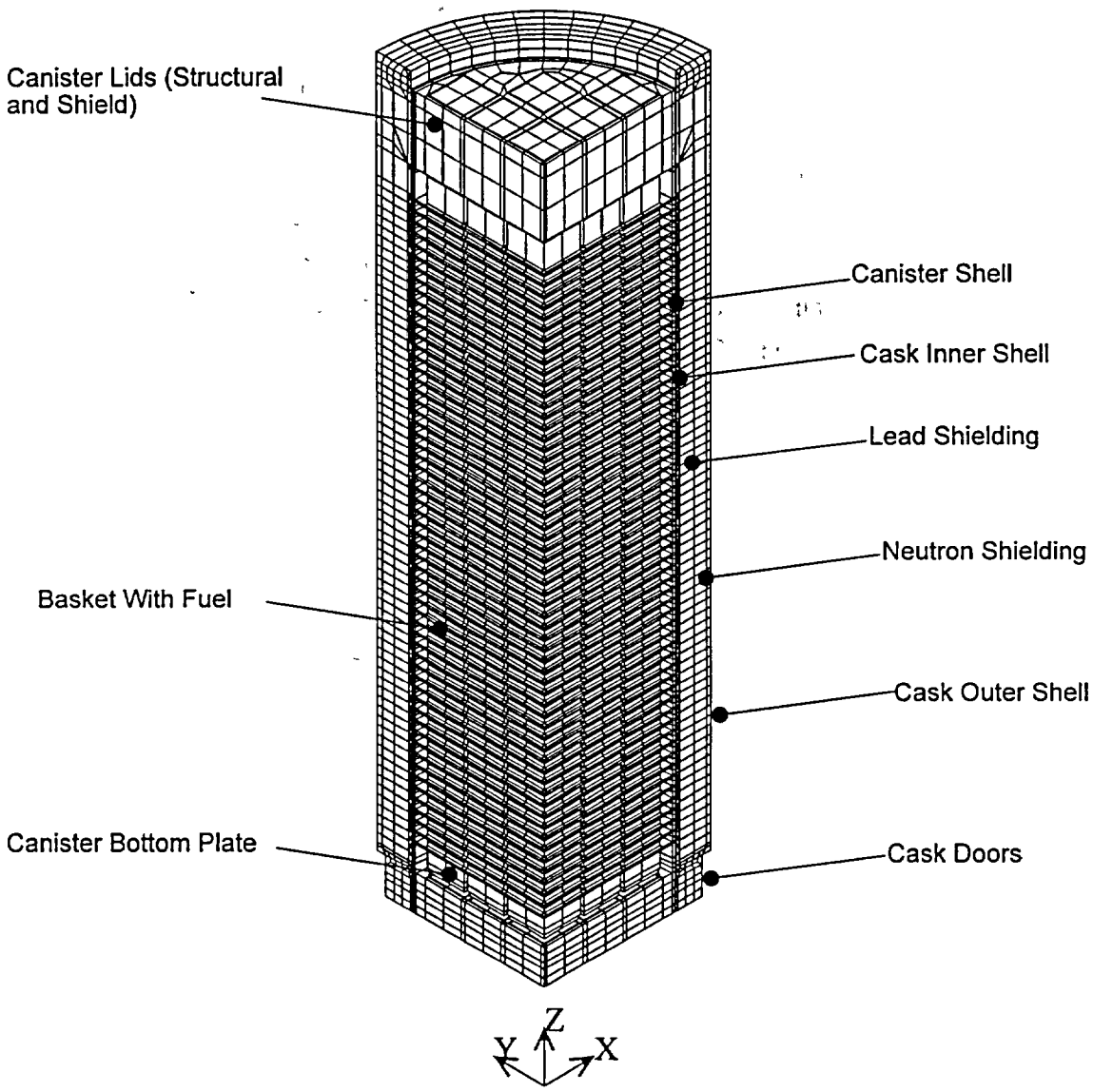
4.4.1.3 Three-Dimensional Transfer Cask and Canister Models

The three-dimensional quarter-symmetry transfer cask model is a representation of the PWR canister and transfer cask assembly. A half-symmetry model is used for the BWR canister and transfer cask. The model is used to perform a transient thermal analysis to determine the maximum water temperature in the canister for the period beginning immediately after removing the transfer cask and canister from the spent fuel pool. The model is also used to calculate the maximum temperature of the fuel cladding, the transfer cask and canister components during the vacuum drying condition and after the canister is back-filled with helium. The transfer cask is evaluated separately for PWR or BWR fuel using two models. For each fuel type, the class of fuel with the shortest associated canister and transfer cask is modeled in order to maximize the contents heat generation rate per unit volume and minimize the heat rejection from the external surfaces. The models for PWR and BWR fuel are shown in Figures 4.4.1.3-1 and 4.4.1.3-2, respectively. ANSYS SOLID70 three-dimensional conduction elements, LINK31 (PWR model) and MATRIX50 (BWR model) radiation elements are used. The model includes the transfer cask and the canister and its internals. The details of the canister and contents are modeled using the same methodology as that presented in Section 4.4.1.2 (Three-Dimensional Canister Models). Effective thermal properties for the fuel regions and the fuel tube regions are established using the fuel models and fuel tube models presented in Sections 4.4.1.5 and 4.4.1.6 respectively. The effective specific heat and density are calculated on the basis of material mass and volume ratio, respectively.

Radiation across the gaps was represented by the LINK31 elements or the MATRIX50 elements, which used the gray body emissivities for stainless and carbon steels. Convection is considered at the top of the canister lid, the exterior surfaces of the transfer cask, as well as at the annulus between the canister and the inner surface of the transfer cask. The combination of radiation and convection at the transfer cask exterior vertical surfaces and canister lid top surface is taken into account in the model using the same method described in Section 4.4.1.2 for the three-dimensional canister models. The bottom of the transfer cask is modeled as being in contact with the concrete floor. Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region based on a total heat load of 23 kW for both PWR and BWR fuel. The model considers the active fuel length of 144 inches and an axial power distribution, as shown in Figure 4.4.1.1-3 and 4.4.1.1-4 for PWR and BWR fuel, respectively.

An initial temperature of 100°F is considered in the model on the basis of typical maximum average water temperature in the spent fuel pool. For the design basis heat loads, the water inside the canister is drained within 17 hours and the canister is back-filled with helium immediately after the vacuum drying and transferred to the concrete cask. The design basis heat load transient analysis is performed for 17 hours with the water inside the canister, 27 hours (PWR) and 25 hours (BWR) for the vacuum condition, and 20 hours (PWR) and 16 hours (BWR) for the helium condition, followed by a steady state analysis (in helium condition). Different time durations are used for the transient analyses for the reduced heat load cases, as specified in Section 4.4.3.1. The temperature history of the fuel cladding and the basket components, as well as the transfer cask components, is determined and compared with the short-term temperature limits presented in Table 4.1-3.

Figure 4.4.1.3-1 Three-Dimensional Transfer Cask and Canister Model - PWR



Note: Canister and transfer cask media not shown for clarity.

4.4.3 Maximum Temperatures for PWR and BWR Fuel

Temperature distribution and maximum component temperatures for the Universal Storage System under the normal conditions of storage and transfer, based on the use of the transfer cask, are provided in this section. Components of the Universal Storage System containing PWR and BWR fuels are addressed separately. Temperature distributions for the evaluated off-normal and accident conditions are presented in Sections 11.1 and 11.2.

Figure 4.4.3-1 shows the temperature distribution of the Vertical Concrete Cask and the canister containing the PWR design basis fuel for the normal, long-term storage condition. The air flow pattern and air temperatures in the annulus between the PWR canister and the concrete cask liner for the normal condition of storage are shown in Figures 4.4.3-2 and 4.4.3-3, respectively. The temperature distribution in the concrete portion of the concrete cask for the PWR assembly is shown in Figure 4.4.3-4. The temperature distribution for the BWR design basis fuel is similar to that of the PWR fuel and is, therefore, not presented. Table 4.4.3-1 shows the maximum component temperatures for the normal condition of storage for the PWR design basis fuel. The maximum component temperatures for the normal condition of storage for the BWR design basis fuel are shown in Table 4.4.3-2.

As shown in Figure 4.4.3-3, a high-temperature gradient exists near the wall of the canister and the liner of the concrete cask, while the air in the center of the annulus exhibits a much lower temperature gradient, indicating significant boundary layer features of the air flow. The temperatures at the concrete cask steel liner surface are higher than the air temperature, which indicates that salient radiation heat transfer occurs across the annulus. As shown in Figure 4.4.3-4, the local temperature in the concrete, directly affected by the radiation heat transfer across the annulus, can reach 186°F (less than the 200°F allowable temperature). The bulk temperature in the concrete, as determined using volume average of the temperatures in the concrete region, is 135°F, less than the allowable value of 150°F.

Under typical operations, the transient history of maximum component temperatures for the transfer conditions (canister, inside the transfer cask, containing water for 17 hours, vacuum for 27 hours for PWR and 25 hours for BWR and for 20 hours in helium for PWR and 16 hours in helium for BWR) is shown in Figures 4.4.3-5 and 4.4.3-6 for PWR and BWR fuels, respectively. The maximum component temperatures for the transfer conditions (vacuum and helium conditions) are shown in Tables 4.4.3-3 and 4.4.3-4, for PWR and BWR fuels, respectively.

The maximum calculated water temperature is 203°F for both the PWR and BWR fuels at the end of 17 hours based on an initial water temperature of 100°F.

4.4.3.1 Maximum Temperatures at Reduced Total Heat Loads

This section provides the evaluation of component temperatures for fuel heat loads less than the design basis heat load of 23 kW. Transient thermal analyses are performed for PWR fuel heat loads of 20, 17.6, 14, 11 and 8 kW to establish the allowable time limits for the vacuum condition in the canister as described in the Technical Specifications (Chapter 12) for the Limiting Conditions of Operation (LCO), LCOs 3.1.1 and 3.1.4. The time limits ensure that the allowable temperatures of the limiting components – the heat transfer disks and the fuel cladding – are not exceeded. A steady state evaluation is also performed for all the heat load cases in the vacuum condition and all the heat load cases in the helium condition. If the steady state temperature calculated is less than the limiting component allowable temperature, then the allowable time duration in the vacuum or helium conditions is defined to be 600 hours (25 days) based on the 30 day time test for abnormal regimes as described in PNL-4835 [34].

The three-dimensional transfer cask and canister model for the PWR fuel configuration, described in Section 4.4.1.3, is used for the transient and steady state thermal analysis for the reduced heat load cases. To obtain the bounding temperatures for all possible loading configurations, thermal analyses are performed for a total of fourteen (14) cases as tabulated below. The basket locations are shown in Figure 4.4.3-7. Since the maximum temperature for the limiting components (fuel cladding and heat transfer disk) always occurs at the central region of the basket, hotter fuels (maximum allowable heat load for 5-year cooled fuel: 0.958 kW = 23 kW/24) are specified at the central basket locations. The bounding cases for each heat load condition are noted with an asterisk (*) in the tabulation which follows. Six cases (cases 3 through 8) are evaluated for the 17.6 kW heat load condition. The first four cases (cases 3 through 6) represent standard UMS system fuel loadings. The remaining two cases (cases 7 and 8) account for the preferential loading configuration for Maine Yankee site specific fuel (Section 4.5.1.2), with case 8 being the bounding case for the Maine Yankee fuel. Based on the analysis results of the 17.6 kW heat load cases, only two loading cases are required to establish the bounding condition for the 20, 14, 11 and 8 kW heat loads.

Canister Heat Load (kW)	Heat Load Case	Heat Load (kW) Evaluated in Each Basket Location (See Figure 4.4.3-7)					
		1	2	3	4	5	6
20	1	0.958	0.958	0.709	0.958	0.709	0.709
20*	2	0.958	0.958	0.958	0.958	0.958	0.210
17.6	3	0.958	0.958	0.509	0.958	0.509	0.509
17.6*	4	0.958	0.958	0.568	0.958	0.958	0.000
17.6	5	0.958	0.958	0.958	0.958	0.568	0.000
17.6	6	0.958	0.958	0.284	0.958	0.958	0.284
17.6	7	0.958	0.146	1.050	0.146	1.050	1.050
17.6	8	0.958	0.958	1.050	0.384	1.050	0.000
14	9	0.958	0.958	0.209	0.958	0.209	0.209
14*	10	0.958	0.958	0.000	0.958	0.626	0.000
11	11	0.958	0.896	0.000	0.896	0.000	0.000
11*	12	0.958	0.958	0.000	0.834	0.000	0.000
8	13	0.958	0.521	0.000	0.521	0.000	0.000
8*	14	0.958	0.958	0.000	0.084	0.000	0.000

The heat load (23 kW/24 Assemblies = 0.958 kW) at the four (4) central basket locations corresponds to the maximum allowable canister heat load for 5-year cooled fuel (Table 4.4.7-8). The non-uniform heat loads evaluated in this section bound the equivalent uniform heat loads, since they result in higher maximum temperatures of the fuel cladding and heat transfer disk.

Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region in each fuel assembly location of the model using the axial power distribution for PWR fuel (Figure 4.4.1.1-3) in the axial direction.

The thermal analysis results for the closure and transfer of a loaded PWR fuel canister in the transfer cask for the reduced heat load cases are shown in Table 4.4.3-5, with a comparison to the results for the design basis heat load case. The temperatures shown are the maximum temperatures for the limiting components (fuel cladding and heat transfer disk). The maximum temperatures of the fuel cladding and the heat transfer disk are less than the allowable temperatures (Table 4.1-3) of these components for the short-term conditions of vacuum drying and helium backfill. As shown in Table 4.4.3-5, a time limit of 600 hours is specified for moving the canister out of the transfer cask after the canister is filled with helium. This time limit is for the heat load cases where the maximum fuel cladding/heat transfer disk temperatures for the steady state condition are below the short-term allowable temperatures. Note that the maximum water temperature at the end of the "water period" is considered to be the volumetric average temperature of the calculated cladding temperatures in the active fuel region of the hottest fuel assembly. The

results indicate that the volumetric average water temperature is below 212°F for all cases evaluated. This is consistent with the thermal model that only considers conduction in the fuel assembly region and between the disks. This approach does not include consideration of convection of the water or the energy absorbed by latent heat of vaporization.

The Technical Specifications specify the remedial actions, either in-pool or forced air cooling, required to ensure that the fuel cladding and basket component temperatures do not exceed their short-term allowable temperatures, if the time limits are not met. LCOs 3.1.1 and 3.1.4 incorporate the operating times for heat loads that are less than the design basis heat loads as evaluated in this section.

Using the same three-dimensional transfer cask/canister models, analysis is performed for the conditions of in-pool cooling and forced air cooling followed by the vacuum drying and helium backfill operation (LCO 3.1.1). The conditions at the end of the vacuum drying as shown in Tables 4.4.3-5 (PWR) and 4.4.3-8 (BWR) are used as the initial conditions of the analyses. The LCO 3.1.1 "Action" analysis results are shown in Tables 4.4.3-6 and 4.4.3-7 for the PWR configuration and Tables 4.4.3-9 and 4.4.3-10 for the BWR configuration. Note that the duration of the second vacuum (after completion of the in-pool or forced air cooling) is limited (calculated based on the heat-up rate of the first vacuum), so the maximum temperatures at the end of the second vacuum cycle will not exceed those at the end of the first vacuum cycle. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR) are conservatively presented as the maximum temperatures for the second vacuum condition. The maximum temperatures for the fuel cladding and the heat transfer disk are below the short-term allowable temperatures.

The in-pool cooling and the forced-air cooling followed by the helium backfill operation in LCO 3.1.4 are also evaluated for the PWR configuration for the 23 kW case and the BWR configuration for the 23 kW and 20 kW cases. The temperature profiles at the end of the helium condition, as shown in Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR, are used as the initial condition. The results for the BWR are shown in Tables 4.4.3-11 and 4.4.3-12 for the in-pool cooling and forced-air cooling, respectively. The results for the PWR are shown in Tables 4.4.3-13 and 4.4.3-14 for the in-pool cooling and forced-air cooling, respectively. Note that the time limit for the first helium backfill condition is used for the second helium backfill condition (after completion of the in-pool or forced-air cooling). Based on the heat-up rate of the first helium condition, the maximum component temperatures at the end of the second helium condition are well below the maximum temperatures at the end of the first helium condition. The maximum

temperatures at the end of the first helium condition (Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR) are conservatively presented as the maximum temperatures for the second helium backfill condition, as shown in Tables 4.4.3-11 and 4.4.3-12 for the BWR configuration and Tables 4.4.3-13 and 4.4.3-14 for the PWR configuration.

Figure 4.4.3-1 Temperature Distribution (°F) for the Normal Storage Condition: PWR Fuel

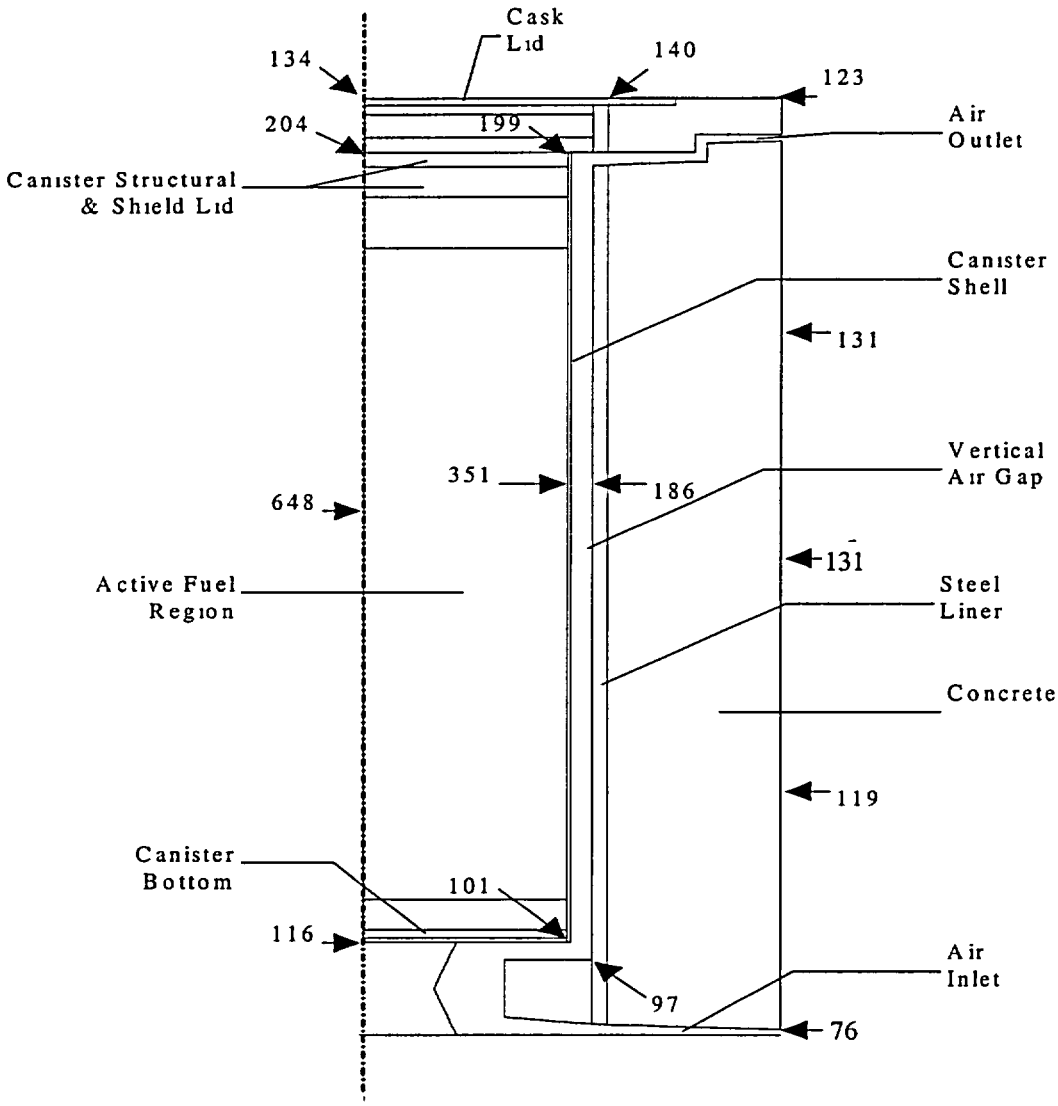


Figure 4.4.3-2 Air Flow Pattern in the Concrete Cask in the Normal Storage Condition: PWR Fuel

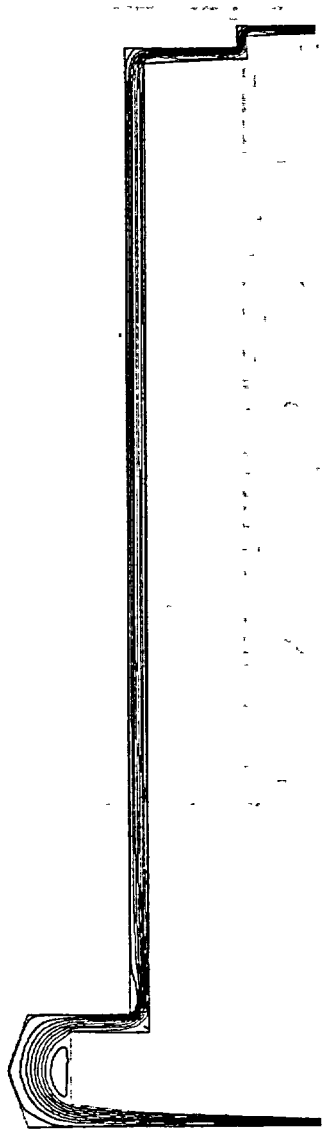


Figure 4.4.3-3 Air Temperature (°F) Distribution in the Concrete Cask During the Normal Storage Condition: PWR Fuel

