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SAFETY EVALUATION REPORT

Docket No. 71-9276
Model No. FuelSolutions™ TS125 Transportation Package
Certificate of Compliance No. 9276
Revision No. 0

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SUMMARY

By application dated April 20, 2001, as supplemented, BNFL Fuel Solutions (BFS), requested that the Nuclear Regulatory Commission (NRC) approve the Model No. TS125 Transportation Package as a Type B(U)-85 package. Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concludes that the package meets the requirements of 10 CFR Part 71.

The FuelSolutions™ TS125 Transportation Package consists of a TS125 Transportation Cask and impact limiters, together with a FuelSolutions™ W21 or W74 canister and its payload. Due to the complexity of the review, some of the following SER sections address the TS125 cask and the W21 and W74 canisters separately, while others are combined.

References:

BFS application dated April 20, 2001.

Supplements dated June 7, 2001, and January 22, February 5, February 28, April 11, and April 30, 2002.

1.0 GENERAL INFORMATION

1.1. Packaging

The FuelSolutions™ TS125 Transportation Package consists of a TS125 Transportation Cask and impact limiters, together with a FuelSolutions™ W21 or W74 canister and its payload. The FuelSolutions™ canister and its payload are contained inside the TS125 Transportation Cask cavity. The TS125 Transportation Cask cavity is sized to accommodate one FuelSolutions™ long canister, or alternatively, one FuelSolutions™ short canister with a cask cavity spacer. The approximate dimensions and weights of the package are as follows:

Package Length:	342.4 inches
Package Outside Diameter:	143.5 inches
Cask Length (w/o impact limiters):	210.4 inches
Cask Outside Diameter (w/o impact limiters):	94.2 inches
Cask Cavity Length:	193.0 inches
Cask Cavity Diameter (section at rails):	66.88 inches
Canister Outside Diameter:	66.0 inches

Maximum Long Canister Length: .... 192.25 inches
Maximum Short Canister Length: .... 182.25 inches
Cask Cavity Spacer Length: .... 10.0 inches
Max. Package Weight: .... 285,000.0 pounds
Max. Cask Payload Weight (incl. Canister and Cavity Spacer): .... 85,000.0 pounds

The TS125 Transportation cask body is an assembly composed of stainless steel components of an inner shell, an outer shell, a top ring forging, a closure lid with a seal test port and a cavity vent port, a bottom plate forging, and a cavity drain port. The inner and outer shells are welded to the bottom plate forging and the top ring forging. The cask body also includes an annular lead gamma shield; an annular neutron shield with cask tie-down rings, support angles, and jacket; a bottom end neutron shield with a support ring and jacket; a longitudinal shear block; and lifting trunnion mounting bosses. The inner and outer shells form the annular cavity for the lead gamma shield. The outer shell and the neutron shield jacket form the annular cavity for the solid neutron shield. The neutron shield support angles facilitate heat rejection through the solid neutron shielding material to the outer surface of the cask body. The cask closure lid includes a thick recessed plate with two concentric “Helicoflex” silver-jacketed metallic O-ring seals, the cavity vent port, and the seal test port. The closure lid is secured to the cask body during transport with 60 – 2 inch diameter closure bolts. The vent and drain ports are closed by a plug assembly to maintain containment integrity during transportation.

The transportation cask’s containment boundary consists of: the inner cylindrical shell, the bottom plate forging (which forms the bottom closure of the cask), the top ring forging and sealing surfaces, the closure lid and sealing surfaces, the welds associated with the above components, the closure bolts, the innermost closure lid O-ring seal, the cavity vent port seal gland and O-ring seal, and the cavity drain port seal gland and O-ring seal. The package is designed to be “leaktight” as defined by ANSI N14.5 (leakage rate less than or equal to 1×10^{-7} ref-cm^3/s). The structural components of the transportation cask are made of high-strength austenitic stainless steel. The gamma shielding is made of lead and is completely enclosed within the annular region between the inner and outer steel shells. The neutron shielding is solid hydrogenous material that is completely enclosed within the annular region between the cask outer shell and neutron shield jacket with tie-down rings at each end.

The FuelSolutions™ TS125 Transportation Cask has identical energy-absorbing impact limiters at both ends. Each impact limiter assembly consists of crushable aluminum honeycomb energy-absorbing core segments that are encased in a sealed stainless steel shell. In addition to confining the aluminum honeycomb core segments in the event of a free drop, the impact limiter shell protects the aluminum honeycomb material from the weather. Both the top and bottom impact limiters are attached to the transportation cask body tie-down rings with 12, one inch diameter bolts. A tamper-indicating device is provided which connects each impact limiter to the transportation cask to assure that the package has not been opened by unauthorized personnel during transport.

A FuelSolutions™ canister consists of a steel shell assembly and an internal basket assembly. The shell assembly maintains a helium atmosphere for transport conditions. Credit is not taken for containment provided by the canister shell for transport conditions. The shell assembly also provides radiological shielding in both the radial the axial directions. The internal basket assembly provides geometric spacing, structural support, and criticality control for the spent nuclear fuel (SNF) assemblies for transport conditions.

There are two classes of W21 canisters (W21T and W21M), differing primarily in materials of construction. Each W21 canister class includes four different canister types, as follows. The W21T canister class includes a long canister with lead shield plugs (W21T-LL), a long canister with carbon steel shield plugs (W21T-LS), a short canister with lead shield plugs (W21T-SL), and a short canister with carbon steel shield plugs (W21T-SS). The W21M canister class includes a long canister with depleted uranium shield plugs (W21M-LD), a long canister with carbon steel shield plugs (W21M-LS), a short canister with depleted uranium shield plugs (W21M-SD), and a short canister with carbon steel shield plugs (W21M-SS). There are also two classes of W74 canisters (W74T and W74M), differing primarily in materials of construction. Both the W74T and W74M canister classes include only a long canister with carbon steel shield plugs.

A FuelSolutions™ canister shell assembly consists of a steel cylindrical shell, bottom end closure, bottom shield plug, bottom shell extension, bottom outer plate, top shield plug, top inner closure plate, and top outer closure plate. The closure plates at the top and bottom are welded to the cylindrical shell. All structural components of the canister shell assembly are constructed of austenitic stainless steel, with the exception of the shield plugs. The shield plug materials may be composed of lead, depleted uranium or carbon steel, depending upon the specific canister variant. To prevent any corrosion, galvanic, or chemical reactions between the shield plug materials and the cask environment or contents, the shield materials are isolated from the environment and cask interior. The lower shield plugs are encased within stainless steel. The upper shield plugs that are made of lead or depleted uranium are encased in stainless steel. The carbon steel upper shield plug is electroless nickel-plated.

A FuelSolutions™ W21 canister basket assembly consists of 21 guide tubes that are positioned and supported by a series of circular spacer plates, which are in turn positioned and supported by support rod assemblies. The W21 guide tubes include neutron absorber sheets on all four sides.

The W74 canister includes two stackable basket assemblies with a capacity to accommodate up to 64 Big Rock Point fuel assemblies. Each basket includes 37 cell locations, with the center five cell locations mechanically blocked to prevent fuel loading in these locations. The W74 basket assembly consists of a series of circular spacer plates that are positioned and supported by four support tubes that run through the spacer plates and support sleeves between the spacer plates. Each basket cell location, with the exception of the four support tubes and the five blocked-out center cells, contain a guide tube assembly. The W74 guide tube assemblies include borated stainless steel neutron absorber sheets on either one side or two opposite sides. The guide tubes are arranged in the basket to position at least one poison sheet between adjacent fuel assemblies, with the exception of intact fuel assemblies placed in the support tubes.

In the W74 basket, damaged fuel is placed in damaged fuel cans that are accommodated in the support tube cell locations. The W74 damaged fuel cans are similar to the W74 guide tubes, but include a screened bottom end, a screened removal lid, and borated stainless steel neutron absorber sheets on all four sides.

1.2 Type and Form of Material

Shipment of spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies, is not authorized.

<u>W21 Canister</u>

The contents of the W21 canister are limited to 21 pressurized water reactor (PWR) SNF assemblies meeting the requirements of Table 1 and Table 2. Two different loading configurations, designated as W21-1 and W21-2, are permitted in the W21 canister. The W21-2 loading configuration, which accommodates SNF with higher initial ^{235}U enrichments, consists of up to 20 PWR SNF assemblies meeting the requirements of Table 1 and Table 2. The W21-2 loading configuration requires that the center guide tube be mechanically blocked to prevent inadvertent loading of a SNF assembly. If less than the maximum number of PWR assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the PWR assemblies they are replacing, must be loaded in the empty guide tubes.

The SNF assemblies that are permitted in the W21 canister must meet all of the parameter requirements of at least one criticality class. Table 2 lists the dimensional and initial enrichment limits for each criticality class of PWR fuel assembly. Table 2 provides separate assembly initial ^{235}U enrichment limits for the W21-1 and W21-2 canister loading configurations. The initial enrichment limits presented in Table 2 are bounding for assemblies containing any type of control insert, including assemblies with fuel rods replaced with any type of rod of equal or greater diameter and height.

Table 3 lists minimum required cooling times, as a function of burnup, for PWR assemblies loaded into the W21 canister. For a given fuel burnup level, assembly radiation sources increase with decreasing initial enrichment. Table 3 lists two minimum initial enrichment values for each assembly burnup level. Table 3 also lists two different minimum allowable cooling times, corresponding to the two minimum initial enrichment levels. An assembly must have an initial enrichment level equal to or greater than the value shown in Table 3, to qualify for the corresponding minimum allowable cooling time also shown in Table 3. Assemblies with initial enrichment levels lower than the lowest values shown (for the assembly's burnup level) in Table 3 are not qualified for transportation in the W21 canister.

Table 3 also gives limits on the total amount of initial (pre-irradiation) cobalt that may be present in the assembly active fuel zone (including both assembly and control insert hardware). For assemblies with less than 11 grams of cobalt in the fuel zone, the shorter cooling times shown in Table 3 may be used (provided that the minimum initial enrichment requirement is also met). The longer cooling times shown in Table 3 must be used for assemblies with over 11 grams of cobalt in the fuel zone. Cobalt present in control components that do not extend into the assembly fuel zone (such as thimble plug assemblies) or that do not reside in the core during operation (such as control rod assemblies) do not need to be included in the total fuel zone cobalt content.

All PWR SNF assembly control inserts placed in the W21 canister must be intact, and may contain B_4C, borosilicate glass, silver-indium-cadmium, hafnium, or Gd_2O_3 poison materials. Control insert rod cladding, and other insert hardware may consist of any type of zircaloy, stainless steel, or Inconel. Any PWR assembly control insert that meets these material requirements may be loaded into the W21 canister. Control inserts that employ solid Inconel rods that reside in the core, such as the B&W Grey APSRA, are not qualified for transportation in the W21 canister. Any insert that contains significant quantities of Inconel (such as Inconel rod cladding) requires an evaluation of total assembly fuel zone cobalt quantity. Fuel rods may also be replaced with solid steel or Inconel rods, or rods containing any of the above poison materials, provided that the fuel zone cobalt requirements are met. UO_2 fuel rods containing Gd_2O_3 poison material are also permissible, although the poison is not relied upon to increase allowable ^{235}U initial enrichment levels for the fuel rod or assembly in question.

W74 Canister

The W74 canister contents are limited to 64 Big Rock Point (BRP) SNF assemblies without channels, including intact, partial, and damaged UO_2 and mixed oxide (MOX) fuel assemblies meeting the applicable acceptance criteria specified in Table 4 through Table 9. Specifications W74-1 and W74-2 for intact UO_2 and MOX fuel assemblies are provided in Table 4 and Table 5, respectively. Specifications W74-3 and W74-4 for partial UO_2 and MOX fuel assemblies are provided in Table 6 and Table 7, respectively. Lastly, specifications W74-5 and W74-6 for damaged UO_2 and MOX fuel assemblies are provided in Table 8 and Table 9, respectively. All UO_2 rods may contain any quantity of Gd_2O_3 poison material, provided that the specified ^{235}U initial enrichment limits are satisfied. BRP assemblies containing any amount of plutonium fuel (before irradiation) must meet the requirements of the MOX fuel specifications given in Table 5, Table 7, or Table 9. If less than the maximum number of BRP assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the BRP assemblies they are replacing, must be loaded in the empty guide tubes or support tubes.

The BRP UO_2 fuel assembly types permitted in the W74 canister are identified in Table 10. Any BRP fuel assemblies that do not meet all of the parameter requirements given for any fuel assembly class in Table 10 may only be loaded into the W74 canister damaged fuel can, as long as the requirements given in the applicable damaged fuel loading specification (W74-5 or W74-6) are still met. Any BRP fuel assemblies that meet all of the parameter requirements shown in Table 10, except for the requirement for the number of non-corner water holes, are classified as partial assemblies. The lower initial enrichment limits given in Specification W74-3 apply for those assemblies.

The specific BRP intact MOX fuel assembly types accommodated in the W74 canister are shown in Figure 1 through Figure 4. The specific BRP partial MOX fuel assembly types accommodated in the W74 canister are shown in Figure 5 through Figure 8. These figures show the maximum initial ^{235}U enrichment levels for the uranium present in all UO_2 and MOX fuel rods in each MOX assembly array. The figures also show the maximum overall weight percentage of PuO_2 in the initial MOX fuel rod (metal-oxide) material composition. The limits on maximum burnup, maximum heavy metal loading, and minimum cooling time for each BRP MOX fuel type are shown in Table 11.

Table 1 - Generic Requirements for All W21 Canister PWR SNF Contents

Fuel Assembly Parameter	Requirement
Fuel Rod Cladding Material	Zircaloy 2, 4
Assembly Condition	Intact (1)
Maximum Assembly Width (inch)	8.54
Maximum Burnup Level (MWd/MTU)	60,000 (2)
Maximum Uranium Loading (MTU/assy)	0.471
Axial Uranium Loading (kg/assy-inch)	3.27
Maximum Fuel Zone Height (inch)	150
Maximum Fuel Pellet Stack Density	96.5% (3)
Minimum Bottom Nozzle Height (inch)	1.97 (4)

Notes:

(1) Intact assemblies have no known or suspected fuel rod cladding defects greater than pinhole leaks and hairline cracks. Intact fuel also has no detectable grid spacer damage, or axial shifting in grid spacer location. Fuel assemblies with missing fuel rods (from the standard rod array configuration) may be loaded if all missing fuel rods are replaced with dummy rods that have a height and diameter at least as great as that of a standard fuel rod (i.e., by rods that displace an equal or greater volume of water).

(2) For assembly burnups exceeding 45,000 MWd/MTU, it is necessary to verify that the cladding oxide layer thickness does not exceed 70μm, by measurement of a statistical sample of limiting fuel assemblies. The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

(3) Defined as the average material density within the cylindrical envelope volume covered by the fuel pellets, relative to the theoretical UO_2 density of 10.97 g/cc. Thus, “smearing” over fuel pellet dishes and chamfers to determine the “stack” density is acceptable.

(4) The bottom nozzle height is defined as the distance between the assembly bottom and the bottom of the active fuel.

Table 2 - W21 Canister SNF Assembly Dimensional and Enrichment Limits

Fuel Assembly Class[1]	Criticality Class[1]	Max. Initial Enrichment (w/o ^{235}U)[2]		Number of Fuel Rods	Min. Clad O.D. (in.)	Min. Clad Thickness (in.)	Min. Pellet Diameter (in.)	Fuel Rod Pitch (in.)	No. Guide / Instrument Tube Locations[5]
		W21-1[3]	**W21-2**[4]						
B&W 15x15	B&W 15x15	4.70	5.00	208	0.4300	0.0265	0.3675	0.568	17
B&W 17x17	B&W 17x17	4.60	4.90	264	0.3770	0.0220	0.3232	0.502	25
CE 14x14	CE 14x14	5.00	5.00	176	0.4400	0.0260	0.3700	0.580	5[6]
	CE 14x14 A	5.00	5.00	176	0.4400	0.0260	0.3795	0.568	5[6]
Palisades	CE 15x15 P	5.00	5.00	208 - 216	0.4135	0.0240	0.3500	0.550	1-9
Yankee Rowe	15x16	5.00	5.00	231	0.3650	0.0240	0.3105	0.472	1
	15x16 A	5.00	5.00	237	0.3650	0.0240	0.3105	0.468	1
CE 16x16 CE System 80 St. Lucie 2	CE 16x16	5.00	5.00	236	0.3820	0.0250	0.3250	0.506	5[6]
WE 14x14	WE 14x14	5.00	5.00	179	0.4000	0.0243	0.3444	0.556	17
WE 15x15	WE 15x15	4.70	5.00	204	0.4200	0.0240	0.3569	0.563	21
	WE 15x15 A	4.90	5.00	204	0.4240	0.0300	0.3565	0.563	21
WE 17x17	WE 17x17	4.70	5.00	264	0.3740	0.0225	0.3195	0.496	25
	WE 17x17 A	4.60	4.90	264	0.3600	0.0225	0.3088	0.496	25
	WE 17x17 B	4.60	4.90	264	0.3600	0.0250	0.3030	0.496	25

Notes:

(1) Assembly class defined per Energy Information Administration, *Spent Nuclear Fuel Discharges from U.S. Reactors 1993*, U. S. Department of Energy, 1995. The fuel assembly criticality classes are arbitrary designations given to each set of assembly parameters that are evaluated for criticality.

(2) The maximum allowable enrichments apply for all assemblies that meet the specified physical parameter requirements for the defined assembly class. The maximum allowable enrichments are defined as the maximum planar average enrichment at any axial assembly location. An exception is the CE 15x15 P assembly class, for which the maximum allowable enrichment applies to each individual fuel pin within the assembly.

(3) This enrichment limit applies for up to 21 SNF assemblies, in any W21 canister guide tube.

(4) This enrichment limit applies for up to 20 SNF assemblies, with the center guide tube empty.

(5) Whereas the number of guide tube locations is a specified parameter, the materials and dimensions of the guide tubes are not specified, since any quantity of steel or zircaloy in the guide tube locations will reduce assembly reactivity. Guide tube locations may contain nothing, hollow zircaloy or stainless rods (or rod clusters), solid zircaloy or stainless rods (or rod clusters), or poison rods (or rod clusters).

(6) The CE 14x14 and CE 16x16 assembly guide tubes occupy four fuel rod locations within the assembly array.

Table 3 - W21 Canister Minimum PWR Assembly Cooling Time Requirements

Assembly Burnup Level (GWd/MTU)(1)	Assembly Initial Enrichment (w/o ^{235}U)(1)	Assembly Fuel Zone Cobalt Qty (g/assy)(2)	Required Cooling Time (years)
≤35	≥2.8 %	≤ 11	≥ 6
≤40	≥3.0 %	≤11	≥ 8
≤45	≥3.3 %	≤11	≥ 10
≤50	≥3.5 %	≤11	≥ 12
≤55	≥3.8 %	≤11	≥ 15
≤ 60	≥4.0 %	≤11	≥18
≤35	≥1.5 %	≤50	≥15
≤40	≥1.5 %	≤50	≥20
≤45	≥1.5 %	≤50	≥25
≤50	≥ 2.5 %	≤50	≥25
≤55	≥3.0 %	≤ 50	≥25
≤60	≥3.5 %	≤50	≥25

Notes:

(1) Assembly average values.

(2) Defined as the total initial (pre-irradiation) cobalt mass within the assembly fuel zone, including any cobalt present in inserted control components.

Table 4 - W74 Canister Contents Specification W74-1
Intact UO_2 Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact UO_2 fuel assemblies. [1,2,3] Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-2 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment[4]	≤4.10 w/o ^{235}U.
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U.
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.[5]

W74-1 Notes:

(1) Loaded assemblies must meet all of the assembly geometry requirements specified in Table 10, for any one of the defined assembly classes.

(2) Intact fuel assemblies include those BRP fuel assemblies with 1 to 4 corner rods missing, and BRP 9x9 fuel assemblies with 1 rod missing from a non-corner location. This includes assemblies with partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations. It also includes 9x9 assemblies with 11x11 assembly rods in corner locations.

(3) Intact UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. The empty array or guide tube locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component.

(4) Defined as the maximum array-average enrichment, which is the peak planar average initial enrichment considering all elevations along the assembly axis.

(5) If an intact UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 5 - W74 Canister Contents Specification W74-2
Intact MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact MOX fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-1 and W74-3 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location in the MOX assembly array in Figure 1 through Figure 4.(4,5)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.(6)

<u>W74-2 Notes:</u>

(1) Intact MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. They may also have hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component placed in the empty array or guide tube locations, including all forms of inserts or control components.

(2) J2 (Figure 1) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 9x9 fuel in Table 10. DA and G-Pu (Figure 2 and Figure 3, respectively) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 11x11 fuel in Table 10. One exception is that J2 MOX assemblies with a cladding thickness of 0.05 inches and a fuel pellet diameter of 0.4515 inches are also acceptable. UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) must meet all of the assembly geometry requirements shown for Siemens 9x9 in Table 10.

(3) Intact G-Pu MOX assemblies may have 0 to 4 fuel rods in the array corner locations. G-Pu MOX assemblies may also have partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations.

(4) The maximum ^{235}U enrichment shown in Figure 1 through Figure 4 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have ^{235}U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 1 through Figure 4 for that fuel rod array location.

(5) Figure 4 specifies a maximum total MOX fuel rod plutonium metal mass as opposed to a maximum PuO_2 weight percentage.

(6) If an intact MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 6 - W74 Canister Contents Specification W74-3 Partial UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤64 Big Rock Point BWR partial UO_2 fuel assemblies.(1,2) Partial fuel assemblies are defined as those assemblies having one or more full-length fuel rods missing from the intact fuel assembly array (except as permitted by W74-1 Notes 2 and 3). The affected array locations may contain nothing, partial length rods, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component with a lower length or diameter than a full-length fuel rod. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1, W74-2, and W74-4 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly
Maximum Initial Enrichment(3)	≤3.55 w/o ^{235}U (9x9) ≤3.6 w/o ^{235}U (11x11)
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU
Minimum Cooling Time	≥6.0 years(4)

W74-3 Notes:

(1) Partial UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods.

(2) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the “maximum number of non-corner water holes.”

(3) Defined as the maximum array average initial enrichment, which is the peak planar average initial enrichment considering all elevations along the fuel assembly axis. The averaging is applied only to those fuel rods that are present in the partial array.

(4) If a partial UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 7 - W74 Canister Contents Specification W74-4 Partial MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	$\leq$64 Big Rock Point BWR partial MOX fuel assemblies.[1,2,3] Partial MOX assemblies must conform exactly to one of the four partial assembly array configurations shown in Figure 5 through Figure 8, with respect to the number and location of missing fuel rods within the assembly array. The missing fuel rod array locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-3, W74-5, and W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by fuel assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location (in each of the four allowable partial MOX assembly array configurations) in Figure 5 through Figure 8.[4]
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.

W74-4 Notes:

(1) Partial MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod.

(2) If a partial MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

(3) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(4) The maximum ^{235}U enrichment shown in Figure 5 through Figure 8 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have ^{235}U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 5 through Figure 8 for that fuel rod array location.

Table 8 - W74 Canister Contents Specification W74-5
Damaged UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤8 Big Rock Point BWR damaged UO_2 fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where fuel rod structural integrity cannot be assured, or where grid spacers have moved from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a basket support tube in the upper or lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-4 and W74-6, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial UO_2 fuel assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.
Cladding Material/Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment	≤4.61 w/o ^{235}U peak fuel pellet initial enrichment.
Maximum Pellet Density	≤96.5% (as defined in Table 10, Note 1).
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.[(1)]

W74-5 Note:

(1) If a damaged UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 9 - W74 Canister Contents Specification W74-6
Damaged MOX Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤ 8 Big Rock Point BWR damaged MOX fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where the fuel rod structural integrity cannot be assured, or where the grid spacers have shifted vertically from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a support tube locations in the upper and lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-5, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial MOX assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.(1)
Cladding Material/ Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Pellet Density	96.5% (as defined in Table 10, Note 1)
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	≤4.61 w/o ^{235}U for all UO_2 fuel pellets. All MOX fuel pellets must meet the maximum ^{235}U enrichment and PuO_2 weight percentage requirements for one of the four MOX fuel material compositions described in Figure 1 through Figure 3.
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.(2)

W74-6 Notes:

(1) The UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) may not be loaded into the W74 damaged fuel can.

(2) If a damaged MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 10 - W74 Canister Fuel Geometry Specifications

Fuel Assembly Parameter	Fuel Assembly Class			
	GE 9x9	**Siemens 9x9**	**Siemens 11x11**	**Siemens 11x11A**
Fuel Pellet Stack Density[(1)]	≤ 96.5%	≤96.5%	≤96.5%	≤96.5%
Number of Fuel Rods	≤ 81	≤ 81	≤121	≤ 121
Clad O.D. (in)	0.5625	0.5625	0.449	0.449
Clad Thickness (in)	0.040	0.040	0.034	0.034
Pellet Diameter (in)	0.471	0.4715	0.3715	0.3735
Fuel Rod Pitch (in)	0.707	0.707	0.577	0.577
Active Fuel Length (in)	≤70	≤70	≤70	≤ 70
Number of Array Corner Rods[(2)]	0-4	0-4	0-4	0-4
Number of Non-Corner Water Holes[(2)]	≤ 1	0	0	0
Number of Inert Rods[(2)]	≥0	≥ 0	≥0	≥0
Bottom Tie Plate Height (in)[(3)]	≥1.25	≥1.25	≥1.25	≥1.25

Notes:

(1) The fuel pellet stack density is defined as the average density of the fuel pellet material (within the cylindrical envelope volume covered by the pellet stack) divided by the theoretical UO_2 density of 10.97 g/cc. Thus, smearing the fuel material over the dishing and chamfer voids in the pellet stack is acceptable for determining the stack density.

(2) The definitions of corner rods, non-corner rods, and inert rods are given in the W74-1 and W74-3 assembly loading specifications.

(3) Defined as the distance from the bottom of the assembly to the bottom of the active fuel.

