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Table A.1 (Page 1 of 15)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class |
| b. Maximum initial enrichment: | As specified in Table A.2 for the applicable fuel assembly array/class. |
| c. Post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment per assembly | |
| i. Zr clad: | An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable. |
| ii. SS clad: | An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable. |
| d. Fuel assembly length: | $\leq$ 176.8 inches (nominal design) |
| e. Fuel assembly width: | $\leq$ 8.54 inches (nominal design) |
| f. Fuel assembly weight: | $\leq$ 1,680 lbs |

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain control components.

D. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

Table A.1 (Page 2 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class. |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| d. Post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment per assembly: | |
| i. Zr clad: | An assembly post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}$U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, a decay heat $\leq$ 183.5 Watts, and a minimum initial enrichment $\geq$ 2.4 wt% $^{235}$U. |
| ii. SS clad: | An assembly cooling time after discharge $\geq$ 16 years, an average burnup $\leq$ 22,500 MWD/MTU, and a minimum initial enrichment $\geq$ 3.5 wt% $^{235}$U. |
| e. Fuel assembly length: | $\leq$ 176.2 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 5.85 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 700 lbs, including channels |

Table A.1 (Page 3 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 4 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for fuel assembly array/class 6x6B. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for fuel assembly array/class 6x6B. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the $UO_2$ rods. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 5 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for array/class 6x6B. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for array/class 6x6B. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the $UO_2$ rods. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 6 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods ($ThO_2$ and $UO_2$) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

| | | |
|---|---|---|
| a. | Cladding type: | Zircaloy (Zr) |
| b. | Composition: | 98.2 wt.% $ThO_2$, 1.8 wt. % $UO_2$ with an enrichment of 93.5 wt. % $^{235}U$. |
| c. | Number of rods per Thoria Rod Canister: | ≤ 18 |
| d. | Decay heat per Thoria Rod Canister: | ≤ 115 Watts |
| e. | Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister: | A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤ 16,000 MWD/MTIHM. |
| f. | Initial heavy metal weight: | ≤ 27 kg/canister |
| g. | Fuel cladding O.D.: | ≥ 0.412 inches |
| h. | Fuel cladding I.D.: | ≤ 0.362 inches |
| i. | Fuel pellet O.D.: | ≤ 0.358 inches |
| j. | Active fuel length: | ≤ 111 inches |
| k. | Canister weight: | ≤ 550 lbs, including fuel |

Table A.1 (Page 7 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table A.1 (Page 8 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$. |
| e. Fuel assembly length: | $\leq$ 176.2 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 5.85 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 9 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for the applicable fuel assembly array/class. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 10 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for the applicable original fuel assembly array/class. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for the applicable original fuel assembly array/class. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the original fuel assembly. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 11 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for fuel assembly array/class 6x6B. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for fuel assembly array/class 6x6B. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the $UO_2$ rods. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 12 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembl array/class 6x6B and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for array/class 6x6B. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for array/class 6x6B. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the $UO_2$ rods. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 13 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

| | |
|---|---|
| a. Cladding type: | Zircaloy (Zr) |
| b. Maximum planar-average initial enrichment: | As specified in Table A.3 for original fuel assembly array/class 6x6B. |
| c. Initial maximum rod enrichment: | As specified in Table A.3 for original fuel assembly array/class 6x6B. |
| d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly: | An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% $^{235}U$ for the $UO_2$ rods in the original fuel assembly. |
| e. Fuel assembly length: | $\leq$ 135.0 inches (nominal design) |
| f. Fuel assembly width: | $\leq$ 4.70 inches (nominal design) |
| g. Fuel assembly weight | $\leq$ 400 lbs, including channels |

Table A.1 (Page 14 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods ($ThO_2$ and $UO_2$) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

| | | |
|---|---|---|
| a. | Cladding Type: | Zircaloy (Zr) |
| b. | Composition: | 98.2 wt.% $ThO_2$, 1.8 wt. % $UO_2$ with an enrichment of 93.5 wt. % $^{235}U$. |
| c. | Number of rods per Thoria Rod Canister: | $\leq$ 18 |
| d. | Decay heat per Thoria Rod Canister: | $\leq$ 115 Watts |
| e. | Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister: | A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM. |
| f. | Initial heavy metal weight: | $\leq$ 27 kg/canister |
| g. | Fuel cladding O.D.: | $\geq$ 0.412 inches |
| h. | Fuel cladding I.D.: | $\leq$ 0.362 inches |
| i. | Fuel pellet O.D.: | $\leq$ 0.358 inches |
| j. | Active fuel length: | $\leq$ 111 inches |
| k. | Canister weight: | $\leq$ 550 lbs, including fuel |