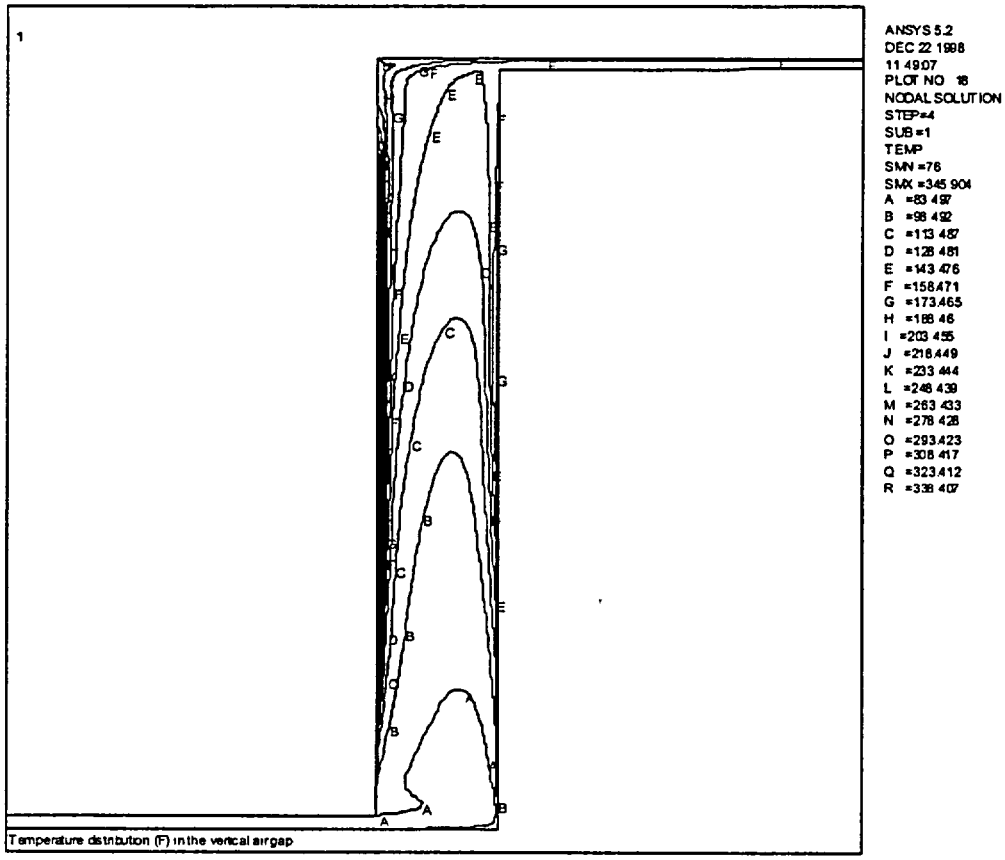


Figure 4.4.3-4 Concrete Temperature (°F) Distribution During the Normal Storage Condition: PWR Fuel

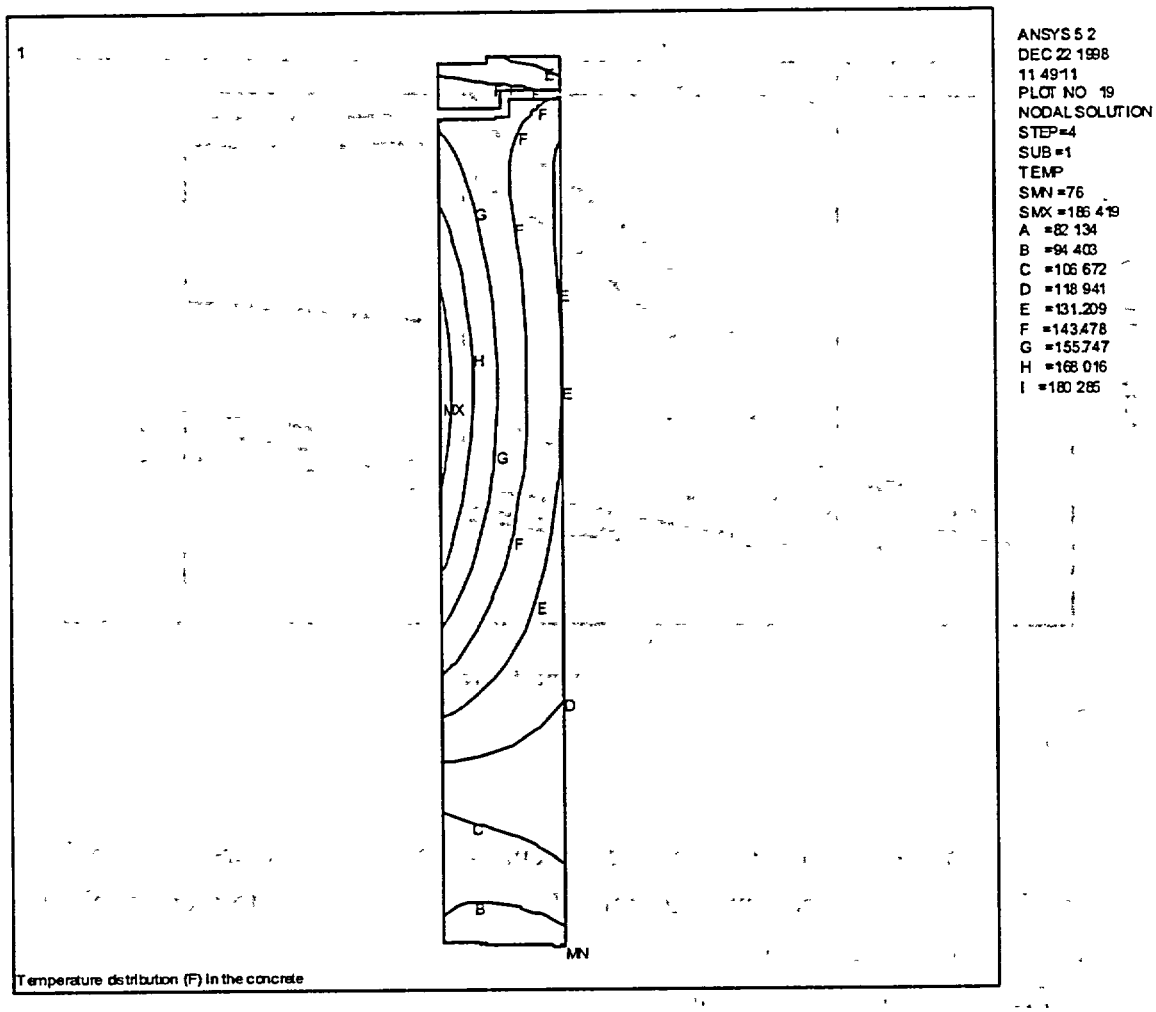
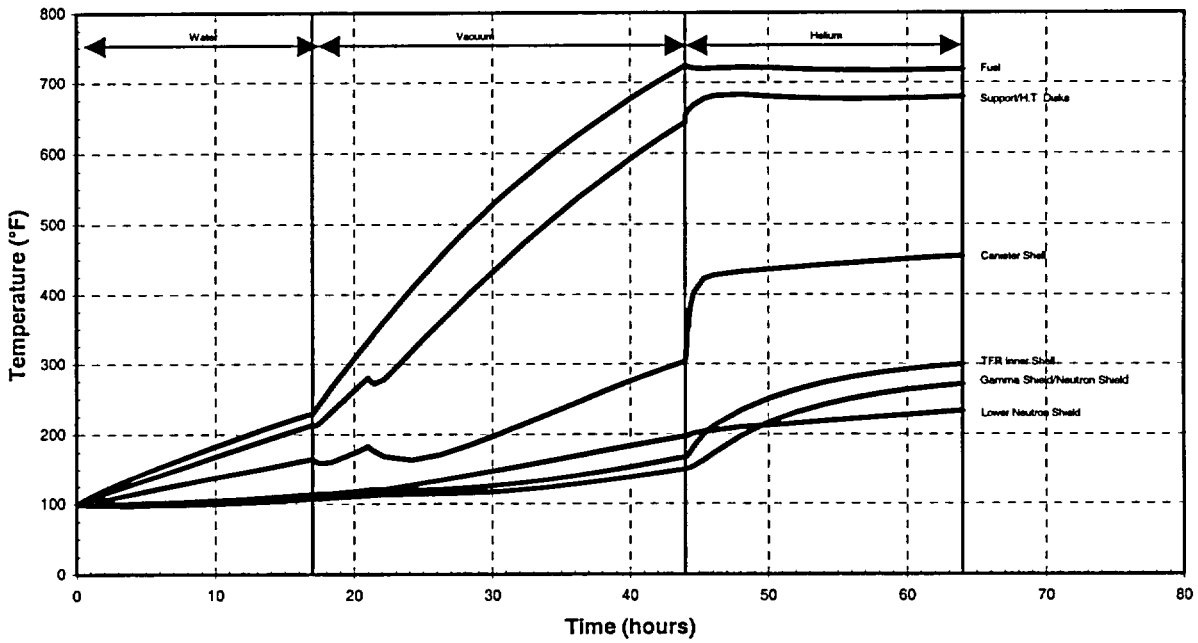


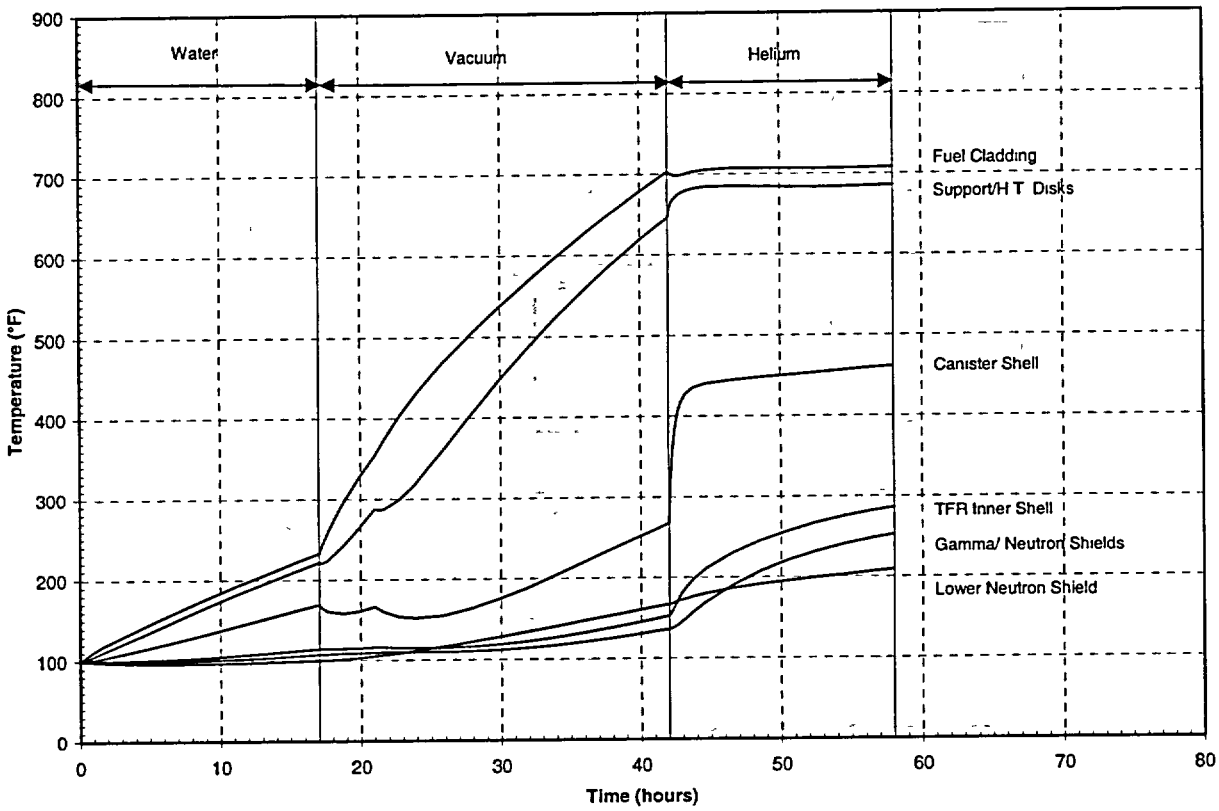
Figure 4.4.3-5 History of Maximum Component Temperature (°F) for Transfer Conditions for PWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load



Notes:

1. This graph corresponds to a canister containing water for 17 hours, vacuum for 27 hours and 20 hours in the helium condition. The results correspond to a uniformly distributed decay heat load of 23 kW.
2. "TFR" refers to the transfer cask.

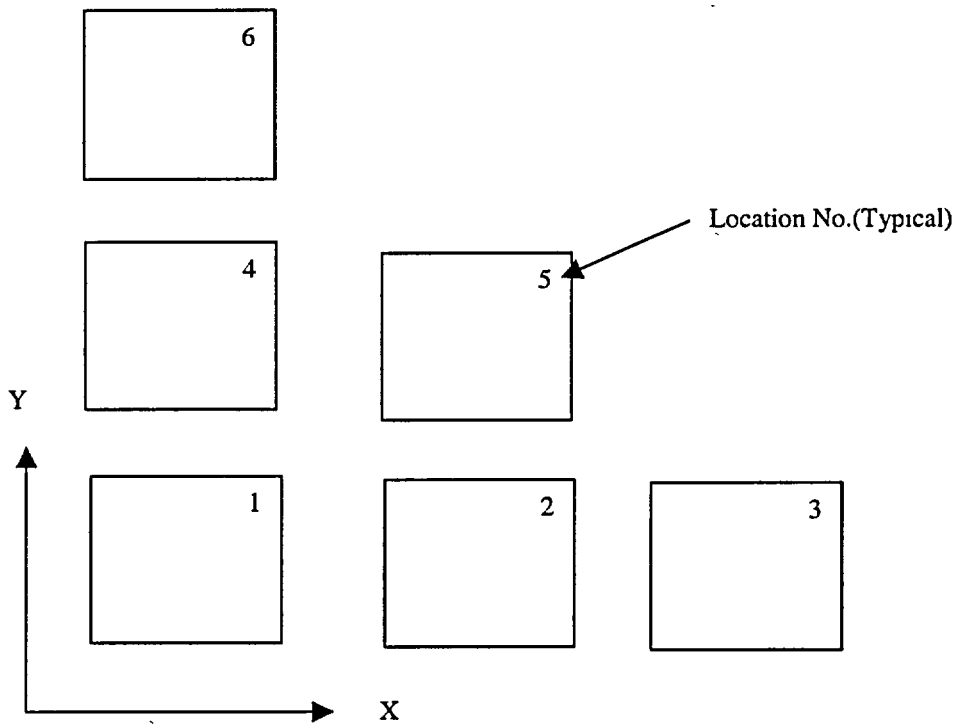
Figure 4.4.3-6 History of Maximum Component Temperature (°F) for Transfer Conditions for BWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load



Notes:

1. This graph corresponds to a canister containing water for 17 hours, vacuum for 25 hours and 16 hours in the helium condition. The results correspond to a uniformly distributed decay heat load of 23 kW.
2. "TFR" refers to the transfer cask.

Figure 4.4.3-7 Basket Location for the Thermal Analysis of PWR Reduced Heat Load Cases



A quarter symmetry configuration is considered. X and Y axes are at the centerlines of the basket.

Figure 4.4.3-8 BWR Fuel Basket Location Numbers

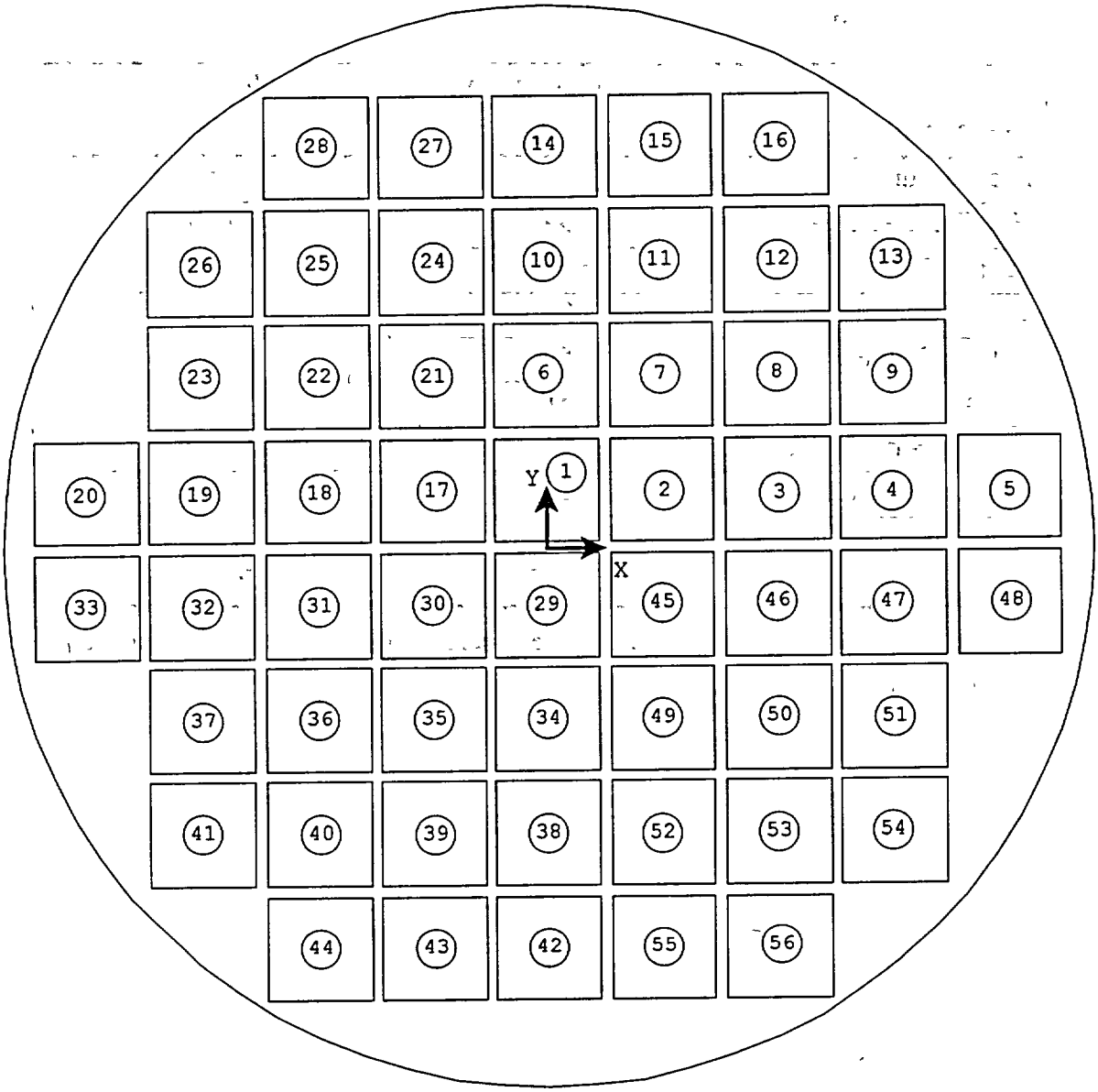


Table 4.4.3-1 Maximum Component Temperatures for the Normal Storage Condition - PWR

Component	Maximum Temperature (°F)	Allowable Temperatures (°F)
Fuel Cladding	648	752
Heat Transfer Disk	599	650
Support Disk	601	650
Top Weldment	399	800
Bottom Weldment	159	800
Canister Shell	351	800
Canister Structural Lid	204	800
Canister Shield Lid	212	800
Concrete	186 (local) 135 (bulk*)	300 (local) 150 (bulk)

* The volume average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.4.3-2 Maximum Component Temperatures for the Normal Storage Condition - BWR

Component	Maximum Temperature (°F)	Allowable Temperatures (°F)
Fuel Cladding	642	752
Heat Transfer Disk	612	650
Support Disk	614	700
Top Weldment	361	800
Bottom Weldment	276	800
Canister Shell	376	800
Canister Structural Lid	180	800
Canister Shield Lid	185	800
Concrete	192 (local) 136 (bulk*)	300 (local) 150 (bulk)

*The volume average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.4.3-3 Maximum Component Temperatures for the Transfer Condition – PWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load

Component	Maximum Temperature (°F)		Allowable Temperature (°F)
	Vacuum[1]	Helium[1]	
Fuel	724	724	752
Lead	151	271	600
Neutron Shield	197	267	300
Heat Transfer Disk	641	680	750
Support Disk	644	683	800
Canister	304	455	800
Transfer Cask Shells	168	300	700

1. See Figure 4.4.3-5 for history of maximum component temperatures.

Table 4.4.3-4 Maximum Component Temperatures for the Transfer Condition – BWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load

Component	Maximum Temperature (°F)		Allowable Temperature (°F)
	Vacuum[1]	Helium[1]	
Fuel	703	708	752
Lead	137	252	600
Neutron Shield	135	249	300
Heat Transfer Disk	645	683	750
Support Disk	646	686	700
Canister	267	462	800
Transfer Cask Shells	153	286	700

1. See Figure 4.4.3-6 for history of maximum component temperatures.

Table 4.4.3-5 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel

Heat Load (kW)	Water			Vacuum			Helium		
	Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	17	230	213	27	724	641	20	724[3]	680[3]
20.0	18	232	214	30	728	628	600[2]	728/708	664/664
17.6	20	239	219	33	731	617	600[2]	731/672	651/624
17.6[1]	20	231	214	33	722	604	600[2]	722/657	635/609
14.0	22	240	219	40	732	596	600[2]	732/613	630/559
11.0	24	237	215	52	730	575	600[2]	730/555	611/495
8.0	37	247	221	103	731	557	600[2]	731/483	595/412

1. Preferential loading configuration, site specific case for Maine Yankee.
2. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835 [34].
3. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed.

Table 4.4.3-6 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	24	491	415	14	724	641	20	724[4]	680[4]
20.0	24	477	397	17	728	628	600[3]	728/708	664/664
17.6	24	465	383	20	731	617	600[3]	731/672	651/624
14.0	24	445	360	26	732	596	600[3]	732/613	630/559
11	24	422	334	38	730	575	600[3]	730/555	611/495
8	24	390	293	89	731	557	600[3]	731/483	595/412

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed.

Table 4.4.3-7 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	24	621	564	5	724	641	20	724[4]	680[4]
20.0	24	591	530	8	728	628	600[3]	728/708	664/664
17.6	24	567	502	11	731	617	600[3]	731/672	651/624
14.0	24	530	458	18	732	596	600[3]	732/613	630/559
11	24	493	415	29	730	575	600[3]	730/555	611/495
8	24	450	363	80	731	557	600[3]	731/483	595/412

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed

Table 4.4.3-8 Maximum Limiting Component Temperatures in Transient Operations for BWR Fuel

	Water			Vacuum			Helium		
		Maximum Temperature (°F)			Maximum Temperature (°F)			Max. Temp. / Temp. at Steady-state (°F)	
Heat Load (kW)	Duration (hours)	Fuel	Heat Transfer Disk	Duration (hours)	Fuel	Heat Transfer Disk	Duration (hours)	Fuel	Heat Transfer Disk
23	17	232	221	25	703	645	16	708[2]	683[2]
20	18	234	222	27	694	627	30	694[2]	661[2]
17	19	234	221	33	701	629	600[1]	701/660	659/631
14	20	232	219	45	719	643	600[1]	719/606	671/574
11	23	234	220	72	733	653	600[1]	733/543	679/508
8	31	236	220	600[1]	724	639	600[1]	724/467	639/427

1. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
2. Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed.