Table 11 - W74 Canister Assembly Specific Requirements for Big Rock Point MOX Fuel

BRP MOX Assembly Type	Maximum Heavy Metal Loading (kg)	Maximum Burnup (MWd/MTIHM)(1)	Minimum Cooling Time (years)
J2 (9x9)	124	22,820	22
DA (11x11)	126	21,850	22
G-Pu (11x11)	127	34,220	15
UO_2 9x9 with 2 inserted MOX rods	142.1	32,000	6

Note:

(1) The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

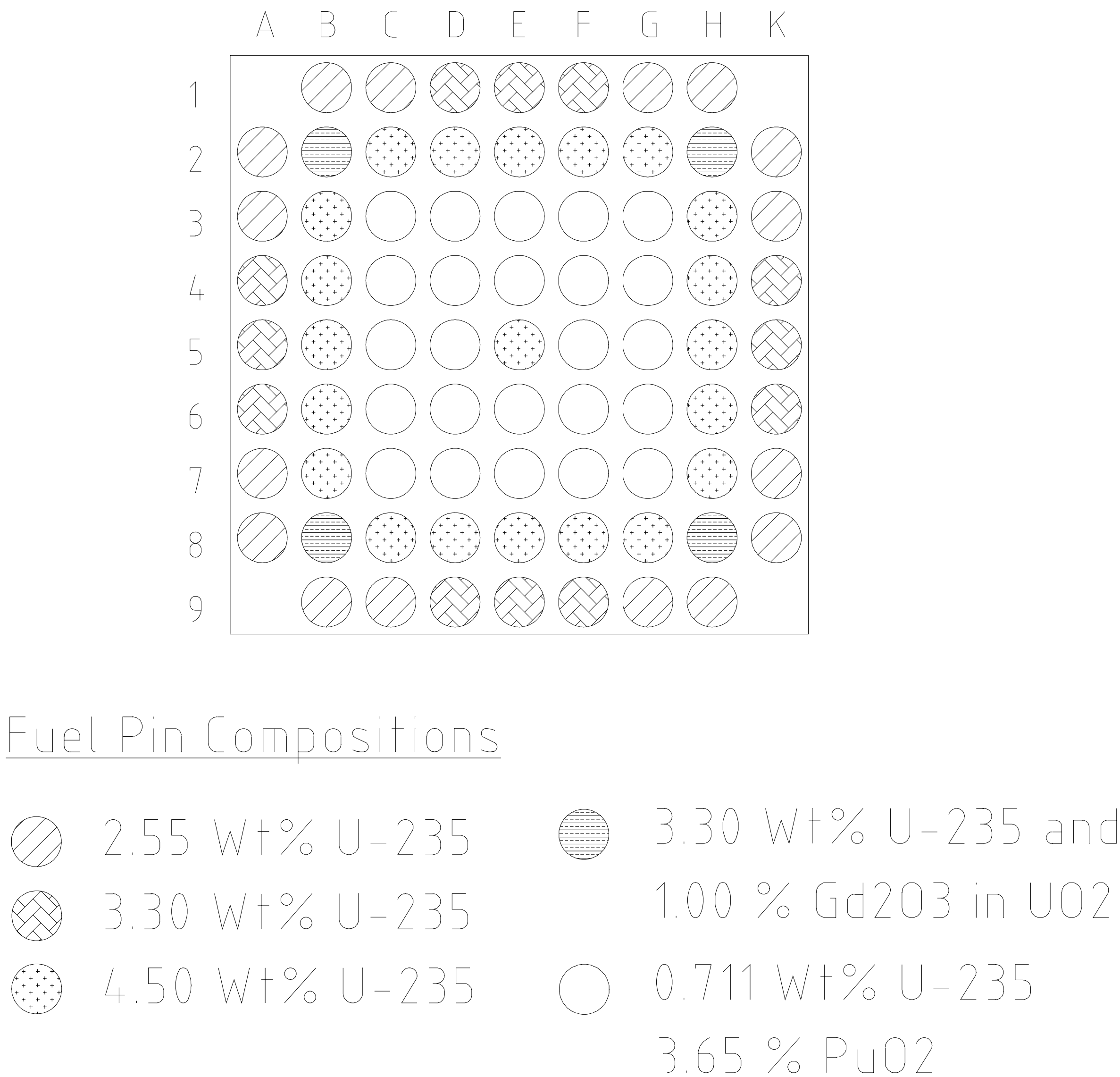
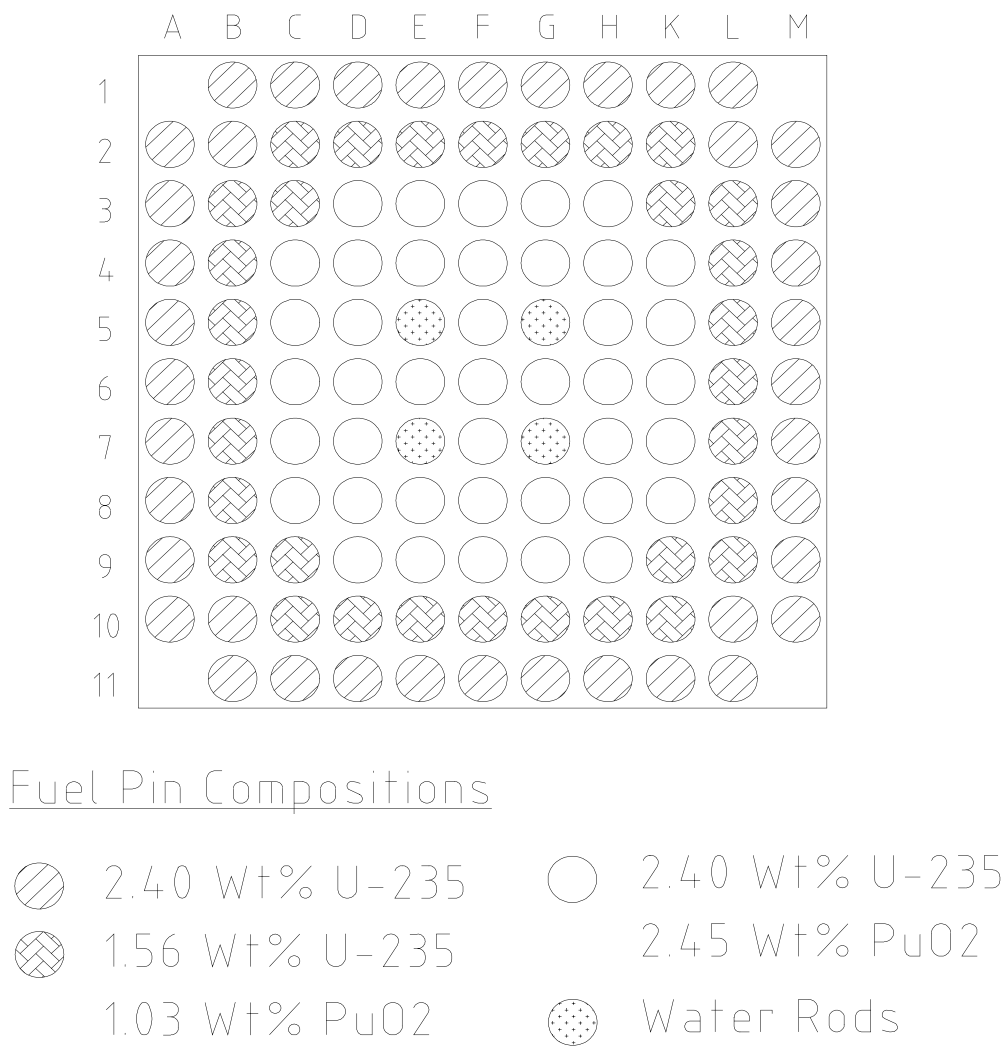
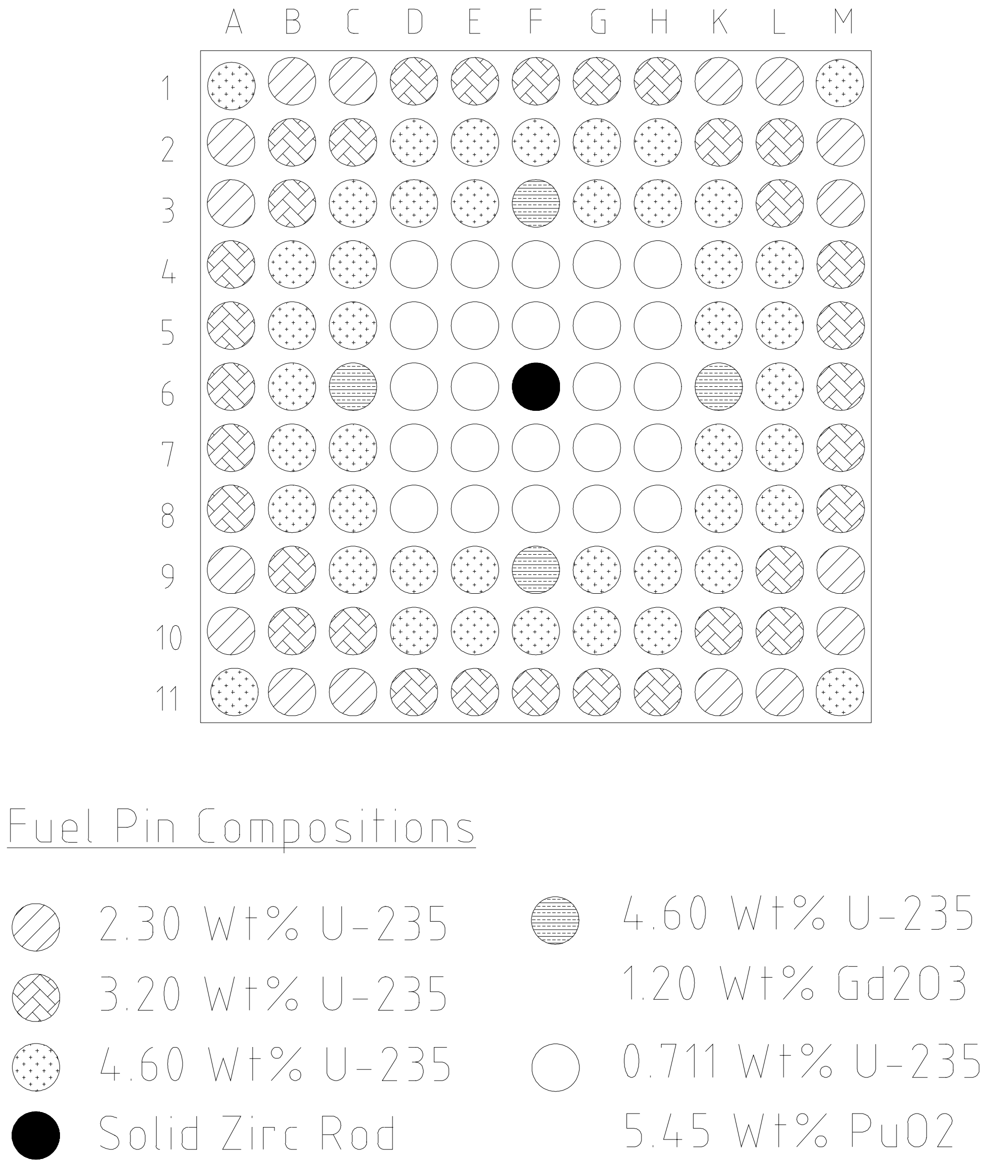


Figure 1 - J2 (9x9) BRP MOX Assembly Array



Note: Water rods are identical to the fuel rods (same diameter and cladding thickness), except that they contain no fuel pellets.

Figure 2 - DA (11x11) BRP MOX Assembly Array



Note: G-Pu assemblies may have any number of fuel rods missing (or present) in the four array corner locations

Figure 3 - G-Pu (11x11) BRP MOX Assembly Array

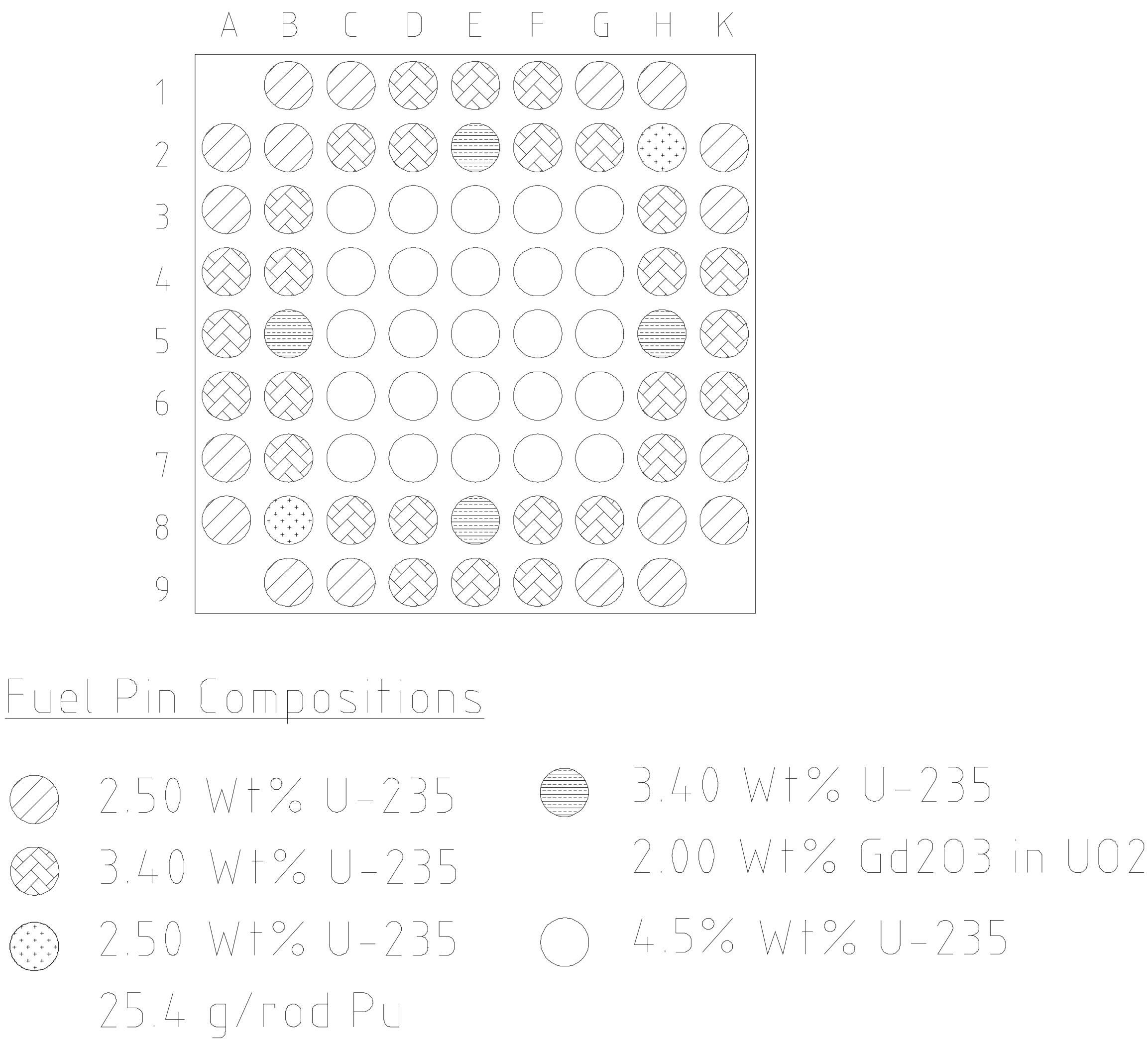


Figure 4 - UO2 9x9 BRP Assembly with Two Inserted MOX Rods

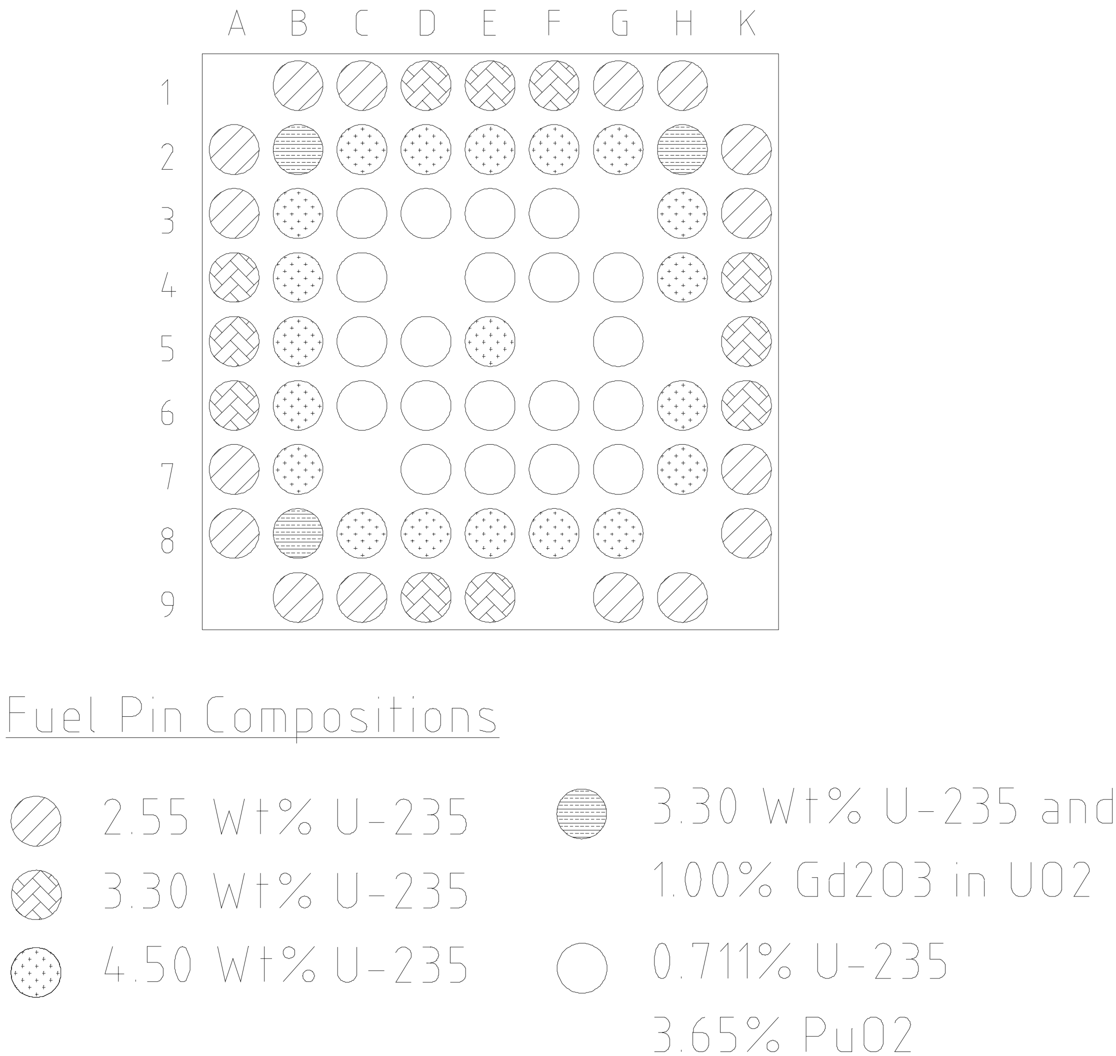


Figure 5 - J2 Partial MOX Assembly Array #1

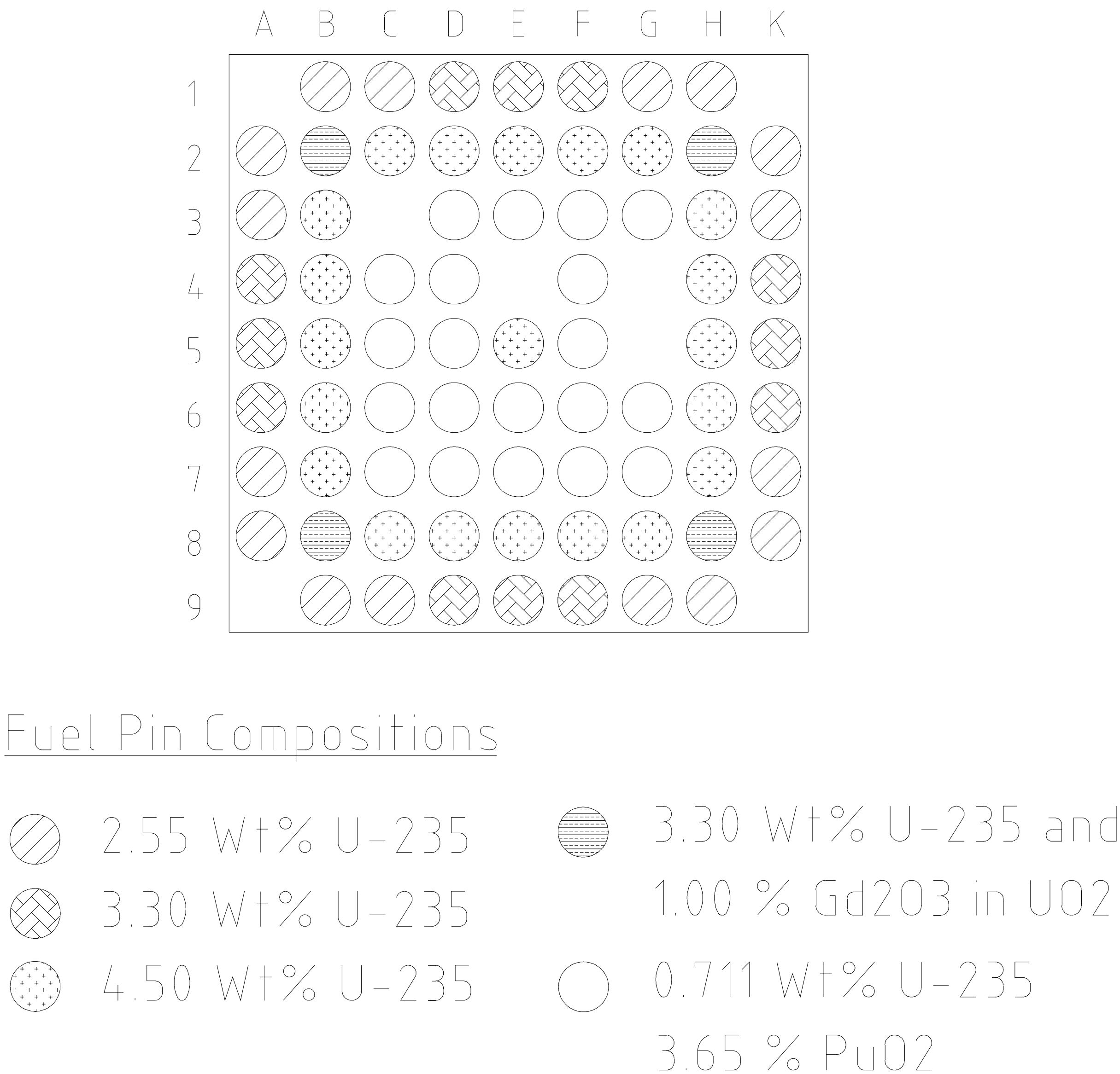


Figure 6 - J2 Partial MOX Assembly Array #2

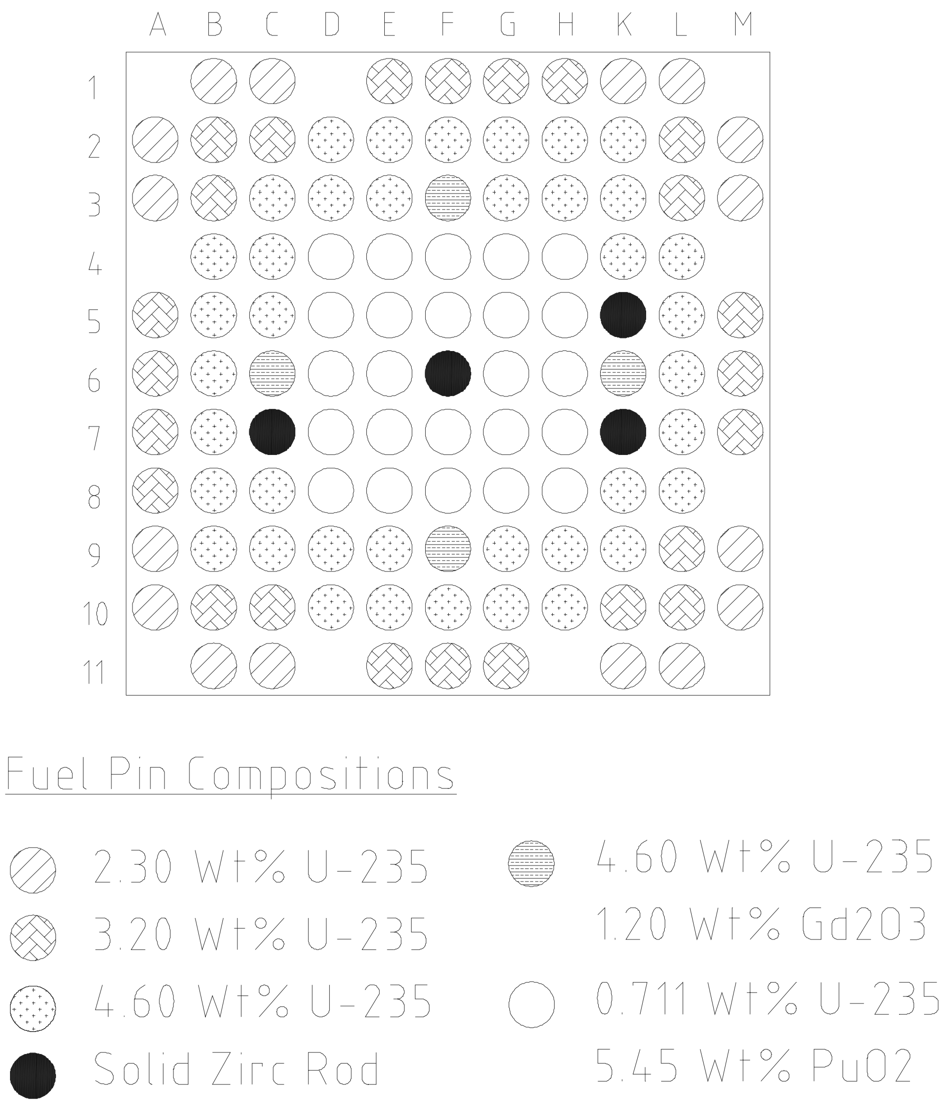


Figure 7 - G-Pu Partial MOX Assembly Array #1

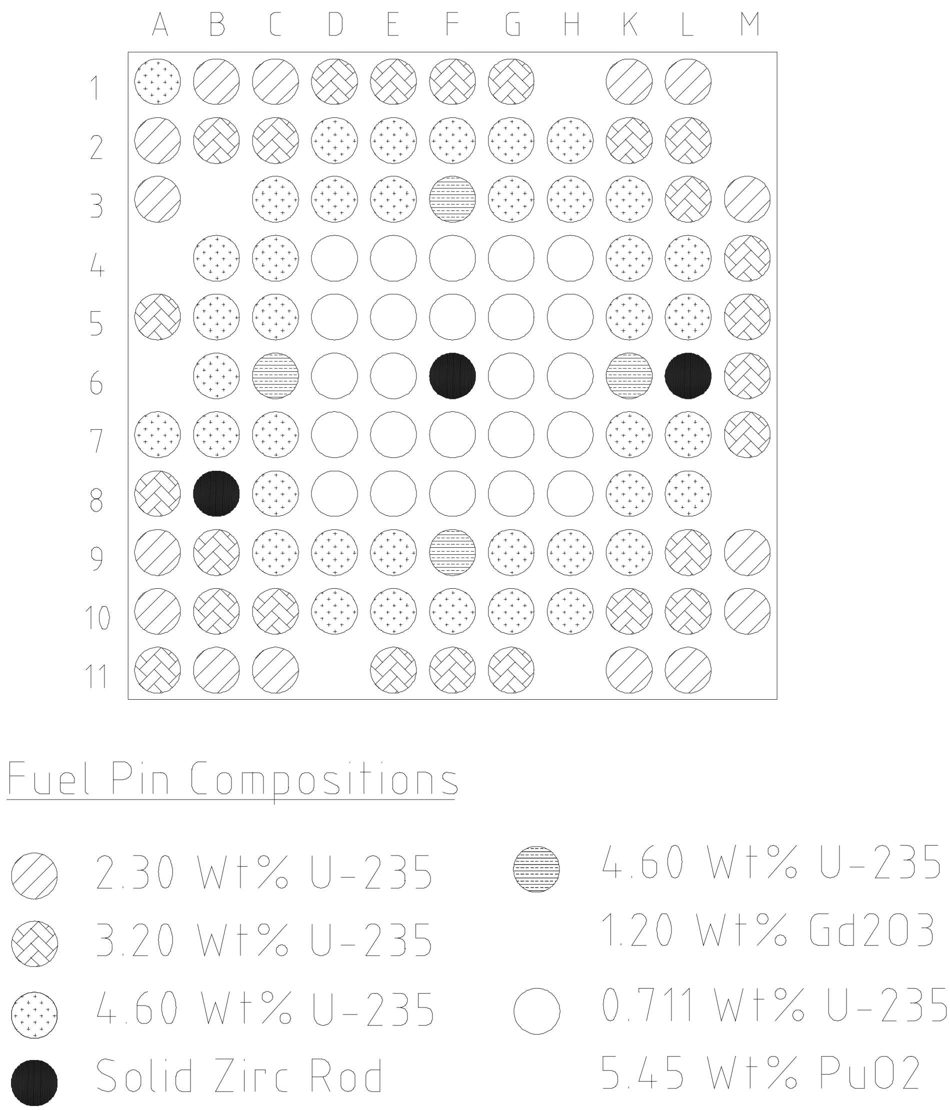


Figure 8 - G-Pu Partial MOX Assembly Array #2

1.2.1 Maximum Quantity of Material Per Package

1.2.1.1 Weight

The maximum payload weight of the TS125 transportation cask is 85,000 pounds. The payload weight includes the weight of the FuelSolutions™ canister and its SNF payload, plus the weight of the cask cavity spacer for short canisters.

1.2.1.2 Decay Heat Limit

The W21 canister loading criteria can be described as follows:

A PWR spent fuel assembly is allowed to be shipped in the canister if:

a) Q (heat generation per assembly) < 0.84 kW, **<u>or</u>**

b) Q $\leq$1.05 kW, **<u>and</u>**
Q $\leq$ 1.15 kW / PF, where PF = fuel assembly maximum peaking factor

When needed, “peaking factors” are to be inferred by ratioing local burnups to the assembly average burnup so that a maximum peaking factor value can then be identified.

The only exception is for the Yankee Rowe fuel assemblies, whose heat generation (active fuel) region is much shorter (91 inches) than that of most PWR fuel assemblies, requiring a lower assembly heat generation limit: Q $\leq$ 0.666 kW/assembly.

The W74 canister loading criteria can be described as follows:

A Big Rock Point spent fuel assembly is allowed to be shipped in the canister if Q (heat generation per assembly) $\leq$ 0.275 kW.

1.3 Conclusions

The applicant adequately described the contents of the package as required by 10 CFR 71.33(b). The staff finds that the package meets the requirements of 10 CFR Part 71.