Table A.1 (Page 15 of 15)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The antimony-Beryllium neutron source material shall be in a water rod location.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 14x14A | 14x14B | 14x14C | 14x14D | 15x15A |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | SS | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 407 | ≤ 407 | ≤ 425 | ≤ 400 | ≤ 464 |
| Initial Enrichment (wt % $^{235}U$) | ≤ 4.6 | ≤ 4.6 | ≤ 4.6 | ≤ 4.0 | ≤ 4.1 |
| No. of Fuel Rods | 179 | 179 | 176 | 180 | 204 |
| Clad O.D. (in.) | ≥ 0.400 | ≥ 0.417 | ≥ 0.440 | ≥ 0.422 | ≥ 0.418 |
| Clad I.D. (in.) | ≤ 0.3514 | ≤ 0.3734 | ≤ 0.3880 | ≤ 0.3890 | ≤ 0.3660 |
| Pellet Dia. (in.) | ≤ 0.3444 | ≤ 0.3659 | ≤ 0.3805 | ≤ 0.3835 | ≤ 0.3580 |
| Fuel Rod Pitch (in.) | ≤ 0.556 | ≤ 0.556 | ≤ 0.580 | ≤ 0.556 | ≤ 0.550 |
| Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 144 | ≤ 150 |
| No. of Guide Tubes | 17 | 17 | 5 (Note 4) | 16 | 21 |
| Guide Tube Thickness (in.) | ≥ 0.017 | ≥ 0.017 | ≥ 0.038 | ≥ 0.0145 | ≥ 0.0165 |

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 15x15B | 15x15C | 15x15D | 15x15E | 15x15F |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 464 | ≤ 464 | ≤ 475 | ≤ 475 | ≤ 475 |
| Initial Enrichment (wt % $^{235}U$) | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 | ≤ 4.1 |
| No. of Fuel Rods | 204 | 204 | 208 | 208 | 208 |
| Clad O.D. (in.) | ≥ 0.420 | ≥ 0.417 | ≥ 0.430 | ≥ 0.428 | ≥ 0.428 |
| Clad I.D. (in.) | ≤ 0.3736 | ≤ 0.3640 | ≤ 0.3800 | ≤ 0.3790 | ≤ 0.3820 |
| Pellet Dia. (in.) | ≤ 0.3671 | ≤ 0.3570 | ≤ 0.3735 | ≤ 0.3707 | ≤ 0.3742 |
| Fuel Rod Pitch (in.) | ≤ 0.563 | ≤ 0.563 | ≤ 0.568 | ≤ 0.568 | ≤ 0.568 |
| Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Guide Tubes | 21 | 21 | 17 | 17 | 17 |
| Guide Tube Thickness (in.) | ≥ 0.015 | ≥ 0.0165 | ≥ 0.0150 | ≥ 0.0140 | ≥ 0.0140 |

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/ Class | 15x15G | 15x15H | 16x16A | 17x17A | 17x17B | 17x17C |
|---|---|---|---|---|---|---|
| Clad Material (Note 2) | SS | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 420 | ≤ 475 | ≤ 443 | ≤ 467 | ≤ 467 | ≤ 474 |
| Initial Enrichment (wt % $^{235}U$) | ≤ 4.0 | ≤ 3.8 | ≤ 4.6 | ≤ 4.0 | ≤ 4.0 | ≤ 4.0 |
| No. of Fuel Rods | 204 | 208 | 236 | 264 | 264 | 264 |
| Clad O.D. (in.) | ≥ 0.422 | ≥ 0.414 | ≥ 0.382 | ≥ 0.360 | ≥ 0.372 | ≥ 0.377 |
| Clad I.D. (in.) | ≤ 0.3890 | ≤ 0.3700 | ≤ 0.3320 | ≤ 0.3150 | ≤ 0.3310 | ≤ 0.3330 |
| Pellet Dia. (in.) | ≤ 0.3825 | ≤ 0.3622 | ≤ 0.3255 | ≤ 0.3088 | ≤ 0.3232 | ≤ 0.3252 |
| Fuel Rod Pitch (in.) | ≤ 0.563 | ≤ 0.568 | ≤ 0.506 | ≤ 0.496 | ≤ 0.496 | ≤ 0.502 |
| Active Fuel Length (in.) | ≤ 144 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Guide Tubes | 21 | 17 | 5 (Note 4) | 25 | 25 | 25 |
| Guide Tube Thickness (in.) | ≥ 0.0145 | ≥ 0.0140 | ≥ 0.0400 | ≥ 0.016 | ≥ 0.014 | ≥ 0.020 |