Table 4.4.3-9 Maximum Limiting Component Temperatures in Transient Operations after Vacuum for BWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	488	444	12	703	645	16	708[4]	683[4]
20	24	476	431	13	694	627	30	694[4]	661[4]
17	24	467	419	19	701	629	600[3]	701/660	659/631
14	24	455	404	28	719	643	600[3]	719/606	671/574
11	24	439	383	54	733	653	600[3]	733/543	679/508

1 The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4 4.3-8) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed

Table 4.4.3-10 Maximum Limiting Component Temperatures in Transient Operations after Vacuum for BWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	623	591	4	703	645	16	708[4]	683[4]
20	24	592	558	5	694	627	30	694[4]	661[4]
17	24	565	528	10	701	629	600[3]	701/660	659/631
14	24	541	503	20	719	643	600[3]	719/606	671/574
11	24	519	477	43	733	653	600[3]	733/543	679/508

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4 4.3-8) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4 Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed.

Table 4.4.3-11 Maximum Limiting Component Temperatures in Transient Operations after Helium for BWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	489	444	16	708	683
20	24	477	431	30	694	661

1. The maximum temperatures at the end of helium in Table 4.4.3-8 are conservatively used.

Table 4.4.3-12 Maximum Limiting Component Temperatures in Transient Operations after Helium for BWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	630	598	16	708	683
20	24	601	566	30	694	661

1. The maximum temperatures at the end of helium in Table 4.4.3-8 are conservatively used.

Table 4.4.3-13 Maximum Limiting Component Temperatures in Transient Operations after Helium for PWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	489	413	20	724	680

1. The maximum temperatures at the end of helium in Table 4.4.3-5 are conservatively used.

Table 4.4.3-14 Maximum Limiting Component Temperatures in Transient Operations after Helium for PWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	626	569	20	724	680

1. The maximum temperatures at the end of helium in Table 4.4.3-5 are conservatively used.

4.4.5 Maximum Internal Pressures

The maximum internal operating pressures for normal conditions of storage are calculated in the following sections for the PWR and BWR Transportable Storage Canisters.

4.4.5.1 Maximum Internal Pressure for PWR Fuel Canister

The internal pressures within the PWR fuel canister are a function of fuel type, fuel condition (failure fraction), burnup, UMS® canister type, and the backfill gases in the canister cavity. Gases included in the canister pressure evaluation include rod-fill, rod fission and rod backfill gases, canister backfill gases and burnable poison generated gases. Each of the fuel types expected to be loaded into the UMS® canister system is separately evaluated to arrive at a bounding canister pressure.

Fission gases include all fuel material generated gases including long-term actinide decay generated helium. Based on detailed SAS2H calculations of the maximum fissile material mass assemblies in each canister class, the quantity of gas generated by the fuel rods rises as burnup and cool time is increased and enrichment is decreased. To assure the maximum gas is available for release, the PWR inventories are extracted from 60,000 MWD/MTU burnup cases at an enrichment of 1.9 wt. % ^{235}U and a cool time of 40 years. Gas inventories at 60,000 MWD/MTU bound those calculated at 45,000 MWD/MTU, the maximum allowable burnup. Gases included are all krypton, iodine, and xenon isotopes in addition to helium and tritium (^{3}H). Molar quantities for each of the maximum fissile mass assemblies are summarized in Table 4.4.5-1. Fuel generated gases are scaled by fissile mass to arrive at molar contents of other UMS® fuel types.

Fuel rod backfill pressure varies significantly between the PWR fuel types. The maximum reported backfill pressure is listed for the Westinghouse 17×17 fuel assembly at 500 psig. With the exception of the B&W fuel assemblies, which are limited to 435 psig, all fuel assemblies evaluated are set to the maximum 500 psig backfill reported for the Westinghouse assembly. Backfill quantities are based on the free volume between the pellet and the clad and the plenum volume. The fuel rod backfill gas temperature is conservatively assumed to have an initial temperature of 68°F.

Burnable poison rod assemblies (BPRAs) placed within the UMS® storage canister may contribute additional molar gas quantities due to (n,alpha) reactions of fission generated neutrons with ^{10}B during in-core operation. ^{10}B forms the basis of a portion of the neutron poison population. Other neutron poisons, such as gadolinium and erbium, do not produce a significant amount of helium nuclides (alpha particles) as part of their activation chain. Primary BPRAs in existence include Westinghouse Pyrex (borosilicate glass) and WABA (wet annular burnable absorber) configurations, as well as B&W BPRAs and shim rods employed in CE cores. The CE shim rods replace standard fuel rods to form a complete assembly array. The quantity of helium available for release from the BPRAs is directly related to the initial boron content of the rods and the release fraction of gas from the matrix material in question. Release from either of the low temperature, solid matrix materials is likely to be limited, but no release fractions were available in open literature. As such, a 100% release fraction is assumed based on a boron content of 0.0063 g/cm ^{10}B per rod, with the maximum number of rods per assembly. The maximum number of rods is 16 for Westinghouse core 14×14 assemblies, 20 rods for Westinghouse and B&W 15×15 assemblies, and 24 rods for Westinghouse and B&W 17×17 assemblies. The length of the absorber is conservatively taken as the active fuel length. CE core shim rods are modeled at 0.0126 g/cm ^{10}B for 16, 12, and 12 rods applied to CE manufactured 14×14, 15×15 and 16×16 cores, respectively.

The canister backfill gases are conservatively assumed to be at 250°F, which is significantly below the canister shell maximum initial temperature of 304°F at the end of vacuum drying. The initial pressure of the canister backfill gas is 1 atm (0.0 psig). Free volume inside each PWR canister class is listed in Table 4.4.5-2. The listed free volumes do not include fuel assembly components since these components vary for each assembly type and fuel insert. Subtracting out the rod and guide tube volumes and all hardware components arrives at free volume of the canisters including fuel assemblies and a load of 24 BPRAs. For the Westinghouse BPRAs, the Pyrex volume is employed since it displaces more volume than the WABA rods.

The total pressure for each of the UMS® payloads is found by calculating the releasable molar quantity of each gas (30% of the fission gas and 100% of the rod backfill adjusted for the 1% fuel failure fraction), and summing the quantities directly. The quantity of gas is then employed in the ideal gas equation in conjunction with the average gas temperature at normal operating conditions to arrive at system pressures. The normal condition average temperature of the gas within the PWR canister is conservatively considered to be 420°F. This temperature bounds the

calculated gas temperature (418°F) for normal conditions of storage using the three-dimensional canister models. Each of the UMS® PWR fuel types is individually evaluated for normal condition pressure, and sets the maximum normal condition pressure at 4.21 psig. A summary of the maximum pressure in each PWR canister class is shown in Table 4.4.5-3. The table also includes the fuel type producing the listed maximum pressures.

4.4.5.2 Maximum Internal Pressure for BWR Fuel Canister

BWR canister maximum pressures are determined in the same manner as those documented for the PWR canister cases. Primary differences between PWR and BWR analysis include a maximum normal condition average gas temperature of 410°F, rod backfill gas pressures of 132 psig, and limits pressurizing gases to fission gases (including helium actinide decay gas), rod backfill gases, and canister backfill gas. The 132 psig employed in this analysis is significantly higher than the 6 atmosphere maximum pressure reported in open literature. BWR assemblies do not contain an equivalent to the PWR BPRAs and, therefore, do not require ^{10}B helium generated gases to be added. Fissile gas inventories for the maximum fissile material assemblies in each of the three BWR lattices configurations (7×7, 8×8, and 9×9) are shown in Table 4.4.5-4. Free volumes, without fuel components, in UMS® canister classes 4 and 5 are shown in Table 4.4.5-5. Maximum pressures for each canister class are listed in Table 4.4.5-6. The maximum normal condition pressure of 3.97 psig is based on a GE 7×7 assembly, designed for a BWR/2-3 reactor, with gas inventories conservatively taken from a 60,000 MWD/MTU source term. The normal condition pressure for a UMS® storage canister containing the GE 9×9 fuel assembly with 79 fuel rods is 3.96 psig. Similar fuel masses and displaced volume account for similar canister pressures.

Table 4.4.5-1 PWR Per Assembly Fuel Generated Gas Inventory

Array	Assy Type	MTU	Moles
14×14	WE Standard	0.4144	35.52
15×15	B&W	0.4807	41.32
16×16	CE (System 80)	0.4417	38.10
17×17	WE Standard	0.4671	40.18

Table 4.4.5-2 PWR Canister Free Volume (No Fuel or Inserts)

Canister Class	1	2	3
Basket Volume (in^3)	69800	74490	77460
Canister Height (inch)	175.05	184.15	191.75
Canister Free Volume w/o Fuel (liter)	7970	8400	8770

Table 4.4.5-3 PWR Maximum Normal Condition Pressure Summary

Canister Class	Fuel Type	Pressure (psig)
Class 1	WE 17×17 Standard	4.20
Class 2	B&W 17×17 Mark C	4.21
Class 3	CE 16×16 System 80	4.11

4.4.7 Evaluation of System Performance for Normal Conditions of Storage

Results of thermal analysis of the Universal Storage System containing PWR or BWR fuel under normal conditions of storage are summarized in Tables 4.4.3-1 through 4.4.3-4. The maximum PWR and BWR fuel rod cladding temperatures are below the allowable temperatures; temperatures of safety-related components during storage and transfer operations under normal conditions are maintained within their safe operating ranges; and thermally induced stresses in combination with pressure and mechanical load stresses are shown in the structural analysis of Chapter 3.0 to be less than the allowable stresses. Therefore, the Universal Storage System performance meets the requirements for the safe storage of design basis fuel under the normal operating conditions specified in 10 CFR 72.

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4.5 Thermal Evaluation for Site Specific Spent Fuel

This section presents the thermal evaluation of fuel assemblies or configurations, which are unique to specific reactor sites or which differ from the UMS® Storage System design basis fuel. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibit defects greater than pinhole leaks or hairline cracks.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation.

4.5.1 Maine Yankee Site Specific Spent Fuel

The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14×14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14×14 fuel assembly is included in the population of the design basis PWR fuel assemblies for the UMS® Storage System (See Table 2.1.1-1). The maximum decay heat for the standard Maine Yankee fuel is the design basis heat load for the PWR fuels (23 kW total, or 0.958 kW per assembly). This heat load is bounded by the thermal evaluations in Section 4.4 for the normal conditions of storage, Section 4.4.3.1 for less than design basis heat loads and Chapter 11 for off-normal and accident conditions.

Some Maine Yankee site specific fuel has a burnup greater than 45,000 MWD/MTU, but less than 50,000 MWD/MTU. As shown in Table 12B2-6 in Chapter 12, loading of fuel assemblies in this burnup range is subject to preferential loading in designated basket positions in the Transportable Storage Canister. Certain fuel assemblies in this burnup range must be loaded in a Maine Yankee fuel can.

The site specific fuels included in this evaluation are:

1. Consolidated fuel rod lattices consisting of a 17 × 17 lattice fabricated with 17 × 17 grids, 4 stainless steel support rods and stainless steel end fittings. One of these

lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the remaining rod position locations either empty or containing stainless steel dummy rods.

2. Standard fuel assemblies with a Control Element Assembly (CEA) inserted in each one.
3. Standard fuel assemblies that have been modified by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.
4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.
5. Standard fuel assemblies with in-core instrument thimbles stored in the center guide tube.
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.
7. Standard fuel assemblies that have some fuel rods removed.
8. Standard fuel assemblies that have damaged fuel rods.
9. Standard fuel assemblies that have some type of damage or physical alteration to the cage (fuel rods are not damaged).
10. Two (2) rod holders, designated CF1 and CA3. CF1 is a lattice having approximately the same dimensions as a standard fuel assembly. It is a 9×9 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 in which damaged fuel rods have been inserted.
11. Standard fuel assemblies that have damaged fuel rods stored in their guide tubes.
12. Standard fuel assemblies with inserted startup sources and other non-fuel items.

The Maine Yankee site specific fuels are also described in Section 1.3.2.1.

The thermal evaluations of these site specific fuels are provided in Section 4.5.1.1. Section 4.5.1.2 presents the evaluation of the Maine Yankee preferential loading of fuel exceeding the design basis heat load (0.958 kW) per assembly on the basket periphery.

assembly with this configuration. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.6 Standard Fuel Assemblies with Variable Enrichment and Axial Blankets

The Maine Yankee variably enriched fuel assemblies are limited to two batches of fuel, which have a maximum burnup less than 30,000 MWD/MTU. The variably enriched rods in the fuel assemblies have enrichments greater than 3.4 wt % ^{235}U, except that the axial blankets on one batch are enriched to 2.6 wt % ^{235}U. As shown in Table 12B2-8, fuel at burnups less than or equal to 30,000 MWD/MTU with any enrichment greater than, or equal to, 1.9 wt % ^{235}U may be loaded with 5 years cool time.

The thermal conductivities of the fuel assemblies with variable enrichment (radial) and axial blankets are considered to be essentially the same as those of the standard fuel assemblies. Since the heat load per assembly is limited to the design basis heat load, there is no effect on the thermal performance of the system due to this loading configuration.

4.5.1.1.7 Standard Fuel Assemblies with Removed Fuel Rods

Except for assembly number EF0046, the maximum number of missing fuel rods from a standard fuel assembly is 14, or 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 34% less than the design basis heat load of 0.958 kW. Fuel assembly EF0046 was used in the consolidated fuel demonstration program and has only 69 rods remaining in its lattice. This fuel assembly has a heat load of 70 watts, or 7% of the design basis heat load of 0.958 kW. Therefore, the thermal performance of fuel assemblies with removed fuel rods is bounded by that of the standard fuel assemblies.

4.5.1.1.8 Fuel Assemblies with Damaged Fuel Rods

Damaged fuel assemblies are standard fuel assemblies with fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Fuel, classified as damaged, will be placed in a Maine Yankee fuel can. The primary function of the fuel can is to confine fuel material within the can and to facilitate handling and retrievability. The Maine Yankee fuel can is shown in Drawings 412-501 and 412-502. The placement of the loaded fuel cans is restricted

by the operating procedures and/or Technical Specifications to loading into the four corner positions at the periphery of the fuel basket as shown in Figure 12B2-1. The heat load for each damaged fuel assembly is considered to be the design basis heat load of 0.958 kW (23 kW/24).

A steady-state thermal analysis is performed using the three-dimensional canister model described in Section 4.4.1.2 simulating 100% failure of the fuel rods, fuel cladding, and guide tubes of the damaged fuel held in the Maine Yankee fuel can. The canister is assumed to contain twenty (20) design basis PWR fuel assemblies and damaged fuel assemblies in fuel cans in each of the four corner positions.

Two debris compaction levels are considered for the 100% failure condition: (Case 1) 100% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 52-inch debris level in the bottom of each fuel can, and (Case 2) 50% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 104-inch debris level in the bottom of each fuel can. The entire heat generation rate for a single fuel assembly (i.e., 0.958 kW) is concentrated in the debris region with the remainder of the active fuel region having no heat generation rate applied. To ensure the analysis is bounding, the debris region is located at the lower part of the active fuel region in lieu of the bottom of the fuel can. This location is closer to the center of the basket where the maximum fuel cladding temperature occurs. The effective thermal conductivities for the design basis PWR fuel assembly (Section 4.4.1.5) are used for the debris region. This is conservative since the debris (100% failed rods) is expected to have higher density (better conduction) and more surface area (better radiation) than an intact fuel assembly. In addition, the thermal conductivity of helium is used for the remainder of the active fuel length. Boundary conditions corresponding to the normal condition of storage are used at the outer surface of the canister model (see Section 4.4.1.2). A steady-state thermal analysis is performed. The results of the thermal analyses performed for 100% fuel rod, fuel cladding, and guide tube failure are:

Description	Maximum Temperature (°F)			
	Fuel Cladding	Damaged Fuel	Support Disk	Heat Transfer Disk
Case 1 (100% Compaction)	654	672	598	594
Case 2 (50% Compaction)	674	594	620	616
Design Basis PWR Fuel	670	N/A	615	612
Allowable	752	N/A	650	650

As demonstrated, the extreme case of 100% fuel rod, fuel cladding, and guide tube failure with 50% compaction of the debris results in temperatures that are less than 1% higher than those calculated for the design basis PWR fuel. The maximum temperatures for the fuel cladding, damaged fuel assembly, support disks, and heat transfer disks remain within the allowable temperature range for both 100% failure cases. Additionally, the temperatures used in the structural analyses of the fuel basket envelop those calculated for both 100% failure cases.

Additionally, the above analysis has been repeated to consider a maximum heat load of 1.05 kW/assembly (see Section 4.5.1.2) in the Maine Yankee fuel cans. To maintain the 23 kW total heat load per canister, the model considers a heat load of 1.05 kW/assembly in the four (4) Maine Yankee fuel cans and 0.94 kW/assembly in the rest of the twenty (20) basket locations. The analysis results indicate that the maximum temperatures for the fuel cladding and basket components are slightly lower than those for the case with a heat load of 0.958 kW in the damaged fuel can, as presented above. The maximum fuel cladding temperature is 650°F (< 654°F) and 672°F (< 674°F) for 100% and 50% compaction ratio cases, respectively. Therefore, the case with 1.05 kW/assembly in the Maine Yankee fuel can is bounded by the case with 0.958 kW/assembly in the fuel cans.

4.5.1.1.9 Standard Fuel Assemblies with Damaged Lattice

Certain standard fuel assemblies may have damage or physical alteration to the lattice or cage that holds the fuel rods, but not exhibit damage to the fuel rods. Fuel assemblies with lattice damage are evaluated in Section 11.2.16. The structural analysis demonstrates that these assemblies retain their configuration in the design basis accident events and loading conditions.

The effective thermal conductivity for the fuel assembly used in the thermal analyses in Section 4.4 is determined by the two-dimensional fuel model (Section 4.4.1.5). The model conservatively ignores the conductance of the steel cage of the fuel assembly. Therefore, damage or physical alteration to the cage has no effect on the thermal conductivity of the fuel assembly used in the thermal models. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.10 Damaged Fuel Rod Holders

The Maine Yankee site specific fuel inventory includes two (2) damaged fuel rod holders designated CF1 and CA3. CF1 is a 9×9 array of tubes having roughly the same dimensions as a fuel assembly. Some of the tubes hold damaged fuel rods. CA3 is a previously used fuel assembly cage (i.e., a fuel assembly with all of the fuel rods removed), into which damaged fuel rods have been inserted.

Similar to the fuel assemblies that have damaged fuel rods, the damaged fuel rod holders will be placed in Maine Yankee fuel cans and their location in the basket is restricted to one of the four corner fuel tube positions of the basket. The decay heat generated by the fuel in each of these rod holders is less than one-fourth of the design basis heat load of 0.958 kW. Therefore, the thermal performance of the damaged fuel rod holders is bounded by that of the standard fuel assemblies.

4.5.1.1.11 Assemblies with Damaged Fuel Rods Inserted in Guide Tubes

Similar to fuel assemblies that have damaged fuel rods, fuel assemblies that have up to two damaged fuel rods or poison rods stored in each guide tube are placed in Maine Yankee fuel cans and their loading positions are restricted to the four corner fuel tubes in the basket. The rods inserted in the guide tubes can not be from a different fuel assembly (i.e., any rod in a guide tube originally occupied a rod position in the same fuel assembly). Storing fuel rods in the guide tubes of a fuel assembly slightly increases the axial conductance of the fuel assembly (helium replaced by solid material). The design basis heat load bounds the heat load for these assemblies. Therefore, the thermal performance of fuel assemblies with rods inserted in the guide tubes is bounded by that of the standard fuel assemblies.

4.5.1.1.12 Standard Fuel Assemblies with Inserted Start-up Sources and Other Non-Fuel Items

Five Control Element Assembly (CEA) fingertips and a 24-inch ICI segment may be placed into the guide tubes of a fuel assembly. In addition, four irradiated start-up neutron sources and one unirradiated source, having a combined total heat load of 15.4 watts, will be loaded into separate fuel assemblies. With the CEA fingertips and the neutron sources inserted into the guide tubes of the fuel assemblies, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective

4.5.1.2 Preferential Loading with Higher Heat Load (1.05 kW) at the Basket Periphery

The Maine Yankee fuel inventory includes fuel assemblies that will exceed the initial per assembly heat load of 0.958 kW. To enable loading of these assemblies into the storage cask, a higher peripheral heat load is evaluated. The maximum heat load for peripheral assemblies is set at 1.05 kW. The maximum basket heat load for this configuration remains restricted to 23 kW.

To ensure that these fuel assemblies do not exceed their allowable cladding temperatures, a loading pattern is shown that places higher heat load assemblies at the periphery of the basket (Positions "A" in Figure 4.5.1.2-1) and compensates by placing lower heat load assemblies in the basket interior positions (Positions "B" in Figure 4.5.1.2-1). There are 12 interior basket locations and 12 peripheral basket locations in the UMS® PWR basket design. The maximum total basket heat load of 23 kW is maintained for these peripheral loading scenarios.

Given the higher than design basis heat load in peripheral basket locations, an evaluation is performed to assure that maximum cladding temperature does not exceed the allowable temperature of 400°C (752°F) per ISG-11, Revision 2 [38].

A parametric study is performed using the three-dimensional periodic model, as described in Section 4.5.1.1 (Figure 4.5.1.1-2), to demonstrate that placing a higher heat load in the peripheral locations does not result in heating of the fuel assemblies in the interior locations beyond that found in the uniform heat loading case. The side surface of the model is assumed to have a uniform temperature of 350°F.