1.4 Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:	0.0

1.5 Drawings

The package is constructed and assembled in accordance with BFS drawings:

FS-200, Sheets 1 through 3, Revision 1
FS-205, Sheets 1 through 3, Revision 2
FS-210, Sheets 1 through 9, Revision 2
FS-220, Sheets 1 through 7, Revision 1
FS-230, Sheets 1 and 2, Revision 1
W21-110, Sheets 1 through 9, Revision 4
W21-120, Sheets 1 through 10, Revision 5
W21-121, Sheet 1, Revision 5
W21-122, Sheets 1 and 2, Revision 3
W21-130, Sheets 1 through 9, Revision 4
W21-131, Sheets 1 and 2, Revision 3
W21-140, Sheets 1 through 4, Revision 5
W21-150, Sheets 1 and 2, Revision 4
W21-190, Sheet 1, Revision 4
W74-110, Sheets 1 and 2, Revision 5
W74-120, Sheets 1 through 6, Revision 5
W74-121, Sheet 1, Revision 7
W74-122, Sheet 1, Revision 6
W74-130, Sheets 1 and 2, Revision 6
W74-140, Sheets 1 through 4, Revision 5
W74-150, Sheets 1 and 2, Revision 5
3319, Sheets 1 through 5, Revision 5

2.0 STRUCTURAL EVALUATION

This section presents the results of the structural design review of the FuelSolutions TS125 Transportation Package, which includes a TS125 Transportation Cask, and a W21 Canister or a W74 Canister, containing the specified spent nuclear fuel. The purpose of this review is to verify that the transportation package meets the structural requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

2.1 Structural Design Description for the TS125 Transportation Cask

The TS125 Transportation Package is designed to meet the requirements of a 10 CFR Part 71, Subparts E and F, which are applicable to a Type B(U)F transportation package. The primary components of the TS125 Transportation Package include the TS125 Transportation Cask and steel canisters W21 or W74 containing the spent nuclear fuel. The cask is 94.2 inches in diameter and 210.4 inches long without impact limiters, and the package is 342.4 inches long with 143.5 inches diameter impact limiters, one at each end of the package. The cask is designed to accommodate short canisters using a cask Cavity Spacer Assembly. The weight of the package with the canister and the spent nuclear fuel is approximately 278,000 lbs, however the analyzed maximum allowable weight is 285,000 pounds.

The TS125 Transportation Cask is constructed and assembled in accordance with BFS General Arrangement drawings listed below:

FS-200, TS125 Transportation Cask Transportation Configuration, Sheets 1-3, Rev. 1.
FS-205, TS125 Transportation Cask Transfer Configurations, Sheets 1-3, Rev. 2.
FS-210, TS125 Transportation Cask Body Assembly, Sheets 1-9, Rev. 2.
FS-220, TS125 Transportation Cask Impact Limiter Assembly, Sheets 1-7, Rev. 1.
FS-230, TS125 Transportation Cask Cavity Spacer Assembly, Sheets 1-2, Rev. 1.

2.1.1 Structural Evaluation of the TS125 Transportation Cask

The TS125 Transportation Cask body is made up of the inner shell (67 inches inside diameter, 1.5 inches thick stainless steel, SA-240, Type XM-19 plate), and the outer shell (81.8 inches outside diameter, 2.65 inches thick stainless steel, SA-240, Type XM-19 plate). The inner and outer shell are welded at the top to the stainless steel top ring forging (SA-336, Type FXM-19), and welded at the bottom to the bottom plate forging (6 inches thick, SA-336, Type FXM-19). The cask body is enclosed on the top by a 6 inches thick Closure Lid Assembly (SA-240, XM-19) bolted to the top ring forging with 60- 2 inches diameter bolts (SB-637, Alloy UNS N07718) and two metallic Helicoflex HN-series O-rings (inner and outer). The cask cavity is approximately 66.8 inches diameter and 193.0 inches long.

The space between the inner shell and the outer shell is filled with ASTM B29 Chemical Copper Lead (UNS L51121) to provide gamma shielding. Two circumferential tie-down rings are attached to the cask, one at each end, and the space between the rings is filled with solid neutron shielding material (GESC NS-4-FR), approximately 6 inches wide around the cask. The neutron shield steel jacket with support angles (3/16 inches thick, A516, Grade 70) provide the structural support for the neutron shielding material and facilitate heat rejection through the solid neutron shielding materials. The bottom of the cask is provided with a steel support ring and a jacket (1/4 inches thick, A516, Gr 70) for approximately 5 inches thick solid neutron shielding material. The Cavity Spacer assembly for short canisters is made of stainless steel (A240, Type 304). The assembly is bolted to the bottom plate forging with 6 - 1.5 inches diameter screws (SA-193, Grade B6).

The cask exterior shell includes four mounting bosses located near the closure-end of the cask for attaching removable lifting trunnions (A564 , Grade 630) and spaced at 90^o to each other. The exterior shell also has bolt holes near the bottom end for attaching two removable rotation trunnions (A564, Grade 630) during the cask handling operations. The cask exterior is coated with Epoxy Enamel.

Impact Limiters attached to the cask body during transportation are corrugated Aluminum 5052 Alloy sheets, bonded together to form a honeycombed material with equal crush strength in both in-plane directions. The strength in the direction normal to the honeycombed sheets varies from 15 to 30 percent of the strength in the in-plane directions. The Impact Limiters are manufactured by Hexcel Corporation, California, USA, as a product called CROSS-CORE. The Impact Limiters use materials with in-plane crush strengths of 1200 psi and 2250 psi. The tolerance on the crush strength is specified as ± 10 percent. Impact Limiter shells are made of 304 stainless steel sheets, 0.135 inch thick. The Impact Limiter for the TS125 Transportation Cask has outside diameter of 143.5 inches and the over-all length is 80 inches. Each Impact Limiter is attached to the cask body with 12-1 inch diameter (A193 Grade B8S) bolts. Each Impact Limiter weighs approximately 17,700 lbs.

The cask is transported horizontally with the bottom supported on two semi-circular saddles. The cask is restrained in lateral and vertical directions by tie-down straps, and by a shear block in the longitudinal direction. A personnel barrier over the cask provides a physical barrier between personnel and the cask. The personnel barrier is not a part of the 10 CFR 71 licensed transportation package.

The applicant has evaluated the structures using the material properties based on the service temperatures, as described in Section 2.0 of the SAR. The applicant has also evaluated the fracture toughness of ferrous components for cyclic loads for a service life of 40 years in SAR Section 2.1.2.3.1. Environmental effects on the materials, such as the potential for corrosion and material degradation due to irradiation have been addressed by the applicant in SAR Section 2.4.9.

The Transportation Cask serves as the primary containment boundary for the spent nuclear fuel transported in the canister. The structural components included in the primary containment boundary are the interior shell, the bottom plate forging, the top ring forging, closure lid and bolts, inner metallic O-ring, and the Port Plug assembly. These components and associated welds are designed, fabricated and inspected in accordance with the applicable requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code), Section III, Division 3, Subsection WB. The Closure Lid bolts are designed in accordance with the recommendations of NUREG/CR-6007. The weld connecting the inner shell to the top ring forging is not examined by the radiographic methods, as required by the ASME Code, Section III, Division 3, Article WB-5231, because of inaccessibility for inspection. The alternative acceptable examination method, as described in the NRC Interim Staff Guidance -4 (ISG), Cask Closure Weld Inspections, Revision 1, is used by the applicant to verify the structural integrity of the weld. This method includes a multi-pass welding process with approximately 1/8 inch thick layers, and a non-destructive examination of the root layer, the final layer and sufficient intermediate layers to detect critical flaws, using a Dye Penetrant (PT) method (ASME Section V, Article 6). The critical flaw size is calculated in accordance with ASME Section XI methodology to verify that the critical flaw size is more than 1/8 inch. Additionally, the weld is designed with allowable stresses reduced by a factor of 0.8.

The containment boundary components are tested for leak tightness at a pressure of 112.5 psi (1.5 times the maximum normal operating pressure of 75 psig), to comply with the requirements of 10 CFR 71.85(b). The applicant has analyzed the containment structure, as an axisymmetric shell using the ANSYS finite-element computer program, for the test pressure. As shown in SAR Table 2.9-1, the stresses are less than the yield strength of the materials. The staff has reviewed the applicant's analysis, and finds that the cask meets the 10 CFR 71.85(b) requirements.

The Closure Lid has one Vent Port and one Seal Test Port. The Vent Port is used for draining, inerting, or venting the cask, and for sampling of cask cavity gases. The Seal Test Port is used for testing the leak-tightness of the Lid's containment O-ring. The bottom plate forging has a Drain Port, which is used for draining the cask after the canister loading. The Vent Port and the Drain Port are made of stainless steel (SA-240 Type 316), and are parts of the containment boundary. The ports are sealed using the Helicoflex metallic seals with port plugs torqued to maintain the seals in compressed position for NCT and HAC conditions, and comply with the containment requirements.

The structural steel components and welds, not associated with the primary containment function of the transportation cask, including the outer shell, neutron shield support tie-down rings and shield jacket, rotation trunnions, cavity spacer assembly, and the impact limiter attachment studs, are designed, fabricated and inspected in accordance with the applicable requirements of the ASME Code, Section III, Division 1, Subsection NF.

The TS125 Transportation Cask is evaluated in the following sections for compliance to the requirements of 10 CFR Part 71.

2.1.1.1 General Standards for all Packages (10 CFR 71.43)

Minimum Package Size

The package meets the 10 CFR 71.43(a) requirement for the smallest overall dimension of 4 inches because the smallest dimension of the package is 143.5 inches.

Tamper-Proof Feature

The outside of the package incorporates two wire-seals, one at each end of the package, which connect the impact limiters to the tie-down rings. The wire-seal is not readily breakable, and while intact, would be evidence that the package has not been opened by unauthorized persons, as required by 10 CFR 71.43(b).

Positive Closure

The source is contained within a canister, which is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. The sections for W21 and W74 canisters address the structural requirements of the canisters, while Section 4.0 addresses the containment requirements. Therefore, the package design meets the requirements of 10 CFR 71.43(c).

Chemical and Galvanic Reactions

The TS125 Transportation Cask is constructed of stainless steel, coated carbon steel, solid neutron shield, metallic and elastomeric seals, and chemical copper lead. The materials exposed to the cask interior or exterior environment are close in the galvanic series and would not cause galvanic reactions. These materials also would not react chemically and are not affected adversely under irradiation. Therefore, the package design meets the requirements of 10 CFR 71.43(d).

2.1.1.2 Lifting and Tie-down Standards (10 CFR 71.45)

The package is lifted using lifting trunnions at the top of the cask, which are designed for the requirements of 10 CFR 71.45(a). Each lifting trunnion is designed to take 50 percent of the maximum weight of the cask with the allowable stresses limited to the lesser of one-third of the yield strength of the material or one-fifth of the tensile strength of the material in accordance with the NUREG-0612 requirements for lifting heavy loads, and 10 CFR 71.45(a). To provide redundancy during the lifting of critical loads, four trunnions are used for lifting the cask. The staff reviewed the applicant's design calculations using equations for stresses in the lifting trunnions and the exterior shell and agree with the inputs, method of analysis, and conclusions. Based on this review, the staff has concluded that the lifting devices meet the requirements of 10 CFR 71.45(a). Rotation Trunnions are not used to lift the cask, and are designed to ASME Section III, Division 1, Subsection NF Level A allowable values.

The tie-down system for the cask consists of tie-down rings at the cask ends and the shear block at the center of the cask for transferring the longitudinal loads to the intermodal skid saddles. The shear block is designed for the load of 10 times the weight of the cask, or 10 g acceleration, while the tie-down rings are designed to transfer 2 g vertical and 5 g lateral acceleration loads in accordance with the requirements of 10 CFR 71.45(b). The applicant performed the evaluation using manual calculations. The staff reviewed the input, method of analyses and conclusions, and agree with the applicant that the tie-down system meets meet the requirements of 10 CFR 71.45(b).

2.1.1.3 Normal Conditions of Transport (10 CFR 71.71)

The TS125 Transportation Cask has been evaluated for conditions and tests for "Normal Conditions of Transport," as required by 10 CFR 71.71(c), and as described below:

Heat

The applicant analyzed the effects of solar heat (insolation) on the package, as specified in 10 CFR 71.71(c)(1), using a linearly elastic method for computed temperatures, as described in Section 3.0 of this SER. The ANSYS computer program and an axi-symmetric finite-element model were used for the analysis. The finite-element model for the cask includes the exterior and interior shells, top and bottom forgings and the lead shielding material. The neutron shielding support angles and the steel support jacket, and the closure bolts are not included in the finite-element model, but are analyzed separately using computerized and manual calculations. Since these elements are relatively flexible, the approach of decoupling them from the major cask structural elements for the thermal analysis is reasonable. Stresses in the exterior shell, tie-down rings and neutron shielding support angles, and steel jacket due to thermal expansion in the radial direction are computed using a plane strain two-dimensional finite element model. The stresses due to thermal expansion of the neutron shielding material on the outer shell in the longitudinal direction are computed using manual calculations, considering displacement compatibility of the cask exterior shell, the support angles and the steel jacket. Since the neutron shielding material has a higher thermal coefficient of expansion than the steel material, the shell and the neutron shielding jacket and support angles would experience tensile stresses, while the neutron shield would experience the compressive stresses.

Stresses due to the existing internal pressure are added to the thermal stresses to determine the combined stress intensities in various components. The combined stress intensities at critical locations are compared to the ASME III Code Service Level A allowable values (SAR Table 2.6-2, 2.6-3) to verify compliance to the design criteria requirements.

The stresses in the closure bolts due to thermal and other load conditions are described in SAR Section 2.12.7 and are evaluated later in this section of the SER.

Based on the above and the review of the applicant's input, method of analyses and conclusions, the staff finds that the Transportation Cask meets the requirements of 10 CFR 71.71(c)(1).

Cold

The applicant has evaluated the condition of the cask using an ambient temperature of -40^{o}F in still air and shade, as required by 10 CFR 71.71(c)(2). Based on the temperatures computed using the procedures described in Section 3 of this SER, thermal stresses were determined using the procedures discussed in the Heat section above. Stresses due to the existing internal pressure are added to the thermal stresses to determine the combined stress intensities in various components. The combined stress intensities at critical locations are then compared to the ASME III Code Service Level A allowable values (SAR Table 2.6-2, 2.6-3) to verify compliance to the design criteria requirements. The material properties of the cask are not affected adversely at these temperatures because the major structural components, such as the shells and the bottom plate forging and the top closure lid, are constructed of austenitic stainless steel. This steel has a face-centered cubic lattice structure, and is commonly used for structures in low temperature service. Mechanical properties of these materials (yield and tensile strength) are higher at low temperatures. The impact resistance of these steels is also excellent, despite a slight decrease as the temperature decreases (see Mark's Standard Handbook for Mechanical Engineers, Avallone, E. A. and Baumeister III, T., Tenth Edition, p. 19-32).

Based on the detailed review of the applicant's evaluation, the staff has concluded that the package would not be adversely affected by cold temperatures to -40^{o}F, and it would maintain structural integrity to meet the 10 CFR 71.71(c)(2) requirements.

Reduced External Pressure

The reduced external pressure to 3.5 psi would increase the internal pressure effects, because the maximum differential pressure would be (75 + 14.7-3.5) = 86.2 psi, instead of the maximum normal differential pressure of 75 psi. This would increase the stresses due to the internal pressure by 15 percent. Since the cask components have significant design margins (SAR Table 2.6-2), the cask will maintain structural integrity for this condition. Therefore, it is concluded that the cask meets the requirements of 10 CFR 71.71(c)(3).

Increased External Pressure

The increased external pressure to 20 psi would reduce the maximum differential pressure on the cask, and would be bounded by the cases with the maximum normal operating pressure of 75 psig. Therefore, it is concluded that the cask meets the requirements of 10 CFR 71.71(c)(4).

Vibration

Loads on the cask due to vibration are computed based on Table 2 of ANSI N14.23 and are 2.0 g in the vertical direction and 0.1g in lateral directions. Stress intensities in the cask components for the vibration loads, in combination with the internal pressure, NCT thermal and bolt pre-load, are determined by the applicant using the finite-element analysis, and compared with the ASME III Code Service Level A allowable values to verify compliance to the requirements of 10 CFR 71.71(c)(5). Based on the detailed review of the analysis methods and conclusions, the staff concluded that the cask meets the requirements of 10 CFR 71.71(c)(5).

Water Spray

The TS125 Transportation Cask is a sealed package and water spray due to rainfall of approximately 2 in/h for at least 1 hour would not affect adversely the leak tightness. Therefore, it is concluded that the cask meets the requirements of 10 CFR 71.71(c)(6).

Free Drop

To minimize the adverse impact on the structural integrity of the cask and its contents due to a design hypothetical drop accident during transportation, the applicant has provided two impact limiters, one at each end of the transportation cask to absorb the drop impact energy. For the NCT, a 1 foot drop analysis is required to be considered in compliance with 10 CFR 71.71(c)(7).

The applicant has evaluated the cask for the NCT free drop in the horizontal orientation only because the lifting and handling operations that are governed by the 10 CFR Part 71 regulations, are performed with the cask being in the horizontal orientation only (SAR, Chapter 7), and a drop at other orientations is not credible. All other lifting and handling operations with the cask being in the other orientations are governed by the 10 CFR Part 50 or 10 CFR Part 72 regulations. The staff finds that the NCT drop at orientations other than the horizontal orientation is not credible. Based on the evaluation described later in this section in the Hypothetical Accident Conditions Free Drop, the staff finds that the maximum design acceleration of 15 g for the NCT free drop is reasonable.

Stress intensities in the cask body components were calculated by the applicant using the ANSYS computer program, and a half-symmetrical finite-element model, while the stress intensities in the neutron shielding jacket and the shield support angles were computed using manual calculations with equations. The governing values of stress intensities are summarized in the SAR Table 2.6-7, and compared with the ASME III Code Service Level A allowable values to verify compliance to the requirements of 10 CFR 71.71(c)(7). The applicant has also evaluated the exterior and interior shells for buckling due the most critical NCT condition using the ASME Code Case N-284-1. Based on the detailed review of the analysis methods and conclusions, the staff concluded that the cask meets the requirements of 10 CFR 71.71(c)(5).

2.1.1.4 Hypothetical Accident Conditions (10 CFR 71.73)

The applicant has evaluated the TS125 Transportation Cask and the impact limiters for the Hypothetical Accident Conditions (HAC) of 10 CFR 71.73(c), as discussed below, including the cumulative damage based on sequential application of the free drop, puncture, and thermal tests. Additionally, the applicant has demonstrated that the undamaged containment system of the cask can withstand an external water pressure of 290 psi for a period of not less than one hour without collapse, buckling, or in-leakage of water(10 CFR 71.61).

Free Drop

The TS125 Transportation Cask with the impact limiters is required to be tested in accordance with 10 CFR 71.73(c)(1), for a free drop through a distance of 30 feet onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which the maximum damage is expected. The applicant has evaluated the compliance to the 10 CFR 71.73(c)(1) requirements by dropping 1/4 scale models of the cask from a height of 30 feet in various orientations. Sandia National Laboratory conducted the tests with the cask impacting the unyielding target on the end, side, center of gravity over the corner, and oblique drop with the cask longitudinal at 75^{o} from the vertical. Test results demonstrated that for the specific drop orientations tested, the cask maintained the structural integrity, and confirmed the adequacy of the analytical methods and computer programs used. The analytical methods used for pre-test predictions are described in detail below.

Evaluation of the maximum forces on the cask and its contents during the drop test are expressed in terms of the maximum deceleration of the cask due to an impact on the unyielding surface. The maximum cask deceleration is a direct function of the force-displacement characteristics of the impact limiters, which depend on the drop orientation, crush strength of the impact limiter material, and the construction of the honeycomb layers. The applicant has used special purpose computer programs to predict the maximum deceleration values for the free drops. The computer programs are based on the assumption that the cask is rigid and that the resistance forces and displacements are based on the impacted area of the honeycomb material, the crush strength (ultimate strength remaining constant with deflection), and the direction of the impact loads with reference to the layer orientation. Hexcel Corporation performed static specimen bench tests, dynamic drop weight tests, and scaled sub-model tests to develop the material properties input to the drop loads evaluation, including pre-test predictions for the static crush testing, and dynamic confirmatory drop testing. The material properties include the crush strengths in two in-plane directions (T1, T2), a ratio (*wtrat*) of the crush strength in the normal direction (W) to the in-plane directions, and the take-up deflection (*deltake*), which models the initial lower resistance due to shifting of the impact limiter shell and aluminum honeycomb segments.

Based on the results of the testing by the Hexcel Corporation for the impact limiter material properties, the applicant developed the force displacement relationships analytically using the special purpose programs, for various drop orientations, such as the side drop, the end drop, and a drop with the cask center of gravity (c.g.) over the corner. Scaled quasi-static tests (1/8 scale model for the end drop and the c.g. at the corner and, 1/4 scale model for the side drop) were performed by the Sandia National Laboratory (SNL) for the applicant to verify and validate the adequacy of the analytical tools, assumptions, and inputs for the drop analyses. Based on the results of the scaled model static tests for the impact limiter characteristics, a series of 1/4 scale model 30-foot dynamic drop tests were performed by the SNL to verify the design adequacy of the impact limiters and bolted attachments to the cask.

Results for the quasi-static, and 30-foot dynamic drop tests performed by Sandia, showed good correlation with the pre-test analytical predictions except for the force displacement relationship for the 1/4 scale static side drop test, and the deceleration time-histories for the slap-down effects for the 30-foot drop at 75^{o} from vertical. The differences in the pre-test predictions and the test results were due to incorrect values of the parameters, *deltake* and *wtrat* , used in the analytical predictions. The applicant had used the values of 2.5 inches and 0.475 inches for the *deltake* and *wtrat* respectively for the full model, based on the material block testing performed by the Hexcel Corporation. Based on correlating the results of the 1/4 scale model static tests, the *deltake* and *wtrat* parameters were adjusted to 8.0 inches and 1.0 inches respectively. The analytical results based on using the revised values of *deltak* and *wtrat* correlate reasonably well with the test results for the 30-foot side drop and the slap-down effects for the 30-foot drop at 75^{o} from vertical.

The staff finds the parameters *deltake* and *wtrat* used in the special purpose computer programs can be adjusted to correlate the static test results, and the same parameter values should be used to predict the 30-foot drop test results. The *deltake* and *wtrat* parameters should be based on the scaled static tests which simulate the impact limiter behavior for various drops, and then should be used to predict the 30-foot drop confirmatory test results. Additionally, during a side drop or the slap-down in a drop at an angle, a significant portion of the resistance depends on the in-plane strength of the material, and not on the normal direction strength. Therefore, the *wtrat* value of 1.0 is reasonable.

Since the analytical methods for computation of the maximum deceleration values during a drop are based on the assumption of a rigid cask, the results are multiplied by a factor, called a dynamic load factor (DLF), to determine the statically equivalent design decelerations. The DLF accounts for the dynamic amplification of deceleration depending on the component frequency. Structural evaluations for the cask and the content are then based on statically equivalent decelerations.

The staff has reviewed the applicant's method of analyses for pre-test predictions, and test results for NCT 1-foot and HAC 30-foot drop events, and finds that the design deceleration values of 15 g for the NCT horizontal drop, 60 g for the HAC end and side drops, and 40 g for the corner drops, are reasonable. The staff's evaluation of the adequacy of the results was based on engineering judgement supported by approximate calculations using the conservation of energy and the force displacement relationships of the impact limiters for various drop conditions, derived from the static tests.

Based on the design deceleration values for various drop orientations as described above, the applicant has performed detailed stress analyses to determine the critical areas of stress intensities, and combined those with the stress intensities due to the maximum normal operating pressure of 75 psig and bolt pre-load conditions. Results of the stress analyses are given in SAR Tables 2.7-2, 2.7-6, and 2.7-10, and are compared to the ASME III Code allowable values for Service Level D. The applicant has also evaluated the potential for shell buckling for the analyzed drop events, using the ASME Code Case N-284-1.

Based on the detailed review of the applicant's methods of analyses, inputs and conclusions, and engineering judgement, the staff has concluded that the TS125 Transportation Cask meets the 10 CFR 71.73(c)(1) requirements.

Crush

The crush test of 10 CFR 71.73(c)(2) is not required for the TS125 Transportation Cask because the mass of the cask is greater than 1100 lbs.

Puncture

The damaged transportation cask is required by 10 CFR 71.73(c)(3) to be evaluated for a free drop through a distance of 40 inches in a position for which the maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. Since the cask is shown to maintain structural integrity during the HAC 30-feet drop in any orientation without a significant change in the original geometry, the applicant has used an analytical approach for the undamaged cask, to demonstrate that the cask will not be adversely affected by the puncture test.

Puncture resistance of the cask at the critical locations (closure lid, and the bottom plate forging) is evaluated by applying the maximum bar failure force at the center of each component. The maximum bar failure force is taken as the average of the tensile and the yield strength of the 6 inch diameter mild steel bar. The stress intensities in each component are limited to the ASME III Code Service Level D allowable values. For evaluation of the cask outer shell, the thickness required to penetrate the outer shell is computed using the ASME III Code, Division 3, Subsection WB-3324 requirements. The staff finds the applicant's method of analyses and conclusion meet the requirements of 10 CFR 71.73(c)(3).

Thermal

The applicant has calculated the temperatures (SAR Section 3) in various parts of the cask for an engulfing fire, as required by 10 CFR 71.71(c)(4) for the initial conditions at -20^{o}F or 100^{o}F Using these temperatures as input, the applicant analyzed the cask as an axi-symmetrical structure in the ANSYS computer program to determine the stress intensities in various components. Results of the stress analyses are shown in SAR Table 2.7-14. The areas in the outer shell at the bottom plate forging and the tie-down rings are likely to yield locally and deform. However, the deformations would relieve the thermal stresses, and the solid neutron shielding material would remain attached to the support angles and the steel jacket. Two pressure relief valves, one in the top tie-down ring and the other in the bottom support ring, are provided to relieve the pressure build-up in the neutron shielding material above 35 psi.

Based on the review of the applicant's method of analysis, inputs and conclusions, the staff finds that the cask design meets the 10 CFR 71.73(c)(4) requirements.

Immersion - fissile material

The requirements of 10 CFR 71.73(c)(5) do not apply to the TS125 Transportation Cask because the criticality analysis discussed in SAR Chapter 6 considers the water in-leakage.

Immersion - all packages

The requirements of 10 CFR 71.73(c)(6) for the cask to be evaluated for 21.7 psi is bounded by the 10 CFR 71.61 requirements of 290 psi external pressure for the containment evaluation.

2.1.1.5 Special Requirements for Irradiated Nuclear Fuel Shipments

The applicant has evaluated the cask containment for 290 psi external water pressure in accordance with 10 CFR 71.61 requirements. The evaluation was performed using an axi-symmetric finite element model in the ANSYS computer program. Stress intensities at critical locations are compared to the ASME III Service Level D allowable values. To ensure that the water will not leak into the containment, the stress intensities in the closure lid, top forging ring, and the closure bolts are limited to the yield strength of the materials.

Based on the review of the applicant's method of analysis, inputs and conclusions, the staff finds that the cask design meets the 10 CFR 71.61 requirements.

2.1.1.6 Lead Pour Stress

The applicant has evaluated the cask exterior and interior shells for stresses due to pouring of molten lead into the gamma shield annulus during fabrication of the cask. The calculations are based on using standard formulas for the behavior of cylindrical shells. The evaluation considers the hydrostatic pressures of molten lead, and pressures on the shells as the lead cools. The resulting stresses accounting for the lead creep behavior are approximately 1200 psi, which are not significant when compared to the ultimate strength of the stainless steel material of 110,000 psi.

Based on the review of the applicant's method of analyses, input and conclusions, the staff finds that the fabrication stresses in the cask due to lead pour would not adversely affect the structural integrity and the ability of the cask to withstand NCT and HAC tests required by 10 CFR 71.71 and 10 CFR 71.73.

2.1.1.7 Closure Bolts

The Transportation Cask closure bolts are evaluated in accordance with the guidelines of NUREG/CR-6007. Since the closure lid and the closure bolts are protected by the lip of the top forging ring, the bolts are not expected to transfer shear, and are designed for tensile loads only. The closure lid is also provided with a raised-face flange to eliminate prying effects on bolts due to internal pressure and the impact due to drop events. The closure bolts are evaluated for the tensile loads due to installation torque pre-load, the internal pressure of 75 psig, thermal effects, the impact loads due to drop loads, puncture loads, and loads due to immersion. Results of the evaluation are given in SAR Tables 2.12-9 through 2.12-13. These stresses are compared to the allowable stresses recommended in the NUREG/CR-6007, which are based on ASME III Code, Division 1, Subsection NB.