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. Zr. Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

Table A.3 (Page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 6x6A | 6x6B | 6x6C | 7x7A | 7x7B | 8x8A |
|---|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 110 | ≤ 110 | ≤ 110 | ≤ 100 | ≤ 195 | ≤ 120 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 2.7 | ≤ 2.7 for the $UO_2$ rods. See Note 4 for MOX rods | ≤ 2.7 | ≤ 2.7 | ≤ 4.2 | ≤ 2.7 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 4.0 | ≤ 4.0 | ≤ 4.0 | ≤ 5.5 | ≤ 5.0 | ≤ 4.0 |
| No. of Fuel Rods | 35 or 36 | 35 or 36 (up to 9 MOX rods) | 36 | 49 | 49 | 63 or 64 |
| Clad O.D. (in.) | ≥ 0.5550 | ≥ 0.5625 | ≥ 0.5630 | ≥ 0.4860 | ≥ 0.5630 | ≥ 0.4120 |
| Clad I.D. (in.) | ≤ 0.5105 | ≤ 0.4945 | ≤ 0.4990 | ≤ 0.4204 | ≤ 0.4990 | ≤ 0.3620 |
| Pellet Dia. (in.) | ≤ 0.4980 | ≤ 0.4820 | ≤ 0.4880 | ≤ 0.4110 | ≤ 0.4910 | ≤ 0.3580 |
| Fuel Rod Pitch (in.) | ≤ 0.710 | ≤ 0.710 | ≤ 0.740 | ≤ 0.631 | ≤ 0.738 | ≤ 0.523 |
| Active Fuel Length (in.) | ≤ 120 | ≤ 120 | ≤ 77.5 | ≤ 80 | ≤ 150 | ≤ 120 |
| No. of Water Rods (Note 11) | 1 or 0 | 1 or 0 | 0 | 0 | 0 | 1 or 0 |
| Water Rod Thickness (in.) | ≥ 0 | ≥ 0 | N/A | N/A | N/A | ≥ 0 |
| Channel Thickness (in.) | ≤ 0.060 | ≤ 0.060 | ≤ 0.060 | ≤ 0.060 | ≤ 0.120 | ≤ 0.100 |

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 8x8B | 8x8C | 8x8D | 8x8E | 8x8F | 9x9A | 9x9B |
|---|---|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 185 | ≤ 177 | ≤ 177 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.2 | ≤ 4.2 | < 3.6 | ≤ 4.2 | ≤ 4.2 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 |
| No. of Fuel Rods | 63 or 64 | 62 | 60 or 61 | 59 | 64 | 74/66 (Note 5) | 72 |
| Clad O.D. (in.) | ≥ 0.4840 | ≥ 0.4830 | ≥ 0.4830 | ≥ 0.4930 | ≥ 0.4576 | ≥ 0.4400 | ≥ 0.4330 |
| Clad I.D. (in.) | ≤ 0.4295 | ≤ 0.4250 | 0.4230 | ≤ 0.4250 | ≤ 0.3996 | ≤ 0.3840 | ≤ 0.3810 |
| Pellet Dia. (in.) | ≤ 0.4195 | ≤ 0.4160 | ≤ 0.4140 | ≤ 0.4160 | ≤ 0.3913 | ≤ 0.3760 | ≤ 0.3740 |
| Fuel Rod Pitch (in.) | ≤ 0.642 | ≤ 0.641 | ≤ 0.640 | ≤ 0.640 | ≤ 0.609 | ≤ 0.566 | ≤ 0.572 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Water Rods (Note 11) | 1 or 0 | 2 | 1 - 4 (Note 7) | 5 | N/A (Note 12) | 2 | 1 (Note 6) |
| Water Rod Thickness (in.) | ≥ 0.034 | > 0.00 | > 0.00 | ≥ 0.034 | ≥ 0.0315 | > 0.00 | > 0.00 |
| Channel Thickness (in.) | ≤ 0.120 | ≤ 0.120 | ≤ 0.120 | ≤ 0.100 | ≤ 0.055 | ≤ 0.120 | ≤ 0.120 |