Two cases are considered (total heat load per cask = 20 kW for both cases):

1. Uniform loading: Heat load = 0.833 (20/24) kW per assembly for all 24 assemblies
2. Non-uniform loading:
   Heat load = 0.958 (23/24) kW per assembly for 12 peripheral assemblies
   Heat load = 0.708 (17/24) kW per assembly for 12 interior assemblies

The analysis results (maximum temperatures) are:

	Case 1 Uniform Loading (°F)	Case 2 Non-Uniform Loading (°F)
Fuel (Location 1)	675	648
Fuel (Locations 2 & 4)	632	611
Fuel (Location 5)	577	588
Fuel (Locations 3 & 6)	563	576
Basket	611	592

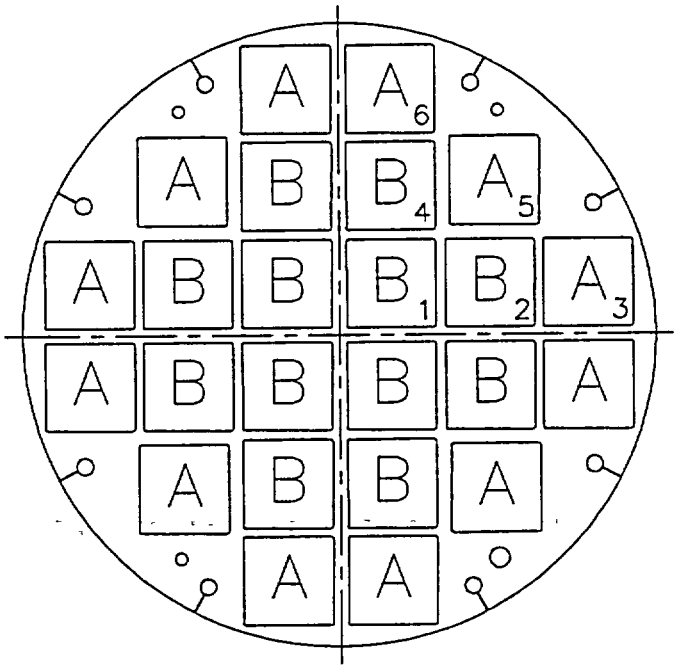
Locations are shown in Figure 4.5.1.2-1.

The maximum fuel cladding temperature for Case 2 (non-uniform loading pattern) is well below that for Case 1 (uniform loading pattern). The comparison shows that placing hotter fuel in the peripheral locations of the basket and cooler fuel in the interior locations (while maintaining the same total heat load per basket) reduces the maximum fuel cladding temperature (which occurs in the interior assembly), as well as the maximum basket temperature.

Based on the parametric study (uniform versus non-uniform analysis) of the 20 kW basket, a 15% redistribution of heat load resulted in a maximum increase of 13°F (576-563=13) in a peripheral basket location. Changing the basket peripheral location heat load from 0.958 kW maximum to 1.05 kW is a less than 10% redistribution for the 23 kW maximum basket heat load. The highest temperature of a peripheral basket location may, therefore, be estimated by adding 13°F to 566°F (maximum temperature in peripheral assemblies for the 23 kW basket with uniform heat load distribution). The 579°F (304°C) temperature is well below the allowable cladding temperature of 400°C .

Therefore, the maximum fuel cladding temperature for the preferential loading configuration with the higher heat load of 1.05 kW at the periphery basket locations will not exceed the allowable fuel cladding temperature.

Figure 4.5.1.2-1 Canister Basket Preferential Loading Plan



"A" indicates peripheral locations.
"B" indicates interior locations.
Numbered locations indicate positions where maximum fuel temperatures are presented.

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5.5 Minimum Allowable Cooling Time Evaluation for PWR and BWR Fuel

Sections 5.1 through 5.4 include the source term and shielding analyses for the design basis UMS® PWR and BWR assemblies with a burnup of 40 GWD/MTU and a 5-year cool time. The shielding evaluation design basis fuel assemblies source term are based on an initial minimum enrichment of 3.7 wt % ^{235}U for PWR and 3.25 wt % ^{235}U for BWR fuel assemblies. The source terms for the design basis assemblies represent a maximum heat load of 25.2 kW for the PWR cask and 24 kW for the BWR cask. The maximum allowable heat load for the UMS® storage system is 23 kW.

This section determines minimum cooling times for PWR and BWR assemblies at burnups ranging from 30 to 45 GWD/MTU with corresponding minimum initial enrichments from 1.9 wt% ^{235}U to 4.9 wt % ^{235}U. For each combination of initial enrichment and burnup, the minimum cooling times necessary to meet the maximum allowable decay heat, maximum transfer cask dose rate and maximum storage cask dose rate are determined. The listed minimum cooling times are the most limiting time required to meet either the canister maximum allowable heat load, the transfer cask design basis radial dose rates or storage cask design basis radial dose rates.

To address differences in the fissile material loading between assemblies, the assemblies are grouped by fuel pin array size. The BWR fuel types evaluated are 7×7, 8×8 and 9×9 assemblies and the PWR fuel types evaluated are 14×14, 15×15, 16×16 and 17×17 fuel assemblies.

5.5.1 Selection of Limiting PWR and BWR Fuel Types for Minimum Cooling Time Determination

The bounding PWR and BWR fuel assemblies are listed in Table 5.5-1. The selection of the limiting PWR and BWR assemblies is made based upon bounding maximum initial uranium loadings at 95% theoretical fuel density. Detailed PWR and BWR fuel characteristics, including the maximum MTU loadings, are documented in Table 6.2-1 and Table 6.2-2 for a wide range of PWR and BWR fuel assemblies.

Bounding uranium loadings produce the maximum heat loads and fuel radiation source terms. To ensure that fuel hardware such as grid spacers and burnable poison rods are fully considered in selecting the shielding limited fuel types the Westinghouse 15×15, GE 8×8-62 fuel rod and GE 8×8 60 fuel rod fuel assembly types are also evaluated.

5.5.2 Decay Heat Limit

The maximum allowable heat load, or decay heat limit as used in the context of this chapter, is based on the overall maximum decay heat limit of 23 kW. The maximum allowable heat load on a per assembly basis is 0.958 kW.

As documented in Section 5.4.1, the SAS2H sequence of SCALE 4.3 is used to determine source term magnitudes for each fuel assembly type, initial enrichment and burnup combination. Source term in this context implies both heat load and radiation sources for both fuel and activated hardware.

5.5.3 Storage Cask and Standard Transfer Cask Dose Rate Limits and Dose Calculation Method

Storage cask and standard transfer cask surface radial surface dose rates for the design basis assemblies are presented in Tables 5.1-1 through 5.1-4. The design basis radial storage and transfer cask (dry cavity) dose rates are used as an upper bound dose rate limit for any other fuel assembly type, burnup, and enrichment combination.

To avoid the significant effort required to prepare and execute hundreds of one-dimensional cases for all fuel configurations and burnups under consideration, a unique device is employed which permits the ready calculation of dose rates at a given location using a dose rate response function. The dose rate response function for a given source type at a given detector location is a collection of values, one for each energy group, each of which gives the contribution to the dose rate at the detector location from a unit source strength in that energy group. With this response function, the dose rate, d, at the corresponding detector location is determined for any given fuel type by vector multiplying the unnormalized source spectrum, f, by the response function, r:

$$d = r \cdot f$$

The dose rate response function is computed by solving a series of one-dimensional cases, one for each energy group, with a unit source strength in each energy group. In practice, the source strength is normalized to some large value (here, 10^{10}) in order to avoid numeric underflow in the calculation.

Sample response functions for the PWR and BWR storage casks and standard transfer cask are listed in Table 5.5-3 and Table 5.5-4 for neutron and gamma sources, respectively. Only seven energy groups are presented for the fuel neutron source since the complete SAS2H neutron source is located in these energy groups.

With the dose rate response method a convenient and simple method for determining storage and transfer cask surface dose rates is available.

5.5.4 Minimum Allowable Cooling Time Determination

The following strategy is used to determine limiting cooling times for each combination of fuel type, initial enrichment, and burnup:

a) Determine decay heat and dose rate values at each cooling time step.
b) Interpolate in the resulting collection of data to find minimum cooling time required to meet each limiting value, decay heat and transfer and storage cask dose rate, individually.
c) Select the maximum of this collection of minimum required cooling times, rounded up to the next whole year, as the minimum required cooling time for this combination of burnup, enrichment and cooling time.

5.5.4.1 PWR and BWR Assembly Minimum Cooling Times

Minimum allowable cooling times are established for each of the fuel type, burnup, and enrichment combinations based on the cask decay heat limit of 23 kW and the one-dimensional dose rate limits in Table 5.5-2. A sample of the calculated cooling times required to reach each of the limits for Westinghouse 17×17 and GE 9×9 fuel assemblies at 40 GWD/MTU are shown in Tables 5.5-5 and 5.5-6, respectively. The identical calculation sequence is repeated for all the assembly types and burnups indicated in Section 5.5-1. The limiting cooling times are then collapsed to array size specific limiting values as listed in Table 5.5-7 and Table 5.5-8.

Table 5.5-1 Limiting PWR and BWR Fuel Types Based on Uranium Loading

Reactor	Array	Fuel Assembly
PWR	17×17	WE 17×17 Standard
PWR	16×16	CE 16×16 System 80
PWR	15×15	BW 15×15
PWR	14×14	WE 14×14
BWR	9×9	GE 9×9-79 Fuel Rods (GE 9×9-2L)
BWR	8×8	GE 8×8-63 Fuel Rods
BWR	7×7	GE 7×7

Table 5.5-2 Design Basis Assembly Dose Rate Limit

Configuration	Neutron	Gamma	Hardware Gamma	Total[1]
PWR Storage	0.6	22.5	11.1	34.2
BWR Storage	0.9	16.5	0.1	17.6
PWR Transfer	68.1	127.4	82.4	277.8
BWR Transfer	108.0	92.6	1.1	201.6

1. Measurements in mrem/hr.

Table 5.5-3 Radial Surface Response to Neutrons

Group	E_{avg} (MeV)	Storage Cask[1]		Standard Transfer Cask[1]	
		WE17×17	GE9×9-2L	WE17×17	GE9×9-2L
1	1.32E+01	1.4090E+07	1.4092E+07	1.6120E+09	1.5776E+09
2	4.72E+00	8.0067E+06	8.3652E+06	1.0621E+09	1.0579E+09
3	2.43E+00	6.9413E+06	7.4255E+06	9.9278E+08	1.0147E+09
4	1.63E+00	5.4135E+06	5.9542E+06	6.9518E+08	7.2993E+08
5	1.15E+00	4.7208E+06	5.2315E+06	5.1436E+08	5.4559E+08
6	6.50E-01	4.4771E+06	5.0154E+06	3.6579E+08	4.0159E+08
7	2.50E-01	3.3522E+06	3.8567E+06	1.0275E+08	1.1960E+08

1. millirem/hour per 10^{10} neutrons/second.

Table 5.5-4 Radial Surface Response to Gammas

Group	E_{avg} (MeV)	Storage Cask[1]		Standard Transfer Cask[1]	
		WE17×17	GE9×9-2L	WE17×17	GE9×9-2L
1	9.00E+00	2.1642E+05	2.0388E+05	8.8476E+04	8.4060E+04
2	7.25E+00	1.6963E+05	1.6008E+05	1.0748E+05	1.0204E+05
3	5.75E+00	1.1013E+05	1.0387E+05	1.1130E+05	1.0542E+05
4	4.50E+00	6.0237E+04	5.6658E+04	1.0110E+05	9.5390E+04
5	3.50E+00	2.8958E+04	2.7132E+04	8.0031E+04	7.5091E+04
6	2.75E+00	1.1556E+04	1.0761E+04	5.2075E+04	4.8507E+04
7	2.25E+00	4.8220E+03	4.4662E+03	2.9488E+04	2.7289E+04
8	1.83E+00	1.6449E+03	1.5139E+03	1.2645E+04	1.1617E+04
9	1.50E+00	5.2977E+02	4.8508E+02	4.3386E+03	3.9611E+03
10	1.17E+00	1.1402E+02	1.0395E+02	7.2955E+02	6.6298E+02
11	9.00E-01	1.6902E+01	1.5317E+01	4.6940E+01	4.2324E+01
12	7.00E-01	2.8772E+00	2.6068E+00	1.5996E+00	1.4442E+00
13	5.00E-01	2.5056E-01	2.2771E-01	8.5258E-04	7.7577E-04
14	3.50E-01	6.0028E-03	5.3421E-03	1.8977E-11	1.6941E-11
15	2.50E-01	2.1429E-04	1.8519E-04	1.1956E-32	1.0398E-32
16	1.50E-01	4.3116E-08	2.8984E-08	0.0000E+00	0.0000E+00
17	7.50E-02	2.5041E-35	2.2809E-36	0.0000E+00	0.0000E+00
18	3.00E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

1. millirem/hour per 10^{10} γ/second.

Table 5.5-5 Westinghouse 17×17 Minimum Cooling Time Evaluation

Enrichment (wt % ^{235}U)	Minimum Cooling Time (Years)[1]				Active Constraint
	Decay Heat	Storage Dose	Transfer Dose	Limiting	
1.9	6.3	6.0	9.5	10	Transfer Dose
2.1	6.2	5.8	8.5	9	Transfer Dose
2.3	6.1	5.7	7.7	8	Transfer Dose
2.5	6	5.6	7	7	Transfer Dose
2.7	5.9	5.5	6.5	7	Transfer Dose
2.9	5.9	5.4	6.1	7	Transfer Dose
3.1	5.8	5.3	5.8	6	Decay Heat
3.3	5.8	5.2	5.5	6	Decay Heat
3.5	5.7	5.1	5.2	6	Decay Heat
3.7	5.7	5	5	6	Decay Heat
3.9	5.6	5	5	6	Decay Heat
4.1	5.6	5	5	6	Decay Heat
4.3	5.5	5	5	6	Decay Heat
4.5	5.5	5	5	6	Decay Heat
4.7	5.4	5	5	6	Decay Heat
4.9	5.4	5	5	6	Decay Heat

1. 40,000 MWD/MTU burnup.

Table 5.5-6 GE 9×9-2L Minimum Cooling Time Evaluation

Enrichment (wt % ^{235}U)	Minimum Cooling Time (Years)[1]				Active Constraint
	Decay Heat	Storage Dose	Transfer Dose	Limiting	
1.9	5.8	5.6	14.3	15	Transfer Dose
2.1	5.7	5.5	11.7	12	Transfer Dose
2.3	5.6	5.4	9.5	10	Transfer Dose
2.5	5.5	5.3	7.9	8	Transfer Dose
2.7	5.4	5.2	6.7	7	Transfer Dose
2.9	5.3	5.2	5.9	6	Transfer Dose
3.1	5.2	5.1	5.4	6	Transfer Dose
3.3	5.2	5	5	6	Decay Heat
3.5	5.1	5	5	6	Decay Heat
3.7	5	5	5	5	Decay Heat
3.9	5	5	5	5	Decay Heat
4.1	5	5	5	5	Decay Heat
4.3	5	5	5	5	Decay Heat
4.5	5	5	5	5	Decay Heat
4.7	5	5	5	5	Decay Heat
4.9	5	5	5	5	Decay Heat

1. 40,000 MWD/MTU burnup.

Table 5.5-7 Loading Table for PWR Fuel

Minimum Initial Enrichment	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]				30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
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 < 3.9	5	5	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5	5	5

Minimum Initial Enrichment	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]				40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	7	9	14	13	9	13
2.3 ≤ E < 2.5	8	8	6	8	12	12	8	12
2.5 ≤ E < 2.7	8	7	6	7	11	11	7	11
2.7 ≤ E < 2.9	7	7	6	7	10	10	7	10
2.9 ≤ E < 3.1	7	6	6	7	9	9	7	9
3.1 ≤ E < 3.3	6	6	6	6	9	8	7	8
3.3 ≤ E < 3.5	6	6	6	6	8	8	7	8
3.5 ≤ E < 3.7	6	6	6	6	7	8	7	7
3.7 ≤ E < 3.9	6	6	6	6	7	8	7	7
3.9 ≤ E < 4.1	6	6	6	6	7	7	7	7
4.1 ≤ E < 4.3	5	6	6	6	6	7	7	7
4.3 ≤ E < 4.5	5	6	6	6	6	7	7	7
4.5 ≤ E < 4.7	5	6	6	6	6	7	6	7
4.7 ≤ E < 4.9	5	6	5	6	6	7	6	7
E ≥ 4.9	5	6	5	6	6	7	6	7

Table 5.5-8 Loading Table for BWR Fuel

Minimum Initial Enrichment	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]			30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]		
wt % ^{235}U (E)	7×7	8×8	9×9	7×7	8×8	9×9
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	6	5	6
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 < 3.9	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5

Minimum Initial Enrichment	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]			40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]		
wt % ^{235}U (E)	7×7	8×8	9×9	7×7	8×8	9×9
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	11	9	10	20	18	19
2.5 ≤ E < 2.7	9	8	8	18	16	17
2.7 ≤ E < 2.9	8	7	7	15	13	14
2.9 ≤ E < 3.1	7	6	6	13	11	12
3.1 ≤ E < 3.3	6	6	6	11	10	10
3.3 ≤ E < 3.5	6	5	6	9	8	9
3.5 ≤ E < 3.7	6	5	6	8	7	7
3.7 ≤ E < 3.9	6	5	5	7	6	7
3.9 ≤ E < 4.1	5	5	5	7	6	7
4.1 ≤ E < 4.3	5	5	5	7	6	6
4.3 ≤ E < 4.5	5	5	5	6	6	6
4.5 ≤ E < 4.7	5	5	5	6	6	6
4.7 ≤ E < 4.9	5	5	5	6	6	6
E ≥ 4.9	5	5	5	6	6	6

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5.6.1.3 Shielding Evaluation

The shielding evaluation consists of a loading table analysis of the CE 14×14 fuel following the methodology developed in Section 5.5 (Minimum Allowable Cooling Time Evaluation for PWR and BWR fuel). Fuel assemblies which include non-fuel hardware are addressed explicitly. The results of the analysis are loading tables which give the required cool time for a particular fuel configuration.

No restrictions are placed on the loading locations for any of the non-fuel assembly hardware components. This implies that a canister may contain up to 24 CEAs, 24 ICI thimbles, or 24 steel substitute rod assemblies or any combination thereof as long as the most limiting cool time is selected for any of the components in the canister. Neither CEAs or ICI thimbles may be placed into an assembly containing steel substitute rods that have received core exposure. ICI thimbles and CEAs may be inserted in fuel assemblies that also have hollow Zircaloy rods replacing burnable poison rods, solid steel rods replacing fuel rods provided there has been no reactor core exposure of the steel rods, fuel assemblies with fuel rods removed from the lattice, fuel assemblies with variable enrichment or low enrichment replacement fuel rods, or axial blanket fuel assemblies. Due to physical constraints, ICI thimbles and CEAs cannot be located in the same assembly.

5.6.1.4 Standard Fuel Source Term

Results are obtained, for CE 14×14 fuel with no additional non-fuel material included, by following the minimum allowable cooling time evaluation (loading table analysis) methodology developed in Section 5.5. CE 14×14 source terms at various combinations of initial enrichment and burnup are computed using the CE 14×14 SAS2H model described in Section 5.6.1.1.

Following the methodology developed in Section 5.5, one-dimensional shielding calculations are performed for CE 14×14 fuel region sources at various combinations of initial enrichment, burnup, and cool time. The resulting dose rate and source term data is interpolated to determine the cool time required for each combination of enrichment and burnup to decay below the design basis limiting values of dose and heat generation rate.

The resulting loading table for CE 14×14 fuel with no additional non-fuel material is shown in Table 5.6.1-10.

In addition to the standard fuel evaluation, a preferential loading strategy is analyzed. The preferential loading configuration relies on placing higher heat load fuel assemblies on the periphery of the basket than would be allowed with a uniform loading strategy. Peripheral loadings are evaluated with decay heats of up to 1.05 kW per peripheral assembly. To maintain the maximum allowable heat load per basket of 23 kW, the maximum allowable per assembly heat load in the interior location of the basket is reduced to compensate for the higher heat load peripheral elements. Burnup and cool time combinations for peripheral and interior assemblies are listed in Table 5.6.1-10 as a function of initial enrichment. The cool time column for peripheral element and interior assembly loading is indicated by the "P" and "I" indicators in the column headings.

5.6.1.4.1 Control Element Assemblies (CEA)

The result of the analysis is a set of loading tables for Maine Yankee fuel giving the cool time required for a fuel assembly with a specified burnup and enrichment combination to contain a design basis CEA with a cool time of 5, 10, 15, or 20 years. Fuel assemblies containing CEAs will be loaded into Class 2 canisters, which are slightly longer than the Class 1 canisters used for bare fuel assemblies. The additional length is required to accommodate the CEA, which is inserted in the top of the fuel assembly.

The approach taken is to compute downward adjustments to the design basis one-dimensional dose rate limiting value for the storage cask (as specified in Table 5.5-3) which ensures that the fuel sources have decayed adequately to cover the effect of the additional source added as a result of CEA containment. The adjustment is determined on the basis of a conservative comparison of three-dimensional shielding analysis results for the original Class 1 canister containing CE 14×14 fuel assemblies and the Class 2 canister containing either no CEA or CEAs cooled to 5, 10, 15, or 20 years. Results for CEA cool times longer than 20 years are bounded by the 20 year results.

Assuming design basis CE 14×14 fuel with a burnup of 40,000 MWD/MTU, 3.7 wt % ^{235}U enrichment and a 5-year cool time, the additional CEA source results in a localized peak near the bottom of the transfer cask that results in a surface dose rate that is less than 500 mrem/hr. Since this is comparable to the no-CEA case, it is not necessary to extend cool time of fuel assemblies with CEAs inserted to account for an increased transfer cask surface dose.

5.6.1.4.1.1 Establishment of Limiting Values

Since the additional activated material in the CEA analysis is assumed present in the lower 8 inches of the active fuel source region, the one-dimensional dose methodology is not appropriate to address the additional source term due to its small axial extent. The one-dimensional analysis is based on the response from the full-length fuel region source. To account for the additional source, the one-dimensional normal conditions dose rate limit is adjusted by an amount that ensures that the contribution from the additional activated material is bounded.

By adjusting the one-dimensional dose rate limit, we require the fuel to cool to a point where the decrease in fuel region dose rate matches the increased dose rate due to the additional CEA material. Hence, it is necessary to determine the amount by which the dose rate increases as a result of the added material. A one-dimensional calculation of this additional dose rate is not reasonable due to the small axial extent of the CEA source. One-dimensional buckling corrections are inaccurate for a cylindrical source where the ratio of height to diameter of the source is less than unity, as is the case here.

Instead, the additional contribution to dose rate due to the activated material is computed by a detailed three-dimensional shielding model. The model is based on the three-dimensional models described in Section 5.3. However, the fuel is modeled in a Class 2 canister since that canister will be used to store/transfer CEA-bearing assemblies.

The three-dimensional shielding evaluation is conducted for the CE 14×14 fuel at a burnup of 40,000 MWD/MTU and initial enrichment of 3.7 wt % ^{235}U. According to the cool time analysis conducted for PWR fuels in Table 5.6.1-10, this fuel will require a 5-year cool time before it is acceptable for transfer or storage in the UMS® vertical concrete cask. Hence, the 5-year cooled CE 14×14 at 40,000 MWD/MTU and 3.7 wt % ^{235}U initial enrichment provides the base case for the dose rate limit adjustment calculation.

Additional three-dimensional models are defined based on the base case fuel configuration in a Class 2 canister and either containing a design basis CEA assumed to be cooled for 5, 10, 15, or 20 years or containing no CEA at all (no CEA case below).