The closure bolts are screwed into stainless steel Helicoil thread inserts (manufactured by Emhart Industries) in the top ring forging. The engagement length for the bolt is provided as 2.5 times the 2 inch bolt diameter, or 5 inches. The closure bolts are torqued to provide the pre-load sufficient to withstand the forces resulting from NCT or HAC drop tests, and maintain the metallic seals in compressed position for containment leak-tight integrity.

The staff reviewed the applicant's evaluation and finds that the closure bolts will maintain the structural integrity during the NCT and HAC conditions and tests, and thus meet the 10 CFR 71.71 and 10 CFR 71.73 requirements.

2.1.1.8 Impact Limiter Attachment Studs

The applicant has evaluated the impact limiter attachment stud loads arising from NCT and HAC drop tests, using the impact forces from the analysis of each test. Maximum forces in the studs are determined by manual calculations using the equilibrium of forces and moments. The studs experience tensile loads during the HAC 75^{o} oblique drop and the NCT and HAC side drop tests only. The maximum stresses are then compared to the ASME III Code, Division 1, Subsection NF, allowable values. The attachment studs are threaded into the tie-down rings for a length of 2.0 inches, which develops the capacity of the attachment studs.

The staff has reviewed the method of analysis, input and conclusions, and has concluded that the Impact Limiter Attachment Studs will maintain the structural integrity during the NCT and HAC conditions and tests, and thus meet the 10 CFR 71.71 and 10 CFR 71.73 requirements.

2.1.1.9 Cavity Spacer Assembly

The cavity spacer assembly is made up of a 3 inches thick circular stainless steel base plate, supported by two concentric 0.75 inches thick support rings. The assembly is bolted to the bottom plate forging with 6-1.5 inch diameter bolts (SA-193, Grade B6). The applicant has analyzed various components of the assembly for a maximum load of 60 times (maximum deceleration of the cask for the HAC end drop test) the bounding load of 85000 lbs for the weight of the canister and the fuel content. Since the weight of the canister and the fuel content is transferred to the base plate through a rigid canister, the loads on the interior and exterior rings are assumed to be proportional to the area of each ring. The stress intensities are compared to the ASME III Code, Division 1, Subsection NF allowable values.

The staff has reviewed the method of analysis, input and conclusions, and has concluded that the cavity spacer assembly will maintain the structural integrity during the NCT and HAC conditions and tests, and thus meet the 10 CFR 71.71 and 10 CFR 71.73 requirements.

2.1.2 Materials

The staff reviewed the materials information presented for the TS125 Transportation Cask to determine whether the materials of construction of the FuelSolutions™ TS125 Transportation Package meet the requirements of 10 CFR Part 71. Additionally, the FuelSolutions™ W21 and W74 Storage Canister SAR's, as amended for transportation, were reviewed for any changes in the operating conditions that may impact the previous analyses and evaluations performed by the staff when these canisters were previously licensed for storage. In particular, the following aspects were reviewed: material selection; applicable codes and standards; proposed alternatives to the Code; weld specifications; inspections; bolt specifications; chemical and galvanic reactions; coatings; and long term performance issues such as corrosion and thermal aging.

The materials, applicable codes, and specifications for the FuelSolutions™ TS125 Transportation Package components are listed in tables in Chapter 2 and 3 of the respective SARs. Additional materials information and applicable specification information are also identified on the general arrangement drawings in Chapter 1 of the respective SAR and the Engineering drawings.

The staff examined the TS125 transportation cask and associated canisters for compliance with the recently issued ISG-11, rev. 2, although the design was submitted under ISG-11, rev. 1. The previously licensed storage canisters (W21 and W74) were noted to be compliant with the temperature limits of ISG-11, rev.2. For NCT, the TS125 transportation cask was also noted to be compliant with the temperature limits of ISG-11, rev.2.

Since the FuelSolutions storage casks were previously certified for high burn-up fuel under rev.1 of ISG-11, the applicant has committed to requiring cladding oxide thickness measurements in accordance with Revision1 of ISG-11. The 70-80 micron oxide limit from that guidance will ensure that the average hydrogen concentration in the cladding is less than 400-500 ppm. At this hydrogen level, with the temperature limits previously imposed (400C), the amount of hydride reorientation will be negligible. Thus, the cladding properties are not expected to be adversely affected by hydride reorientation or creep. Consequently, the staff concludes that the cladding will reasonably maintain its integrity during the hypothetical accident condition of transport.

The TS125 transportation cask is fabricated of high strength type XM-19 austenitic stainless steel. This material has a long history of industrial use. It has excellent impact toughness at all design temperatures, is impervious to atmospheric corrosion and weather, is unaffected by long-term exposure to temperatures of 800°F, and can readily withstand shorter term temperature excursions to 1000°F and higher.

The cask body, except the lid, is of all-welded construction. The welds are fabricated and inspected in accordance with the provisions of the ASME Code, Section III, Division 3, Subsection WB. The only exception to the welding provisions of the Code is the outer shell to top ring forging weld. This weld is performed using the provisions of the NRC staff guidance contained in ISG-4, “Cask Closure Weld Inspections,” for situations where a full volumetric examination is not feasible. This alternative substitutes a progressive dye penetrant (PT) examination for the volumetric examination in addition to imposing a joint efficiency factor of 0.80, and requires demonstration that a postulated maximum possible undetected flaw size will not result in joint failure under postulated design accident conditions.

The transport cask lid is attached to the cask body with high strength nickel-base alloy bolts (ASME SB-637, grade N07718). The bolts, which are a precipitation hardened austenitic alloy, have excellent impact toughness at all design temperatures, are impervious to atmospheric corrosion and weather, resist long term exposure to temperatures up to 700°F and can withstand exposure to higher temperatures for short durations.

The TS125 lid is sealed with two concentric metal O-ring seals. The seals are fabricated with a silver jacket, stainless steel liner and Inconel spring. This combination of materials has excellent resistance to corrosion and can withstand all design basis accident temperature conditions. The staff further notes that the vendor has benefitted from prior industry experience with metallic seals and has selected silver coated seals, instead of aluminum, due to their superior aqueous corrosion resistance under long term exposure to pooled, chloride-contaminated rainwater or condensation.

The materials of the lid O-rings are able to withstand long times at elevated temperatures. The specific seals employed are designed for long term temperature capability from below the cask design minimum temperature of minus 40°F to 932°F. Under a hypothetical fire accident condition, the seal area temperature would remain below 550°F, thus assuring their integrity.

The cask impact limiters are fabricated of stainless steel-encased aluminum honeycomb. The stainless cover comprises the permanent weather and mechanical damage resisting element of the impact limiters. Stainless steel has excellent resistance to atmospheric corrosion and weather. Its excellent toughness makes it very resistant to handling damage.

Aluminum is another metal that is unaffected by ductile-to-brittle toughness changes at low temperatures. Consequently, the impact absorbing properties of the aluminum honeycomb impact limiters are unaffected by long term exposure to very low temperatures and up to 300° to 350°F. This temperature range exceeds the design normal and off-normal temperature extremes. Since the impact limiter is not credited with performing any function in a postulated post-accident fire, its performance in a fire is not relevant.

Aluminum has good atmospheric corrosion resistance, although this aspect of the metal is not credited due to the presence of the stainless weather cover which would preclude exposure to the elements during shipping.

Aluminum and stainless steel are relatively unreactive when coupled, thus, no adverse galvanic reactions will occur. This is borne out by extensive industrial experience.

The cask impact limiter attachment studs are fabricated from an austenitic stainless steel. Thus, the attachment hardware is immune to brittle fracture or atmospheric corrosion concerns.

Gamma shielding is provided by a poured lead (ASTM B29) shield that is contained and sealed within the double-walled construction of the cask body. Since the lead is contained within a stainless steel shell, corrosion concerns are precluded. In the event of a hypothetical fire, potential melting and subsequent loss of the lead is precluded by its containment within the double wall of the cask. Lead does not suffer a brittle-to-ductile transformation, so fracture of the shield is not a concern.

Neutron shielding is provided by a proprietary, cast-in-place, boron carbide filled polymeric shield material called NS-4-FR. This material is contained within a carbon steel jacket around the exterior and bottom of the transportation cask. An epoxy coating is applied to the steel jacket to prevent corrosion. Other carbon steel parts of the neutron shield assembly are electroless nickel plated to prevent corrosion.

The NS-4-FR absorber material has been previously evaluated in detail by the staff. Under design normal and off-normal conditions, it is inert and not susceptible to any significant degradation in service due to thermal or radiation effects. In a hypothetical fire the polymeric matrix may smoulder or burn. However, the boron carbide neutron absorber would not be consumed or chemically altered and would be held in place by the steel cover.

After the polymer is cast into place and catalyzes, it is an essentially inert material with regard to potential reactions with the environment or stainless or carbon steel materials of the cask structure and protective shield.

A commercial epoxy coating is used on the outer surfaces of the neutron absorber shield. This material is inert in the normal design and off-normal design environments. No adverse reactions with any other components of the transportation cask would occur.

Electroless nickel plating is used for some of the smaller components that make up the neutron absorber shield. This kind of plating is essentially inert and will not cause any adverse chemical or galvanic reactions with any components of the transportation cask, under all design conditions.

The TS125 requires the use of separate, detachable lifting trunnions for handling prior to/after shipping. The trunnions are fabricated from a precipitation-hardened stainless steel. The trunnions, attaching hardware and ancillary lifting devices are designed, analyzed, and tested in accordance with 10 CFR 71.45, NUREG-0612, and ANSI N14.6 to be single failure proof.

The TS125 Transportation Cask acts as a structural and confinement “overpack” for either a FuelSolutions™ W21 or W74 storage canister. As such, the TS125 design does not therefore depend upon the structural or containment integrity of the spent fuel loaded W21 or W74 storage canisters it is designed to transport. Consequently these payloads are not credited in the structural or containment analysis of the TS125 as providing these functions.

2.2 Structural Design Description for the W21 Canister

The primary components of the W21 Canister include the stainless steel canister shell assembly, the fuel basket assembly, and shield plugs. The canister shell assembly is a right cylindrical shell with top end inner and outer closure plates, bottom shell extension, and a bottom end plate. The shell assembly also includes top and bottom shield plugs. The W21 canister assemblies are of two different classes, one is called the W21M class, while the other is called the W21T class. Each of the W21 canister classes has four different types of canisters, depending on the length of the canister and the type of material used in the shield plug, such as Lead, Depleted Uranium (DU), or carbon steel. The overall length of the canister varies from a minimum of 182.3 inches to a maximum of 192.3 inches, while the exterior diameter is 66 inches. The cavity length in the canister varies from 163.0 inches to 180 inches, while the cavity diameter is 64.75 inches. The gross weight of the canister including the fuel varies from 69722 lbs. to 79873 lbs.

The W21 Canister is constructed and assembled in accordance with BNFL Fuel Solutions General Arrangement drawings listed below:

W21-110, W21 Canister Field Assembly, Sheets 1-9, Rev. 4.
W21-120, W21 Canister Basket Assembly, Sheets 1-10, Rev. 5.
W21-121, W21 Canister Spacer Plate, Sheets 1-1, Rev. 5.
W21-122, W21 Canister Basket Guide Tube Assembly, Sheets 1-2, Rev. 3.
W21-130, W21 Canister Shell Assembly, Sheets 1-9, Rev. 4.
W21-131, W21 Canister Shield Plug Support Vent/Drain Body & Top, Sheets 1-2, Rev. 3.
W21-140, W21 Canister Shield Plug Assembly, Sheets 1-4, Rev. 5.
W21-150, W21 Canister Top Closure Plates and Port Cover, Sheets 1-2, Rev. 4.
W21-190, W21 Canister Fuel Spacer Assembly, Sheets 1-1, Rev. 4.

2.2.1 Structural Evaluation of the W21 Canister

The W21 Canister shell assembly is made up of a right cylindrical shell, 0.625 inch thick austenitic stainless steel material (SA-240, Type 304 or 316), 1 inch thick bottom closure plate (SA-336, Type F304 or F316), 1 inch thick top inner closure plate (SA-240, Type 304 or 316), and 2 inches thick top outer closure plate (SA-240, Type 304 or 316). All plates are welded to the cylindrical shell using full penetration welds. The cylindrical shell is extended at the bottom for enclosing the bottom shield plug (which is either 2.125 inches thick DU, or 5.75 inches thick A36 steel, or 3.1 inches thick B29 Chemical Lead, or a combination of 1.625 inches thick DU and 1.875 inches thick A36 steel), and for attaching 1.75 inches thick bottom end plate (SA-240, Type 304 or 316). The Top Shield Plug (which is either 1.25 inches thick DU, or 2.125 inches thick DU, or 7.25 inches thick A36 steel, or 3.4 inches thick B29 Chemical Lead, or 3.8 inches thick B29 Chemical Lead) is supported by the support ring.

The pressure retaining components of the canister shell assembly are designed, fabricated, and inspected in accordance with the requirements of American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME III Code) Section III, Division 1, Subsection NB. The non-pressure retaining components, such as the bottom end plate, the shell extension at the bottom, top and bottom shield plugs, top shield support ring, and top shield lug casing, are designed, fabricated, and inspected to ASME Code, Section III, Division 1, Subsection NF.

The W21 Canister fuel basket assembly consists of 64.38 inches diameter circular spacer plates (0.75 inches thick Type B, SA-517 Grade P carbon steel, or 2 inches thick Type A, SA-240, Type XM-19 stainless steel, or 0.75 inches thick Type A, SA-517 Grade P carbon steel), spaced at 3 to 5 inches along the length of the assembly. The Type A spacer plates are kept in position axially by eight 3 inches diameter support rods (SA-564, Grade 630 or SA-479, Type XM-19). Each rod consists of 5 pieces with threaded ends, and are joined together with Type A spacer plates and the top and bottom ends, to form a basket structure. Support sleeves (3 inch Schedule 40 pipe, SA-312, Type 304L, or SA-106, Grade C), placed around the rods to support the Type B spacer plates. The support rods are torqued (150 ft-lbs), to keep the assembly from becoming loose during the transportation due to vibration and thermal effects.

The spacer plates have 21 machined openings (9.43 inches square) for the 9.28 inches square stainless steel guide tubes (SA-240 Type 316). The guide tubes are welded at the bottom spacer plate to two 11 gage attachment brackets. The 14 gage neutron absorber sheets are welded to the guide tubes at top and bottom.

The fuel basket assembly is designed, fabricated, and inspected in accordance with the requirements of American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME Code) Section III, Division 1, Subsection NG.

The applicant has evaluated the structures using the material properties based on the service temperatures, as described in Section 2.0 of the SAR. The applicant has also evaluated the fracture toughness of ferrous components for cyclic loads for a service life of 40 years in SAR Section 2.1.2.3.1. Environmental effects on the materials, such as the potential for corrosion and the material degradation due to irradiation have been addressed by the applicant in SAR Section 2.4.4.

The canister shell assembly is designed as the confinement boundary for on-site storage, but is not relied on as a confinement boundary during transportation. The TS125 Transportation Cask serves as the primary containment boundary for the spent nuclear fuel transported in the canister.

The W21 Canister is evaluated in the following section for compliance to the requirements of 10 CFR Part 71.

2.2.2.1 General Standards for all Packages (10 CFR 71.43)

Minimum Package Size

The canister is contained within the TS125 transportation cask, which meets the 10 CFR 71.43(a) requirement for the smallest overall dimension of 4 inches because the smallest dimension of the cask package is 143.5 inches.

Tamper-Proof Feature

The canister is contained within the TS125 Transportation Cask, which incorporates two wire-seals, one at each end of the package, which connect the impact limiters to the tie-down rings. The wire-seal is not readily breakable, and while intact, would be evidence that the package has not been opened by unauthorized persons, as required by 10 CFR 71.43(b).

Positive Closure

The source is contained within a canister, which provides a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. Section 4.0 of this SER addresses the containment requirements. Therefore, the package design meets the requirements of 10 CFR71.43(c).

Chemical and Galvanic Reactions

The W21 canister is constructed of stainless steel or coated carbon steel. These materials are close in the galvanic series and would not cause galvanic reactions. These materials also would not react chemically and are not affected adversely under irradiation. Therefore, the package design meets the requirements of 10 CFR 71.43(d).

2.2.2.2 Lifting and Tie-down Standards (10 CFR 71.45)

The W21 Canister is contained within the TS125 Transportation Cask, and is lifted and transported as a part of the cask. Therefore, the lifting and tie-down requirements of 10 CFR 71.45(a) and 10 CFR 71.45(b), addressed in the SER for the TS125 Transportation Cask, are not applicable to the W21 Canister.

2.2.2.3 Normal Conditions of Transport (10 CFR 71.71)

The W21 Canister has been evaluated for conditions and tests for Normal Conditions of Transport, as required by 10 CFR 71.71(c), and as described below:

Heat

The applicant analyzed the effects of solar heat (insolation) on the package, as described in Section 2.0 of the SER for the TS125 Transportation Cask. Temperatures in various components, as described in Section 3.0 of this SER, are used to evaluate the stresses based on equations and allowing for the gaps between different components. The spacer plates are analyzed using a two-dimensional plane stress analyses using the ANSYS finite element computer program. The results of the evaluation are given in SAR Table 2.6-5.

The staff has reviewed the applicant's input and methods of analysis, and finds that since the various canister components are free to expand due to changes in temperature during the NCT Heat conditions, the stresses would be insignificant, and thus would meet the ASME III Code requirements. Based on this, the staff has concluded that the W21 Canister meets the requirements of 10 CFR 71.71(c)(1).

Cold

The applicant has evaluated the condition of the cask using an ambient temperature of -40^{o}F in still air and shade, as required by 10 CFR 71.71(c)(2). The evaluation was performed as a part of the evaluation for NCT Heat condition. The results of the evaluation are given in the SAR Table 2.6-5.

Based on the review of the applicant's evaluation, the staff has concluded that the package would not be adversely affected by cold temperatures to -40^{o}F, and it would maintain structural integrity to meet the 10 CFR 71.71(c)(2) requirements.

Reduced and Increased External Pressure

The W21 canister is contained within the TS125 Transportation Cask, and is not affected by the changes in the atmospheric pressures. Therefore, it is concluded that the cask meets the requirements of 10 CFR 71.71(c)(3) and 10 CFR 71.71(c)(4).

Vibration

Loads on the canister due to vibration are computed based on Table 2 of ANSI N14.23 and are 2.0 g in the vertical direction and 0.1 g in lateral directions. Stress intensities in the canister components for the vibration loads are determined by the applicant using a finite-element analysis, and compared with the ASME III Code Service Level A allowable values (SAR Table 2.6-6), to verify compliance to the requirements of 10 CFR 71.71(c)(5). Based on the detailed review of the analysis methods and conclusions, the staff concluded that the canister meets the requirements of 10 CFR 71.71(c)(5).

Water Spray

The canister is contained within the TS125 TS125 Transportation Cask, which protects the canister from the water spray effects. Therefore, it is concluded that the canister meets the requirements of 10 CFR 71.71(c)(6).

Free Drop

To minimize the adverse impact on the structural integrity of the cask and its contents due to a potential drop during transportation, the applicant has provided two impact limiters, one at each end of the transportation cask to absorb the drop impact energy. For the NCT, a one foot drop is required to be considered in accordance with 10 CFR 71.71(c)(7).

Stress intensities in the canister components were calculated by the applicant using the ANSYS computer program, and a half-symmetrical finite-element model. The governing values of stress intensities are summarized in the SAR Table 2.6-8, and compared with the ASME III Code Service Level A allowable values to verify compliance to the requirements of 10 CFR 71.71(c)(7).

Based on the evaluation described in the Hypothetical Accident Conditions Free Drop for the TS125 Transportation Cask SER, the staff finds that the maximum acceleration of 15 g for the NCT free drop is acceptable.

2.2.2.4 Hypothetical Accident Conditions (10 CFR 71.73)

The applicant has evaluated the W21 Canister for the Hypothetical Accident Conditions (HAC) of 10 CFR 71.73(c), as discussed below, including the cumulative damage based on sequential application of the free drop, crush, puncture, and thermal tests.

Free Drop

The W21 Canister is evaluated for the end drop, the side drop, the corner drop, and the oblique drops at various angles from a height of 30 feet, in accordance with the requirements of 10 CFR 71.73(c)(1). The maximum deceleration values for various drops are based on the evaluation of the TS125 Transportation Cask, and are as discussed below.

End Drop:

For the end drop, a bounding deceleration design value of 60 g is used to evaluate the canister shell and the fuel basket. The canister shell components, except for the top shield plug and the ring support, are evaluated for the top and bottom end drops using an axi-symmetric finite element model of the ANSYS computer program. Effects of the internal pressure loading of 12 psig are added to the stress intensities due to the end drops. Thermal stresses are considered as secondary for the accident conditions and are not considered. The critical stress intensity values are compared to the ASME Code Service Level D allowable values, as shown in the SAR Table 2.7-8. The canister shell is also evaluated for buckling in accordance with the ASME Code Case N-284-1. The top shield plug and the support ring are evaluated for the bottom end drop using hand calculations.

The stresses in the spacer plates of the fuel basket assembly at 60 g's are determined by scaling the results of the analysis performed for the spacer plates in Section 3.7.3.2.1 of the W21 Canister Storage Final Safety Analysis Report (FSAR), for 50 g. All of the spacer plates, except the bottom spacer plate, are loaded by the self-weight only because the guide tubes are attached to the bottom spacer plate. The bottom spacer plate is designed for the weight of all guide tubes at an acceleration value of 20 g, which would be the maximum load the spacer plate would see because the attachment brackets fail in shear at this design load. The support rod assemblies are evaluated by scaling the results of the analysis performed for the W21 Canister Storage FSAR. The W21M-LD support rod assembly is used as a bounding assembly because it is the heaviest and the longest, and the strength of the materials in the assembly are lower than those of the other W21 support rod assemblies.

The guide tubes are evaluated for stresses and buckling potential using hand calculation. Stress intensities for each of the components of the fuel basket assembly are compared to the ASME III Code Service Level D allowable values, as shown in the Transportation SAR Table 2.7-9.

Side Drop:

The canister shell and the fuel basket assembly are evaluated for a bounding design acceleration loading of 60 g for the side drop. The evaluation is performed by scaling the results of the W21 Canister Storage FSAR for a similar evaluation, where appropriate. Results of the evaluation are shown in SAR Table 2.7-11.

Corner Drop:

The canister shell and the fuel basket assembly are evaluated for a bounding design acceleration loading of 40 g for the corner drop. The evaluation is performed by scaling the results of the W21 Canister Storage FSAR for a similar evaluation, where appropriate, and using the results of the evaluation for the end drop and side drop. Results of the evaluation are shown in SAR Table 2.7-12.

Oblique Drop:

The canister shell and the fuel basket assembly are evaluated for oblique drop primary impact and slap-down impact loadings, as shown in the TS125 Transportation Cask SAR Table 1.12-4. The evaluation is performed by scaling the results of the W21 Canister Storage FSAR for a similar evaluation, where appropriate, and using the results of the evaluation for the end drop and side drop. Results of the evaluation are shown in SAR Table 2.7-26.

Based on the detailed review of the applicant's methods of analyses, inputs and conclusions, and engineering judgement, the staff has concluded that the W21 Canister meets the 10 CFR 71.73(c)(1) requirements.

Crush

The W21 Canister is contained within the TS125 Transportation Cask. Since the mass of the cask is greater than 1100 lbs, the crush test is not required in accordance with the 10 CFR 71.73(c)(2) requirements.

Puncture

The W21 Canister is contained within the TS125 Transportation Cask, which protects the canister from being adversely affected by this test.

Thermal

The W21 Canister is contained within the TS125 Transportation Cask and thus protects the canister from being adversely affected by this test.

Immersion - fissile material

The requirements of 10 CFR 71.73(c)(5) do not apply to the W21 canister because the criticality analysis discussed in SAR Chapter 6 considers the water in-leakage.

Immersion - all packages

The W21 Canister is contained within the TS125 Transportation Cask and thus protects the canister from being adversely affected by this test.

2.2.2.5 Special Requirements for Irradiated Nuclear Fuel Shipments

The W21 Canister is contained within the TS125 Transportation Cask, which protects the canister from being adversely affected by this test. The TS125 Transportation Cask is evaluated for the 10 CFR 71.61 requirements in the SAR for the cask.

2.2.2.6 Fuel Rods

The applicant has evaluated the fuel rod buckling and the post-buckling behavior using a simplified approach. Even if the fuel rod buckles during a 30-feet drop event, the rod is laterally supported by the guide tube, and the rod would continue to maintain the fuel pellets in position. Additionally, the fuel rod assemblies are flexible and the impact load due to a 30-foot drop is of short duration. Based on this, the staff concluded that the fuel rods will maintain the structural integrity during the NCT and HAC conditions and tests, and thus meet the 10 CFR 71.71 and 10 CFR 71.73 requirements.

2.2.3 Materials

The staff reviewed the materials of construction for the W21 canister to evaluate what, if any, effect the conditions of transport might have upon the materials of this storage canister. The W21 was previously evaluated and licensed for storage under the regulations of 10 CFR 72.

The primary materials analysis for the W21 canister under transportation conditions was concerned with potential chemical or galvanic reactions with the TS125 transport cask and an evaluation of the transport design conditions upon the canister structural and confinement materials.

The W21 canister shell which forms the structural and confinement boundary is fabricated from austenitic stainless steel. Since this is the same class of material as used for the TS125 cask, there is no potential for adverse chemical or galvanic reactions. Similarly, there is no temperature dependent brittle-to-ductile fracture issue. Additionally, a brittle type failure response under transportation design loadings, particularly impacts, are precluded. Consequently, the staff finds there are no outstanding materials issues with the structural and confinement performance of the W21 canister during transportation.

The allowable cladding temperatures for the transportation mode were compared to the previously reviewed and approved cladding temperature limits established for the W21 storage canister. The regulations consider transportation as a short-term condition, and thereby allow the cladding temperature limit to be 570° C (based upon PNL-4835 and NUREG-1536). However, for consistency with the previous storage analyses, a temperature limit of 400° C was adopted as the limit for normal conditions of transport. This value is the same as that used in the storage analyses for off-normal conditions of storage.

For the hypothetical accident condition fire and post-fire conditions, the analyses showed that the short term limit of 570° C would be maintained. Thus, the vendor has shown that very conservative temperature conditions will exist for normal transportation conditions and that the normal limit of 570° C for transportation will be met under the hypothetical accident conditions.

2.3 Structural Design Description of the W74 Canister

The primary components of the W74 Canister include the stainless steel canister shell assembly, the fuel basket assembly, and shield plugs. The canister shell assembly is a right cylindrical shell with top end inner and outer closure plates, bottom shell extension, and a bottom end plate. The shell assembly also includes top and bottom shield plugs. The W74 canister assemblies are of two different classes, one is called the W74M class, while the other is called the W74T class. The overall length of the canister is 192.25 inches, while the exterior diameter is 66 inches. The cavity length in the canister is 173 inches, while the cavity diameter is 64.75 inches The gross weight of the canister including the fuel varies from 7,575 lbs. to 7,739 lbs.

The W74 Canister is constructed and assembled in accordance with BFS General Arrangement drawings listed below:

W74-110, W74 Canister Assembly, Sheets 1-2, Rev. 5.
W74-120, W74 Canister Basket Assembly, Sheets 1-6, Rev. 5.
W74-121, W74 Canister Spacer Plates, Sheets 1-1, Rev. 7.
W74-122, W74 Canister Basket Guide Tube Assembly, Sheets 1-1, Rev. 6.
W74-130, W74 Canister Shell Assembly, Sheets 1-2, Rev. 6.
W74-140, W74 Canister Shield Plug Assembly, Sheets 1-4, Rev. 5.
W74-150, W74 Canister Top Closure Plates and Port Covers, Sheets 1-2, Rev. 5.
3319, W74 Assembly and Detail Damaged Fuel Can, Sheets 1-5, Rev. 5.

2.3.1 Structural Evaluation of the W74 Canister

The W74 Canister shell assembly is made up of a right cylindrical shell, 0.625 inch thick austenitic stainless steel (SA-240, Type 304 or 316), 1 inch thick bottom closure plate (SA-336, Type F304 or F316), 1 inch thick top inner closure plate (SA-240, Type 304 or 316), and a 2 inches thick top outer closure plate (SA-240, Type 304 or 316). All plates are welded to the cylindrical shell using full penetration welds. The cylindrical shell is extended at the bottom for enclosing a 5.75 inches thick bottom shield plug (A36 steel), and for attaching a 1.75 inches thick bottom end plate (SA-24, Type 304 or 316). The top shield plate contains 37 independent shield plugs (7.25 inches thick, SA-516, Grade 55 or 60 steel), or a solid top shield plate (7.25 inches thick, TSA-36 steel). The top shield plug assembly is supported by eight shield plug support bars that are welded to the inside of the canister shell.