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 9x9C | 9x9D | 9x9E (Note 13) | 9x9F (Note 13) | 10x10A |
|---|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | Zr | Zr | Zr |
| Design Initial U (kg/assy.) (Note 3) | ≤ 177 | ≤ 177 | ≤ 177 | ≤ 177 | ≤ 186 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.1 | ≤ 4.1 | ≤ 4.2 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 |
| No. of Fuel Rods | 80 | 79 | 76 | 76 | 92/78 (Note 8) |
| Clad O.D. (in.) | ≥ 0.4230 | ≥ 0.4240 | ≥ 0.4170 | ≥ 0.4430 | ≥ 0.4040 |
| Clad I.D. (in.) | ≤ 0.3640 | ≤ 0.3640 | ≤ 0.3640 | ≤ 0.3860 | ≤ 0.3520 |
| Pellet Dia. (in.) | ≤ 0.3565 | ≤ 0.3565 | ≤ 0.3530 | ≤ 0.3745 | ≤ 0.3455 |
| Fuel Rod Pitch (in.) | ≤ 0.572 | ≤ 0.572 | ≤ 0.572 | ≤ 0.572 | ≤ 0.510 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 | ≤ 150 |
| No. of Water Rods (Note 11) | 1 | 2 | 5 | 5 | 2 |
| Water Rod Thickness (in.) | ≥ 0.020 | ≥ 0.0300 | ≥ 0.0120 | ≥ 0.0120 | ≥ 0.0300 |
| Channel Thickness (in.) | ≤ 0.100 | ≤ 0.100 | ≤ 0.120 | ≤ 0.120 | ≤ 0.120 |

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

| Fuel Assembly Array/Class | 10x10B | 10x10C | 10x10D | 10x10E |
|---|---|---|---|---|
| Clad Material (Note 2) | Zr | Zr | SS | SS |
| Design Initial U (kg/assy.) (Note 3) | ≤ 186 | ≤ 186 | ≤ 125 | ≤ 125 |
| Maximum planar-average initial enrichment (wt.% $^{235}U$) | ≤ 4.2 | ≤ 4.2 | ≤ 4.0 | ≤ 4.0 |
| Initial Maximum Rod Enrichment (wt.% $^{235}U$) | ≤ 5.0 | ≤ 5.0 | ≤ 5.0 | ≤ 5 |
| No. of Fuel Rods | 91/83 (Note 9) | 96 | 100 | 96 |
| Clad O.D. (in.) | ≥ 0.3957 | ≥ 0.3780 | ≥ 0.3960 | ≥ 0.3940 |
| Clad I.D. (in.) | ≤ 0.3480 | ≤ 0.3294 | ≤ 0.3560 | ≤ 0.3500 |
| Pellet Dia. (in.) | ≤ 0.3420 | ≤ 0.3224 | ≤ 0.3500 | ≤ 0.3430 |
| Fuel Rod Pitch (in.) | ≤ 0.510 | ≤ 0.488 | ≤ 0.565 | ≤ 0.557 |
| Design Active Fuel Length (in.) | ≤ 150 | ≤ 150 | ≤ 83 | ≤ 83 |
| No. of Water Rods (Note 11) | 1 (Note 6) | 5 (Note 10) | 0 | 4 |
| Water Rod Thickness (in.) | > 0.00 | ≥ 0.031 | N/A | ≥ 0.022 |
| Channel Thickness (in.) | ≤ 0.120 | ≤ 0.055 | ≤ 0.080 | ≤ 0.080 |

Table A.3 (Page 5 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Zr designates cladding material made from Zirconium or Zirconium alloys.
3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.
4. $\leq$ 0.635 wt. % $^{235}U$ and $\leq$ 1.578 wt. % total fissile plutonium ($^{239}Pu$ and $^{241}Pu$), (wt. % of total fuel weight, i.e., $UO_2$ plus $PuO_2$).
5. This assembly class contains 75 total fuel rods; 66 full length rods and 8 partial length rods.
6. Square, replacing nine fuel rods.
7. Variable
8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.
10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
11. These rods may be sealed at both ends and contain Zr material in lieu of water.
12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.
13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 10 | ≤ 24,500 | ≥ 2.3 | ≤ 411 |
| ≥ 12 | ≤ 29,500 | ≥ 2.6 | ≤ 473 |
| ≥ 14 | ≤ 34,500 | ≥ 2.9 | ≤ 540 |
| ≥ 15 | ≤ 37,500 | ≥ 3.2 | ≤ 579 |

Note 1: Linear interpolation between points is permitted.