5.6.1.4.1.2 Three-Dimensional Model Results

Table 5.6.1-11 gives the three-dimensional UMS® vertical concrete cask and transfer bottom model results for each case. Only the bottom model is considered because the top model is not sensitive to changes in the CEA description. The parameter Delta shown in the table is the difference between the base case maximum (from Table 5.5-3 for the storage cask) dose rate and the value computed for each remaining case. This quantity is directly applied to the one-dimensional design basis normal conditions dose rate limit, as specified in Table 5.6.1-11 for the storage cask to determine a modified limiting value applicable to each CEA decay case. The resulting dose rate limits are shown in the "Limit" column of the table.

Note that direct application of the "Delta" to the one-dimensional dose rate limit is somewhat conservative. The three-dimensional maximum dose rate results are significantly higher than the one-dimensional results, hence a given difference between three-dimensional results represents a larger percentage of the corresponding one-dimensional results.

Also note that the dose rate delta for the "No-CEA" case in Table 5.6.1-11 is zero. Unlike the UMS® transport cask, where a spacer positions the canister in the cask, the UMS® standard transfer and storage casks are extended to accommodate the longer Class 2 canister. These cask extensions maintain the spacing of the fuel assembly with respect to the points of minimum shielding in the bottom cask model, and thereby result in identical cask bottom half dose rates for fuel assemblies in Class 1 and Class 2 canisters.

5.6.1.4.1.3 Decay Heat Limits

As discussed in Section 5.6.1.1.1, the additional decay heat associated with a full cask of CEAs is conservatively taken as 0.35 kW/cask. This additional heat load is accounted for by reducing the fuel assembly decay heat limit to 22.65 kW/cask.

5.6.1.4.1.4 Loading Table Analysis

With the adjusted one-dimensional dose and heat generation rate limits established above, the loading table analysis proceeds following the methodology developed in Section 5.5. Each combination of initial enrichment and burnup is analyzed to determine the minimum required cool time in order for an assembly to either 1) contain a design basis CEA cooled 5, 10, 15, or 20

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU – Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	7	7	5
2.1 ≤ E < 2.3	6	6	5
2.3 ≤ E < 2.5	6	6	5
2.5 ≤ E < 2.7	5	6	5
2.7 ≤ E < 2.9	5	6	5
2.9 ≤ E < 3.1	5	6	5
3.1 ≤ E < 3.3	5	6	5
3.3 ≤ E < 3.5	5	6	5
3.5 ≤ E < 3.7	5	6	5
3.7 ≤ E ≤ 4.2	5	6	5

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern, interior basket locations: allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable (Continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	11	11	6
2.1 ≤ E < 2.3	9	9	6
2.3 ≤ E < 2.5	8	8	6
2.5 ≤ E < 2.7	7	7	6
2.7 ≤ E < 2.9	7	7	6
2.9 ≤ E < 3.1	6	7	6
3.1 ≤ E < 3.3	6	7	5
3.3 ≤ E < 3.5	6	7	5
3.5 ≤ E < 3.7	6	7	5
3.7 ≤ E ≤ 4.2	6	7	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	Not allowed	Not allowed	7
2.1 ≤ E < 2.3	Not allowed	Not allowed	7
2.3 ≤ E < 2.5	Not allowed	Not allowed	7
2.5 ≤ E < 2.7	Not allowed	Not allowed	7
2.7 ≤ E < 2.9	Not allowed	Not allowed	7
2.9 ≤ E < 3.1	Not allowed	Not allowed	7
3.1 ≤ E < 3.3	Not allowed	Not allowed	7
3.3 ≤ E < 3.5	Not allowed	Not allowed	6
3.5 ≤ E < 3.7	Not allowed	Not allowed	6
3.7 ≤ E ≤ 4.2	Not allowed	Not allowed	6

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern, interior basket locations: allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

Table 5.6.1-11 Three-Dimensional Shielding Analysis Results for Various Maine Yankee CEA Configurations Establishing One-Dimensional Dose Rate Limits for Loading Table Analysis

CEA Cool Time [years]	Dose Rate [mrem/hr]	FSD	Delta [mrem/hr]	Limit [mrem/hr]
Class 1 Result	32.0	0.85%	-	34.2
No CEA	32.0	0.85%	-0.0	34.2
05y	43.8	0.59%	-11.8	22.4
10y	33.1	0.69%	-1.1	33.1
15y	32.0	0.85%	-0.0	34.2
20y	32.0	0.85%	-0.0	34.2

Table 5.6.1-12 Loading Table for Maine Yankee CE 14 × 14 Fuel Containing CEA Cooled to Indicated Time

Enrichment	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	6	5	5	5
2.7 ≤ E < 2.9	5	6	5	5	5
2.9 ≤ E < 3.1	5	6	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	7	7	7	7	7
2.7 ≤ E < 2.9	7	7	7	7	7
2.9 ≤ E < 3.1	6	6	6	6	6
3.1 ≤ E < 3.3	6	6	6	6	6
3.3 ≤ E < 3.5	6	6	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6

Note: The NoCEA (Class 2) column is provided for comparison. Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

Table 5.6.1-13 Establishment of Dose Rate Limit for Maine Yankee ICI Thimble Analysis

Case	Top Model	
	Rate (mrem/hr)	FSD
No ICI Thimble	33.3	1.4%
4 Year Cooled ICI Thimble	33.3	1.4%
Delta	0.0	

Table 5.6.1-14 Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assembly Number	Burnup [MWD/MTU]	Enrichment [wt %]	SSR Source [g/s/assy]	Cool Time [years]	Earliest Loadable	Loading Configuration
N420	45,000	3.3	2.1602E+13	6	Jan 2001	Standard
N842	35,000	3.3	3.1396E+12	5	Jan 2001	Standard
N868	40,000	3.3	5.2444E+12	5	Jan 2001	Standard
R032	45,000	3.5	1.4550E+13	6	Jan 2002	Standard
R439	50,000	3.5	1.3998E+13	7	Jan 2003	Standard
R444	50,000	3.5	5.5993E+13	8	Jan 2004	Standard
R439	50,000	3.5	1.3998E+13	6	Jan 2002	Pref(1.050)
R444	50,000	3.5	5.5993E+13	6	Jan 2002	Pref(1.050)

Table 5.6.1-15 Maine Yankee Consolidated Fuel Model Parameters

Lattice	Assy	Num Rods	Actual Burnup [MWD/MTU]	Actual Enrichment [wt %]	Modeled Burnup [MWD/MTU]	Modeled Enrichment [wt %]	Required Cool Time [y]	Cool Time 1/1/01 [y]
CN-1	EF0039	172	5150	1.929	30000	1.9	6	26
CN-10	EF0045	176	17150	1.953	30000	1.9	6	24
	EF0046	107	17150	1.953	30000	1.9	6	24

Table 5.6.1-16 Maine Yankee Source Rate Analysis for CN-10 Consolidated Fuel Lattice

Cool Time [years]	Num Rods Present	Decay Heat [kW/cask]	Fuel Neutron [n/s/assy]	Fuel Gamma [g/sec/assy]	Fuel Hardware [g/sec/assy]
6	176	13.9	1.63E+08	3.16E+15	9.28E+12
24	283	7.42	8.41E+07	1.28E+15	8.67E+11
Src Ratio 24/6		0.86	0.83	0.65	0.15

Table 5.6.1-17 Additional Maine Yankee Non-Fuel Hardware Characterization – Non-Neutron Sources

Non Fuel Material	Waste Volume [ft^3]	Total Curies	Co-60 Curies
Sb-Be Source 1H1	0.020	4.15E+02	2.22E+02
Sb-Be Source 6H4	0.020	4.32E+02	2.31E+02
CEA Tips	0.100	1.06E+02	8.90E+01
ICI	0.007	2.82E+01	1.76E+01

Chapter 12

Spent fuel having a burnup from 45,000 to 50,000 MWD/MTU is assigned to peripheral locations, and may require loading in a Maine Yankee fuel can. 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 for the basket and canister.

The Fuel Assembly Limits for the Maine Yankee SITE SPECIFIC FUEL are shown in Table 12B2-7. Part A of the table lists the STANDARD, INTACT FUEL ASSEMBLY and SITE SPECIFIC FUEL that does not require preferential loading.

Part B of the table lists the SITE SPECIFIC FUEL configurations that require preferential loading due to the criticality, shielding or thermal evaluation. The loading pattern for Maine Yankee SITE SPECIFIC FUEL that must be preferentially loaded is presented in Section B 2.1.2. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation.

Fuel assemblies with a Control Element Assembly (CEA) or a CEA plug inserted are loaded in a Class 2 canister and basket due to the increased length of the assembly with either of these components installed. However, these assemblies are not restricted as to loading position within the basket.

The Transportable Storage Canister loading procedures for Maine Yankee SITE SPECIFIC FUEL are administratively controlled in accordance with the requirements of Section B 2.1.3 for the loading of: (1) a fuel configuration with removed fuel or poison rods, (2) a MAINE YANKEE FUEL CAN, or (3) fuel with burnup between 45,000 MWD/MTU and 50,000 MWD/MTU.

Table 12-1 NAC-UMS® System Controls and Limits

Control or Limit	Applicable Technical Specification	Condition or Item Controlled
1. Fuel Characteristics	Table 12B2-1	Type and Condition
	Table 12B2-2	Class, Dimensions and Weight for PWR
	Table 12B2-3	Class, Dimensions and Weight for BWR
	Table 12B2-4	Minimum Cooling Time for PWR Fuel
	Table 12B2-5	Minimum Cooling Time for BWR Fuel
	Table 12B2-7	Maine Yankee Site Specific Fuel Limits
	Table 12B2-8	Minimum Cooling Time for Maine Yankee Fuel – No CEA
	Table 12B2-9	Minimum Cooling Time for Maine Yankee Fuel – With CEA
2. Canister	LCO 3.1.4	Time in Transfer Cask (fuel loading)
Fuel Loading	Table 12B2-1	Weight and Number of Assemblies
	Table 12B2-7	Maine Yankee Site Specific Fuel Limits
	Table 12B2-4	Minimum Cooling Time for PWR Fuel
	Table 12B2-5	Minimum Cooling Time for BWR Fuel
Drying	LCO 3.1.2	Vacuum Drying Pressure
Backfilling	LCO 3.1.3	Helium Backfill Pressure
Sealing	LCO 3.1.5	Helium Leak Rate
Vacuum	LCO 3.1.1	Time in Vacuum Drying
External Surface	LCO 3.2.1	Level of Contamination
Unloading	Note 1	Fuel Cooldown Requirement
3. Concrete Cask	LCO 3.2.2	Surface Dose Rates
	Note 1	Cask Spacing
	Note 2	Cask Handling Height
4. Surveillance	LCO 3.1.6	Heat Removal System
5. Transfer Cask	12B 3.4(8)	Minimum Temperature
6. ISFSI Concrete Pad	B3.4.1(6) B3.4.2(7)	Seismic Event Performance

1. Procedure and/or limits are presented in the Operating Procedures of Chapter 8.
2. Lifting height and handling restrictions are provided in Section A5.6 of Appendix 12A.

APPENDIX 12A

TECHNICAL SPECIFICATIONS FOR THE NAC-UMS® SYSTEM

AMENDMENT 2

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A 1.0 USE AND APPLICATION

A 1.1 Definitions

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout this Chapter 12.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CANISTER	See TRANSPORTABLE STORAGE CANISTER
CANISTER HANDLING FACILITY	The CANISTER HANDLING FACILITY includes the following components and equipment: (1) a canister transfer station that allows the staging of the TRANSFER CASK with the CONCRETE CASK or transport cask to facilitate CANISTER lifts involving spent fuel handling not covered by 10 CFR 50; and (2) either a stationary lift device or mobile lifting device used to lift the TRANSFER CASK and CANISTER.
CONCRETE CASK	See VERTICAL CONCRETE CASK
CONSOLIDATED FUEL	A nonstandard fuel configuration in which the individual fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is similar to a fuel assembly. CONSOLIDATED FUEL is stored in a MAINE YANKEE FUEL CAN.

(continued)

DAMAGED FUEL	A fuel assembly or fuel rod with known or suspected cladding defects greater than pinhole leaks or hairline cracks. DAMAGED FUEL must be placed in a MAINE YANKEE FUEL CAN.
FUEL DEBRIS	An intact or a partial fuel rod or an individual intact or partial fuel pellet not contained in a fuel rod. Fuel debris is inserted into a 9 × 9 array of tubes in a lattice that has approximately the same dimensions as a standard fuel assembly. FUEL DEBRIS is stored in a MAINE YANKEE FUEL CAN.
HIGH BURNUP FUEL	A fuel assembly having a burnup between 45,000 and 50,000 MWD/MTU, which must be preferentially loaded in peripheral positions of the basket.
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within the perimeter fence licensed for storage of spent fuel within NAC-UMS® SYSTEMs (see also 10 CFR 72.3).
INITIAL PEAK PLANAR-AVERAGE ENRICHMENT	THE INITIAL PEAK PLANAR-AVERAGE ENRICHMENT is the maximum planar-average enrichment at any height along the axis of the fuel assembly. The 4.0 wt % ^{235}U enrichment limit for BWR fuel applies along the full axial extent of the assembly. The INITIAL PEAK PLANAR-AVERAGE ENRICHMENT may be higher than the bundle (assembly) average enrichment.

(continued)

INTACT FUEL (ASSEMBLY OR ROD) (Undamaged Fuel)	A fuel assembly or fuel rod with no fuel rod cladding defects, or with known or suspected fuel rod cladding defects not greater than pinhole leaks or hairline cracks.
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a NAC-UMS® SYSTEM while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the CANISTER and end when the NAC-UMS® SYSTEM is secured on the transporter. LOADING OPERATIONS does not include post-storage operations, i.e., CANISTER transfer operations between the TRANSFER CASK and the CONCRETE CASK or transport cask after STORAGE OPERATIONS.
MAINE YANKEE FUEL CAN	A specially designed stainless steel screened can sized to hold INTACT FUEL, CONSOLIDATED FUEL, DAMAGED FUEL or FUEL DEBRIS. The screens preclude the release of gross particulate from the can into the canister cavity. The MAINE YANKEE FUEL CAN may only be loaded in a Class 1 canister.

(continued)

NAC-UMS® SYSTEM	NAC-UMS® SYSTEM includes the components approved for loading and storage of spent fuel assemblies at the ISFSI. The NAC-UMS® SYSTEM consists of a CONCRETE CASK, a TRANSFER CASK, and a CANISTER.
OPERABLE	The CONCRETE CASK heat removal system is OPERABLE if the difference between the ISFSI ambient temperature and the average outlet air temperature is ≤ 102°F for the PWR CANISTER or ≤ 92°F for the BWR CANISTER.
SITE SPECIFIC FUEL	Spent fuel configurations that are unique to a site or reactor due to the addition of other components or reconfiguration of the fuel assembly at the site. It includes fuel assemblies, which hold nonfuel-bearing components, such as a control element assembly, a burnable poison rod insert, an in-core instrument thimble or a flow mixer, or which are modified as required by expediency in reactor operations, research and development or testing. Modification may consist of individual fuel rod removal, fuel rod replacement of similar or dissimilar material or enrichment, the installation, removal or replacement of burnable poison rods, or containerizing damaged fuel. Site specific fuel includes irradiated fuel assemblies designed with variable enrichments and/or axial blankets, fuel that is consolidated and fuel that exceeds design basis fuel parameters.

(continued)

STANDARD FUEL	Irradiated fuel assemblies having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For PWR fuel, a flow mixer, an in-core instrument thimble or a burnable poison rod insert is considered to be a component of standard fuel. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the STANDARD FUEL configuration.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI, while an NAC-UMS® SYSTEM containing spent fuel is located on the storage pad within the ISFSI perimeter.
TRANSFER CASK	TRANSFER CASK is a shielded lifting device that holds the CANISTER during LOADING and UNLOADING OPERATIONS and during closure welding, vacuum drying, leak testing, and non-destructive examination of the CANISTER closure welds. The TRANSFER CASK is also used to transfer the CANISTER into and from the CONCRETE CASK and into the transport cask. TRANSFER CASK refers to either the standard or advanced transfer cask.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involved in transferring a loaded CANISTER from a CONCRETE CASK to another CONCRETE CASK or to a TRANSPORT CASK.

(continued)

TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities involved in moving a loaded NAC-UMS® CONCRETE CASK and CANISTER to and from the ISFSI. TRANSPORT OPERATIONS begin when the NAC-UMS® SYSTEM is first secured on the transporter and end when the NAC-UMS® SYSTEM is at its destination and no longer secured on the transporter.
TRANSPORTABLE STORAGE CANISTER (CANISTER)	TRANSPORTABLE STORAGE CANISTER is the sealed container that consists of a tube and disk fuel basket in a cylindrical canister shell that is welded to a baseplate, shield lid with welded port covers, and structural lid. The CANISTER provides the confinement boundary for the confined spent fuel.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a NAC-UMS® SYSTEM to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the NAC-UMS® SYSTEM is no longer secured on the transporter and end when the last fuel assembly is removed from the NAC-UMS® SYSTEM.
VERTICAL CONCRETE CASK (CONCRETE CASK)	VERTICAL CONCRETE CASK is the cask that receives and holds the sealed CANISTER. It provides the gamma and neutron shielding and convective cooling of the spent fuel confined in the CANISTER.

2. The time duration from the end of 24 hours of in-pool cooling or of forced air cooling of the CANISTER through completion of vacuum dryness testing and the introduction of helium backfill shall not exceed the following limits:

PWR Forced Air		PWR In-Pool	
Total Heat Load (L) (kW)	Time Limit (Hours)	Total Heat Load (L) (kW)	Time Limit (Hours)
20 < L ≤ 23	3	20 < L ≤ 23	12
17.6 < L ≤ 20	6	17.6 < L ≤ 20	15
14 < L ≤ 17.6	9	14 < L ≤ 17.6	18
11 < L ≤ 14	16	11 < L ≤ 14	24
8 < L ≤ 11	27	8 < L ≤ 11	36
L ≤ 8	78	L ≤ 8	87

BWR Forced Air		BWR In-Pool	
Total Heat Load (L) (kW)	Time Limit (Hours)	Total Heat Load (L) (kW)	Time Limit (Hours)
20 < L ≤ 23	2	20 < L ≤ 23	10
17 < L ≤ 20	3	17 < L ≤ 20	11
14 < L ≤ 17	8	14 < L ≤ 17	17
11 < L ≤ 14	18	11 < L ≤ 14	26
8 < L ≤ 11	41	8 < L ≤ 11	52

APPLICABILITY: During LOADING OPERATIONS

(continued)

CANISTER Maximum Time in Vacuum Drying
A 3.1.1

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO time limits not met	A.1 Commence filling CANISTER with helium	2 hours
	AND	
	A.2.1.1 Place TRANSFER CASK with helium filled loaded CANISTER in spent fuel pool.	2 hours
	AND	
	A.2.1.2 Maintain TRANSFER CASK and CANISTER in spent fuel pool for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS
	OR	
	A.2.2.1 Commence supplying air to the TRANSFER CASK annulus fill/drain lines at a rate of 375 CFM and a maximum temperature of 76°F	2 hours
	AND	
	A.2.2.2 Maintain airflow for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Monitor elapsed time from completion of CANISTER draining operations until start of helium backfill	Once within 1 hour of completion of CANISTER draining AND 2 hours thereafter.
SR 3.1.1.2	Monitor elapsed time from the end of in-pool cooling or of forced-air cooling until restart of helium backfill	Once within 1 hour of completion of in-pool cooling or forced-air cooling AND 2 hours thereafter.

A 3.1 NAC-UMS® SYSTEM Integrity

A 3.1.4 CANISTER Maximum Time in TRANSFER CASK

LCO 3.1.4 The following limits for CANISTER time in TRANSFER CASK shall be met, as appropriate:

1. The time duration from completion of backfilling the CANISTER with helium through completion of the CANISTER transfer operation from the TRANSFER CASK to the CONCRETE CASK shall not exceed the following time limits for PWR and BWR fuel.

Total PWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	20
$L \leq 20$	600

Total BWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	16
$17 < L \leq 20$	30
$L \leq 17$	600

2. The time duration from completion of in-pool or external forced air cooling of the CANISTER through completion of the CANISTER transfer operation from the TRANSFER CASK to the CONCRETE CASK shall not exceed the following time limits for PWR and BWR fuel after 24 hours of in-pool cooling or forced air cooling.

Total PWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	20
$L \leq 20$	600

Total BWR Heat Load (L) (kW)	Time Limit (Hours)
$20 < L \leq 23$	16
$17 < L \leq 20$	30
$L \leq 17$	600

3. The time duration for holding a loaded and closed CANISTER, with the structural lid installed, in the TRANSFER CASK with forced air cooling in operation shall not exceed 600 hours for PWR and BWR fuel heat loads less than or equal to 23 kW.

APPLICABILITY: During LOADING OPERATIONS and TRANSFER OPERATIONS

(continued)

CANISTER Maximum Time in TRANSFER CASK
A 3.1.4

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO time limits not met	A.1.1	Place TRANSFER CASK with helium filled loaded CANISTER in spent fuel pool	2 hours
	AND		
	A.1.2	Maintain TRANSFER CASK and CANISTER in spent fuel pool for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS
	OR		
	A.2.1	Commence supplying air to the TRANSFER CASK annulus fill/drain lines at a rate of 375 CFM and a maximum temperature of 76°F	2 hours
	AND		
	A.2.2	Maintain airflow for a minimum of 24 hours	Prior to restart of LOADING OPERATIONS
	OR		
	A.3.1	Load CANISTER into operable CONCRETE CASK	24 hours
		OR	
	A.3.2	Load CANISTER into operable TRANSPORT CASK	24 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.4.1	Monitor elapsed time from completion of helium backfill until completion of transfer of loaded CANISTER into CONCRETE CASK	Once at completion of helium backfill AND 4 hours thereafter
SR 3.1.4.2	Monitor elapsed time from completion of in-pool or forced-air cooling until completion of transfer of loaded CANISTER into CONCRETE CASK	Once at completion of cooling operations AND 4 hours thereafter
SR 3.1.4.3	Monitor elapsed time from closing of the TRANSFER CASK shield doors until unloading of the CANISTER from the TRANSFER CASK. AND Monitor continuous forced air cooling operation until unloading of the CANISTER from the TRANSFER CASK	Once at closing of the TRANSFER CASK shield doors AND 24 hours thereafter Once at start of cooling operations AND 4 hours thereafter

A 3.1 NAC-UMS® SYSTEM Integrity

A 3.1.5 CANISTER Helium Leak Rate

LCO 3.1.5 There shall be no indication of a helium leak at a test sensitivity of 1×10^{-7} cm^3/sec (helium) through the CANISTER shield lid to CANISTER shell confinement weld to demonstrate a helium leak rate equal to or less than 2×10^{-7} cm^3/sec (helium).