The pressure retaining components of the canister shell assembly are designed, fabricated, and inspected in accordance with the requirements of American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME Code) Section III, Division 1, Subsection NB. The non-pressure retaining components, such as the bottom end plate, the shell extension at the bottom, and top and bottom shield plugs, are designed, fabricated, and inspected to ASME Code Section III, Division 1, Subsection NF.

The W74 canister fuel basket assembly consists of two baskets stacked on each other. Each basket has multiple 64.38 inches diameter circular spacer plates (0.75 inches thick SA-517 Grade P carbon steel, or SA-240, Type XM-19 stainless steel), spaced at 5 to 8 inches along the length of the assembly. The spacer plates are kept in position axially by four welded support tubes that run through 3/16 inches thick support sleeves placed between the spacer plates. Attachment sleeves welded to the support tubes at the top and bottom spacer plates, provide the axial support to the spacer plates. A 2 inches thick engagement spacer plate (SA-240, Type XM-19), bolted at the bottom to the upper basket assembly support tubes, supports the fuel assemblies, while the basket is in the vertical orientation. The upper basket assembly sits freely on top of the lower basket assembly.

The spacer plates have 37 square machined openings (7.25 inches x 7.40 inches) for placing the 13 gage, 6.9 inches square stainless steel guide tubes (SA-240 Type 316). For the W74T canister, the guide tubes are not mechanically fastened to the spacer plates, but are placed free between the engagement spacer plate and the canister cavity. The 14 gage neutron absorber sheets are welded to the guide tubes on the sides.

The fuel basket assembly is designed, fabricated, and inspected in accordance with the requirements of American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, (ASME Code) Section III, Division 1, Subsection NG.

The applicant has evaluated the structures using the material properties based on the service temperatures, as described in Section 2.0 of the SAR. The applicant has also evaluated the fracture toughness of ferrous components for cyclic loads for a service life of 40 years in SAR Section 2.1.2.3.1. Environmental effects on the materials, such as the potential for corrosion and the material degradation due to irradiation have been addressed by the applicant in SAR Section 2.4.4.

The canister shell assembly is designed as the confinement boundary for on-site storage, but is not relied on as a confinement boundary during transportation. The TS125 Transportation Cask serves as the primary containment boundary for the spent nuclear fuel transported in the canister.

The W74 Canister is evaluated in the following section for compliance to the requirements of 10 CFR Part 71.

2.3.1.1 General Standards for all Packages (10 CFR 71.43)

Minimum Package Size

The canister is contained within the TS125 Transportation Cask, which meets the 10 CFR 71.43(a) requirement for the smallest overall dimension of 4 inches because the smallest dimension of the cask package is 143.5 inches.

Tamper-Proof Feature

The canister is contained within the TS125 Transportation Cask, which incorporates two wire-seals, one at each end of the package, which connect the impact limiters to the tie-down rings.

The wire-seal is not readily breakable, and while intact, would be evidence that the package has not been opened by unauthorized persons, as required by 10 CFR 71.43(b).

Positive Closure

The source is contained within a canister, which provides a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. Section 4.0 of this SER addresses the containment requirements. Therefore, the package design meets the requirements of 10 CFR 71.43(c).

Chemical and Galvanic Reactions

The W74 Canister is constructed of stainless steel or coated carbon steel. These materials are close in the galvanic series and would not cause galvanic reactions. These materials also would not react chemically and are not affected adversely under irradiation. Therefore, the package design meets the requirements of 10 CFR 71.43(d).

2.3.1.2 Lifting and Tie-down Standards (10 CFR 71.45)

Lifting Devices and Tie-down Device

The W74 Canister is contained within the TS125 Transportation Cask, and is lifted and transported as a part of the cask. Therefore, the lifting and tie-down requirements of 10 CFR 71.45(a) and 10 CFR 71.45(b), addressed in the SER for the TS125 Transportation Cask, are not applicable to the W74 Canister.

2.3.1.3 Normal Conditions of Transport (10 CFR 71.71)

The W74 Canister has been evaluated for conditions and tests for “Normal Conditions of Transport,” as required by 10 CFR 71.71(c), and as described below:

Heat

The applicant analyzed the effects of solar heat (insolation) on the package, as described in section 2.0 of the SER for the TS125 Transportation Cask. Temperatures of various components, as described in Section 3.0 of this SER, are used to evaluate the stresses based on equations and allowing for the gaps between different components. The spacer plates are analyzed using a two-dimensional plane stress analyses using the ANSYS finite element computer program. The results of the evaluation are given in SAR Table 2.6-4.

The staff has reviewed the applicant’s input and methods of analysis, and finds that the various canister components are free to expand due to changes in temperatures during the NCT Heat conditions, and that the stresses should meet the ASME III Code requirements. Based on this, the staff has concluded that the W74 Canister meets the requirements of 10 CFR 71.71(c)(1).

Cold

The applicant has evaluated the condition of the cask using an ambient temperature of -40°F in still air and shade, as required by 10 CFR 71.71(c)(2). The evaluation was performed as a part of the evaluation for NCT Heat condition. The results of the evaluation are given in the SAR Table 2.6-4.

Based on the review of the applicant's evaluation, the staff has concluded that the package would not be adversely affected by cold temperatures to -40°F, and it would maintain structural integrity to meet the 10 CFR 71.71(c)(2) requirements.

Reduced and Increased External Pressure

The W74 canister is contained within the TS125 Transportation Cask, and is not affected by the changes in the atmospheric pressures. Therefore, it is concluded that the cask meets the requirements of 10 CFR 71.71(c)(3) and 10 CFR 71.71(c)(4).

Vibration

Loads on the canister due to vibration are computed based on Table 2 of ANSI N14.23 and are 2.0 g in the vertical direction and 0.1 g in lateral directions. Stress intensities in the canister components for the vibration loads are determined by the applicant using a finite-element analysis, and compared with the ASME III Code Service Level A allowable values (SAR Table 2.6-5), to verify compliance to the requirements of 10 CFR 71.71(c)(5). Based on the detailed review of the analysis methods and conclusions, the staff concluded that the canister meets the requirements of 10 CFR 71.71(c)(5).

Water Spray

The canister is contained within the TS125 Transportation Cask, which protects the canister from the water spray effects. Therefore, it is concluded that the canister meets the requirements of 10 CFR 71.71(c)(6).

Free Drop

To minimize the adverse impact on the structural integrity of the cask and its contents due to a potential drop during transportation, the applicant has provided two impact limiters, one at each end of the transportation cask to absorb the drop impact energy. For the NCT, a one foot drop is required to be considered in accordance with 10 CFR 71.71(c)(7).

Stress intensities in the canister components were calculated by the applicant using the ANSYS computer program, and a half-symmetrical finite-element model. The governing values of stress intensities are summarized in the SAR Table 2.6-8, and compared with the ASME III Code Service Level A allowable values to verify compliance to the requirements of 10 CFR 71.71(c)(7).

Based on the evaluation described in the Hypothetical Accident Condition Free Drop for the TS125 Transportation Cask, the staff finds that the maximum design acceleration of 15 g for the NCT free drop is acceptable.

2.3.1.4 Hypothetical Accident Conditions (10 CFR 71.73)

The applicant has evaluated the W74 Canister for the Hypothetical Accident Conditions (HAC) of 10 CFR 71.73(c), as discussed below, including the cumulative damage based on sequential application of the free drop, crush, puncture, and thermal tests.

Free Drop

The W74 Canister is evaluated for the end drop, the side drop, the corner drop, and the oblique drops at various angles from a height of 30 feet, in accordance with the requirements of 10 CFR 71.73(c)(1). The maximum deceleration values for various drops are based on the evaluation of the TS125 Transportation Cask, and are as discussed below.

End Drop:

For the end drop, a bounding deceleration design value of 60 g is used to evaluate the canister shell and the fuel basket. The canister shell components, except for the top shield plug assembly, are evaluated for the top and bottom end drops using an axi-symmetric finite element model in the ANSYS computer program. Effects of the internal pressure loading of 12 psig are added to the stress intensities due to the end drops. Thermal stresses are considered secondary to the accident conditions and are not considered. The critical stress intensity values are compared to the ASME Code Service Level D allowable values, as shown in the SAR Table 2.7-2. The canister shell is also evaluated for buckling in accordance with the ASME Code Case N-284-1. The top shield plug assembly is evaluated by scaling the maximum stresses for the top shield plate in Section 3.7.3.1 of the W74 Canister Storage FSAR.

For the end drop, the spacer plates are supported by the four support tube assemblies. The stresses in the spacer plates of the fuel basket assembly for 60 g's are determined by scaling the results of the analysis performed for the spacer plates in Section 3.5.3.2.1 of the W74 Canister Storage FSAR, for 50 g. All of the spacer plates, except the bottom long-performance spacer plate, are loaded by the self-weight only because the guide tubes are attached to the bottom spacer plate. The bottom spacer plate is designed for the weight of all guide tubes at an acceleration value of 20 g, which is the maximum load the spacer plate would see because the attachment brackets fail in shear at this design load.

The support tubes nearest the impacting end provide longitudinal support for the self-weight, the weight of the spacer plates and sleeves, and the weight of the basket assembly, including the spent fuel assemblies. The support tubes for the W74M canister and the welds to the spacer plates were evaluated for 60 g's by scaling the maximum stresses calculated for a 50 g bottom end drop load in Section 3.7.3.2.3 of the W74 Canister Storage FSAR. Support tubes were also evaluated for buckling potential. The support sleeves were evaluated for the self-weight, and the weight of the spacer plates by hand calculations for stresses and buckling potential.

The guide tubes support their self-weight and are evaluated using hand calculations for stresses and buckling potential.

Stress intensities for the canister shell assembly are shown in the SAR Table 2.7-2. For each of the components of the fuel basket assembly, the stress intensities are shown in the SAR Table 2.7-1.

Side Drop:

The canister shell and the fuel basket assembly are evaluated for a bounding design acceleration loading of 60 g's for the side drop. The evaluation is performed by scaling the results of the W74 Canister Storage FSAR for a similar evaluation, where appropriate. Results of the evaluation are shown in SAR Table 2.7-4.

Corner Drop:

The canister shell and the fuel basket assembly are evaluated for a bounding design acceleration loading of 40 g's for the corner drop. The evaluation is performed by scaling the results of the W74 Canister Storage FSAR for a similar evaluation, where appropriate, and using the results of the evaluation for the end drop and side drop. Results of the evaluation are shown in SAR Table 2.7-5.

Oblique Drop:

The canister shell and the fuel basket assembly are evaluated for oblique drop primary impact and slap-down impact loadings, as shown in the TS125 Transportation Cask SAR Table 1.12-4. The evaluation is performed by scaling the results of the W74 Canister Storage FSAR for a similar evaluation, where appropriate, and using the results of the evaluation for the end drop and side drop. Results of the evaluation are shown in SAR Table 2.7-16.

Based on the detailed review of the applicant's methods of analyses, inputs and conclusions, and engineering judgement, the staff has concluded that the W74 Canister meets the 10 CFR 71.73(c)(1) requirements.

<u>Crush</u>

The W74 Canister is contained within the TS125 Transportation Cask. Since the mass of the cask is greater than 1100 lbs, the crush test is not required in accordance with the 10 CFR 71.73(c)(2) requirements.

<u>Puncture</u>

The W74 Canister is contained within the TS125 Transportation Cask which protects the canister from being adversely affected by this test.

Thermal

The W74 Canister is contained within the TS125 Transportation Cask, which protects the canister from being adversely affected by this test.

Immersion - fissile material

The requirements of 10 CFR 71.73(c)(5) do not apply to the W74 canister because the criticality analysis discussed in SAR Chapter 6 considers the water in-leakage.

Immersion - all packages

The W74 Canister is contained within the TS125 Transportation Cask, which protects the canister from being adversely affected by this test.

2.3.1.5 Special Requirements for Irradiated Nuclear Fuel Shipments

The W74 Canister is contained within the TS125 Transportation Cask, which protects the canister from being adversely affected by this test. The TS125 Transportation Cask is evaluated for the 10 CFR 71.61 requirements in the SAR for the cask.

2.3.1.6 Fuel Rods

The applicant has evaluated the fuel rod buckling and the post-buckling behavior using a simplified approach. Even if the fuel rod buckles during a 30 feet drop event, the rod is laterally supported by the guide tube, and the rod would continue to maintain the fuel pellets in its position. Additionally, the fuel rod assemblies are flexible and the impact load due to 30 foot drop is of short duration. Based on this, the staff finds the fuel rods will maintain the structural integrity during the NCT and HAC conditions and tests, and thus meet the 10 CFR 71.71 and 10 CFR 71.73 requirements.

2.3.2 Materials

The staff reviewed the materials of construction for the W74 canisters to evaluate what, if any effect the conditions of transport might have upon the materials of this storage canister. The W74 was previously evaluated and licensed for storage under the regulations of 10 CFR 72.

The primary materials analysis for the W74 canister under transportation conditions was concerned with potential chemical or galvanic reactions with the TS125 transport cask and an evaluation of the transport design conditions upon the canister structural and confinement materials.

As for the case of the W21 canister, the W74 is primarily fabricated of austenitic stainless steel. Thus the materials performance of the W74 will be similar to that of the W21 in transportation.

The allowable cladding temperatures for the transportation mode were compared to the previously reviewed and approved cladding temperature limits established for W74 storage canister. The regulations consider transportation as a short-term condition, and thereby allow the cladding temperature limit to be 570° C (based upon PNL-4835 and NUREG-1536). However, for consistency with the previous storage analyses, a temperature limit of 400° C was adopted as the limit for normal conditions of transport. This value is the same as that used in the storage analyses for off-normal conditions of storage.

For the hypothetical accident condition fire and post-fire conditions, the analyses showed that the short term limit of 570° C would be maintained. Thus, the vendor has shown that very conservative temperature conditions will exist for normal transportation conditions and that the normal limit of 570 °C for transportation will be met under the hypothetical accident conditions.

2.4 Evaluation of Findings

Based on the review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated, and that the structural performance of the TS125 Transportation package meets the structural requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The objective of this review is to verify that the thermal performance of the transportation cask, and W21 and W74 canisters, has been adequately evaluated for the tests specified under normal conditions of transport (NCT) and hypothetical accident conditions (HAC), and that the transportation package design satisfies the thermal requirements of 10 CFR Part 71.

On January 29, 2001, the NRC staff issued a Safety Evaluation Report (SER) for the FuelSolutions™ Canister Storage System (Docket No. 72-1026). Included in this SER was a complete thermal review of the W21 and W74 fuel canisters that are used for both storage and transportation. Evaluations were completed of the methodology for calculating the maximum allowable cladding temperatures, including peak rod pressure, and the estimated pressures and fuel and component temperatures for NCT and HAC. In addition, all calculated temperatures and pressures were confirmed by the staff. That review will not be repeated here, but will be referred to as necessary.

3.1 Description of the TS125 Transportation Cask

3.1.1 Thermal Packaging Design Features

The thermal design features of the cask include the following:

> The cask cavity length is designed to accommodate both the long (192 inches) and short (182 inches) FuelSolutions™ canister configurations. A 10-inch cask cavity spacer must be inserted into the bottom of the cask prior to loading the short canisters.
>
> The 0.5-inch annulus between the loaded canister and the transportation cask is backfilled with helium gas at 1 atm to enhance heat transfer from the canister.
>
> The side wall region of the transportation cask consists of a series of concentric cylinders or shells. The inner stainless steel shell is surrounded by a cylinder of chemical copper lead for attenuating gamma radiation. A thick outer stainless steel shell surrounds the lead gamma shield and provides additional structural support. Neutron shielding is provided by a 6-inch thick shell of NS-4-FR material surrounding the outer shell. The most external layer is the neutron shield jacket, a 3/16-inch thick layer of carbon steel enveloping/protecting the resin-like neutron shield material.
>
> Thirty-two carbon steel angles are spaced lengthwise between the cask outer shell and the neutron shield jacket to enhance heat transfer through the solid NS-4-FR neutron shield material to the ambient environment. These support ribs are welded to the cask outer shell and outer jacket using a continuous weld. The relatively high thermal conductivity of the carbon steel then distributes the heat around the exterior of the cask for efficient transfer to the ambient environment.
>
> The outer surface of the neutron shield jacket is coated with an epoxy-based coating to protect the jacket from corrosion, raise its emissivity, and lower its solar absorptance.

The impact limiters are formed of Cross-Core® aluminum honeycomb with an outer skin of stainless steel. They provide a heat transfer path between the cask ends and the ambient environment, and protection from high impact loads.

The personnel barrier is fabricated of an uncoated stainless steel mesh, with approximately 28 inches of clearance between the cask exterior and the barrier. The design provides a minimum 60 percent free opening, which yields a nearly unobstructed flow of air around the transportation cask for convective as well as radiative heat transfer to the ambient environment.

3.1.1.1 Codes and Standards

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant. Special materials are referenced to the manufacturer's data sheet or recent studies/papers.

3.1.1.2 Content Heat Load Specification

The envisioned multiple applications (W21 canisters, W74 canisters, short or long canister configurations) of the transportation cask supported the applicant's development of a bounding axial heat flux profile, as shown in Figure 3.1-4 in the SAR as the "maximum thermal profile". Assuming a limiting steady-state condition (design basis) and using the thermal evaluation code SINDA/FLUINT, this axial thermal profile was then used to determined the maximum heat load Q_{max} that would result in component temperatures still below the appropriate material allowable thermal limits. Acknowledging the fact that spent fuel elements have a wide range of active fuel lengths (91 inches for Yankee Rowe PWR fuel vs.150 inches for CE 16x16 PWR fuel), the applicant addresses the effect of loading a "more concentrated" (i.e., shorter) heat source into the transportation cask and develops the thermal rating criteria of a maximum linear heat generation rate ($LHGR_{max}$) on a per unit length basis. The following table summarizes the TS125 Transportation Cask Thermal Ratings:

Component	**Axial Heat Profile**	**max. peaking factor**	**Q_{max} (kW)**	**$LHGR_{max}$ (kW/in)**
Transportation Cask	max. thermal profile	1.095	22.0	0.211

where LHGR is defined as: $LHGR \equiv (Q \cdot PF) / AFL$

Q = heat load (kW)
PF = peaking factor (= 1.095)
AFL = active fuel length (inches)

If the thermal characteristics of a proposed fuel-loaded canister are bounded by the ratings above, it can then be inserted into the TS125 transportation cask. Otherwise, further analyses will be required.

3.1.1.3 Summary Tables of Temperatures

Table 3.4-1 in the SAR shows maximum temperature values obtained when simulating Normal Conditions of Transport (NCT). Not only are the package components (impact limiters, containment vessel, seals, shielding, neutron absorbers) identified with their corresponding maximum temperature values, but the design temperature limits are also identified. Table 3.4-2 provides details about the temperature distributions throughout the cask for NCT thermal load conditions. For the Hypothetical Accident Conditions (HAC, 30 minutes fire), Table 3.5-1 shows the temperature values (initial & peak) obtained from the simulations. All components remained below their limits during the fire. These temperature values are consistent with those presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions.

3.1.1.4 Summary Tables of Pressures in the Containment System

Due to the fact that transportation cask internal pressure calculation strongly depends on the specific canister and SNF payload, detailed calculations are only presented in the appropriate FuelSolutions™ Canister Transportation SARs. Since the transportation cask design pressure of 75 psig bounds the maximum normal operating pressure (MNOP) with any loaded FuelSolutions™ canister, this value is used in the SAR for structural evaluations.

3.1.2 Material Properties and Component Specifications

3.1.2.1 Material Properties

The application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. Also provided are the thermal emissivities and absorptivities used to model the radiative heat transfer to and away from appropriate internal and external surfaces within the package. The thermal properties used for the analysis are taken from sound technical references and are appropriate for the materials specified. Additionally, the fluid properties of the surrounding air and the backfill helium were provided for the evaluation of thermal conduction and convection parameters. These properties were appropriate for the conditions of the cask required by 10 CFR Part 71.

3.1.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant. All components were shown to satisfactorily perform under normal conditions of transport with an ambient temperature of -40°F.

3.1.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified.

3.1.3 Thermal Evaluation under Normal Conditions of Transport

3.1.3.1 Model

The 3-D thermal model was developed with the help of the Thermal Desktop® computer program to define a 180° model of the entire cask assembly (i.e., cask body, impact limiters, personnel barrier and a generic canister shell). The computer code SINDA/FLUINT® is then used for simulating the following steady-state conditions.

The following conservative assumptions are noted:

- The canister is assumed to be axisymmetrically positioned within the cask. In reality, during transportation, the canister would be in an eccentric position, laying on the two high rails that are attached to the cask inner shell. These two contacting surfaces would provide a better way for heat removal; even if only localized at the lower side of the cask.

- A small gap may exist between the lead shielding and the outer shell, due to different thermal expansion of the two concentric layers. Air is assumed to be the filler for this gap. The varying width of the gap (as a function of neighboring temperatures) as well as its heat transfer characteristics (conduction and radiation) are part of the thermal model.

- The neutron shield, together with the carbon steel support angles and the steel jacket are replaced/modeled with an equivalent conductivity and heat capacity. Air is conservatively assumed, instead of the resin NS-4-FR.

The following steady-state conditions were simulated with the computer code SINDA/FLUINT® :

Case 1: T_{air} = 100°F, with sun, maximum decay heat

Case 3: T_{air} = -20°F, no sun, maximum decay heat

Case 5: T_{air} = -40°F, no sun, maximum decay heat

3.1.3.2 Limiting Hot Conditions

The maximum allowable decay heat conditions (thermal ratings) are in fact obtained by exercising the SINDA/FLUINT® thermal model for the NCT hot (Case 1: 100°F ambient w/ solar) condition with increasing values of decay heat (Q) until one of the material thermal limits is barely reached; in this case, the allowed maximum NS-4-FR temperature of 338°F. Two bounding axial power profiles (see Figure 3.1-4 in the SAR) are considered so that the large differences in the length of existing spent fuels can be accounted for. This "search" process results in the establishment of the transportation cask thermal ratings:

Q_{max} = 22.0 (kW) $LHGR_{max}$ = 0.211 (kW/in)

Table 3.4-1 in the SAR summarizes the results from the hot condition calculation (Case 1). The most limiting component is the solid neutron material (NS-4-FR) whose long-term maximum allowable temperature pf 338°F was barely reached with the thermal rating heat loads Q_{max} and $LHGR_{max}$ (i.e., 333 and 329°F respectively). In the SAR, Figures 3.4-10 through 3.4-13 provide axial and radial temperature profiles resulting from the hot condition calculations.

Also in Table 3.4-1 of the SAR are the results from the simulations considering maximum heat load but cold ambient conditions (Cases 3 and 5). In the SAR, Figures 3.4-14 through 3.4-17 provide axial and radial temperature profiles resulting from the cold ambient condition calculations.

Under normal conditions of transport, all of the materials used remain below their respective failure temperatures.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(1).

3.1.3.3 Limiting Cold Conditions

With no decay heat, no insolation and ambient temperatures of -20°F and -40°F, these steady-state cases result in the entire package at the same temperature as the surrounding ambient. These temperature levels are within the allowable minimum temperature levels for all components.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(2).

3.1.3.4 Accessible Surface Temperature

Under normal conditions of transport, the package is enclosed by a protective screen to ensure that the accessible surface remains well below a temperature of 185°F. The applicant indicates that, even when accounting for the solar contribution, the personnel barrier surface reaches a maximum temperature value of 139°F.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.43(g) and must be shipped as exclusive use.

3.1.3.5 Maximum Normal Operating Pressure (MNOP)

Since cask internal pressures are strongly dependent on the specific canister and SNF payload to be loaded, the applicant defers the internal pressure calculation (MNOP) task to the SARs dealing with canister designs. Instead, the transportation cask design internal pressure of 75 psig was used for structural considerations, since it bounds any MNOP that will have to be calculated when analyzing the canister designs themselves.

3.1.3.6 Maximum Thermal Stresses

Maximum thermal stresses for NCT load conditions are evaluated in Section 2 of this SER.

3.1.3.7 Confirmatory Analysis

The confirmatory analysis performed by the staff was a simple 1-D ANSYS model for the transportation cask. The main reason for this effort was to simulate the fire scenario. However, since steady-state conditions had to be specified at the onset of the transient, this simple model allowed some comparisons against the applicant's NCT results; more specifically, radial temperature profiles. The staff found reasonable agreement with the applicant's predictions.

3.1.3.8 Summary

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71.

3.1.4 Thermal Evaluation under Hypothetical Accident Conditions

3.1.4.1 Model

The same SINDA/FLUINT® thermal model used for steady-state evaluations was applied, with the exception of a few conservative modifications made to account for the postulated damage sustained from the HAC 30-foot drop and puncture event:

- The impact limiters are assumed absent during the 30-minutes fire, but are in place and undamaged during the post-fire cool down. This approach maximizes the heat input into the cask structure from the fire and minimizes the post-fire cool down rate. Conservative structural analyses had indicated that the impact limiters would have been crushed and punctured, resulting in higher thermal conductivity in the vicinity of the crushed honeycomb and an opening up of a portion of the interior of the impact limit to direct fire.

- Credit is taken for the presence of NS-4-FR in the neutron shield during the fire, but the resin material is replaced by "air" during the cool down period. This results in the equivalent conductivity to be increased by approximately 32 percent during the fire, conservatively drawing more heat into the cask structure during the fire and dampening the heat removal during the post-fire period.

- The neutron shield jacket external surface emissivity and absorptivity are assumed to be 0.9 during and after the fire, accounting for the loss of the coating layer and soot build up during the fire

- The personnel barrier is assumed to be lost (not present).

- Quiet air conditions are assumed before and after the fire; however, during the fire, the wind speed is conservatively considered as 33 ft/sec (10 m/sec) .

The following fire-transient conditions were simulated with the computer code SINDA/FLUINT® :

Case 7: T_{air} = -20/1475/-20°F, no sun, maximum decay heat

Case 8: T_{air} = 100/1475/100°F, with sun, maximum decay heat

Following the 30-minutes fire event, the transient analyses are continued for a sufficient time to determine the maximum temperatures reached for all components.

3.1.4.2 30-Minute Thermal Test

Table 3.5-1 in the SAR summarizes the results from the two fire simulations. Initial and peak temperature values are provided for the transportation cask major components including metallic seals and vent & drain ports. Figures 3.5-1 through 3.5-6 in the SAR show the time evolution for the major component temperatures as well as "snapshots" of the axial temperature distribution at specified times during the transient.

Substantial thermal margins are observed during the fire transient, with peak temperatures well below the allowable short-term values. These findings were expected, due to the large thermal masses displayed by the transportation cask, and the NS-4-FR neutron shield layer which acts like a radiation shield to limit the amount of heat transferred into the cask during the 30-minutes event. The generic canister wall (most internal component in the model being analyzed) sees a temperature rise in the range of only 60°F.

Due to the conservative assumption of eliminating the impact limiters during the fire, the maximum inner shell, gamma shield, and outer shell temperatures occur at the junction with the top and bottom forgings. These temperature peaks occur only at the very ends of these components and are not representative of the general temperature levels achieved during the fire event. Appropriately, the applicant reports peak conditions for both inner and outer shell from locations that lie underneath the neutron shield. The maximum temperature reported for the gamma shield is the maximum value that occurs within the model.

3.1.4.3 Maximum Internal Pressure

For the steady-state calculations, cask internal pressures are strongly dependent on the specific canister and SNF payload to be loaded. The applicant defers the internal pressure calculation task to the SARs dealing with canister designs. Instead, the transportation cask design internal pressure of 75 psig was used for structural considerations, since it bounds any pressure that will have to be calculated when analyzing the canister designs themselves.

3.1.4.4 Maximum Thermal Stresses

Maximum thermal stresses for HAC load conditions are evaluated in Section 2 of this SER.

3.1.4.5 Confirmatory Analysis

The confirmatory analysis performed by the staff was a simple 1-D ANSYS model for the transportation cask in order to explore the thermal response to the fire. The main focus of this effort was to evaluate the role of the steel angles (within the neutron shield layer) in conducting heat to the outer shell. Effective material densities were derived so that masses were preserved for each component (inner shell, gamma shield, outer shell, neutron shield, jacket) that was considered for the 1-D model. The generic canister was replaced by a constant heat source. The ANSYS simulation confirms that the cask proportions (large thermal masses) are big enough to minimize the effects of a 30-minutes fire. The staff find the overall results acceptable.

3.1.4.6 Summary

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.73(4).