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 7 | ≤ 24,500 | ≥ 2.3 | ≤ 496 |
| ≥ 8 | ≤ 29,500 | ≥ 2.6 | ≤ 562 |
| ≥ 10 | ≤ 34,500 | ≥ 2.9 | ≤ 610 |
| ≥ 12 | ≤ 39,500 | ≥ 3.2 | ≤ 667 |
| ≥ 15 | ≤ 44,100 | ≥ 3.4 | ≤ 704 |

Note 1: Linear interpolation between points is permitted.

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH STAINLESS STEEL CLAD (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 19 | ≤ 30,000 | ≥ 3.1 | ≤ 377 |
| ≥ 24 | ≤ 40,000 | ≥ 3.1 | ≤ 475 |

Note 1: Linear interpolation between points is permitted.

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68 (Note 1)

| Post-irradiation Cooling Time (years) | Assembly Burnup (MWD/MTU) | Assembly Minimum Enrichment (wt. % U-235) | Decay Heat (Watts) |
|---|---|---|---|
| ≥ 8 | ≤ 24,500 | ≥ 2.1 | ≤ 179 |
| ≥ 9 | ≤ 29,500 | ≥ 2.4 | ≤ 208 |
| ≥ 12 | ≤ 34,500 | ≥ 2.6 | ≤ 222 |
| ≥ 15 | ≤ 39,100 | ≥ 2.9 | ≤ 238 |

Note 1: Linear interpolation between points is permitted.

REFERENCE:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9263 | 3 | USA/9263/B(U)-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Source Production and Equipment Company Inc., application dated March 22, 1999, as supplemented.

c. DOCKET NUMBER 71-9263

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SPEC-150

(2) Description

A welded titanium encased, uranium shielded, radiographic exposure device. Primary components consist of an outer titanium shell, internal supports, depleted uranium shield, and a titanium, titanium alloy or zircalloy S-tube. The contents are securely positioned in the S-tube by a source cable lock assembly and source safety plug assembly. The unit resembles a rectangular box approximately 5.4 inches wide, 5.6 inches high and 14.5 inches long. The maximum weight of the package is 53 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 15B000, Rev. 6; 15B001-3, Rev. 2; 15B002A, Rev. 5; 15B008, Rev. 4; 15B625, Rev. 1; 19B005, Rev. 0; 19B006, Rev. 0; and 190909, Rev. 0.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

150 curies (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography".

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly used must be fabricated of materials capable of resisting a 1475 °F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

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

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7, of the application, as supplemented, and

(b) Each packaging must meet the Acceptance Tests and Maintenance Program in Section 8, of the application, as supplemented.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration date: June 30, 2005.

## REFERENCES

Source Production and Equipment Company, Inc., application dated April 22, 1999.

Supplements dated: May 6, 1999; March 22, June 6, and June 19, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 23, 2000

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9269 | 3 | 71-9269 | USA/9269/B(U)-85 | 1 | OF | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AEA Technology/QSA Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology/QSA Inc. application dated July 23, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: 650L

(2) Description

A welded stainless steel encased, uranium shielded, iridium-192 source changer. Primary components consist of a steel or stainless steel housing, internal supports, depleted uranium shield, and a titanium "U" tube. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly. Additionally, the Model No. 650L has two source locking assemblies mounted on the top cover plate. These assemblies are used to secure the radioactive source in a shielded position during transport. The unit resembles a rectangular box approximately 10-inches long, 13.25-inches high and 8.25-inches wide. The maximum weight of the package is 90 pounds.

(3) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA Inc. Drawing No. R65006, Rev. G, Sheets 1-4.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed sources which meet the requirements of special form radioactive material.

5. (b) Contents (continued)

(2) Maximum quantity of material per package

240 curies (output)

Output curies are determined in accordance with American National Standard N432–1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly. The source assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining its positioning function. The cable of the source assembly must engage the source hold-down assembly. The flexible cable of the source assembly must be of sufficient length and diameter to provide positive positioning of the source at the crimp of the "U" tube.

7. The nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

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

(a) The package shall be prepared for shipment in accordance with the Operating Procedures of Chapter 7 of the application, and

(b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: November 30, 2005.

REFERENCES

AEA Technology/QSA Inc. application dated July 23, 1999.