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER helium leak rate limit not met	A.1 Establish CANISTER helium leak rate within limit	25 days
B. Required Action and associated Completion Time not met	B.1 Remove all fuel assemblies from the NAC-UMS® SYSTEM	5 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify CANISTER helium leak rate is within limit	Once prior to TRANSPORT OPERATIONS.

A 3.1 NAC-UMS® SYSTEM

A 3.1.6 CONCRETE CASK Heat Removal System

LCO 3.1.6 The CONCRETE CASK Heat Removal System shall be OPERABLE.

APPLICABILITY: During STORAGE OPERATIONS

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK Heat Removal System inoperable	A.1 Restore CONCRETE CASK Heat Removal System to OPERABLE status	8 hours
B. Required Action A.1 and associated Completion Time not met	B.1 Perform SR 3.1.6.1 AND B.2 Restore CONCRETE CASK Heat Removal System to OPERABLE status	Immediately and every 6 hours thereafter 12 hours

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify the difference between the average CONCRETE CASK air outlet temperature and ISFSI ambient temperature is ≤ 102°F (for the PWR CANISTER) and ≤ 92°F (for the BWR CANISTER)	24 hours

A 3.2 NAC-UMS® SYSTEM Radiation Protection

A 3.2.1 CANISTER Surface Contamination

LCO 3.2.1 Removable contamination on the accessible exterior surfaces of the CANISTER or accessible interior surfaces of the TRANSFER CASK shall each not exceed:

a. 10,000 dpm/100 cm^2 from beta and gamma sources; and

b. 100 dpm/100 cm^2 from alpha sources.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER or TRANSFER CASK removable surface contamination limits not met	A.1 Restore CANISTER and TRANSFER CASK removable surface contamination to within limits	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify that the removable contamination on the accessible exterior surfaces of the CANISTER is within limits	Once, prior to TRANSPORT OPERATIONS
SR 3.2.1.2	Verify that the removable contamination on the accessible interior surfaces of the TRANSFER CASK does not exceed limits	Once, prior to TRANSPORT OPERATIONS

A 3.2 NAC-UMS® SYSTEM Radiation Protection

A 3.2.2 CONCRETE CASK Average Surface Dose Rates

LCO 3.2.2 The average surface dose rates of each CONCRETE CASK shall not exceed the following limits unless required ACTIONS A.1 and A.2 are met.

a. 50 mrem/hour (neutron + gamma) on the side (on the concrete surfaces);

b. 50 mrem/hour (neutron + gamma) on the top;

c. 100 mrem/hour (neutron + gamma) at air inlets and outlets.

APPLICABILITY: During STORAGE OPERATIONS

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK average surface dose rate limits not met	A.1 Administratively verify correct fuel loading AND	24 hours

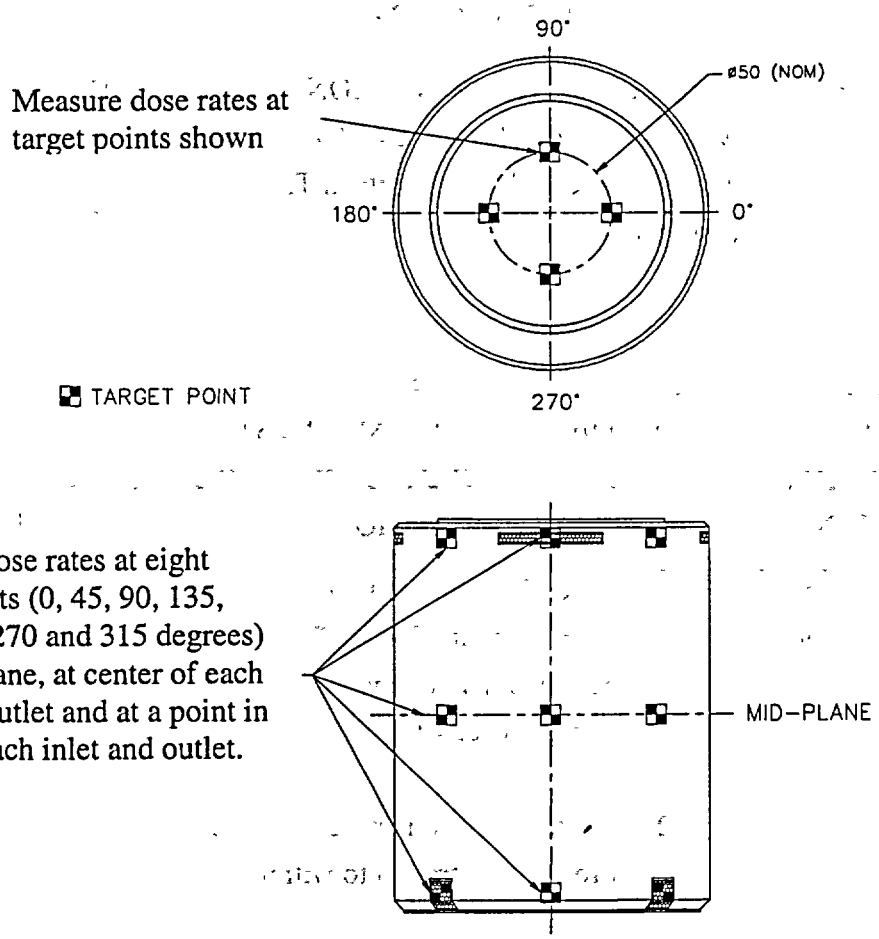
(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Perform analysis to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR 20 and 10 CFR 72	7 days
B. Required Action and associated Completion Time not met.	B.1 Remove all fuel assemblies from the NAC-UMS® SYSTEM	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify average surface dose rates of CONCRETE CASK loaded with a CANISTER containing fuel assemblies are within limits. Dose rates shall be measured at the locations shown in Figure 12A3-1.	Once after completion of transfer of CANISTER into CONCRETE CASK and prior to beginning STORAGE OPERATIONS.

Figure 12A3-1 CONCRETE CASK Surface Dose Rate Measurement



A 3.3 NAC-UMS® SYSTEM Criticality Control

A 3.3.1 Dissolved Boron Concentration

LCO 3.3.1 The dissolved Boron concentration in the water in the CANISTER cavity shall be ≥ 1,000 ppm.

APPLICABILITY: During LOADING OPERATIONS and UNLOADING OPERATIONS with water and at least one fuel assembly in the CANISTER that exceeds the enrichment limits in Table 12B2-2 for fuel assemblies taking no boron credit.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Dissolved Boron concentration not met.	A.1 Suspend loading of fuel assemblies into CANISTER and any other actions that increase reactivity.	Immediately
	AND	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately
	AND	
	A.3 Remove all fuel assemblies from the CANISTER.	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	Verify the dissolved Boron concentration is met using two independent measurements.	Within 4 hours prior to commencing LOADING or UNLOADING OPERATIONS. AND Every 48 hours thereafter while the CANISTER is in the spent fuel pool or while water is in the CANISTER, except when no water is being introduced into the CANISTER cavity.

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A 5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

A 5.1 Training Program

A training program for the NAC-UMS® Universal Storage System shall be developed under the general licensee's systematic approach to training (SAT). Training modules shall include comprehensive instructions for the operation and maintenance of the NAC-UMS® Universal Storage System and the independent spent fuel storage installation (ISFSI).

A 5.2 Pre-Operational Testing and Training Exercises

A dry run training exercise on loading, closure, handling, unloading, and transfer of the NAC-UMS® Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the CANISTER. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the CONCRETE CASK into its designated loading area
b. Moving the TRANSFER CASK containing the empty CANISTER into the spent fuel pool
c. Loading one or more dummy fuel assemblies into the CANISTER, including independent verification
d. Selection and verification of fuel assemblies requiring preferential loading
e. Installing the shield lid
f. Removal of the TRANSFER CASK from the spent fuel pool
g. Closing and sealing of the CANISTER to demonstrate pressure testing, vacuum drying, helium backfilling, welding, weld inspection and documentation, and leak testing
h. TRANSFER CASK movement through the designated load path
i. TRANSFER CASK installation on the CONCRETE CASK
j. Transfer of the CANISTER to the CONCRETE CASK

(continued)

A 5.2 Pre-Operational Testing and Training Exercises (continued)

k. CONCRETE CASK shield plug and lid installation
l. Transport of the CONCRETE CASK to the ISFSI
m. CANISTER unloading, including reflooding and weld removal or cutting
n. CANISTER removal from the CONCRETE CASK

Appropriate mockup fixtures may be used to demonstrate and/or to qualify procedures, processes or personnel in welding, weld inspection, vacuum drying, helium backfilling, leak testing and weld removal or cutting.

A 5.3 Special Requirements for the First System Placed in Service

The heat transfer characteristics and performance of the NAC-UMS® SYSTEM will be recorded by air inlet and outlet temperature measurements of the first system placed in service with a heat load equal to or greater than 10 kW. A letter report summarizing the results of the measurements will be submitted to the NRC in accordance with 10 CFR 72.4 within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® SYSTEM heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® SYSTEMs that are subsequently loaded, provided that the performance of the first system placed in service with a heat load ≥ 10 kW is demonstrated by the comparison of the calculated and measured temperatures.

A 5.4 Surveillance After an Off-Normal, Accident, or Natural Phenomena Event

A Response Surveillance is required following off-normal, accident or natural phenomena events. The NAC-UMS® SYSTEMs in use at an ISFSI shall be inspected within 4 hours after the occurrence of an off-normal, accident or natural phenomena event in the area of the ISFSI. This inspection must specifically verify that all the CONCRETE CASK inlets and outlets are not blocked or obstructed. At least one-half of the inlets and outlets on each CONCRETE CASK must be cleared of blockage or debris within 24 hours to restore air circulation.

The CONCRETE CASK and CANISTER shall be inspected if they experience a drop or a tipover.

(continued)

A 5.5 Radioactive Effluent Control Program

The program implements the requirements of 10 CFR 72.126.

a. The NAC-UMS® SYSTEM does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. LCO 3.1.5, CANISTER Helium Leak Rate, provides assurance that there are no radioactive effluents from the NAC-UMS® SYSTEM.

b. This program includes an environmental monitoring program. Each general license user may incorporate NAC-UMS® SYSTEM operations into their environmental monitoring program for 10 CFR Part 50 operations.

A 5.6 NAC-UMS® SYSTEM Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or CONCRETE CASK using devices that are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or CONCRETE CASK is in the fuel building or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.).

Pursuant to 10 CFR 72.212, this program shall evaluate the site specific transport route conditions.

(continued)

A 5.6 NAC-UMS® SYSTEM Transport Evaluation Program (continued)

a. The lift height above the transport surface prescribed in Section B3.4.1(6) of Appendix B to Certificate of Compliance (CoC) No. 1015 shall not exceed the limits in Table 12A5-1. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than those prescribed for the reference pad surface which forms the basis for the values cited in the NAC-UMS® FSAR, Sections 11.2.12.3 and 11.2.15.1.1.

b. For site specific transport conditions which are not bounded by the surface characteristics in Section B3.4.1(6) of Appendix B to CoC No. 1015, the program may evaluate the site specific conditions to ensure that the impact loading due to design basis drop events does not exceed 60g. This alternative analysis shall be commensurate with the drop analyses described in the Safety Analysis Report for the NAC-UMS® SYSTEM. The program shall ensure that these alternative analyses are documented and controlled.

c. The TRANSFER CASK and CONCRETE CASK may be lifted to those heights necessary to perform cask handling operations, including CANISTER transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section B3.5 of Appendix B to CoC No. 1015, as applicable.

A 5.7 Control of Boron Concentration in Water in the CANISTER and in the Spent Fuel Pool During Loading or Unloading

The criticality analysis shows that PWR fuel with certain combinations of initial enrichment and fuel content require credit for the presence of at least 1,000 parts per million of boron in solution in the water in the CANISTER (see Section B3.2.1 for the requirements for the pool soluble boron concentration during loading). This water must be used to flood the canister cavity during underwater PWR fuel loading or unloading. The boron in the pool water ensures sufficient thermal neutron absorption to preserve criticality control during fuel loading in the basket. Consequently, if boron credit is required for the fuel being loaded or unloaded, the canister must be flooded with water that contains boron in the proper concentration in accordance with the requirements of LCO 3.3.1. Concentration of boron must also be measured and maintained in accordance with LCO 3.3.1.

Table 12A5-1 TRANSFER CASK and CONCRETE CASK Lifting Requirements

Item	Orientation	Lifting Height Limit
TRANSFER CASK	Horizontal	None Established
TRANSFER CASK	Vertical	None Established[1]
CONCRETE CASK	Horizontal	Not Permitted
CONCRETE CASK	Vertical	< 24 inches

Note:

1. See Technical Specification A5.6(c).

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APPENDIX 12B

APPROVED CONTENTS AND DESIGN FEATURES FOR THE NAC-UMS® SYSTEM

AMENDMENT 2

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B 2.0 APPROVED CONTENTS

B 2.1 Fuel Specifications and Loading Conditions

The NAC-UMS® System is designed to provide passive dry storage of canistered PWR and BWR spent fuel. The system requires few operating controls. The principal controls and limits for the NAC-UMS® SYSTEM are satisfied by the selection of fuel for storage that meets the Approved Contents presented in this section and in Tables 12B2-1 through 12B2-5 for the standard NAC-UMS® SYSTEM design basis spent fuels.

This section also permits the loading of fuel assemblies that are unique to specific reactor sites. SITE SPECIFIC FUEL assembly configurations are either shown to be bounded by the analysis of the standard NAC-UMS® System design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

The separate specific evaluation may establish different limits, which are maintained by administrative controls for preferential loading. The preferential loading controls allow the loading of fuel configurations that may have higher burnup, additional hardware material or unique configurations as compared to the standard NAC-UMS® System design basis spent fuels.

Unless specifically excepted, SITE SPECIFIC FUEL must meet all of the controls and limits specified for the NAC-UMS® System, as presented in Table 12-1.

If any Fuel Specification or Loading Conditions of this section are violated, the following actions shall be completed:

- The affected fuel assemblies shall be placed in a safe condition.
- Within 24 hours, notify the NRC Operations Center.
- Within 30 days, submit a special report that describes the cause of the violation and actions taken to restore or demonstrate compliance and prevent recurrence.

(continued)

B 2.1.1 Fuel to be Stored in the NAC-UMS® SYSTEM

INTACT FUEL ASSEMBLIES meeting the limits specified in Tables 12B2-1 through 12B2-5 may be stored in the NAC-UMS® SYSTEM.

B 2.1.2 Maine Yankee SITE SPECIFIC FUEL Preferential Loading

The estimated Maine Yankee SITE SPECIFIC FUEL inventory is shown in Table 12B2-6. As shown in this table, certain of the Maine Yankee fuel configurations must be preferentially loaded in specific basket fuel tube positions.

Corner positions are used for CONSOLIDATED FUEL, certain HIGH BURNUP FUEL and DAMAGED FUEL or FUEL DEBRIS loaded in a MAINE YANKEE FUEL CAN, for fuel assemblies with missing fuel rods, burnable poison rods or fuel assemblies with fuel rods that have been replaced by hollow Zircaloy rods. Designation for placement in corner positions results primarily from shielding or criticality evaluations of these fuel configurations. CONSOLIDATED FUEL is conservatively designated for a corner position, even though analysis shows that these lattices could be loaded in any basket position. Corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1.

Preferential loading is also used for HIGH BURNUP fuel not loaded in the MAINE YANKEE FUEL CAN. This fuel is assigned to peripheral locations, positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24 in Figure 12B2-1. The interior locations, positions 4, 5, 8, 9, 10, 11, 14, 15, 16, 17, 20, and 21, must be loaded with fuel that has lower burnup and/or longer cool times to maintain the design basis heat load (23 kW per canister).

One of the two loading patterns (Standard or Preferential) shown in Table 12B2-8 must be used to load each canister. For the Standard loading pattern, the heat load of each fuel assembly is limited to 0.958 kW. For the Preferential loading pattern, the heat load of the fuel assemblies at the basket periphery locations is limited to 1.05 kW, and the heat load of the fuel assemblies at the basket interior locations is limited to 0.867 kW. Once selected, all of the spent fuel in that canister must be loaded in accordance with that pattern. Within a pattern, mixing of enrichment and cool time is allowed, but no mixing of loading patterns is permitted. Choosing a Preferential pattern restricts the interior fuel to the cool times shown in the Preferential (I) column, and the peripheral fuel to the cool times shown in the Preferential (P) column.

(continued)

Fuel assemblies with a control element assembly (CEA) inserted will be loaded in a Class 2 canister and basket due to the increased length of the assembly with the CEA installed. However, these assemblies are not restricted as to loading position within the basket. Fuel assemblies with non-fuel items installed in corner guide tubes of the fuel assembly must also have a flow mixer installed and must be loaded in a basket corner fuel position in a Class 2 canister.

The Transportable Storage Canister loading procedures indicates that loading of a fuel configuration with removed fuel or poison rods, CONSOLIDATED FUEL, or a MAINE YANKEE FUEL CAN with DAMAGED FUEL, FUEL DEBRIS or HIGH BURNUP FUEL, is administratively controlled in accordance with Section B 2.1.

Figure 12B2-1 PWR Basket Fuel Loading Positions

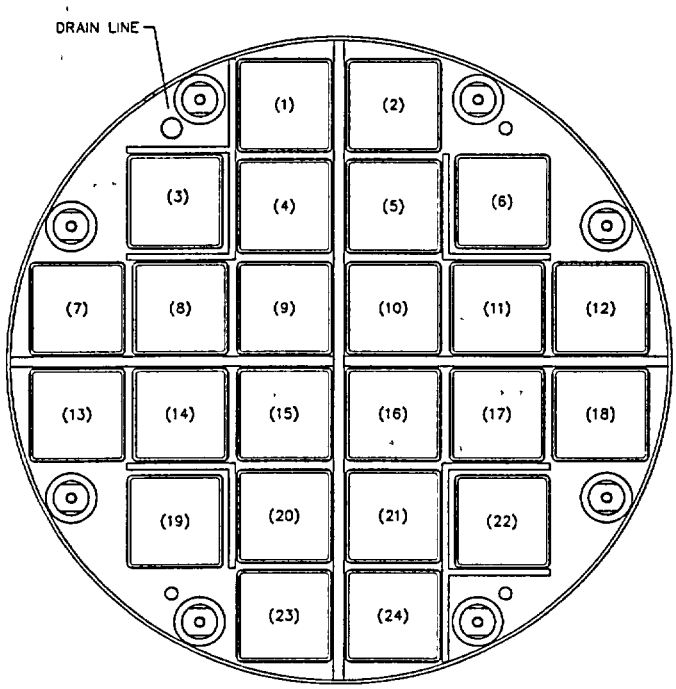

Figure 12B2-2 BWR Basket Fuel Loading Positions

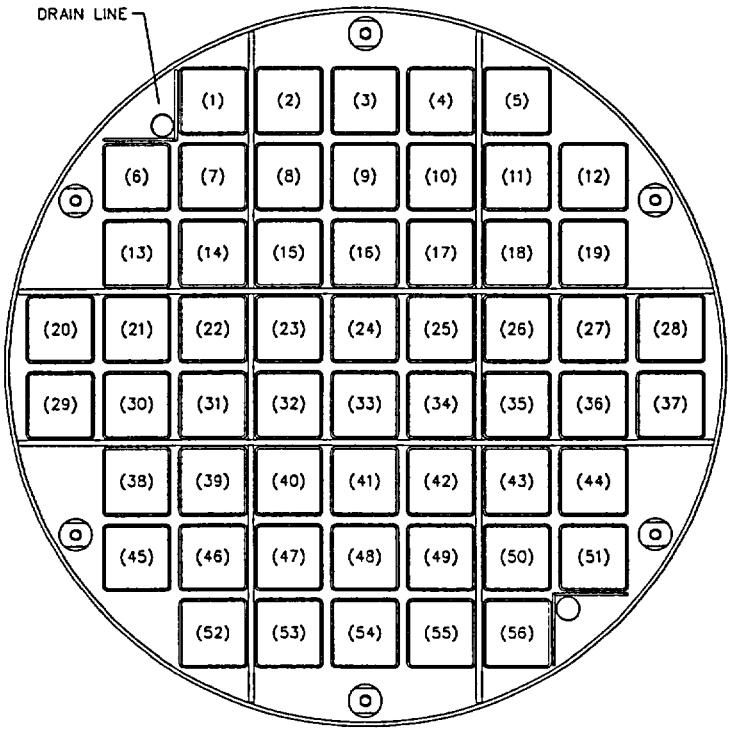

Table 12B2-1

Fuel Assembly Limits

I. NAC-UMS® CANISTER: PWR FUEL

A. Allowable Contents

1. Uranium oxide PWR INTACT FUEL ASSEMBLIES listed in Table 12B2-2 and meeting the following specifications:

a.	Cladding Type:	Zircaloy with thickness as specified in Table 12B2-2 for the applicable fuel assembly class.
b.	Enrichment, Post-irradiation Cooling Time and Average Burnup Per Assembly:	Maximum enrichment limits are shown in Table 12B2-2. For variable enrichment fuel assemblies, maximum enrichments represent peak rod enrichments. Combined minimum enrichment, maximum burnup and minimum cool time limits are shown in Table 12B2-4.
c.	Decay Heat Per Assembly:	≤ 958.3 watts †
d.	Nominal Fresh Fuel Assembly Length (in.):	≤ 178.3
e.	Nominal Fresh Fuel Assembly Width (in.):	≤ 8.54
f.	Fuel Assembly Weight (lbs.):	≤ 1,602 ‡

† Decay heat may be higher for site-specific configurations, which control fuel loading position.

‡ Includes the weight of nonfuel-bearing components.

B. Quantity per CANISTER: Up to 24 PWR INTACT FUEL ASSEMBLIES.

C. PWR INTACT FUEL ASSEMBLIES may contain a flow mixer, an in-core instrument thimble or a burnable poison rod insert (Class 1 and Class 2 contents) consistent with Table 12B2-2.

D. PWR INTACT FUEL ASSEMBLIES shall not contain a control element assembly, except as permitted for site-specific fuel.

E. Stainless steel spacers may be used in CANISTERS to axially position PWR INTACT FUEL ASSEMBLIES that are shorter than the available cavity length to facilitate handling.