3.2 Description of the W21 Canister

3.2.1 Thermal Packaging Design Features

The thermal design features of the canister include the following:

> In order to accommodate a large fraction of the designated SNF PWR assemblies, two standard canister lengths: short (182 inches) and long (192 inches) and two canister classes: W21M and W21T are proposed. The main differences between the two classes are the stainless steel used for the canister shell (type 316 and type 304, respectively) and the use of stainless steel spacer plates at some locations in the W21M design. The multiple shield plug designs (thickness and materials: depleted uranium, steel, or lead) also reflect the envisioned diverse use for the canisters.
>
> The canister can hold up to 21 PWR SNF assemblies.
>
> Carbon steel spacer plates are used for increased thermal conductance.
>
> Helium gas is used to backfill the canister to an internal pressure of 24 atm (normal operation) in order to enhance heat transfer (conduction and convection) within the canister.

3.2.1.1 Codes and Standards

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant. Special materials are referenced to the manufacturer's data sheet or recent studies/papers.

3.2.1.2 Content Heat Load Specification

Since the bounding thermal ratings derived for the TS125 transportation cask apply, the applicant follows the same methodology of searching for maximum allowed values of heat load (Q_{max} and $LHGR_{max}$). First, bounding axial heat profiles are identified, as shown in Figure 3.1-1 in the SAR. Using the thermal evaluation code SINDA/FLUINT®, these axial profiles are then applied to a uniformly loaded model of the canister inside the transportation cask.   Limiting (design basis: NCT hot with sun) steady-state ambient conditions are assumed. Successive SINDA/FLUINT® simulations allow the determination of the maximum heat load Q_{max} that result in all component temperatures remaining below the appropriate material allowable thermal limits. Acknowledging the fact that spent fuel elements have a wide range of active fuel lengths (91 inches for Yankee Rowe PWR fuel vs.150 inches for CE 16x16 PWR fuel), the applicant addressed the effect of loading a “more concentrated” (i.e., shorter) heat source into the transportation cask and develops the thermal rating criteria of a maximum linear heat generation rate ($LHGR_{max}$) on a per unit length basis. The following table summarizes the W21 Canister within the TS125 Transportation Cask Thermal Ratings:

Component	Axial Heat Profile	max. peaking factor	Q_{max} (kW)	$LHGR_{max}$ (kW/in)
W21 Canister	max. thermal profile	1.095	1.05/assembly	0.0101/assembly
		Total:	22.0	0.211

where LHGR is defined as: $LHGR \equiv (Q \cdot PF) / AFL$

Q = heat load (kW)
PF = peaking factor (= 1.095)
AFL = active fuel length (inches)

The SAR does not address any situation where preferential loading (hotter assemblies in the center and cooler assemblies in the periphery, so that the total heat limit is still achieved) of SNF assemblies might occur. For this reason the staff considers it important to specify the thermal ratings on a per assembly basis.

If the thermal characteristics of a proposed fuel-loaded canister are bounded by the thermal ratings above, it can then be inserted into the W21 canister that is to be put inside a TS125 transportation cask. Otherwise, further analyses will be required.

3.2.1.3 Summary Tables of Temperatures

Tables 3.4-1 and 3.4-2 in the SAR show maximum temperature values obtained when simulating Normal Conditions of Transport under a heat load of Q_{max} and $LHGR_{max}$, respectively. Not only are the package components (fuel cladding, canister shell, containment vessel, shielding, neutron absorbers, seals, impact limiters) identified with their corresponding maximum temperature values, but the design temperature limits are also identified. For the Hypothetical Accident Conditions (30-minutes fire), Table 3.5-1 shows the temperature values (initial & peak) obtained from the simulations with a Q_{max} heat load. Given the similar peak temperature levels seen for the NCT evaluations under Q_{max} and $LHGR_{max}$ thermal ratings, the applicant only investigated/simulated the HAC scenarios for the Q_{max} thermal rating. All components remained below their limits during the fire. These temperature values are consistent with those presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions.

3.2.1.4 Summary Tables of Pressures in the Containment System

The pressure calculations of the containment system (canister + transportation cask) under normal conditions of transport and hypothetical accident conditions were reviewed and found inconsistent with the pressures presented throughout the appropriate sections of the SAR.

Both General Information and Containment Evaluation report a MNOP of 11.7 psig (instead of 11.9 psig) for NCT conditions and an internal pressure of 67.1 psig (instead of 68.2 psig). These discrepancies are small and do not affect the considerable safety margin. Tables 3.4-3 and 3.5-2 in the SAR summarize the calculations for NCT and HAC situations, respectively. The transportation design pressure (75 psig) is identified in Table 3.1-5. For all possible situations (including fuel rod failure with release of fission gases), the evaluations indicate a considerable margin in containment pressure.

3.2.2 Material Properties and Component Specifications

3.2.2.1 Material Properties

The application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the canister and cask. Also provided are the thermal emissivities and absorptivities used to model the radiative heat transfer to and away from appropriate internal and external surfaces within the package. The thermal properties used for the analysis are taken from sound technical references and are appropriate for the materials specified. Additionally, the fluid properties of the surrounding air and the backfill helium were provided for the evaluation of thermal conduction and convection parameters. These properties were appropriate for the conditions of the package required by 10 CFR Part 71.

3.2.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant in the TS125 Transportation Cask SAR. All components were shown to satisfactorily perform under normal conditions of transport with an ambient temperature of -40°F. References for the technical specifications for BORAL® and electroless nickel plating (for coating the carbon steel parts) were also provided by the applicant in the W21 transportation canister SAR.

3.2.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified. The fuel cladding temperature limits (752°F for NCT scenarios and 1058°F for the HAC fire test) were reviewed in Section 2 of this SER.

3.2.3 Thermal Evaluation under Normal Conditions of Transport

3.2.3.1 Model

The 3-D thermal model developed for the transportation cask (TS125 SAR) is now coupled with a 3-D thermal model of the canister and its constituents, developed with the SINDA/FLUINT® computer program. Submodels are developed for a typical spacer plate region (including associated sections of SNF and guide tubes), the bottom end of the canister (including the bottom end shield plug and the associated basket assembly spacer plate sections), and the top end of the canister (including the closure end shield plug and the associated basket assembly spacer plate sections). By properly combining these submodels, a full length description of the canister assembly is achieved. Most of the canister is modeled with 90° submodels except for the portion of the canister where the temperature gradient within the spacer plates is the highest. There is a lot of detail in each submodel component. The thermal response of the spent fuel assemblies is equivalently represented by slightly modified Manteufel & Todreas correlations $T_{fuel\ max}$ = function ($T_{wall\ guide\ tube}$). The same computer code SINDA/FLUINT® is then used for simulating limiting steady-state conditions.

The following conservative assumptions are noted:

- The W21M version of the FuelSolutions™ W21 canister design is used for the model development since it presents slightly lower thermal conductance in the radial direction. In the axial direction, the two designs (W21M & W21T) are basically thermally equivalent: the principle path of heat removal is radial and not axial, the differences in shield plug materials thermal properties (conductivity and heat capacity) are not that large and geometric differences tend to make them equivalent (e.g., although carbon steel has twice the conductivity compared to depleted uranium, the required carbon steel shield plugs are thicker). Also, since thermal loads are expected to be independent of the canister length, the short version of the W21M design is used for the bounding calculations that follow.

- The guide tubes are assumed to be centered in their respective spacer plate cutouts, and not in direct contact with the spacer plates.

- The conservative assumptions identified in the SER for the TS125 transportation cask are still applicable here, except for the canister now being eccentrically located within the transportation cask.

The following steady-state conditions were simulated with the computer code SINDA/FLUINT® :

Case 1: T_{air} = 100°F, with sun, maximum decay heat

Case 3: T_{air} = -20°F, no sun, maximum decay heat

Case 5: T_{air} = -40°F, no sun, maximum decay heat

3.2.3.2 Limiting Hot Conditions

The maximum allowable decay heat conditions (thermal ratings) are obtained by exercising the SINDA/FLUINT® thermal model for the NCT hot (Case 1: 100°F ambient w/ solar) condition with increasing values of decay heat (Q) until one of the material thermal limits is barely reached, or the TS125 Transportation Cask thermal ratings are reached. Two bounding axial power profiles (see Figure 3.1-1 in the SAR) are considered so that the large differences in the length of existing spent fuels can be accounted for. This "search" process results in the establishment of the W21 canister thermal ratings:

Q_{max} = 22.0 (kW) $LHGR_{max}$ = 0.211 (kW/in)

Tables 3.4-1 and 3.4-2, in the SAR, summarize the results from the hot condition calculations (Case 1). The most limiting component is the solid neutron material (NS-4-FR) whose long-term peak radial average temperature is within 10°F of the maximum allowable radial average temperature (300°F). The $LHGR_{max}$ thermal rating analysis yields even greater margin. In the SAR, Figures 3.4-11 through 3.4-13 provide axial and radial temperature profiles resulting from the hot condition calculations.

SAR Tables 3.4-1 and 3.4-2 show results from the simulations consideration of maximum heat load with cold ambient conditions (Cases 3 and 5). SAR Figures 3.4-14 through 3.4-19 provide axial and radial temperature profiles resulting from the cold ambient condition calculations.

Under normal conditions of transport, all of the materials remain below their respective temperature limits.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(1).

3.2.3.3 Limiting Cold Conditions

With no decay heat, no insolation and ambient temperatures of -20°F and -40°F, these steady-state cases result in the entire package at the same temperature as the surrounding ambient. These temperature levels are within the allowable minimum temperature levels for all components.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(2).

3.2.3.4 Accessible Surface Temperature

Under normal conditions of transport, the package is enclosed by a protective screen to ensure that the accessible surface remains well below a temperature of 185°F. The applicant indicates that, even when accounting for the solar contribution, the personnel barrier surface reaches a maximum temperature value of 139°F.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.43(g) and must be shipped as exclusive use.

3.2.3.5 Maximum Normal Operating Pressure (MNOP)

Table 3.4-3 in the SAR summarizes the internal pressure calculations performed by the applicant. Acknowledging the strong dependence between the SNF fill and fission gas quantities with the specific SNF assembly types, the maximum pressure is calculated for each SNF assembly type that can be accommodated by the W21 canister. Conservatively, the smallest loaded W21 canister free volume situation was considered for each fuel class. The MNOP is based on the initial cask helium backfill, the canister backfill, SNF rod fill gas, SNF fission gases, PWR control component gases, and no containment by the canister shell pressure boundary. Three percent of the SNF rods and three percent of the external burnable poison rods (if applicable) are considered failed. For each postulated rod failure, 100 percent of the rod fill gas, 30 percent of the SNF fission gases, and 30 percent of the gases generated within the PWR control rods are assumed to be released into the cavity. The MNOP was determined to be 11.9 psig (26.6 psia) which is less than the design basis value of 75 psig.

3.2.3.6 Maximum Thermal Stresses

Maximum thermal stresses for NCT load conditions are evaluated in Section 2 of this SER.

3.2.3. Confirmatory Analyses

The confirmatory analyses mentioned in the issuing of the SER for the FuelSolutions™ Canister Storage System (Docket No. 72-1026) are also applicable here, since the W21 canister design is the same.

3.2.3.8 Summary

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71.

3.2 Thermal Evaluation under Hypothetical Accident Conditions

3.2.4.1 Model

The same SINDA/FLUINT® thermal model used for steady-state evaluations was applied, with the exception of a few conservative modifications made to account for the postulated damage sustained from the HAC 30-foot drop and puncture event:

- The impact limiters are assumed absent during the 30-minutes fire, but are in place and undamaged during the post-fire cool down. This approach maximizes the heat input into the cask structure from the fire and minimizes the post-fire cool down rate. Conservative structural analyses had indicated that the impact limiters would have been crushed and punctured, resulting in higher thermal conductivity in the vicinity of the crushed honeycomb and an opening up of a portion of the interior of the impact limit to direct fire.

- Credit is taken for the presence of NS-4-FR in the neutron shield during the fire, but the resin material is replaced by "air" during the cool down period. This results in the equivalent conductivity to be increased by approximately 32 percent during the fire, conservatively drawing more heat into the cask structure during the fire and dampening the heat removal during the post-fire period.

- The neutron shield jacket external surface emissivity and absorptivity are assumed to be 0.9 during and after the fire, accounting for the loss of the coating layer and soot build up during the fire

- The personnel barrier is assumed to be lost (not present).

- Quiet air conditions are assumed before and after the fire; however, during the fire, the wind speed is conservatively considered as 33 ft/sec (10 m/sec).

The following fire-transient conditions were simulated with the computer code SINDA/FLUINT® :

Case 7: T_{air} = -20/1475/-20°F, no sun, maximum decay heat

Case 8: T_{air} = 100/1475/100°F, with sun, maximum decay heat

Following the 30-minute fire event, the transient analyses are continued for a sufficient time to determine the maximum temperatures reached for all components.

3.2.4.2 30-Minute Thermal Test

Table 3.5-1 in the SAR summarizes the results from the two fire simulations. Initial and peak temperature values are provided for the transportation package major components including metallic seals and vent & drain ports . Figures 3.5-1 through 3.5-2 in the SAR show the time evolution for the major component temperatures during the transient.

Substantial thermal margins are observed during the fire transient, with peak temperatures well below the allowable short-term values. Such findings were, in fact, expected due mostly to the large thermal masses displayed by the transportation cask, and the NS-4-FR neutron shield layer which acts like a radiation shield to limit the amount of heat transferred into the cask during the 30-minutes event. The canister shell experiences a temperature rise in the range of 20°F, whereas the peak fuel cladding temperature rises by only 16°F.

Due to the conservative assumption of eliminating the impact limiters during the fire, the maximum inner shell, gamma shield, and outer shell temperatures occur at the junction with the top and bottom forgings. These temperature peaks occur only at the very ends of these components and are not representative of the general temperature levels achieved during the fire event. Appropriately, the applicant reports peak conditions for both inner and outer shell from locations that lie underneath the neutron shield. The maximum temperature reported for the gamma shield is the maximum value that occurs within the model.

3.2.4.3 Maximum Internal Pressure

Table 3.5-2 in the SAR summarizes the internal pressure calculations performed by the applicant for the fire event. Acknowledging the strong dependence between the SNF fill and fission gas quantities with the specific SNF assembly types, the internal pressure is calculated for each SNF assembly type that can be accommodated by the W21 canister. Conservatively, the smallest loaded W21 canister free volume situation was considered for each fuel class. The internal pressure is based on the initial cask helium backfill, the canister backfill, SNF rod fill gas, SNF fission gases, PWR control component gases, and no containment by the canister shell pressure boundary. One hundred percent of both the SNF rods and the external burnable poison rods (if applicable) are considered failed. For each postulated rod failure, 100 percent of the rod fill gas, 30 percent of the SNF fission gases, and 30 percent of the gases generated within the PWR control rods are assumed to be released into the cavity. The evolution of the internal gases temperature during the fire dictates the level of pressure that is achieved. The peak internal pressure was determined to be 68.2 psig (82.9 psia) which is below the design basis value of 75 psig.

3.2.4.4 Maximum Thermal Stresses

Maximum thermal stresses for HAC load conditions are evaluated in Section 2 of this SER.

3.2.4.5 Confirmatory Analyses

The confirmatory analyses mentioned in the issuing of the SER for the FuelSolutions™ Canister Storage System (Docket No. 72-1026) are also applicable here, since the W21 canister design is the same.

3.2.4.6 Summary

The staff reviewed the applicant’s evaluation and finds that the package meets the requirements of 10 CFR 71.73(4).

3.3 Description of the W74 Canister

3.3.1 Thermal Packaging Design Features

The thermal design features of the canister include the following:

> Only one canister length (long, 192 inches) but two canister classes (W74M and W74T) are proposed. The main differences between the two classes are the stainless steel used for the canister shell (type 316 and type 304, respectively) and the use of stainless steel spacer plates at some locations in the W74M design.
>
> The canister is designed to hold up to 64 Big Rock Point BWR SNF assemblies, including a maximum of eight damaged fuel assemblies in damaged fuel cans. Upper and lower baskets, each containing 32 SNF assemblies, stack on top of each other. The ten cell locations (five per basket) at the center of the basket are mechanically blocked to prevent fuel assemblies from being loaded.
>
> Carbon steel spacer plates are used for increased thermal conductance.
>
> Helium gas is used to backfill the canister to an internal pressure of 24 atm (normal operation) in order to enhance heat transfer (conduction and convection) within the canister.

3.3.1.1 Codes and Standards

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant. Special materials are referenced to the manufacturer’s data sheet or recent studies/papers.

3.3.1.2 Content Heat Load Specification

Since the bounding thermal ratings derived for the TS125 Transportation Cask apply, the applicant follows the same methodology of searching for maximum allowed values of heat load (Q_{max} and $LHGR_{max}$). First, a bounding axial heat profile is identified, as shown in Figure 3.1-1 in the SAR. Using the thermal evaluation code SINDA/FLUINT®, these axial profiles are then applied to a uniformly loaded model of the canister inside the transportation cask. Limiting (design basis: NCT hot with sun) steady-state ambient conditions are assumed. Successive SINDA/FLUINT® simulations allow the determination of the maximum heat load Q_{max} that results in all component temperatures still below the appropriate material allowable thermal limits. In order to keep similarity with the TS125 Transportation Cask SAR and the transportation canister W21 SAR, a maximum linear heat generation rate thermal limit is also specified, using the fact that the only fuel expected to be loaded into W74 canisters is the Big Rock Point BWR SNF (70 inches active fuel length). The following table summarizes the W74 Canister within the TS125 Transportation Cask Thermal Ratings:

Component	**Axial Heat Profile**	**max. peaking factor**	**Q_{max} (kW)**	**$LHGR_{max}$ (kW/in)**
W74 Canister	max. thermal profile	1.220	0.344/assembly	0.0030/assembly
		Total:	22.0	0.192

where LHGR is defined as: LHGR ≡ (Q·PF) / AFL

Q = heat load (kW)
PF = peaking factor (= 1.220)
AFL = active fuel length (2x70 = 140 inches)

The SAR does not address any situation where preferential loading (hotter assemblies in the center and cooler assemblies in the periphery, so that the total heat limit is still achieved) of SNF assemblies might occur. For this reason the staff considers it important to specify the thermal ratings on a per assembly basis.

If the thermal characteristics of a proposed fuel-loaded canister are bounded by the thermal ratings above, it can then be inserted into the W74 canister that is to be put inside a TS125 Transportation Cask. Otherwise, further analyses will be required.

3.3.1.3 Summary Tables of Temperatures

Table 3.4-1 in the SAR shows maximum temperature values obtained when simulating Normal Conditions of Transport under a heat load of Q_{max} (and $LHGR_{max}$). Not only are the package components (fuel cladding, canister shell, containment vessel, shielding, neutron absorbers, seals, impact limiters) identified with their corresponding maximum temperature values, but the design temperature limits are also identified. For the Hypothetical Accident Conditions (30-minutes fire), Table 3.5-1 shows the temperature values (initial & peak) obtained from the simulations with a Q_{max} heat load. All components remained below their limits during the fire. These temperature values are consistent with those presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions.

3.3.1.4 Summary Tables of Pressures in the Containment System

The pressure calculations of the containment system (canister + transportation cask) under normal conditions of transport and hypothetical accident conditions were reviewed and found to be slightly inconsistent with the pressures presented throughout the appropriate sections of the SAR. Both General Information and Containment Evaluation report a MNOP of 11.7 psig (instead of 11.9 psig) for NCT conditions and an internal pressure of 67.1 psig (instead of 68.2 psig). However, these discrepancies are small and do not affect the considerable safety margin. Tables 3.4-3 and 3.5-2 in the SAR summarize the calculations for NCT and HAC situations, respectively. The transportation design pressure (75 psig) is identified in Table 3.1-5. For all possible situations (including fuel rod failure with release of fission gases), the evaluations indicate a considerable margin in containment pressure.

3.3.2 Material Properties and Component Specifications

3.3.2.1 Material Properties

The application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the canister and cask. Also provided are the thermal emissivities and absorptivities used to model the radiative heat transfer to and away from appropriate internal and external surfaces within the package. The thermal properties used for the analysis are taken from sound technical references and are appropriate for the materials specified. Additionally, the fluid properties of the surrounding air and the backfill helium were provided for the evaluation of thermal conduction and convection parameters. These properties were appropriate for the conditions of the package required by 10 CFR Part 71.

3.3.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant in the TS125 Transportation Cask SAR. All components were shown to satisfactorily perform under normal conditions of transport with an ambient temperature of -40°F. References for the technical specifications for borated stainless steel and electroless nickel plating (for coating the carbon steel parts) were also provided by the applicant in the W74 transportation canister SAR.

3.3.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified. The fuel cladding temperature limits (752°F for NCT scenarios and 1058°F for the HAC fire tests) were reviewed in Section 2 of this SER.

3.3.3 Thermal Evaluation under Normal Conditions of Transport

3.3.3.1 Model

The 3-D thermal model developed for the transportation cask (TS125 SAR) is now coupled with a 3-D thermal model of the canister and its constituents, developed with the help of the SINDA/FLUINT® computer program. Submodels are developed for a typical spacer plate region (including associated sections of SNF and guide tubes), the bottom end of the canister (including the bottom end shield plug and the associated basket assembly spacer plate sections), and the top end of the canister (including the closure end shield plug and the associated basket assembly spacer plate sections). By properly combining these submodels, a full length description of the canister assembly is achieved. Most of the canister is modeled with 90° submodels except for the portion of the canister where the temperature gradient within the spacer plates is the highest. There is a lot of detail in each submodel component. The thermal response of the spent fuel assemblies is equivalently represented by slightly modified Manteufel & Todreas correlations $T_{fuel\ max}$ = function ($T_{wall\ guide\ tube}$). The same computer code SINDA/FLUINT® is then used for simulating limiting steady-state conditions.

The following conservative assumptions are noted:

- The W74M version of the FuelSolutions™ W74 canister design is used for the model development since the two designs (W74M & W74T) are essentially thermally equivalent (in the radial and axial directions).

- The guide tubes are assumed to be centered in their respective spacer plate cutouts, and not in direct contact with the spacer plates.

- The conservative assumptions identified in the SER for the TS125 Transportation Cask are still applicable here, EXCEPT for the canister now being eccentrically located within the transportation cask.

The following steady-state conditions were simulated with the computer code SINDA/FLUINT® :

<u>Case 1</u>: T_{air} = 100°F, with sun, maximum decay heat

<u>Case 3</u>: T_{air} = -20°F, no sun, maximum decay heat

<u>Case 5</u>: T_{air} = -40°F, no sun, maximum decay heat

3.3.3.2 Limiting Hot Conditions

The maximum allowable decay heat conditions (thermal ratings) are in fact obtained by exercising the SINDA/FLUINT® thermal model for the NCT hot (Case 1: 100°F ambient w/ solar) condition with increasing values of decay heat (Q) until one of the material thermal limits is barely reached, or the TS125 Transportation Cask thermal ratings are reached. The bounding axial power profile (see Figure 3.1-1 in the SAR) reflects the unique loading of the Big Rock Point BWR SNF assemblies (two baskets axially stacked).

This "search" process results in the establishment of the W74 canister thermal ratings:

Q_{max} = 22.0 (kW) $LHGR_{max}$ = 0.192 (kW/in)

Table 3.4-1, in the SAR, summarizes the results from the hot condition calculations (Case 1). The most limiting component is the solid neutron material (NS-4-FR) whose long-term peak radial average temperature is within 11°F of the maximum allowable radial average temperature (300°F). In the SAR, Figures 3.4-11 through 3.4-13 provide axial and radial temperature profiles resulting from the hot condition calculations.

Also in Table 3.4-1, in the SAR, are the results from the simulations considering maximum heat load but cold ambient conditions (Cases 3 and 5). In the SAR, Figures 3.4-14 through 3.4-15 provide axial and radial temperature profiles resulting from the cold ambient condition calculations.

Under normal conditions of transport, all of the materials remain below their respective temperature limits.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(1).

3.3.3.3 Limiting Cold Conditions

With no decay heat, no insolation and ambient temperatures of -20°F and -40°F, these steady-state cases result in the entire package at the same temperature as the surrounding ambient. These temperature levels are within the allowable minimum temperature levels for all components.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71(c)(2).

3.3.3.4 Accessible Surface Temperature

Under normal conditions of transport, the package is enclosed by a protective screen to ensure that the accessible surface remains well below a temperature of 185°F. The applicant indicates that, even when accounting for the solar contribution, the personnel barrier surface reaches a maximum temperature value of 139°F.

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.43(g) and must be shipped as exclusive use.

3.3.3.5 Maximum Normal Operating Pressure (MNOP)

Table 3.4-2 in the SAR summarizes the internal pressure calculations performed by the applicant. Acknowledging the strong dependence between the SNF fill and fission gas quantities with the specific SNF assembly types, the maximum pressure is calculated for each SNF assembly type that can be accommodated by the W74 canister. The smallest loaded W74 canister free volume situation was conservatively considered. The MNOP is based on the initial cask helium backfill, the canister backfill, SNF rod fill gas, SNF fission gases, and no containment by the canister shell pressure boundary. Three percent of the SNF rods are considered failed. For each postulated rod failure, 100 percent of the rod fill gas and 30 percent of the SNF fission gases are assumed to be released into the cavity. The MNOP was determined to be 10.7 psig (25.4 psia) which is less than the design basis value of 75 psig.

3.3.3.6 Maximum Thermal Stresses

Maximum thermal stresses for NCT load conditions are evaluated in Section 2 of this SER.

3.3.3.7 Confirmatory Analyses

The confirmatory analyses mentioned in the issuing of the SER for the FuelSolutions™ Canister Storage System (Docket No. 72-1026) are also applicable here, since the W74 canister design is the same.

3.3.3.8 Summary

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.71.

3.3.4 Thermal Evaluation under Hypothetical Accident Conditions

3.3.4.1 Model

The same SINDA/FLUINT® thermal model used for steady-state evaluations was applied, with the exception of a few conservative modifications made to account for the postulated damage sustained from the HAC 30-foot drop and puncture event:

- The impact limiters are assumed absent during the 30-minutes fire, but are in place and undamaged during the post-fire cool down. This approach maximizes the heat input into the cask structure from the fire and minimizes the post-fire cool down rate. Conservative structural analyses had indicated that the impact limiters would have been crushed and punctured, resulting in higher thermal conductivity in the vicinity of the crushed honeycomb and an opening up of a portion of the interior of the impact limiter to direct fire.

- Credit is taken for the presence of NS-4-FR in the neutron shield during the fire, but the resin material is replaced by "air" during the cool down period. This results in the equivalent conductivity to be increased by approximately 32 percent during the fire, conservatively drawing more heat into the cask structure during the fire and dampening the heat removal during the post-fire period.

- The neutron shield jacket external surface emissivity and absorptivity are assumed to be 0.9 during and after the fire, accounting for the loss of the coating layer and soot build up during the fire.

- The personnel barrier is assumed to be lost (not present).

- Quiet air conditions are assumed before and after the fire; however, during the fire, the wind speed is conservatively considered as 33 ft/sec (10 m/sec) .

The following fire-transient conditions were simulated with the computer code SINDA/FLUINT® :

Case 7: T_{air} = -20/1475/-20°F, no sun, maximum decay heat

Case 8: T_{air} = 100/1475/100°F, with sun, maximum decay heat

Following a 30-minutes fire event, the transient analyses are continued for a sufficient time to determine the maximum temperatures reached for all components.

3.3.4.2 30-Minute Thermal Test

Table 3.5-1 in the SAR summarizes the results from the two fire simulations. Initial and peak temperature values are provided for the transportation package major components including metallic seals and vent & drain ports . Figures 3.5-1 through 3.5-2 in the SAR show the time evolution for the major component temperatures during the transient.

Substantial thermal margins are observed during the fire transient, with peak temperatures well below the allowable short-term values. Such findings were, in fact, expected due mostly to the large thermal masses displayed by the transportation cask, and the NS-4-FR neutron shield layer which acts like a radiation shield to limit the amount of heat transferred into the cask during the 30-minute event. The canister shell sees a temperature rise in the range of 32°F, whereas the peak fuel cladding temperature rises by only 15°F.

Due to the conservative assumption of eliminating the impact limiters during the fire, the maximum inner shell, gamma shield, and outer shell temperatures occur at the junction with the top and bottom forgings. These temperature peaks occur only at the very ends of these components and are not representative of the general temperature levels achieved during the fire event. Appropriately, the applicant reports peak conditions for both inner and outer shell from locations that lie underneath the neutron shield. The maximum temperature reported for the gamma shield is the maximum value that occurs within the model.

3.3.4.3 Maximum Internal Pressure

Table 3.5-2 in the SAR summarizes the internal pressure calculations performed by the applicant for the fire event. Acknowledging the strong dependence between the SNF fill and fission gas quantities with the specific SNF assembly types, the internal pressure is calculated for each SNF assembly type that can be accommodated by the W74 canister. Conservatively, the smallest loaded W74 canister free volume situation was considered for each fuel class. The internal pressure is based on the initial cask helium backfill, the canister backfill, SNF rod fill gas, SNF fission gases, and no containment by the canister shell pressure boundary. One hundred percent of the SNF rods are considered failed. For each postulated rod failure, 100 percent of the rod fill gas and 30 percent of the SNF fission gases are assumed to be released into the cavity. The evolution of the internal gases temperature during the fire dictates the level of pressure that is achieved. The peak internal pressure was determined to be 29.3 psig (44.0 psia) which is below the design basis value of 75 psig.