F. Unenriched fuel assemblies are not authorized for loading.

G. The minimum length of the PWR INTACT FUEL ASSEMBLY internal structure and bottom end fitting and/or spacers shall ensure that the minimum distance to the fuel region from the base of the CANISTER is 3.2 inches.

H. PWR INTACT FUEL ASSEMBLIES with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches. End fitting damage including damaged or missing hold-down springs is allowed, as long as the assembly can be handled safely by normal means.

Table 12B2-1
Fuel Assembly Limits (continued)

II. NAC-UMS® CANISTER: BWR FUEL

A. Allowable Contents

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES listed in Table 12B2-3 and meeting the following specifications:

a. Cladding Type:	Zircaloy with thickness as specified in Table 12B2-3 for the applicable fuel assembly class.
b. Enrichment:	Maximum and minimum INITIAL PEAK PLANAR-AVERAGE ENRICHMENTS are 4.0 and 1.9 wt % ^{235}U, respectively. Fuel enrichment, burnup and cooling time are related as shown in Table 12B2-5.
c. Decay Heat per Assembly:	$\leq$ 410.7 watts
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	As specified in Table 12B2-5 and for the applicable fuel assembly class.
e. Nominal Fresh Fuel Design Assembly Length (in.):	$\leq$ 176.1
f. Nominal Fresh Fuel Design Assembly Width (in.):	$\leq$ 5.51
g. Fuel Assembly Weight (lbs):	$\leq$ 702, including channels

Table 12B2-1
Fuel Assembly Limits (continued)

B. Quantity per CANISTER: Up to 56 BWR INTACT FUEL ASSEMBLIES

C. BWR INTACT FUEL ASSEMBLIES can be unchanneled or channeled with Zircaloy channels.

D. BWR INTACT FUEL ASSEMBLIES with stainless steel channels shall not be loaded.

E. Stainless steel fuel spacers may be used in CANISTERS to axially position BWR INTACT FUEL ASSEMBLIES that are shorter than the available cavity length to facilitate handling.

F. Unenriched fuel assemblies are not authorized for loading.

G. The minimum length of the BWR INTACT FUEL ASSEMBLY internal structure and bottom end fitting and/or spacers shall ensure that the minimum distance to the fuel region from the base of the CANISTER is 6.2 inches.

Table 12B2-2 PWR Fuel Assembly Characteristics

Fuel Class	Vendor[1]	Array	Max. MTU	W/O Boron Max. wt % ^{235}U [4]	With Boron Max. wt % ^{235}U [5]	No of Fuel Rods	No of Water Holes	Max. Pitch (in)	Min. Rod Dia. (in)	Min. Clad Thick (in)	Max. Pellet Dia.(in)	Max. Active Length (in)	Min. Guide Tube Thick (in)
1	CE	14×14	0.404	4.7	5.0	176	5	0.590	0.438	0.024	0 380	137.0	0 034
1	Ex/ANF	14×14	0.369	5.0	- -	179	17	0.556	0.424	0.030	0.351	142.0	0 034
1	WE	14×14	0 362	5.0	- -	179	17	0.556	0.400	0 024	0.345	144.0	0.034
1	WE	14×14	0.415	5.0	- -	179	17	0 556	0 422	0 022	0.368	145.2	0.034
1	WE, Ex/ANF	15×15	0.465	4.4	5.0	204	21	0.563	0 422	0.024	0.366	144 0	0.015
1	Ex/ANF	17×17	0.413	4.4	5 0	264	25	0.496	0.360	0.025	0 303	144.0	0 016
1	WE	17×17	0.468	4.4	5.0	264	25	0.496	0.374	0.022	0.323	144.0	0 016
1	WE	17×17	0.429	4.3	5.0	264	25	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15×15	0.481	4.4	5 0	208	17	0.568	0.430	0 026	0.369	144.0	0 016
2	B&W	17×17	0.466	4.4	5.0	264	25	0.502	0 379	0.024	0.324	143.0	0.017
3	CE	16×16	0 442	4 8	5 0	236[3]	5	0.506	0 382	0 023	0 3255	150 0	0 035
1	Ex/ANF[2]	14×14	0.375	5.0	- -	179	17	0.556	0.417	0.030	0 351	144 0	0 036
1	CE[2]	15×15	0.432	4 2	5.0	216	9[6]	0.550	0.418	0.026	0.358	132.0	----
1	Ex/ANF[3]	15×15	0.431	4.2	5 0	216	9[6]	0.550	0.417	0 030	0.358	131.8	----
1	CE[2]	16×16	0.403	4.8	5.0	236	5	0.506	0.382	0 023	0 3255	136.7	0 035

Note: Parameters shown are nominal pre-irradiation values.

1. Vendor ID indicates the source of assembly base parameters, which are nominal, pre-irradiation values. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.
2. 14×14, 15×15 and 16×16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively. These are not generic fuel assemblies provided to multiple reactors.
3. Fuel rod positions may be occupied by burnable poison rods or solid filler rods.
4. Maximum initial enrichment without boron credit. Assemblies meeting this limit may contain a flow mixer, an ICI thimble or a burnable poison rod insert.
5. Maximum initial enrichment with taking credit for a minimum soluble boron concentration of 1000-ppm in the spent fuel pool water. Assemblies meeting this limit may contain a flow mixer.
6. 9 non-fuel locations, which may be filled by solid non-fuel rods.

Table 12B2-3 BWR Fuel Assembly Characteristics

Fuel Class[1]	Vendor[4]	Array	Max. MTU	Max. wt % ^{235}U	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	Ex/ANF	7 × 7	0.196	4.5	48	0.738	0.570	0 036	0.490	144 0
4	Ex/ANF	8 × 8	0.177	4.7	63	0 641	0.484	0 036	0 405	145.2
4	Ex/ANF	9 × 9	0.173	4.4	79	0.572	0 424	0 030	0.357	145.2
4	GE	7 × 7	0.199	4.5	49	0.738	0.570	0 036	0.488	144.0
4	GE	7 × 7	0.198	4.5	49	0.738	0.563	0 032	0.487	144.0
4	GE	8 × 8	0.173	4.5	60	0.640	0.484	0 032	0 410	145.2
4	GE	8 × 8	0.179	4.5	62	0 640	0.483	0.032	0 410	145.2
4	GE	8 × 8	0.186	4.7	63	0 640	0.493	0 034	0 416	144 0
5	Ex/ANF	8 × 8	0.180	4.6	62	0 641	0 484	0 036	0.405	150.0
5	Ex/ANF	9 × 9	0.167	4 4	74[3]	0.572	0.424	0.030	0.357	150 0
5	Ex/ANF	9 × 9	0.178	4.5	79[3]	0.572	0.424	0 030	0.357	150 0
5	GE	7 × 7	0.193	4.7	49	0.738	0.563	0 037	0.477	146 0
5	GE	7 × 7	0.198	4.5	49	0.738	0.563	0.032	0.487	144.0
5	GE	8 × 8	0.179	4.5	60	0.640	0 484	0 032	0 410	150 0
5	GE	8 × 8	0.185	4.5	62	0.640	0 483	0 032	0 410	150 0
5	GE	8 × 8	0.188	4.7	63	0.640	0 493	0 034	0 416	146 0
5	GE	9 × 9	0.186	4.5	74[3]	0 566	0.441	0.028	0 376	150.0
5	GE	9 × 9	0.198	4 6	79[3]	0.566	0 441	0 028	0.376	150 0

Note: Parameters shown are nominal pre-irradiation values.
1. 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, which are nominal, pre-irradiation values. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.

Table 12B2-4 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]				30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]			
	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
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 < 3.9	5	5	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5	5	5
E ≥ 4.9	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]**			
	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	6	9	14	13	9	13
2.3 ≤ E < 2.5	8	8	6	8	12	12	8	12
2.5 ≤ E < 2.7	8	7	6	7	11	11	7	11
2.7 ≤ E < 2.9	7	7	6	7	10	10	7	10
2.9 ≤ E < 3.1	7	6	6	7	9	9	7	9
3.1 ≤ E < 3.3	6	6	6	6	9	8	7	8
3.3 ≤ E < 3.5	6	6	6	6	8	8	7	8
3.5 ≤ E < 3.7	6	6	6	6	7	8	7	7
3.7 ≤ E < 3.9	6	6	6	6	7	8	7	7
3.9 ≤ E < 4.1	6	6	6	6	7	7	7	7
4.1 ≤ E < 4.3	5	6	6	6	6	7	7	7
4.3 ≤ E < 4.5	5	6	6	6	6	7	7	7
4.5 ≤ E < 4.7	5	6	5	6	6	7	6	7
4.7 ≤ E < 4.9	5	6	5	6	6	7	6	7
E ≥ 4.9	5	6	5	6	6	7	6	7

Table 12B2-5 Minimum Cooling Time Versus Burnup/Initial Enrichment for BWR Fuel

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]			30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]		
	7×7	8×8	9×9	7×7	8×8	9×9
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	6	5	6
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 < 3.9	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5
E ≥ 4.9	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]**		
	7×7	8×8	9×9	7×7	8×8	9×9
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	11	9	10	20	18	19
2.5 ≤ E < 2.7	9	8	8	18	16	17
2.7 ≤ E < 2.9	8	7	7	15	13	14
2.9 ≤ E < 3.1	7	6	6	13	11	12
3.1 ≤ E < 3.3	6	6	6	11	10	10
3.3 ≤ E < 3.5	6	5	6	9	8	9
3.5 ≤ E < 3.7	6	5	6	8	7	7
3.7 ≤ E < 3.9	6	5	5	7	6	7
3.9 ≤ E < 4.1	5	5	5	7	6	7
4.1 ≤ E < 4.3	5	5	5	7	6	6
4.3 ≤ E < 4.5	5	5	5	6	6	6
4.5 ≤ E < 4.7	5	5	5	6	6	6
4.7 ≤ E < 4.9	5	5	5	6	6	6
E ≥ 4.9	5	5	5	6	6	6

Table 12B2-6 Maine Yankee Site Specific Fuel Canister Loading Position Summary

Site Specific Spent Fuel Configurations[1]	Est. Number of Assemblies[2]	Canister Loading Position
Total Number of Fuel Assemblies[3]	1,434	Any
Inserted Control Element Assembly (CEA)	168	Any
Inserted In-Core Instrument (ICI) Thimble	138	Any
Consolidated Fuel	2	Corner[4]
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[4]
Variable Enrichment[6]	72	Any
Variable Enrichment and Axial Blanket[6]	68	Any
Burnable Poison Rod Replaced by Hollow Zircaloy Rod	80	Corner[4]
Damaged Fuel in MAINE YANKEE FUEL CAN	12	Corner[4]
Burnup between 45,000 and 50,000 MWD/MTU	90	Periphery[5]
MAINE YANKEE FUEL CAN	As Required	Corner[4]
Inserted Start-up Source	4	Corner[4]
Inserted CEA Finger Tip or ICI String Segment	1	Corner[4]

1. 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.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.
3. Includes these site specific spent fuel configurations and standard fuel assemblies. Standard fuel assemblies may be loaded in any canister position.
4. Basket corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1. Corner positions are also periphery positions.
5. 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.
6. Variably enriched fuel assemblies have a maximum burnup of less than 30,000 MWD/MTU and enrichments greater than 1.9 wt %. The minimum required cool time for these assemblies is 5 years.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits

A. Allowable Contents

1. Combustion Engineering 14 × 14 PWR INTACT FUEL ASSEMBLIES meeting the specifications presented in Tables 12B2-1, 12B2-2 and 12B2-4.
2. PWR INTACT FUEL ASSEMBLIES may contain inserted Control Element Assemblies (CEA), In-Core Instrument (ICI) Thimbles or Flow Mixers. CEAs or Flow Mixers may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or Flow Mixer inserted must be loaded in a Class 2 CANISTER and cannot be loaded in a Class 1 CANISTER. Fuel assemblies without an inserted CEA or CEA Plug, including those with inserted ICI Thimbles, must be loaded in a Class 1 CANISTER.
3. PWR INTACT FUEL ASSEMBLIES with fuel rods replaced with stainless steel or Zircaloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt %.
4. PWR INTACT FUEL ASSEMBLIES with fuel rods having variable enrichments with a maximum fuel rod enrichment up to 4.21 wt % ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt % ^{235}U.
5. PWR INTACT FUEL ASSEMBLIES with annular axial end blankets. The axial end blanket enrichment may be up to 2.6 wt % ^{235}U.
6. PWR INTACT FUEL ASSEMBLIES with solid filler rods or burnable poison rods occupying up to 16 of 176 fuel rod positions.
7. PWR INTACT FUEL ASSEMBLIES with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means.

B. Allowable Contents requiring preferential loading based on shielding, criticality or thermal constraints. The preferential loading requirement for these fuel configurations is as described in Table 12B2-6.

1. PWR INTACT FUEL ASSEMBLIES with up to 176 fuel rods missing from the fuel assembly lattice.
2. PWR INTACT FUEL ASSEMBLIES with a burnup between 45,000 and 50,000 MWD/MTU meeting the requirements of Section A 5.7(1).
3. PWR INTACT FUEL ASSEMBLIES with a burnable poison rod replaced by a hollow Zircaloy rod.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits (continued)

4. INTACT FUEL ASSEMBLIES with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 CANISTER. Only one (1) start-up source may be loaded in any fuel assembly or any CANISTER.
5. PWR INTACT FUEL ASSEMBLIES with CEA ends (finger tips) and/or ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a basket corner position and must be loaded in a Class 2 CANISTER.
6. INTACT FUEL ASSEMBLIES may be loaded in a MAINE YANKEE FUEL CAN.
7. FUEL enclosed in a MAINE YANKEE FUEL CAN. The MAINE YANKEE FUEL CAN can only be loaded in a Class 1 CANISTER. The contents that must be loaded in the MAINE YANKEE FUEL CAN are:
   a) PWR fuel assemblies with up to two INTACT or DAMAGED FUEL rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176.
   b) A DAMAGED FUEL ASSEMBLY with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A DAMAGED FUEL ASSEMBLY cannot have an inserted CEA or other non-fuel component.
   c) Individual INTACT or DAMAGED FUEL rods in a rod type structure, which may be a guide tube, to maintain configuration control.
   d) FUEL DEBRIS consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits (continued)

e) CONSOLIDATED FUEL lattice structure with a 17 × 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight ≤ 2,100 pounds. A CONSOLIDATED FUEL lattice cannot have an inserted CEA or other non-fuel component. Only one CONSOLIDATED FUEL lattice may be stored in any CANISTER.

f) HIGH BURNUP FUEL assemblies not meeting the criteria of Section A 5.8(1).

C. Unenriched fuel assemblies are not authorized for loading.

D. A canister preferentially loaded in accordance with Table 12B2-8 may only contain fuel assemblies selected from the same loading pattern.

Table 12B2-8 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU – Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	7	7	5
2.1 ≤ E < 2.3	6	6	5
2.3 ≤ E < 2.5	6	6	5
2.5 ≤ E < 2.7	5	6	5
2.7 ≤ E < 2.9	5	6	5
2.9 ≤ E < 3.1	5	6	5
3.1 ≤ E < 3.3	5	6	5
3.3 ≤ E < 3.5	5	6	5
3.5 ≤ E < 3.7	5	6	5
3.7 ≤ E ≤ 4.2	5	6	5

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern, interior basket locations: allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

Table 12B2-8 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable (Continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	11	11	6
2.1 ≤ E < 2.3	9	9	6
2.3 ≤ E < 2.5	8	8	6
2.5 ≤ E < 2.7	7	7	6
2.7 ≤ E < 2.9	7	7	6
2.9 ≤ E < 3.1	6	7	6
3.1 ≤ E < 3.3	6	7	5
3.3 ≤ E < 3.5	6	7	5
3.5 ≤ E < 3.7	6	7	5
3.7 ≤ E ≤ 4.2	6	7	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for**		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	Not allowed	Not allowed	7
2.1 ≤ E < 2.3	Not allowed	Not allowed	7
2.3 ≤ E < 2.5	Not allowed	Not allowed	7
2.5 ≤ E < 2.7	Not allowed	Not allowed	7
2.7 ≤ E < 2.9	Not allowed	Not allowed	7
2.9 ≤ E < 3.1	Not allowed	Not allowed	7
3.1 ≤ E < 3.3	Not allowed	Not allowed	7
3.3 ≤ E < 3.5	Not allowed	Not allowed	6
3.5 ≤ E < 3.7	Not allowed	Not allowed	6
3.7 ≤ E ≤ 4.2	Not allowed	Not allowed	6

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern, interior basket locations: allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

Table 12B2-9 Loading Table for Maine Yankee CE 14 × 14 Fuel Containing CEA Cooled to Indicated Time

Enrichment	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	6	5	5	5
2.7 ≤ E < 2.9	5	6	5	5	5
2.9 ≤ E < 3.1	5	6	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	7	7	7	7	7
2.7 ≤ E < 2.9	7	7	7	7	7
2.9 ≤ E < 3.1	6	6	6	6	6
3.1 ≤ E < 3.3	6	6	6	6	6
3.3 ≤ E < 3.5	6	6	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6

Note: The No CEA (Class 2) column is provided for comparison Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

APPENDIX 12C

TECHNICAL SPECIFICATION BASES FOR THE NAC-UMS® SYSTEM

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ACTIONS (continued)

A.1
If the LCO time limit is exceeded, the CANISTER will be backfilled with helium to a pressure of 0 psig (+1,-0).

AND

A.2.1.1
The TRANSFER CASK and loaded CANISTER shall be placed in the spent fuel pool with the annulus fill system operating for in-pool cooling operations.

AND

A.2.1.2
The TRANSFER CASK and loaded CANISTER shall be maintained in the spent fuel pool for a minimum of 24 hours prior to the restart of LOADING OPERATIONS.

OR

A.2.2.1
A cooling air flow of 375 CFM at a maximum temperature of 76°F shall be initiated. The airflow will be routed to the annulus fill/drain lines of the TRANSFER CASK and will flow through the annulus and cool the CANISTER.

AND

A.2.2.2
The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The elapsed time shall be monitored from completion of CANISTER draining through completion of the CANISTER vacuum dryness verification testing. Monitoring the elapsed time ensures that helium backfill and in-pool cooling operations can be initiated in a timely manner during LOADING OPERATIONS to prevent fuel cladding and CANISTER materials from exceeding short-term temperature limits.

SR 3.1.1.2

The elapsed time shall be monitored from the end of in-pool cooling through completion of the CANISTER vacuum dryness verification testing. Monitoring the elapsed time ensures that helium backfill and in-pool cooling operations can be initiated in a timely manner during LOADING OPERATIONS to prevents fuel cladding and CANISTER materials from exceeding short-term temperature limits.

REFERENCES

1. SAR Sections 4.4 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.4 CANISTER Maximum Time in the TRANSFER CASK

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. Limiting the total time a loaded CANISTER with a heat load above 20 kW (PWR) or 17 kW (BWR) is in the TRANSFER CASK, prior to its placement in the CONCRETE CASK, ensures that the short-term temperature limits established in the Safety Analysis Report for the spent fuel cladding and CANISTER materials are not exceeded.

For heat loads equal to or less than 20 kW (PWR) or less than 17 kW (BWR), forced air cooling is not required and the CANISTER time in the TRANSFER CASK is limited to 600 hours. This limit ensures that the minimum test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. For these heat loads, entry into the LCO condition is not anticipated. The time limit is established to preclude long-term storage of a loaded CANISTER in the TRANSFER CASK.

(continued)

APPLICABLE SAFETY ANALYSIS

Analyses reported in the Safety Analysis Report conclude that for heat loads greater than 20 kW (PWR) or greater than 17 kW (BWR), spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for the total elapsed times specified in LCO 3.1.4. As shown in the LCO, for total heat loads not specified, the time limit for the next higher specified heat load is conservatively applied. The thermal analysis shows that the fuel cladding and CANISTER component temperatures are below their allowable temperatures for the time durations specified, with the CANISTER in the TRANSFER CASK and backfilled with helium, after completion of 24 hours of in pool cooling with the annulus fill system in operation, or forced air cooling. For lower heat loads, the steady state fuel cladding and component temperatures are below the allowable temperatures.

The basis for forced air cooling is an inlet maximum air temperature of 76°F which is the maximum normal ambient air temperature in the thermal analysis. The specified 375 CFM air flow rate exceeds the CONCRETE CASK natural convective cooling flow rate by a minimum of 10 percent. This comparative analysis conservatively excludes the higher flow velocity resulting from the smaller annulus between the TRANSFER CASK and CANISTER, which would result in improved heat transfer from the CANISTER.

From calculated temperatures reported in the Safety Analysis Report, it can be concluded that spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for a total elapsed time of greater than 20 hours for PWR fuel or 30 hours for BWR fuel for high heat loads, if the loaded CANISTER backfilled with helium is in the TRANSFER CASK. A 2 hour completion time is provided to establish in-pool or forced airflow cooling to ensure cooling of the CANISTER.

For heat loads of 20 kW or less (PWR), or less than 17 kW (BWR), and with the canister closed with the structural lid, the analysis shows that the fuel cladding and CANISTER components reach a steady-state temperature below the short-term allowable temperatures. Therefore, the time in the TRANSFER CASK is limited to 600 hours. For heat loads greater than 20 kW (PWR), or greater than 17 kW (BWR), the analysis shows that if forced air cooling at 375 CFM with air at 76°F is used, the temperatures of the fuel cladding and CANISTER components are at or below the values calculated for the CONCRETE CASK normal conditions.

(continued)

APPLICABLE SAFETY ANALYSIS (Continued)

This limit ensures that the minimum test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. Since the 600 hours is significantly less than the 720 hours considered in PNL-4835 operation in the TRANSFER CASK to this period is acceptable.

Since the cooling provided by the forced air is equivalent to the passive cooling provided by the CONCRETE CASK and TRANSPORT CASK, relocation of a CANISTER closed with its structural lid, to a CONCRETE CASK or TRANSPORT CASK ensures that the fuel cladding and CANISTER component short-term temperature limits are not exceeded.

LCO

For PWR heat loads less than or equal to 20 kW, and BWR heat loads less than or equal to 17 kW, the thermal analysis shows that the presence of helium in the CANISTER is sufficient to maintain the fuel cladding and CANISTER component temperatures below the short term temperature limits. Therefore, forced air cooling is not required for these heat load conditions.

For higher heat loads of these fuels, as shown in the LCO, once forced air cooling is established, the amount of time the CANISTER resides in the TRANSFER CASK is not limited since the cooling provided by the forced air is equivalent to the passive cooling that is provided by the CONCRETE CASK or TRANSPORT CASK. If forced air flow is continuously maintained for a period of 24 hours, or longer, then the temperatures of the spent fuel cladding and CANISTER components are at, or below, the values calculated for the CONCRETE CASK normal conditions. Therefore, forced air cooling may be ended, allowing a new entry into Condition A of this LCO. This provides a new period in which continuation of LOADING OPERATIONS for high heat load PWR and BWR fuel may occur.