3.3.4.4 Maximum Thermal Stresses

Maximum thermal stresses for HAC load conditions are evaluated in Section 2 of this SER.

3.3.4.5 Confirmatory Analyses

The confirmatory analyses mentioned in the issuing of the SER for the FuelSolutions™ Canister Storage System (Docket No. 72-1026) are also applicable here, since the W74 canister design is the same.

3.3.4.6 Summary

The staff reviewed the applicant's evaluation and finds that the package meets the requirements of 10 CFR 71.73(4).

3.4 Evaluation Findings

The applicant states that the effects of material property uncertainties were accounted for by conservatisms in the analytical methods presented for the steady-state and transient simulations. The staff finds that the fire simulations do indeed show a wide thermal margin for all TS125 cask components. However, the staff feels the need to reiterate how the thermal ratings (Q_{max} and $LHGR_{max}$) were determined by searching for a maximum heat load value that would cause an allowable material temperature limit to be barely reached. Of all the proposed conservative assumptions for the steady-state model, the staff considers the evaluation of the effective neutron shield conductivity as the one that most directly affects the radial temperature distributions. By replacing the NS-4-FR with air, the effective conductivity is reduced by approximately 24 percent. Based on the provided temperature profiles (e.g., Figure 3.4-11 in the SAR), the temperature drop across the neutron shield layer is around 120°F. For the same local heat flux, a 24 percent increase in conductivity would result in a temperature drop of roughly 90°F, resulting in a safety margin of 30°F.

On the other hand, the assumption of a 0.30 absorptivity for the epoxy coating definitely lacks conservatism. The applicant, in fact, mentions in the SAR the need for periodic maintenance so that the cask outer surface performs as expected. If the outer surface were to absorb more of the sun's energy, its temperature would rise together with all internal temperatures. The 30°F margin that has just been identified now begins to get challenged. Other details that are not mentioned are the sensitivity of the SINDA/FLUINT® results to the nodalization scheme that was applied for cask modeling as well as the precision expected from this code package. The sole purpose of this paragraph is to highlight the fact that design margins are small due to the way the proposed thermal ratings are derived. As for the W21 and W74 canisters, all calculations indicate a fairly wide margin of safety relative to the allowable temperature limits for the different materials.

As indicated in the SER for the FuelSolutions™ Storage System (Docket No. 72-1026), the confirmatory work[1] (performed by Pacific Northwest National Laboratory (PNNL), while providing technical assistance), was unable to confirm the applicant's evaluation that compensatory buoyance forces enhance the thermal performance of the W21 and W74 canisters. Due to the large margin in the calculations, the staff had reasonable assurance that any degradation in gas convective properties (due to the release of fission gases) will not result in gross cladding failures.

Even though the main body of the thermal evaluation considered the canister loaded with undamaged UO_2 fueled assemblies, the applicant still addresses the use of the W74 canister for storing: MOX, damaged, and partial fuel assemblies. From a thermal point of view (material properties and limits), these assemblies are similar to the intact assemblies considered. Due to the considerable margin observed for the fuel cladding temperature, the presence of an extra thermal "barrier"such as the thin wall of the damaged fuel can, is not an issue. The staff concludes that as long as the SNF assembly (be it damaged or undamaged, UO_2 or MOX, complete or partial) is bounded by the thermal ratings defined earlier in this SER, their loading in the W74 canister is acceptable.

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the thermal performance requirements of 10 CFR Part 71.

3.5 References

Memorandum to: M.W. Hodges from K. Gruss, "PNNL Technical Evaluation Report on Cladding Behavior for High Burnup Fuels," with attachment by E.R. Gilbert, C.E. Beyer, and E.P. Simonen, Pacific Northwest National Laboratory, "Technical Evaluation Report of WCAP-15168 (Dry Storage of High Burnup Spent Fuel)", February 2000 .

4.0 CONTAINMENT EVALUATION

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

4.1 Description of the Containment Boundary

Sections 1.2.1, 1.3.2, and 4.1.1 of the SAR describe the containment boundary. Containment of radioactive materials is provide by the TS125 Transportation Cask and is "leaktight" as defined by ANSI N14.5 (leakage rate less than or equal to 1 x 10^{-7} ref-cm^3/s). The individual transportation cask components that form the containment boundary are: the inner cylindrical shell, the bottom plate forging (which forms the bottom closure of the cask), the top ring forging and sealing surfaces, the closure lid and sealing surfaces, the closure bolts, the inner most closure lid O-ring seal, the cavity vent port seal gland and O-ring seal, and the cavity drain port seal gland and O-ring seal. No credit for containment is taken for the W21 or W74 canister assemblies or welds.

These canisters do not include a separate inner containment system (for loading more than 20 curies of plutonium per canister of damaged fuel debris, etc.). The W74 canister does contain a separate damaged fuel canister to facilitate handling of damaged fuel and confine gross fuel particles to a known subcritical volume under NCT and HAC conditions. This canister is made up of a special overpack and no containment credit is taken for the confinement characteristics offered by this overpack.

All components of the containment boundary are manufactured and welded according to the ASME Code, Section III, Division 3, Subsection WB, or accepted alternatives; as discussed in Section 1.3.3 of the SAR. All non-containment boundary components of the transportation cask body are designed and fabricated as an ASME Section III, Class I component support in accordance with the applicable requirements of Subsection NF, as discussed in Section 1.3.3 of the SAR.

Welding is discussed in Sections 4.1.1.1, 1.3.3, and 2.3 of the SAR. In addition, a welding schedule is presented on SAR page 2.3-4. All containment welds are full penetration welds. Where possible, these welds are 100 percent radiographically examined in accordance with Section III, Division 3, Subsection WB, of the ASME code. Multi-layer liquid penetrant examination is performed where radiographic examination can not be performed. The materials section of this SER further describes welds and welding details that are examined by this process. The staff accepts that the welding practices documented in the SAR are sufficient to assure the structural and containment integrity of the package during NCT and HAC conditions.

4.1.1 Positive Closure

The lid of the containment system is closed by 60 bolts at a torque of 2800 ± 50 ft-lbs. The vent and drain ports are closed by a plug assembly held in place by one bolt at a torque of 250 ± 25 ft-lbs. These bolts insure that no unintentional opening or opening as a result of an overpressure may occur (as described in Section 2 of the SAR and SER). The materials section of the SER provides an evaluation of these bolts. The closure lid bolts are SB-637, grade

N07718 (as described on drawing FS-200), and the port plug assembly bolts are stainless steel 18-8 (as described on drawing FS-210, sheet 1). The closure lid bolts are discussed further in the materials section of the SER, and the port plug assembly bolt material has been previously evaluated and accepted by the staff.

All non-welded containment boundary seals are “Helicoflex” silver metallic O-ring seals, and are replaced after each use. Section 1.3.4.1 of the SAR and the materials evaluation section of the SER, provide product literature and temperature performance capabilities for these metallic O-ring seals. These seals, along with the bolts and applicable torque values specified earlier, assure that the package remains “leaktight” as described in ANSI N14.5, during NCT and HAC conditions.

The leak test procedures are located in Section 7.4 of the SAR. The staff notes that leak testing personnel should be certified in accordance with SNT-TC-1A or an equivalent standard. The staff finds that these leak testing requirements will assure that the package meets the requirements of ANSI N14.5.

The staff finds that the performance capabilities of the seals, bolts, and testing procedures presented are adequate to meet the closure requirements of 10 CFR part 71.

4.1.2 Chemical and Galvanic Reactions

Section 2.4.4 of the SAR and the materials evaluation section of the SER describe the chemical and galvanic evaluations performed on this package. The nickel anti-seize grease on the bolts and the vacuum grease on the seals have been evaluated for these reactions. The electroless nickel plating, and Keeler and Long coatings on outer cask body materials (NS-4-FR jacket) have been evaluated for these reactions. The non-containment boundary seals present are elastomer and have been evaluated for these reactions. The staff finds that no chemical or galvanic reactions are possible between these materials and that the containment characteristics of the package will not be affected by any chemical or galvanic reactions.

4.1.3 Unauthorized Operation of Valves and Similar Devices

As described in the SAR, Section 2.4.5, the package contains only two containment boundary penetrations. These penetrations are the vent port located in the closure lid, and the drain port located in the bottom forging plate. During transportation, these ports are fully recessed, plugged, and covered with a stainless steel protective cover. Further, the ports are located under the impact limiters. The staff finds that these ports are protected against unauthorized operation. The containment system of the package contains no pressure relief valves or similar devices, which could allow for continuous venting.

4.1.4 Transport of Damaged SNF

This design is not approved to ship damaged fuel (e.g., debris, particles, loose pellets, or fragmented rods or assemblies) having more than 20 curies of plutonium per canister.

4.2 Containment Under Normal Conditions of Transport

The staff has reviewed the evaluation of the containment system under normal conditions of transport and concludes that the package is, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71 (normal conditions of transport) the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system.

The TS125 Transportation Cask is designed for a maximum normal operating pressure (MNOP) of 75 psig. Assuming that the W21 and W74 canister is breached, NCT MNOP's are calculated at 11.7 psig and 10.7 psig respectively. Section 3.4.4 of the respective canister SARs provides the details and methodologies of these calculations.

4.3 Containment Under Hypothetical Accident Conditions

The staff has reviewed the evaluation of the containment system under hypothetical accident conditions and concludes that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for hypothetical accident conditions, with no dependence on filters or a mechanical cooling system.

HAC MNOP's are calculated at 67.1 and 29.3 psig for the W21 and W74 canister respectively. Section 3.4.4 of the respective canister SARs provides the details and methodologies of these calculations.

Since the package is leaktight, and gas leakage tests are used to demonstrate compliance, the HAC containment criteria are identical to the NCT criteria discussed in SAR Section 4.2.2 and Table 4.2-1. There are no associated calculations for allowable release rates or allowable leakage rates, and there would be no escape of ^{85}Kr exceeding 10 A_2 in one week and no escape of other radioactive material exceeding a total amount of A_2 in 1 week.

The adequacy of the TS125 Transportation Cask's containment system is demonstrated by analysis for HAC. Detailed structural analyses presented in Chapter 2 of the SAR, and Chapter 2 of the SER, show that there will be no direct release of radioactive material due to HAC loading. Additionally, the closure bolts maximum stress are lower than the bolt yield strength for all HAC loading conditions. Thus, no deformation of the closure bolts is expected. The structural analyses also show that the cask body allowable stress design criteria of the ASME BPVC, Section III, Division 2, Subsection WB is satisfied for all HAC loadings. The thermal evaluation demonstrates that the maximum temperatures of the containment boundary closure and port plug O-rings remain within the allowable temperature range for all HAC thermal loadings. Finally, the structural evaluation shows that no "burping" of the containment system is possible due to the specified bolt torque values. Therefore, the containment system will maintain leaktight conditions under the most severe HAC loading.

4.4 Evaluation Findings

In summary the staff has reviewed the Containment Evaluation section of the SAR and concludes the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR 71, and that the package meets the containment criteria of ANSI N14.5.

4.5 References

1. U.S. Code of Federal Regulations, “Packaging and Transportation of Radioactive Material,” Title 10, Part 71, January 1, 2001.

2. Institute for Nuclear Materials Management, ANSI N14.5, “American National Standard for Leakage Test on Packages for Shipment of Radioactive Materials,” New York, 1997.

3. American Society of Mechanical Engineers, “ASME Boiler and Pressure Vessel Code, Section III, Division 3, Containment Systems and Transport Packagings for Spent Nuclear Fuel and High Level Radioactive Waste,” New York, NY, 1998.

4. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,” NUREG-1617, March, 2000.

5. U.S. Nuclear Regulatory Commission, “Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks,” OMB No. 3150-0011, Bulletin 96-04, U.S. Government Printing Office, Washington, D.C., July 5, 1996.

5.0 SHIELDING EVALUATION

The objective of this review is to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.1 Shielding Design Description

Gamma-radiation shielding for the FuelSolutions TS-125 transportation package is provided by both the shielded canister and the transportation overpack. The overpack is a steel-lead-steel composite on the sides surrounded by an NS-4-FR neutron shield, which is enclosed in a neutron shield jacket. The inner and outer shells are constructed of SA-240, Type XM-19 stainless steel. The inner shell is 1.5 inches thick and the outer shell is 2.65 inches thick. Between the inner and outer shells is a 3.25 inch-thick lead gamma shield. The radial neutron shield is comprised of 6 inches of NS-4-FR enclosed in a 0.1875-inch thick shield jacket made of A516, Grade 70 steel. The overpack lid and bottom forging are constructed of 6-inch thick SA-240, Type XM-19 stainless steel. The bottom neutron shield is composed of a 5.25-inch thick NS-4-FR covered with a 0.25-inch thick neutron shield jacket made of A240, Type 304 stainless steel.

Both of the canisters provide axial shielding. The canister shell of both canisters is 0.63 inches of stainless steel. The axial top shielding consists of an inner and outer closure plate, an optional steel top plate, a shield plug, which varies among steel, lead and depleted uranium, depending on the canister design and an optional steel bottom plate. When the shield plug is a material other than steel, the top and bottom steel plates are used to encase the shield plug. The shielding through the bottom of the canister consists of a steel end plate, shield plug and a closure plate.

The W21 canister has two classes of canister. Each class of canister has four different types. The W21T canister class consists of a long lead (LL), long steel (LS), short lead (SL), and short steel (SS) canister. The W21M canister has long, depleted uranium (LD); long steel (LS); short, depleted uranium (SD), and short steel (SS) designs. Details of the W21 canister shielding are provided in Table 5-1.

The W74 canister has a 5.8-inch thick steel shield plug on the bottom encased in a 1.0 inch thick closure plate and a 1.8-inch thick steel end plate. The top of the canister has a 1.0-inch thick inner closure plate and a 2.0-inch thick outer closure plate. The top shield plug is 7.25 inches of steel. The W74 canister is a long steel (LS) design.

Table 5-1 - W21 Cannister details								
Class	W21M				W21T			
Type	-LD	-LS	-SD	-SS	-LL	-LS	-SL	-SS
Radial Shell	0.63 inches Stainless steel (all Types)							
Top Closure Details								
Top Closure Plate	2.0 inches Stainless steel (all Types)							
Inner Closure Plate	1.0 inch Stainless steel (all Types)							
Shield Plug (Top Sheet)	0.12" SS	N/A	0.12" SS	N/A	0.12" SS	N/A	0.12" SS	N/A
Shield Plug	2.1" DU	7.25" SS	1.3" DU	7.25" SS	3.4" Lead	7.25" SS	3.8" Lead	7.25" SS
Shield Plug (Bottom Sheet)	1.6" Steel	N/A	3.6" Steel	N/A	1.6" Steel	N/A	1.6" Steel	N/A
Bottom Closure Details								
Closure Plate	1.0" Steel	1.0" Steel	1.6" Steel	1.0" Steel	1.0" Steel	1.0" Steel	1.0" Steel	1.0" Steel
Shield Plug	2.1" DU	5.8" Steel	1.9" DU	5.8" Steel	3.1" Lead	5.8" Steel	3.1" Lead	5.8" Steel
End Plate	1.8" Steel	1.8" Steel	1.8" Steel	1.8" Steel	1.0" Steel	1.8" Steel	1.8" Steel	1.8" Steel

5.2 Radiation Source

The applicant evaluated the source term from the contents of both the W21 and the W74 canisters. The radiation source specification is presented in Section 5.2 of the SAR. Generic photon and neutron source terms were generated with the ORIGEN-2.1 computer code and four standard ORIGEN libraries (PWR-US, PWR-UE, BWR-US and BWR-UE).

The applicant performed the source term calculations for the Westinghouse standard 17x17 PWR and GE-5 8X8 BWR fuel assemblies. The applicant has shown that they yield the largest source term per metric ton of initial heavy metal (MTIHM) for each reactor type. A generic source term, in units of gammas/sec-MTIHM, was determined for each burnup and enrichment combination. The source term was determined by multiplying the maximum amount of uranium for a given fuel assembly by the generic source term for each energy group. The required cooling times for the W21 canister are shown in Table 5-2.

Table 5-2 - W21 Required Cooling Times			
Burnup (Gwd/MTU)	Min. Initial Enrichment (w/o ^{235}U)	Max. Assembly Hardware Cobalt (g)	Minimum Cooling Time (years)
≤35	≥1.5	≤50	15
	≥2.8	≤11	6
≤40	≥1.5	≤50	20
	≥3.0	≤11	8
≤45	≥1.5	≤50	25
	≥3.3	≤11	10
≤50	≥2.5	≤50	25
	≥3.5	≤11	12
≤55	≥3.0	≤50	25
	≥3.8	≤11	15
≤60	≥3.5	≤50	25
	≥4.0	≤11	18

The W74 canister source term was determined for a maximum burnup of 32 GWd/MTU, 3 w/o enrichment, a maximum of 2.9 grams of cobalt in the fuel region and a minimum cool time of 6 years.

The calculated source terms include radioactive isotopes in both the active fuel and activated hardware. Source terms were evaluated for the active fuel, plenum zone, top-fitting zone, and bottom-fitting zone of the SNF assemblies. Appropriate axial burnup profile parameters were applied for the design basis fuel in the source term modeling.

The applicant's methods for calculating the radiation source terms were reviewed. The staff used the SCALE-4.4 SAS2H and ORIGEN-S computer modules to perform verification analyses and found acceptable agreement with the applicant's reported values.

5.3 Shielding Model

The models and specifications for shielding are presented in Section 5.3 of the SAR. The applicant's shielding models for normal conditions of transport and hypothetical accident conditions consist of various representations of the overpack and canisters using the design drawings in Section 1.3 of the SAR. The overpack models for normal conditions of transport are shown in Figures 5.3-1 through 5.3-3 and hypothetical accident conditions are depicted in SAR Figures 5.3-4 through 5.3-5 of WSNF-120.

The radiation source is divided into four axial regions: bottom end fitting, fuel region, gas plenum, and top end fitting. The relative positions of these source term regions are depicted in the figures in Section 5.3 of each of the canister SARs. The fuel assembly regions are modeled as homogeneous zones. The end fittings and plenum regions are modeled as homogeneous regions of stainless steel, Inconel, and Zircaloy.

The variation of the maximum peaking factors with burnup is shown in Tables 5.2-10 and 5.2-14 of WSNF-121 for the W21 canister (PWRs). Variation of the maximum peaking factor with burnup for the W74 canister (Big Rock Point fuel assemblies) is shown in Tables 5.2-4 and 5.2-6 of WSNF-123 for the W74 canister. The neutron profile was used to account for the non-linear buildup of neutron source terms (primarily Cm-244) as a function of burnup. The photon source distributions within the plenum, top end fittings, and bottom end fittings were assumed to be uniform.

The composition and densities of the materials used in the shielding analysis are presented in tables located in Section 5.3 of the package SAR and the canister SARs. The accident condition analyses are performed for a complete loss of the hydrogen and boron in the radial neutron shield.

The staff evaluated the SAR shielding models and found them to be acceptable. The model dimensions and material specifications are consistent with the drawings in Section 1 of the SARs and provide the basis for reasonable assurance that the FuelSolutions transportation package was adequately modeled in the shielding analysis.

5.4 Shielding Evaluation

The applicant calculated the dose rates for normal conditions of transport and hypothetical accident conditions using the design basis source terms for both canisters. The dose rates were determined for 86 different locations along the side, top and bottom of the package. The locations were chosen to ensure that the package meets the dose rate limits in 10 CFR Part 71 on the surface and at 2 meters from the surface for normal conditions and 1 meter from the surface for accident conditions.

Additionally, the applicant performed three shielding evaluations that are generic to both canisters. The applicant evaluated the contribution to the dose rate outside the package due to streaming through the fins in the neutron shield, streaming through the shear key for normal conditions, and the effects on the dose rate due to the tests for hypothetical accident conditions. The applicant developed correction factors to be applied to the dose rate calculations based on the results from the evaluations described in Section 5.3.1 for the streaming calculations.

For hypothetical accident conditions, the applicant evaluated streaming through the gamma shield due to lead slump. In the structural evaluation, the applicant determined, using finite element analysis, that the maximum amount of lead slump due to an end drop ranges from 1.06 to 1.12 inches. The applicant conservatively assumed the amount of lead slump to be a maximum of 2 inches in both the top and bottom ends of the lead. The maximum dose rates determined by the applicant are shown in Table 5-3 for normal conditions of transport and Table 5-4 for hypothetical accident conditions.

The applicant also requested that BRP MOX fuel be approved for transport in the W74 canister. There are three BRP MOX fuel assembly designs, the J2 (9x9) assembly, the DA (11x11) assembly and the G-Pu (11x11) assembly. The applicant calculated the gamma and neutron radiation sources using ORIGEN 2.1 with the BWRPUU.LIB cross section library. The MOX fuel assembly parameters are provided in Section 5.5 and Chapter 6. The applicant compared the source terms from the three BRP MOX assembly types with that of design basis BRP UO_2 fuel. This comparison is presented in Table 5.5-3 and 5.5-4 of WSNF-123 for the gamma and neutron source terms, respectively. Table 5.5-2 demonstrates that the BRP MOX fuel assembly source term is bounded by the design basis BRP UO_2 fuel.

Utilizing parameters provided in the SAR, the staff performed confirmatory shielding analyses using the SAS2H/SAS4 shielding sequence in the SCALE-4.4a system of computer codes. The staff homogenized each fuel assembly, but explicitly modeled the spacer plates. The staff's model ignored the poison sheets and their wrappers. The staff results confirmed that the applicant's reported dose rate results are below the applicable 10 CFR Part 71 regulatory limits for exclusive-use transport. The dose rate results calculated by the applicant were found to be below the applicable regulatory limits specified in 10 CFR 71.47 and 71.51.

Table 5-3 - Maximum Dose Rates for Normal Conditions of Transport (mrem/hr)

	Package Surface		Vehicle Edge*		2 Meters From Vehicle	
Location	PWR Fuel	BWR Fuel	PWR Fuel	BWR Fuel	PWR Fuel	BWR Fuel
Side	122.55	115.04	32.68	26.53	9.28	9.2
Top	17.27	20.41	17.27	20.41	4.9	5.44
Bottom	3.91	0.77	3.91	0.77	1.75	0.65
Limit	1000	1000	200	200	10	10

*Applicant defined the outside conveyance as the more restrictive cylindrical surface defined by the impact limiter diameter and the planes at the package top and bottom

Table 5-4 - Maximum Dose Rates for Hypothetical Accident Conditions at 1 meter from Package Surface (mrem/hr)		
Location	PWR Canister	BWR Canister
Side	943.93	509.4
Top	52.43	59.31
Bottom	150.19	152.0

5.5 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the shielding performance requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The staff reviewed the FuelSolutions™ Transportation System criticality analysis to ensure that the transportation package meets the criticality safety requirement of 10 CFR Part 71[1], and in particular, to ensure that the fissile contents remain adequately subcritical for all postulated configurations under the normal conditions of transport specified in 10 CFR 71.71[1] and the hypothetical accident conditions specified in 10 CFR 71.73[1].

6.1 Description of the Criticality Design

The package design criterion for criticality safety is that the sum of the effective neutron multiplication factor (k_{eff}), two standard deviations of the statistical uncertainty and the bias uncertainty (95 percent confidence), and any bias which increases k_{eff}, should not exceed 0.95.

6.1.1 Packaging Design Features

The FuelSolutions™ Transportation System consists of the TS125 Transportation Cask (outer packaging) and either of two inner containers designated as the W21 canister for PWR fuel or the W74 canister for BWR fuel.

6.1.1.1 W21 Canister

The W21 canister has two configurations known as the W21-1 and W21-2. The two configurations differ in that the W21-1 may contain up to 21 PWR fuel assemblies and the W21-2 may contain up to 20 PWR fuel assemblies. The W21 canister design relies on basket geometry and fixed neutron poisons to maintain criticality safety. In addition, the W21-2 configuration limits the fissile mass and arrangement by employing a mechanical block-out which is placed over the center fuel cell in the basket during the initial preparation of the canister for fuel loading. Thus, the center cell position in the W21-2 may not contain a fuel assembly.

The maximum basket fuel cell opening is 9.00 inches by 9.00 inches and the minimum flux trap width between fuel cells is 1.900 inches or 1.300 inches, depending on location. The poison plates are BORAL® with a minimum acceptable areal density of 20 mg/cm^2 of ^{10}B.

6.1.1.2 W74 Canister

The W74 canister design relies on basket geometry, fixed neutron poisons, and fissile mass limits to maintain criticality safety. The basket consists of upper and lower basket sub-assemblies which are loaded axially, end to end, in the canister. Each sub-assembly contains 37 fuel cells; however, the five center cells in each sub-assembly are blocked by mechanical constraints and may not contain fuel assemblies. Thus, the W74 canister may contain up to 64 BWR fuel assemblies.

The fuel cells in the four outer corners of each sub-assembly are sized and positioned to allow loading with damaged fuel in special cans (the other available fuel cells must be loaded with assemblies containing intact fuel rods only). The maximum basket fuel cell opening for fuel with intact rods is 6.95 inches by 6.95 inches and the minimum gap between fuel cells is

1.080 inches or 0.830 inches depending on location. The W74 basket is not a true “flux trap” design since it does not have poison plates on both sides of its water gaps. Damaged fuel is placed in a special can which is then inserted into one of the eight cells for damaged fuel. The damaged fuel can has a maximum inside dimension of 6.95 inches by 6.95 inches and a minimum distance of 0.960 inches to the adjacent cells. The poison plates in the basket are made of borated stainless steel with a minimum acceptable areal density of 3.1 mg/cm^2 of ^{10}B.

The analysis for the W74 canister considered the following classes of BWR fuel assemblies: intact UO_2 assemblies, partial UO_2 assemblies (i.e., with some fuel rods missing), intact MOX assemblies, partial MOX assemblies, and damaged fuel assemblies.

6.1.2 Codes and Standards

The applicant’s criticality analysis is consistent with the appropriate codes and standards for nuclear criticality safety. Also, the criticality analysis is consistent with the recommendations provided in NUREG/CR-5661[2].

6.1.3 Summary Table of Criticality Evaluations

The applicant used the methods recommended in NUREG/CR-6361[3] to establish Upper Subcritical Limits (USLs) for the calculated k_{eff} values of the package as adjusted. The USL value includes the calculational bias and the administrative margin. The final values reported for k_{eff} include a factor of two times the standard deviation of the statistical uncertainty of the calculated value. The reported values of k_{eff} are then compared against their associated USL.

6.1.3.1 W21 Canister

Tables summarizing the final results of the criticality safety analysis are provided in Section 6.4 of the SAR. Table 6.4-11 reports results for a single package as required in 10 CFR 71.55. Tables 6.4-9 and 6.4-10 report results for an array of packages under the normal operating conditions and Tables 6.4-6 and 6.4-8 report results for an array of packages under the hypothetical accident conditions. The array results address the requirements in 10 CFR 71.59.

The maximum k_{eff} and associated USL for each condition, as calculated by the applicant, are summarized in Table 6-1 below. The results are less than their associated USL values and illustrate that the package design meets the criticality safety requirements of 10 CFR Part 71.

Table 6-1
Maximum k_{eff} Results in W21 with USL
MCNP 4a Calculations

Condition	$k_{eff} + 2\sigma$	USL
Single Package, Flooded 10 CFR 71.55(b), (d) and (e)	0.93968	0.94158
Infinite Array of Undamaged Packages, Flooded 10 CFR 71.59(a)(1)	0.94042	0.94224
Infinite Array of Damaged Packages, Flooded 10 CFR71.59(a)(2)	0.94214	0.94224

The maximum results for an array of packages occurred for the W21-2 configuration.

6.1.3.2 W74 Canister

Tables summarizing the final results of the criticality safety analysis are provided in Sections 6.4 and 6.6 of the SAR. The results cover a single package, package arrays under normal and accident conditions, as well as, a range of fuel assembly types including intact and partial UO_2 assemblies, intact and partial MOX assemblies, and fuel with damaged cladding.

The maximum k_{eff} and associated USL for each condition, as calculated by the applicant, are summarized in Table 6-2 below. The results are less than their associated USL values and illustrate that the package design meets the criticality safety requirements of 10 CFR Part 71.

Table 6-2
Maximum k_{eff} Results in W74 with USL
MCNP 4a Calculations

Condition	$k_{eff} + 2\sigma$	USL
Single Package, Flooded 10 CFR 71.55(b), (d) and (e)	0.93937	0.94286
Infinite Array of Undamaged Packages, Flooded 10 CFR 71.59(a)(1)	0.93599	0.94286
Infinite Array of Damaged Packages, Flooded 10 CFR71.59(a)(2)	0.94157	0.94375

6.1.4 Transport Index

The applicant modeled an infinite array of packages for both the normal conditions of transport and the hypothetical accident conditions. Therefore, pursuant to 10 CFR 71.59(b), the transport index based on criticality safety for both the W21 and the W74 package configurations is 0.0.

6.2 Spent Nuclear Fuel Contents

6.2.1 W21 Canister

The W21 canister is designed to contain up to 21 spent PWR fuel assemblies. A wide variety of fuel assemblies of the B&W, CE, and Westinghouse designs and lattice arrays from 14x14 to 17x17 were analyzed for inclusion in the canister. The full range of fuel assembly types was grouped into fuel assembly classes according to the physical characteristics important to criticality. For assembly types within a given assembly class, most physical parameters are the same. However, some parameters vary over a limited range and bounding values were defined for each assembly class. Each assembly class along with the physical parameters important to criticality are specified in Table 6.1-1 of the SAR. Bounding parameter values are presented as a maximum or minimum depending on whether increasing or decreasing the parameter value increases $k_{eff.}$

The W21-1 canister configuration can be loaded with up to 21 fuel assemblies when the initial enrichments are less than the value given in Table 6.1-1 of the SAR for each assembly type. A higher limit on enrichment is allowed in the W21-2 canister but the contents are limited to 20 assemblies with the center fuel cell empty. The limit on enrichment is the maximum planar average enrichment of any axial assembly location.

Fuel assemblies transported in the W21 may not have any known or suspected cladding defects greater than pinhole leaks or hairline cracks. A fuel rod with cladding defects, and any missing fuel rod, must be replaced by a dummy rod that displaces an equal amount of water as the original rod.

The staff reviewed the description of the spent nuclear fuel contents and finds that all relevant specifications have been provided.

6.2.2 W74 Canister

The W74 canister is designed to contain up to 64 BWR fuel assembles from the Big Rock Point (BRP) reactor. This includes: (1) intact UO_2 assemblies of the design classes GE 9x9, Siemens 9x9, and Siemens 11x11 with a peak planar-average enrichment of 4.1 percent ^{235}U, (2) UO_2 assemblies missing only corner rods of the design classes GE 9x9 and Siemens 11x11 with a peak planar-average enrichment of 4.1 percent ^{235}U, (3) UO_2 assemblies with missing interior array fuel rods with a peak planar-average enrichment of 3.55 percent ^{235}U for the GE 9x9 (9x9 BRP assemblies missing no more than one non-corner rod may have up to 4.1 percent enrichment in ^{235}U) and 3.6 percent ^{235}U for the Siemens 11x11, (4) MOX fuel assemblies of the J2(9x9), DA(11x11) and G-Pu(11x11) designs which are intact or missing only corner rods, (5) two UO_2 assemblies with some fuel rods replaced by MOX rods, (6) MOX fuel assemblies of the J2 and G-Pu designs missing non-corner rods, and (7) up to eight damaged UO_2 assemblies with up to 4.61 percent enrichment in ^{235}U or eight MOX fuel assemblies with one of the specified compositions. Damaged fuel assemblies are defined as assemblies whose fuel rods have cladding with damage in excess of pinhole leaks or hairline cracks. Fuel debris or fuel rod fragments are not qualified for loading into the W74 canister. The term “missing fuel rod” refers to a normal fuel rod location where there is neither a fuel rod nor a dummy rod of equivalent moderator displacement.

Five different compositions of isotopic distributions exist in the MOX fuel. The applicant analyzed each composition separately in its specific assembly configuration.

A specification of the allowed contents of the W74 is provided in the Certificate of Compliance.

The staff reviewed the description of the spent nuclear fuel contents and finds that all relevant specifications have been provided.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

6.3.1.1 W21 Canister

There are two basic versions of the W21 canister designated the W21M and W21T. Based on an assessment of the basic canister versions, the W21M configuration was used in the subsequent analyses.

Because of design changes in the TS125 Transportation Cask mid-way in the criticality analysis, much of the results are reported for a model of the older design called a representative transportation cask configuration. Calculations were performed to compare the effect of the older design and the current design. It was determined that the differences did not have a significant impact on k_{eff}.

In the model: (1) the bounding pellet density without any pellet dishing was used, (2) neither ^{234}U nor ^{236}U were included, (3) grid spacers, spacer sleeves, and end fittings were not included, (4) flooding was with pure water at 20 ^{o} C including the pellet-clad gap, (5) worst-case material and fabrication tolerance dimensions were applied, (6) the single package was reflected by 12 inches of water, and (7) worst-case asymmetric placement of the assemblies within the fuel cells was assumed. Axially, the model extended from the top of the bottom closure plate to a point just below the top shield plug support ring. The infinite array was created by reflecting the model boundary.

In addition to the above modeling assumptions, the accident model included a 0.08-inch permanent deformation of the guide tubes between the support plates and the loss of the neutron shield assembly. Also, the model for hypothetical accident conditions included a shifting of the fuel, guide tubes, poison plates, and fuel such that a maximum of 3.28 inches of active fuel was in an unpoisoned part of the basket.

The staff reviewed the applicant’s models and finds that they are consistent with the description of the package and contents given in SAR Sections 1 and 6 including the engineering drawings. Staff also reviewed the applicant’s modeling assumptions and found them to be consistent with the NRC’s acceptance criteria[4] and consistent with the conditions of the package as determined in the other analyses in the SAR.

6.3.1.2 W74 Canister

There are two basic versions of the W74 canister designated the W74M and W74T. Based on an assessment of the basic canister versions, the W74T configuration was used in the subsequent analyses.

Because of design changes in the TS125 Transportation Cask mid-way in the criticality analysis, the intact and partial assembly results are reported for a model of the older design called a representative transportation cask configuration. Calculations were performed to compare the effect of the two designs and it was determined that the differences did not have a significant impact on k_{eff}. The analyses for the MOX and damaged assemblies used a model of the actual TS125 design.

The basic model for the W74 analyses had the following assumptions and features: (1) no credit was taken for fuel pellet dishing, fuel burnup or fuel-related burnable poisons, (2) neither ^{234}U nor ^{236}U were included, (3) fuel assemblies did not have channels, (4) grid spacers, spacer sleeves, and top and bottom tie plates were not included, (5) flooding was with pure water at 20 ^{o}C including the pellet-clad gap, (6) worst-case material and fabrication tolerance dimensions were applied, (7) worst-case asymmetric placement of the assemblies within the fuel cells was assumed, (8) the five center cells in both levels of the basket were modeled as water-filled holes, and (9) the single package was reflected by 12 inches of water. Axially, the model extended from the middle of the bottom end shield plug to just below the top shield plug assembly. The infinite array was created by reflecting the model boundary.

Additional modeling assumptions for the UO_2 assemblies were use of the bounding pellet density of 96.5 percent of theoretical density of UO_2 and use of a uniform assembly enrichment at the peak planar-average enrichment instead of modeling the enrichment of each fuel rod separately. The MOX fuel models explicitly describe the fuel material compositions present in each of the fuel rods (pin-by-pin) in the assembly rather than using assembly average values. The MOX fuel material mixtures assumed the values at the time of fabrication except that the amount of ^{241}Pu was decreased for radioactive decay. The buildup of ^{241}Am after decay of ^{241}Pu was neglected.

The modeling assumptions for the hypothetical accident conditions are the same as for an array of packages under normal conditions of transport with the following exceptions: a 0.08 inch permanent deformation of the fuel cell walls of the basket was included, loss of the transportation cask neutron shield was assumed, and up to 1.63 inches of active fuel is not covered by the poison plates in the upper basket

The staff reviewed the applicant’s models and finds that they are consistent with the description of the package and contents given in SAR Sections 1 and 6 including the engineering drawings. Staff also reviewed the applicant’s modeling assumptions and found them to be consistent with the NRC’s acceptance criteria[4] and consistent with the conditions of the package as determined in the other analyses in the SAR.

6.3.2 Material Properties

6.3.2.1 W21 Canister

The compositions and densities for the materials used in the criticality safety analysis models are provided in Section 6.3 of the SAR. No credit was taken for burnable absorbers in the fuel. The applicant's calculations take credit for only 75 percent of the minimum acceptable ^{10}B areal density in the BORAL® basket absorber material as verified by a testing program.

The basket materials do not degrade during storage such that there is any expected impact on criticality safety. The neutron absorber plates meet all structural and thermal requirements, and can be expected to have no significant erosion or corrosion. The neutron flux in the package is very low such that depletion of the ^{10}B is negligible.

6.3.2.2 W74 Canister

The compositions and densities for the materials used in the criticality models are provided in Section 6.3 of the SAR. The applicant's calculations take credit for only 75 percent of the minimum acceptable ^{10}B areal density in the borated stainless steel basket poison as verified by a testing program.

The basket materials do not degrade during storage such that there is any expected impact on criticality safety. The neutron absorber plates meet all structural and thermal requirements, and can be expected to have no significant erosion or corrosion. The neutron flux in the package is very low such that depletion of the ^{10}B is negligible.

6.3.3 Computer Codes and Cross Section Libraries

The applicant utilized the MCNP (a Monte-Carlo particle transport) code, Version 4a, for the criticality safety analysis of both the W21 and the W74 canisters. The primary cross-section data file used for the analysis is derived from the ENDF/B-V data. The MCNP code is a recognized standard for performing criticality analyses and the staff finds that the code and cross-section set used are appropriate for this particular cask design and contents.

Representative input files for the calculations were reviewed for consistency with the contents specifications and the engineering drawings.

6.3.4 Demonstration of Maximum Reactivity

6.3.4.1 W21 Canister

Parametric variations were analyzed to determine the combination of the allowed fuel parameter values in each fuel class which maximizes k_{eff} for that class. Also, parametric cases were analyzed to determine the most reactive fuel class among the various contents of the cask. The applicant found that the most reactive fuel assembly was the bounding case for the Westinghouse 17x17 B class and this assembly class was used in the subsequent analyses. To further maximize k_{eff}, the applicant neglected the guide and instrument tubes in the fuel assembly models.

To determine the optimum moderation condition, the applicant varied the moderator density in the package interior and between packages in an array. The optimum moderation occurred when the moderator densities in the canister interior and between packages were both at full density. This optimum condition was used in subsequent analyses. The package was flooded under both normal conditions and hypothetical accident conditions.

The effect of material and fabrication tolerances was investigated by performing a calculation at each end of the tolerance range. The worst-case value for each tolerance was then used in the rest of the analyses. As a final look at maximizing k_{eff}, the applicant assessed the result of misloading a fuel assembly with uranium enriched to 5 percent ^{235}U instead of the lower enrichment limits specified in the contents table. In this case, k_{eff} of the package still remains subcritical.

The methods and calculations used to determine the optimum conditions that maximize k_{eff} were reviewed and found acceptable.

6.3.4.2 W74 Canister

The applicant compared three different asymmetric patterns of fuel offset in the fuel assembly cells and found the worst-case pattern was when groups of four adjacent assemblies in the basket were shifted toward each other. The applicant determined worst-case moderator density inside and between the packages and the worst-case tolerances in the same manner as described in Section 6.3.4.1 for the W21 canister. The worst-case conditions were used in the subsequent analyses. The package was flooded under both the normal conditions of transport and hypothetical accident conditions.

The applicant compared the results of calculations for eight different UO_2 assembly patterns when a model of pin-by-pin enrichments was used versus modeling a uniform enrichment at the peak planar-average enrichment. The uniform enrichment cases bounded the pin-by-pin cases and a uniform enrichment model was used in subsequent analyses. The applicant also compared the 9x9 and two 11x11 assembly types to identify the intact assembly with the highest k_{eff}. The highest reactivity assembly types were the GE 9x9 assembly with one water pin and the Siemens 11x11 with 121 fuel rods.

An analysis of partial UO_2 assemblies (assemblies with fuel rod holes at non-corner positions), identified the bounding case by varying the pitch of BRP 9x9 and 11x11 fuel rods in the fuel assembly cells and dropping out fuel rods that would not fit into the basket cell. The pitch between the rods was varied until the pitch which gives the highest k_{eff} was found. This analysis found the maximum reactivity for partial assemblies; however, in order to keep k_{eff} less than its associated USL, the uranium enrichment was limited to 3.55 percent for partial 9x9 assemblies and 3.6 percent for partial 11x11 assemblies. Next, results of calculations for partial fuel assemblies with missing corner rods were compared to the results for fully intact assemblies and it was found that k_{eff} always decreased for partial assemblies where only corner rods are missing.

Intact and partial MOX assemblies were modeled with their exact rod configuration and an explicit description of the fuel composition on a pin-by-pin basis.

The optimum configuration for damaged fuel was determined by performing an analysis similar to that for partial rods but without cladding. Cylindrical fuel rods as well as spherical lumps of fuel were placed in arrays in the damaged fuel cans in the eight corner basket cells. The pitches of the cylinders and spheres were varied for a range of radii to find the k_{eff} values of the optimum combination of pitch and radius. Partial UO_2 fuel assemblies were modeled in the remaining 56 fuel cells because these were found to be more reactive than the intact fuel assemblies.

The methods and calculations used to determine the optimum conditions that maximize k_{eff} were reviewed and found acceptable.

6.3.5 Confirmatory Analyses

6.3.5.1 W21 Canister

The staff used the CSAS/KENO-V code in the SCALE[5] suite of analysis codes to perform confirmatory analyses. These calculations used both the 27-group and 44-group cross sections in CSAS. The CSAS code was developed by the Oak Ridge National Laboratory for performing criticality analyses and is appropriate for this particular application and fuel system.

The staff's calculations were based on the information provided in the SAR. The Westinghouse 17x17 bounding fuel assembly was used in the calculations. The staff's models considered fuel assemblies being centered in each basket cell and also moved toward the center of the cask as much as possible. The value of k_{eff} was higher when the fuel was moved toward the center of the package. One calculation considered a case where the poison plates had shifted to maximize the gap at the corner where four cells meet. All of the staff's calculations gave a value of k_{eff} which meets NRC's acceptance criterion for an adequate level of subcriticality.

In addition, during the review of the Fuel Solutions™ Storage System, the staff performed independent calculations of the W21 canister in its storage overpack and found good agreement with the applicant's analysis.

6.3.5.2 W74 Canister

During the review of the Fuel Solutions™ Storage System, the staff performed independent calculations of the W74 canister in its storage overpack and found good agreement with the applicant's analysis. Calculations were performed for the original application and the amendment to add damaged and MOX fuel. The performance of the transportation package is expected to be very similar to that of the storage configuration and additional calculations were not considered necessary.

6.4 Single Package Evaluation

6.4.1 W21 Canister

The applicant performed single package analyses for the normal conditions of transport and hypothetical accident conditions for both the W21-1 and W21-2 canister configurations loaded with Westinghouse 17x17 FOA fuel assemblies. To satisfy the requirements of 10 CFR

71.55(b)(3), the applicant modeled the three cases of a full cask, a canister surrounded by only the containment shell, and a canister surrounded by the containment shell and the depleted uranium gamma shield. The package was flooded with full density water and reflected by full density water on all sides.

The results of the single package calculations were very close to each other for comparable cases. In addition, the results were compared with the results from the package arrays described below and it was found that the array results bounded the single package in all cases.

The staff reviewed the applicant's evaluation and finds that the package meets the regulatory requirements of 10 CR 71.55(b), (d) and (e).

6.4.2 W74 Canister

The applicant performed single package analyses for the normal conditions of transport and hypothetical accident conditions for the W74 canister loaded with intact Siemens 11x11 UO_2 fuel assemblies. The Siemens assemblies were found to be bounding in the accident array analysis. To satisfy the requirements of 10 CFR 71.55(b)(3), the applicant modeled the three cases of a full cask, a canister surrounded by only the containment shell, and a canister surrounded by the containment shell and the depleted uranium gamma shield. The package was flooded with full density water and reflected by full density water on all sides.

The results of the single package calculations were very close to each other for comparable cases. In addition, the results were compared with the results from the package arrays described below and it was found that the array results were essentially the same as or bounded the single package results. These results showed that the package shell was very effective in isolating the packages in an array.

MOX, partial and damaged fuel packages were only analyzed as arrays under hypothetical accident conditions since the intact UO_2 analysis showed this configuration to be most bounding.

The staff reviewed the applicant's evaluation and finds that the package meets the regulatory requirements of 10 CR 71.55(b), (d) and (e).

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

6.5.1 W21 Canister

An infinite array of undamaged packages was analyzed by the applicant. The modeling assumptions are described in Section 6.3.1.1. The key assumptions in the normal conditions model are: full density water flooding of the package, worst-case asymmetric location of the fuel assembly in each basket cell, worst-case material and fabrication tolerances, and 0.53 inches of active fuel not covered by poison plates. Calculations were performed for both the W21-1 and W21-2 canister configurations. The results are bounded by the results for an array of packages under hypothetical accident conditions, for the W21-1 canister, and all but the Westinghouse 17x17 B class fuel, for the W21-2 canister. The calculated k_{eff} for the one exception was still below its associated USL. Thus, all normal conditions cases are within the NRC's acceptance limits.

6.5.2 W74 Canister

An infinite array of undamaged packages containing the bounding Siemens 11x11 UO_2 fuel was analyzed by the applicant. The modeling assumptions are described in Section 6.3.1.2 and 6.3.4.2. The key assumptions in the normal conditions model are: full density water flooding of the package, worst-case asymmetric location of the fuel assembly in each basket cell, and worst-case material and fabrication tolerances. The intact UO_2 fuel results are bounded by the hypothetical accident cases for the W74 canister. Thus, the normal condition case is within the NRC's acceptance limits.

MOX, partial and damaged fuel packages were only analyzed as arrays under hypothetical accident conditions since the intact UO_2 analysis showed this configuration to be most bounding.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

6.6.1 W21 Canister

The modeling assumptions for the hypothetical accident conditions are described in Section 6.3.1.1. An accident array analysis was performed for each fuel assembly type in each fuel assembly class including the bounding fuel assembly when different from the actual fuel assemblies in that class. Analyses were made for both the W21-1 and W21-2 canisters.

In all cases, the results for an array of packages under hypothetical accident conditions gave a value of k_{eff} which is less than the corresponding USL. Also, this analysis confirmed the selection of the bounding fuel assembly as having the highest k_{eff} in each assembly class.

The staff reviewed the applicant's evaluation and finds that the package meets the regulatory requirements of 10 CFR 71.59(a)(1), 71.59(a)(2) and 71.59(b). The staff also finds that the Transportation Index based on criticality safety is shown to be 0.

6.6.2 W74 Canister

The applicant performed the analysis of an array of packages under hypothetical accident conditions before evaluating the single package or the normal conditions array. The modeling assumptions for the hypothetical accident conditions are described in Section 6.3.1.2 of the SER. In the accident analysis, the range of intact 9x9 and 11x11 UO_2 fuel configurations was included to find the most bounding configuration. The highest reactivity assembly types were the GE 9x9 assembly with one water pin and the Siemens 11x11 with 121 fuel rods. All of the configurations have a value of k_{eff} less than its associated USL, and thus, have an adequate margin of safety. The Siemens 11x11 configuration is the most bounding overall and was used in subsequent analyses. The applicant also found the accident array to have a higher k_{eff} than the normal conditions array or the single package. As the bounding configuration, an array of packages under the hypothetical accident conditions was used in the analysis of the remaining contents, i.e., MOX, partial, and damaged fuel assemblies.

The analysis of partial UO_2 assemblies found that when only corner rods were missing from the assembly lattice, the k_{eff} of the package decreased. The applicant analyzed partial assemblies with non-corner rods missing by finding the optimum pitch for an array of individual fuel rods. The optimum pitch configuration had a k_{eff} that exceeded the value for intact Siemens 11x11 rods but was still less than its associated USL. Intact and partial MOX fuel assemblies were modeled pin-by-pin with the actual isotopic composition. The applicant found that the MOX results were bounded by the UO_2 results in all cases.

In the analysis of damaged assemblies, the applicant performed calculations for both a regular lattice of cylindrical fuel rods without cladding and a lattice of fuel spheres as described in Section 6.3.4.2. The damaged fuel was modeled inside special cans which are placed into the four corner cells of each level of the basket. The bounding partial assembly fuel rods (Siemens 11x11) are then modeled in the remaining 56 available fuel cells. The damaged fuel analysis found that the bounding fuel configuration is a square array of fuel cylinders. The resulting k_{eff} remained within the limits of the associated USL. In a similar manner, damaged MOX assemblies were analyzed and these results did not exceed the associated criticality limit.

The staff reviewed the applicant's evaluation and finds that the package meets the regulatory requirements of 10 CFR 71.59(a)(1), 71.59(a)(2) and 71.59(b). The staff also finds that the Transportation Index based on criticality safety is shown to be 0.

6.7 Benchmark Evaluations

6.7.1 W21 Canister

The applicant performed benchmark comparisons on 49 selected critical experiments that were chosen to represent the range of variables in the cask design. The benchmarks included experiments using BORAL®, borated stainless steel, and steel separator plates in combination with steel or depleted uranium reflecting walls.

The staff reviewed the benchmark experiments chosen and found that the benchmark parameters provided adequate coverage of the range of parameters for the fuel assemblies in the specified contents and for the cask design.

The applicant used the method recommended in NUREG/CR-6361[3] to calculate a USL for each fuel assembly class. As a first step, linear fits were calculated to establish trends in the USL for the four fuel parameters of enrichment, water-to-fuel ratio, hydrogen-to-^{235}U ratio, and pin pitch. Second, the equations for the linear fits were used to determine a minimum USL for each fuel assembly class considered for transport in the W21 canister.

The staff reviewed the applicant's method for determining the USL and concluded that it is sufficient to provide a basis for the criticality evaluation of the package.

6.7.2 W74 Canister

The benchmark evaluation for UO_2 assemblies used the evaluation described in Section 6.7.1 for the W21 canister. Using the UO_2 trend equations for UO_2, USLs for the partial and damaged assemblies were computed separately by substituting the appropriate enrichment and pin configurations into the equations. To benchmark the MOX calculations, the applicant used 24 MOX critical experiments that were chosen to represent the range of variables in the cask design. In addition, the MOX and UO_2 benchmarks were combined to a set of 73 experiments and USLs were calculated for the combined set.

The staff reviewed the benchmark experiments chosen and found that the benchmark parameters provided adequate coverage of the range of parameters for the fuel assemblies in the specified contents and for the cask design.

The applicant used the method recommended in NUREG/CR-6361[3] to calculate a USL for each fuel assembly class. First, linear fits were calculated to establish trends for the four fuel parameters of enrichment, water-to-fuel ratio, hydrogen-to-^{235}U ratio, and pin pitch. For the combined MOX and UO_2 USLs, a linear fit for a fifth parameter, the percentage of fissile material that is plutonium, was added. The equations for the linear UO_2 fits were used to determine a minimum USL for each UO_2 fuel assembly class considered for transport in the W74 canister. For the calculations with MOX fuel, USLs were calculated for the MOX experiment set, the combined set, and the UO_2 set. The minimum USL derived from these three sets was used as the final MOX USL value.

The staff reviewed the applicant's method for determining the USL and concluded that it is sufficient to provide a basis for the criticality evaluation of the package.

6.8 Evaluation Findings

Based on its review of the presentations and information supplied by the applicant, and the analyses performed by staff, the staff finds reasonable assurance that the package design meets the requirements of Part 71.

6.9 References

1. U.S. Code of Federal Regulations, "Packaging and Transportation of Radioactive Material," Title 10, Part 71, January 1, 2001.

2. U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, April, 1997.

3. U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," NUREG/CR-6361, March, 1997.

4. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617, March, 2000.

5. U.S. Nuclear Regulatory Commission, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Vol. 1-5, Rev. 6, May, 2000.

7.0 OPERATING PROCEDURES

Operating procedures for the package are specified in Chapter 7 of the SAR. The chapter includes sections on package loading, unloading, and preparation of an empty package for transport.

The closure lid, vent and drain port penetrations are tested to "leaktight", as defined in ANSI N14.5 as a leak rate of less than or equal to 1 x 10^{-7} ref-cm^3/s.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Acceptance tests and the maintenance program for the package are specified in Chapter 8 of the SAR. The acceptance tests include visual inspections, structural and pressure tests, component tests, shielding tests, and neutron absorber tests.

The acceptance tests are specified in Section 8.1 of the SAR and include structural and pressure tests, leak tests, and impact limiter test. The maintenance program is specified in Section 8.2 of the SAR.

The impact limiter aluminum honeycomb core is completely enclosed in a stainless steel shell assembly for protection from the weather. Per SAR Table 8.1-3, the impact limiter shell assembly welds are examined during fabrication to assure they are weather tight. Furthermore, the SAR specifies that the impact limiters be visually inspected to evaluate the integrity of the welds prior to each use, and, annually for apparent defects that could significantly reduce the effectiveness of the packaging.

Periodic visual inspections of the lid, impact limiter, and trunnion bolts are performed to ensure there is no gross deformation or other deleterious conditions. Additionally, the lid closure bolts were evaluated for cyclical loading associated with normal service in accordance with subsection WB of the ASME Code. The lifting attachment bolts for the lifting trunnions, rotation trunnions, and impact limiters were evaluated in accordance with Subsection NF of the ASME Code. The results of these bolt fatigue evaluations showed that the 5-year replacement interval for the subject bolts is sufficient to preclude the possibility of fatigue failure in all cases.

Fixed neutron absorber plates are used to ensure subcriticality during loading and unloading operations that use water inside the FuelSolutions™ W21 and W74 canisters. The W21 canister uses BORAL® plates and the W74 canister uses borated stainless steel plates.

The BORAL® neutron absorber plates for the W21 canister consist of a core of boron carbide (B_4C)-aluminum mixture encased in a type 1100 aluminum alloy cladding. These plates are attached to the guide tubes by a welded type 316 stainless steel wrapper, and are therefore encased in and completely supported by the stainless steel guide tube assembly. A structural analysis in Section 2 of the W21 SAR demonstrates that the neutron absorber plates remain in the guide tube assembly under the bounding accident conditions.

The borated stainless steel neutron absorber plates in the W74 canister consist of natural boron alloyed with AISI type 304 stainless steel. These neutron absorber plates are attached to the guide tubes by seven 20-gage stainless steel neutron absorber sheet retainers, which are welded to the guide tube through small holes in the absorber plates. A structural analysis is provided in Section 2 of the W74 SAR which demonstrates that the neutron absorber plates will remain in place under the bounding accident conditions.

After manufacturing, each batch of BORAL® is tested using wet chemistry and/or neutron attenuation techniques to verify presence, proper distribution, and minimum ^{10}B content. The test is designed to be representative of each BORAL® panel. The minimum allowable ^{10}B content is 0.02 gm/cm^2 for all plates in the W21 canister. Any panel with a ^{10}B loading less than the minimum allowed is rejected.

Each batch of borated stainless steel for the W74 canister is also tested using wet chemistry and neutron attenuation analysis to verify the presence, proper distribution, and minimum ^{10}B content. The minimum allowable boron content in the stainless steel is 1.25 wt percent natural boron. Any panel with a boron content less than the minimum allowed is rejected.

The staff's acceptance of the neutron absorber tests described above is based, in part, on the fact that the criticality analyses assumed only 75 percent of the minimum required ^{10}B content of the BORAL® or natural boron content of the borated stainless steel.

Installation of the BORAL® and borated stainless steel panels on the W21 and W74 guide tubes is in accordance with written and approved procedures. Quality control procedures are in place to ensure that the canister guide tube walls contain a BORAL® or borated stainless steel panel as specified in each respective canister SAR license drawings.

The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the FuelSolutions™ Storage System. For both designs, the neutron absorber plates meet the thermal requirements and can be expected to have no significant thermally induced degradation, corrosion, or other degradation during the 20-year design ISFSI service. The neutron flux in either canister over the design storage period is also very low such that boron depletion in service is negligible. Thus, the staff finds that the neutron poison will remain effective for the 20-year design storage period. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Based on the review of the statements and representations in the application, the staff concludes that the maintenance and testing procedures meet the requirements of 10 CFR Part 71, and these procedures are adequate to assure that the package will be acceptance tested and maintained in a manner consistent with its evaluation for approval.

9.0 CONDITIONS

In addition to the requirements of Subpart G of 10 CFR Part 71:

Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, “Operating Procedures,” of the application, as supplemented.

Each package must be tested and maintained in accordance with the procedures described in Chapter 8, “Acceptance Tests and Maintenance Program,” of the application, as supplemented.

10.0 CONCLUSION

Based on the review of the statements and representation in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated, and that the package meets the requirements of 10 CFR Part 71.

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