Similarly, in TRANSFER OPERATIONS, for heat loads up to the design basis, continuous forced air cooling maintains the fuel cladding and CANISTER component temperatures below the short term temperature limits. Therefore, the CANISTER may remain in the TRANSFER CASK for up to 600 hours, where the time limit is based on the minimum test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air rather than on temperature limits.

(continued)

CANISTER Maximum Time in the TRANSFER CASK
C 3.1.4

APPLICABILITY	For LOADING OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the CANISTER vacuum dryness verification through completion of the transfer from the TRANSFER CASK to the CONCRETE CASK. For TRANSFER OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the closing of the TRANSFER CASK shield doors through completion of the unloading of the CANISTER from the TRANSFER CASK.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-UMS® SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-UMS® SYSTEM not meeting the LCO. Subsequent NAC-UMS® SYSTEMS that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A.1.1 If either LCO time limit is exceeded, the TRANSFER CASK containing the loaded CANISTER backfilled with helium will be returned to the spent fuel pool with the annulus fill system operating to allow the cooler water to reduce the TRANSFER CASK, CANISTER, and spent fuel cladding temperatures to below the short term temperature limits. AND A.1.2 The TRANSFER CASK and loaded CANISTER shall be kept in the spent fuel pool for a minimum of 24 hours prior to restart of LOADING OPERATIONS. OR A.2.1 A cooling air flow of 375 CFM at a maximum temperature of 76°F shall be initiated. The airflow will be routed to the annulus fill/drain lines in the TRANSFER CASK and will flow through the annulus and cool the CANISTER. AND

(continued)

ACTIONS (continued)	A.2.2 The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS. OR A.3.1 If the CANISTER can be loaded into an operable CONCRETE CASK without the LCO time limit being exceeded, then no further action is required since the spent fuel cladding and CANISTER component short-term temperature limits are not exceeded. OR A.3.2 If the CANISTER can be loaded into a TRANSPORT CASK without the LCO time limit being exceeded, then no further action is required since the spent fuel cladding and CANISTER component short-term temperature limits are not exceeded.
SURVEILLANCE REQUIREMENTS	SR 3.1.4.1 The elapsed time from completion of helium backfill until completion of CANISTER transfer to the CONCRETE CASK shall be monitored. This SR ensures that the fuel cladding and CANISTER component temperature limits are not exceeded. SR 3.1.4.2 The elapsed time from completion of in-pool or forced air cooling until completion of CANISTER transfer to the CONCRETE CASK are completed shall be monitored. This SR is also applicable to the maximum time the CANISTER backfilled with helium can be loaded in the TRANSFER CASK if in-pool or forced air cooling is performed during vacuum drying operations under LCO 3.1.1. SR 3.1.4.3 The elapsed time from closing of the TRANSFER CASK bottom doors until the CANISTER is transferred to an operable CONCRETE CASK or TRANSPORT CASK shall be monitored. This SR ensures that the CANISTER does not remain in the TRANSFER CASK for more than 600 hours based on the minimum test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel. Forced air cooling shall be monitored to ensure that it is continuously operated so that the fuel cladding and CANISTER component temperature limits are not exceeded.
REFERENCES	1. SAR Sections 4.4, 8.1 and 8.2.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.5 CANISTER Helium Leak Rate

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel to the CANISTER shell. The inert atmosphere protects the fuel cladding. Prior to transferring the CANISTER to the CONCRETE CASK, the CANISTER helium leak rate is verified to meet leaktight requirements to ensure that the fuel and helium backfill gas is confined.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles; and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on maintaining an inert atmosphere, and maintaining the cladding temperatures below established long-term limits. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The heat-up of the CANISTER and contents will continue following backfilling the cavity and leak testing the shield lid-to-shell weld, but is controlled by LCO 3.1.4.

(continued)

LCO	Verifying that the CANISTER cavity helium leak rate is below the leaktight limit specified in LCO 3.1.5 ensures that the CANISTER shield lid is sealed. Verifying that the helium leak rate is below leaktight levels will also ensure that the assumptions in the accident analyses and radiological evaluations are maintained.
APPLICABILITY	The leaktight helium leak rate verification is performed during LOADING OPERATIONS before the TRANSFER CASK and integral CANISTER are moved for transfer operations to the CONCRETE CASK. TRANSPORT OPERATIONS would not commence if the CANISTER helium leak rate was not below the test sensitivity. Therefore, CANISTER leak rate testing is not required during TRANSPORT OPERATIONS or STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the helium leak rate limit is not met, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which could cause a helium leak rate in excess of the limit. Actions to correct a failure to meet the helium leak rate limit would include, in ascending order of performance, 1) verification of helium leak test system performance; 2) inspection of weld surfaces to locate helium leakage paths using a helium sniffer probe; and 3) weld repairs, as required, to eliminate the helium leakage. Following corrective actions, the helium leak rate verification shall be reperformed.

(continued)

ACTIONS (continued)

B.1

If the CANISTER leak rate cannot be brought within the limit, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonable based on the time required to re-flood the CANISTER, perform fuel cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

The primary design consideration of the CANISTER is that it is leaktight to ensure that off-site dose limits are not exceeded and to ensure that the helium remains in the CANISTER during long-term storage. Long-term integrity of the stored fuel is dependent on storage in a dry, inert environment.

Verifying that the helium leak rate meets leaktight requirements must be performed successfully on each CANISTER prior to TRANSPORT OPERATIONS. The Surveillance Frequency allows sufficient time to backfill the CANISTER cavity with helium and performs the leak test, while minimizing the time the fuel is in the CANISTER and loaded in the TRANSFER CASK.

REFERENCES

1. SAR Sections 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.6 CONCRETE CASK Heat Removal System

BASES

BACKGROUND

The CONCRETE CASK Heat Removal System is a passive, air-cooled convective heat transfer system, which ensures that heat from the CANISTER is transferred to the environment by the upward flow of air through the CONCRETE CASK. Relatively cool air is drawn into the annulus between the CONCRETE CASK and the CANISTER through the four air inlets at the bottom of the CONCRETE CASK. The CANISTER transfers its heat from the CANISTER surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air flows back into the environment through the four air outlets at the top of the CONCRETE CASK.

APPLICABLE SAFETY ANALYSIS

The thermal analyses of the CONCRETE CASK take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the CONCRETE CASK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and CANISTER component temperatures do not exceed applicable limits. Under normal storage conditions, the four air inlets and four air outlets are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.

Analyses have been performed for the complete obstruction of all of the air inlets and outlets. The complete blockage of all air inlets and outlets stops air cooling of the CANISTER. The CANISTER will continue to radiate heat to the relatively cooler inner shell of the CONCRETE CASK. With the loss of air cooling, the CANISTER component temperatures will increase toward their respective short-term temperature limits. The limiting component is the CANISTER basket support and heat transfer disks, which, by analysis, approach their temperature limits in 24 hours, if no action is taken to restore air flow to the heat removal system.

LCO

The CONCRETE CASK Heat Removal System must be verified to be OPERABLE to preserve the assumptions of the thermal analyses.

(continued)

LCO (continued)	Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environment at a sufficient rate to maintain fuel cladding and CANISTER component temperatures within design limits.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once a CONCRETE CASK containing a CANISTER loaded with spent fuel has been placed in storage, the heat removal system must be OPERABLE to ensure adequate heat transfer of the decay heat away from the fuel assemblies.
ACTIONS	A note has been added to ACTIONS which states that, for this LCO, separate Condition entry is allowed for each CONCRETE CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent CONCRETE CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the heat removal system has been determined to be inoperable, it must be restored to OPERABLE status within 8 hours. Eight hours is reasonable based on the accident analysis which shows that the limiting CONCRETE CASK component temperatures will not reach their temperature limits for 24 hours after a complete blockage of all inlet air ducts. <u>B.1</u> SR 3.1.6.1 is performed to document the continuing status of the operability of the CONCRETE CASK Heat Removal System. <u>AND</u> <u>B.2</u> Efforts must continue to restore the heat removal system to OPERABLE status by removing the air flow obstruction(s).

(continued)

ACTIONS (continued)	B.2 (continued) This Required Action must be completed in 12 hours. The Completion Time reflects a conservative total time period without any cooling of 24 hours. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short-term temperature limit for more than 24 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlets and outlets immediately after the last successful Surveillance.
SURVEILLANCE REQUIREMENTS	SR 3.1.6.1 The long-term integrity of the stored fuel is dependent on the ability of the CONCRETE CASK to reject heat from the CANISTER to the environment. The temperature rise between ambient and the CONCRETE CASK air outlets shall be monitored to verify operability of the heat removal system. Blocked air inlets or outlets will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the CANISTER. Based on the analyses, provided the air temperature rise is less than the limits stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long-term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility. The Frequency of 24 hours is reasonable based on the time necessary for CONCRETE CASK components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of the blockage of the air inlets and outlets.
REFERENCES	1. SAR Chapter 4 and Chapter 11, Section 11.2.13.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.1 CANISTER Surface Contamination

BASES

BACKGROUND	A TRANSFER CASK containing an empty CANISTER is immersed in the spent fuel pool in order to load the spent fuel assemblies. The external surfaces of the CANISTER are maintained clean by the application of clean water to the annulus of the TRANSFER CASK. However, there is potential for the surface of the CANISTER to become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the CONCRETE CASK containing the CANISTER to the ISFSI in order to minimize the radioactive contamination to personnel or the environment. This allows the ISFSI to be entered without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.
APPLICABLE SAFETY ANALYSIS	The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the CANISTER are not contaminated. Failure to decontaminate the surfaces of the CANISTER could lead to higher-than-projected occupational dose and potential site contamination.
LCO	Removable surface contamination on accessible exterior surfaces of the CANISTER and accessible interior surfaces of the TRANSFER CASK are limited to 10,000 dpm/100 cm^2 from beta and gamma sources and 100 dpm/100 cm^2 from alpha sources. Only loose contamination is controlled, as fixed contamination will not result from the CANISTER loading process. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels, which would cause significant personnel skin dose.

(continued)

LCO (continued)

LCO 3.2.1 requires removable contamination to be within the specified limits for the accessible exterior surfaces of the CANISTER and accessible interior surfaces of the TRANSFER CASK. The location and number of CANISTER and TRANSFER CASK surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the CANISTER are the upper portion of the CANISTER external shell wall accessible after draining of the TRANSFER CASK annulus and the top surface of the structural lid. The user shall determine a reasonable number and location of swipes for the accessible portion of the CANISTER. The objective is to determine a removable contamination value representative of the entire upper circumference of the CANISTER and the structural lid, while implementing sound ALARA practices.

Verification swipes and measurements of removable surface contamination levels on the accessible interior surfaces of the TRANSFER CASK shall be performed following transfer of the CANISTER to the CONCRETE CASK. These measurements will provide indirect evidence that the inaccessible surfaces of the CANISTER do not have removable contamination levels exceeding the limit.

APPLICABILITY

Verification that the accessible exterior surface contamination of the CANISTER and accessible interior surface contamination of the TRANSFER CASK are less than the LCO limits is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS and STORAGE OPERATIONS. Measurement of the CANISTER and TRANSFER CASK surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject CANISTER to the ISFSI.

(continued)

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER LOADING OPERATION. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER and TRANSFER CASK not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the removable surface contamination of the CANISTER that has been loaded with spent fuel or the TRANSFER CASK is not within the LCO limits, action must be initiated to decontaminate the CANISTER and TRANSFER CASK, and bring the removable surface contamination to within limits. The Completion Time of 7 days is appropriate, given that the time needed to complete the decontamination is indeterminate and surface contamination does not affect the safe storage of the spent fuel assemblies.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This SR verifies that the removable surface contamination on the accessible exterior surfaces of the CANISTER is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to initiating TRANSPORT OPERATIONS in order to confirm that the CANISTER can be moved to the ISFSI without spreading loose contamination.

(continued)

SURVEILLANCE REQUIREMENTS (continued)	SR 3.2.1.2 This SR verifies that the removable surface contamination on the accessible interior surfaces of the TRANSFER CASK is less than the limits, thereby providing indirect confirmation that the removable surface contamination on the inaccessible surfaces of the CANISTER are within the limits. It also confirms the proper functioning of the annulus clean water fill system. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to TRANSPORT OPERATIONS, which ensures a potentially contaminated CANISTER is not placed at the ISFSI.

REFERENCES	1. SAR Section 8.1. 2. NRC IE Circular 81-07.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.2 CONCRETE CASK Average Surface Dose Rates

BASES

BACKGROUND	The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions in accordance with 10 CFR 72.
APPLICABLE SAFETY ANALYSIS	The CONCRETE CASK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.
LCO	The limits on CONCRETE CASK average surface dose rates are based on the Safety Analysis Report shielding analysis of the NAC-UMS® SYSTEM (Ref. 2). The limits are selected to minimize radiation exposure to the public and to maintain occupational dose ALARA to personnel working in the vicinity of the NAC-UMS® SYSTEM. The LCO specifies sufficient locations for taking dose rate measurements to ensure the dose rates measured are indicative of the effectiveness of the shielding materials.
APPLICABILITY	The CONCRETE CASK average surface dose rates apply during STORAGE OPERATIONS. These limits ensure that the CONCRETE CASK average surface dose rates during STORAGE OPERATIONS are bounded by the shielding safety analyses. Radiation doses during STORAGE OPERATIONS are monitored by the NAC-UMS® SYSTEM user in accordance with the plant-specific radiation protection program as required by 10 CFR 72.212(b)(6) and 10 CFR 20 (Reference 1).
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each loaded CONCRETE CASK. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent NAC-UMS®

(continued)

CONCRETE CASK Average Surface Dose Rate
C 3.2.2

ACTIONS (continued) SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the CONCRETE CASK average surface dose rates are not within limits, it could be an indication that a fuel assembly that did not meet the Approved Contents Limits in Section B2.0 of Appendix 12B was inadvertently loaded into the CANISTER. Administrative verification of the CANISTER fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a misloaded fuel assembly is the cause of the out-of-limit condition. The Completion time is based on the time required to perform such a verification.

A.2

If the CONCRETE CASK average surface dose rates are not within limits and it is determined that the CONCRETE CASK was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the CONCRETE CASK would result in the ISFSI offsite or occupational calculated doses exceeding regulatory limits in 10 CFR Part 72 or 10 CFR Part 20, respectively. If it is determined that the measured average surface dose rates do not result in the regulatory limits being exceeded, STORAGE OPERATIONS may continue.

B.1

If it is verified that the fuel was misloaded, or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the CONCRETE CASK average surface dose rates above the LCO limit, the fuel assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable, based on the time required to transport the CONCRETE CASK, transfer the CANISTER to the TRANSFER CASK, remove the structural lid and vent and drain port cover welds, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

(continued)

SURVEILLANCE REQUIREMENTS	SR 3.2.2.1 This SR ensures that the CONCRETE CASK average surface dose rates are within the LCO limits after transfer of the CANISTER into the CONCRETE CASK and prior to the beginning of STORAGE OPERATIONS. This Frequency is acceptable as corrective actions can be taken before off-site dose limits are compromised. The surface dose rates are measured approximately at the locations indicated on Figure 12A3-1, following standard industry practices for determining average surface dose rates for large containers.

REFERENCES	1. 10 CFR Parts 20 and 72. 2. SAR Sections 5.1 and 8.2.

C 3.3 NAC-UMS® SYSTEM Criticality Control

C 3.3.1 Dissolved Boron Concentration

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into a PWR spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents Limits shown in Table 12B2-2. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

CANISTER cavity vacuum drying is utilized to remove residual moisture from the CANISTER cavity after the water is drained from the CANISTER. Any water not drained from the CANISTER cavity evaporates due to the vacuum. This is aided by the temperature increase, due to the heat generation of the fuel.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of design basis spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium.

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	During loading into, or unloading from, the CANISTER, criticality control of certain PWR fuel requires that the water in the CANISTER contains dissolved boron in a concentration of 1,000 parts per million, or greater. As shown in Table 12B2-2, spent fuel with the enrichments shown in the "without (w/o) boron" column may be loaded with no assured level of boron in the water in the CANISTER. However, spent fuel with the enrichments shown in the "with boron" column must be loaded or unloaded from the CANISTER when the water in the CANISTER has a boron concentration of 1,000 part per million or greater. Since boron concentration varies with water temperature, water temperature must be considered in measuring the boron concentration.
LCO	The criticality analysis shows that PWR fuel with certain combinations of initial enrichment and fuel content requires credit for the presence of at least 1,000 parts per million of boron in solution in the water in the CANISTER (see Section B3.2.1 for the requirements for assuring soluble boron concentration during loading or unloading).. This water must be used to flood the canister cavity during underwater PWR fuel loading or unloading. The boron in the pool water ensures sufficient thermal neutron absorption to preserve criticality control during fuel loading in the basket. Consequently, if boron credit is required for the fuel being loaded or unloaded, the canister must be flooded with water that contains boron in the proper concentration in accordance with the requirements of LCO 3.3.1. Concentration of boron must also be measured and maintained in accordance with LCO 3.3.1. The dissolved boron concentration requirement, and measurement requirement, applies to both the spent fuel pool water and to water in the CANISTER, when pool water is not used to fill the CANISTER.
APPLICABILITY	Control of Boron concentration is required during LOADING or UNLOADING OPERATIONS when the CANISTER holds at least one spent fuel assembly that requires dissolved boron for criticality control as described in Table 12B2-2. This LCO does not apply to spent fuel having an enrichment within the limits specified in the table in the "without (w/o) boron" column.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

A.1

If the required dissolved Boron concentration of the water in the CANISTER is not met, immediate actions must be taken to restore the required dissolved boron concentration. No actions, including continued loading, may be taken that increases system reactivity.

AND

A.2

The required concentration of dissolved Boron must be immediately restored.

AND

A.3

If the required boron concentration in the water in the CANISTER cannot be established, all fuel assemblies must be removed from the CANISTER within 24 hours to bring the system to a safe configuration. The 24 hour period provides adequate time to restore the required boron concentration.

SURVEILLANCE REQUIREMENTS	SR 3.3.1.1 The assurance of an adequate concentration of dissolved boron in the water in the CANISTER must be established within 4 hours of beginning any LOADING or UNLOADING OPERATION, using two different methods of determining boron concentration. During LOADING or UNLOADING OPERATIONS, verification of continued adequate dissolved boron concentration must be performed every 48 hours after the beginning of operations. The 48-hour boron concentration verification is not required when no water is being introduced into the CANISTER cavity. In this situation, no potential exists for the boron in the CANISTER to be diluted, so verification of the boron concentration is not necessary.
REFERENCES	Sections 12B3.2.1 and Table 12B2-2.

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NAC INTERNATIONAL
AFFIDAVIT PURSUANT TO 10 CFR 2.790

Thomas A. Danner (Affiant), Vice President & Chief Engineer, of NAC International, hereinafter referred to as NAC, at 3930 East Jones Bridge Road, Norcross, Georgia 30092, being duly sworn, deposes and says that:

1. Affiant has reviewed the information described in Item 2 and is personally familiar with the trade secrets and privileged information contained therein, and is authorized to request its withholding.

2. The information that is sought to be withheld is the following NAC International calculation package, which is being transmitted with NAC Letter No. ED20020791 dated November 27, 2002:

   - NAC Calculation Package, EA790-3206, Rev. 5, "Thermal Analyses for UMS Transfer Cask/Canister for PWR Fuel."

   NAC International is the owner of this information; the information is considered proprietary to NAC International.

3. NAC International makes this application for withholding of proprietary information based upon the exemption from disclosure set forth in: the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC Regulations 10 CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial financial information obtained from a person, and privileged or confidential" (Exemption 4). The information for which exemption from disclosure is here sought is all "confidential commercial information", and some portions may also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4.

4. Examples of categories of information that fit into the definition of proprietary information are:

   a. Information which discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by NAC's competitors without license from NAC International constitutes a competitive economic advantage over other companies.

   b. Information which, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product.

   c. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of NAC International, its customers, or its suppliers.

d. Information which reveals aspects of past, present or future NAC International customer-funded development plans and programs of potential commercial value to NAC International.

e. Information that discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information that is sought to be withheld is considered to be proprietary for the reasons set forth in Items 4.a and 4.b.

5. The information that is sought to be withheld is being transmitted to the United States Nuclear Regulatory Commission (NRC) in confidence.

6. The information sought to be withheld, including that compiled from many sources, is of a sort customarily held in confidence by NAC International, and is, in fact, so held. This information has, to the best of my knowledge and belief, consistently been held in confidence by NAC International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in Items 7 and 8 following.

7. Initial approval of proprietary treatment of a document is made by the Project Manager and/or the Director of Licensing, the persons most likely to know the value and sensitivity of the information in relation to industry knowledge. Access to proprietary documents within NAC International is limited via "controlled distribution" to individuals on a "need to know" basis. The procedure for external release of NAC proprietary documents typically requires the approval of the Project Manager based on a review of the documents for technical content, competitive effect and accuracy of the proprietary designation. Disclosures of proprietary documents outside of NAC International are limited to regulatory agencies, customers and potential customers and their agents, suppliers, licensees and contractors with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8. NAC International has invested a significant amount of time and money in the research, development, engineering and analytical costs to develop the information that is sought to be withheld as proprietary. This information is considered to be proprietary because it contains detailed descriptions of analytical approaches, methodologies, technical data and evaluation results not available elsewhere. The precise value of the expertise required to develop the proprietary information is difficult to quantify, but it is clearly substantial.

9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of NAC International, as the owner of the information, and reduce or eliminate the availability of profit-making opportunities. The proprietary information is part of NAC International's comprehensive spent fuel storage and transport technology base, and its commercial value extends beyond the original development cost to include the development of the expertise to determine and apply the appropriate evaluation process. The value of this proprietary information and the competitive advantage that it provides to NAC International would be lost if the information were disclosed to the public. Making such information available to other parties, including competitors, without their having to make similar investments of time, labor and money would provide competitors with an unfair advantage and deprive NAC International of the opportunity to seek an adequate return on its large investment.

STATE OF GEORGIA, COUNTY OF GWINNETT

Mr. Thomas A. Danner, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information and belief.

Executed at Norcross, Georgia, this 27th day of November 2002.

Thomas A. Danner
Vice President & Chief Engineer
NAC International

Subscribed and sworn before me this 27th day of November, 2002.

Jacquelyn L Burtle

Notary Public, Gwinnett County, GA
My commission expires May 30, 2006

ED20020791