Supplements dated November 19, 1999, October 2 and October 31, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: November 17, 2000

NRC FORM 618
(6-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9270 | 0 | 71-9270 | USA/9270/B(U)F-85 | 1 | OF | 20 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International, Inc.
3930 East Jones Bridge
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc. application dated April 30, 1997, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UMS Universal Transport Cask Package

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

The UMS is a canister-based system for the storage and transportation of spent nuclear fuel. The transportation component of the UMS system, designated the Universal Transport System, consists of a Universal Transport cask body with a closure lid and energy-absorbing impact limiters loaded with a Transportable Storage Canister (TSC) containing either spent Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) nuclear fuel or Maine Yankee site specific contents including Greater than Class C (GTCC) waste.

The NAC-UMS is designed to transport up to 24 intact PWR spent fuel assemblies, 56 intact BWR spent fuel assemblies, GTCC waste, or site specific spent nuclear fuel with associated component hardware. Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Class 4 and 5 BWR fuel assemblies include the Zircaloy channels. The loading of site specific fuels that include control component hardware may require the use of a TSC that is longer than if the hardware were excluded. The spent fuel is loaded into a TSC which contains a stainless steel grid work referred to as a basket.

The cask body of the UMS is a right-circular cylinder of multi wall construction which consists of 304 stainless steel inner and outer shells separated by lead gamma radiation shielding which is poured in place. The inner and outer shells are welded to a 304 stainless steel top forging which mates to the cask lid. The inner shell is also welded to a 304 stainless steel bottom forging and the outer shell is welded to the bottom plate. The cask bottom consists of the bottom forging and bottom plate with neutron shield material sandwiched between them. Layers of 4.5 inches thick 304

5.(a)(2) Description (Continued)

stainless steel ring and two 0.75 inch stainless steel disks are located at the bottom lead annulus between the bottom forging and the outer shell.

Neutron shield material is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates.

The containment boundary of the UMS consists of the inner shell; bottom forging; top forging; cask lid and lid inner O-ring; vent port cover plate and vent port cover plate inner O-ring; and drain port cover plate and drain port cover plate inner O-ring.

There are five TSCs of different lengths, each to accommodate a different class of PWR or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and the lengths vary from about 175 to 192 inches long. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The TSC contains the basket and fuel assemblies or GTCC waste. Spacers are placed below each Class 1, 2, 4 or 5 canisters to locate and support the canister in the cask cavity.

The spacers are free standing structures that are confined in place by the bottom of the canister and the cask bottom inner surface. The spacer(s) ensure that the canister lid is laterally supported by the cask top forging when the cask is horizontal and minimizes axial movement of the canister. Each Class 1 or PWR canister is positioned by a stainless steel spacer that is 16.75 inches or 7.65 inches in length, respectively. The Class 5 BWR canister is located with a 1.5 inch aluminum spacer and the Class 4 BWR canister is located by four 1.5 inch aluminum spacers. No spacers are used with the Class 3 PWR canister.

The spent fuel basket design uses a series of high strength stainless steel PWR or carbon steel BWR support disks to support the fuel assemblies in stainless steel tubes. The PWR fuel tubes contain BORAL neutron poison on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing BORAL on two sides of the tubes; (2) tubes containing BORAL on one side; and (3) tubes containing no BORAL. Aluminum heat transfer disks are provided in both the PWR and BWR fuel baskets to enhance thermal performance of the basket. The heat transfer disks are supported by stainless steel tie rods and split spacers that maintain the basket assembly configuration.

The GTCC waste canister is essentially identical to the Class 1 TSC, except for the placement of lifting lugs and the placement of a key way within the canister. The GTCC basket is constructed of Type 304 stainless steel and consists primarily of a cylinder with a 3-inch thick wall closed at the bottom end with a 3-inch thick plate. The cylinder is centered in the GTCC waste canister by 14 Type 304 stainless steel support plates along its length. A 3-inch thick 304 stainless steel separator fixture divides the cylinder into two vertically stacked components, each 77 inches deep with a diameter of 47.8 inches.

The package has impact limiters at each end of the cask body. The impact limiters consist of a combination of redwood and balsa wood encased in Type 304 stainless steel. The impact limiters

5.(a)(2) Description (Continued)

limit the g-loads acting on the cask during a transport drop load condition due to crushing of the redwood and balsa wood. The upper and lower impact limiters are bolted to the cask body by 16 equally spaced attachment rods with nuts.

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Overall length (with impact limiters, in) | 273.3 |
| Overall length (without impact limiters, in) | 209.3 |
| Impact Limiter Outside diameter (in) | 124.0 |
| Outside diameter (without impact limiters, in) | 92.9 |
| Cavity diameter (in) | 67.6 |
| Cavity length (in) | 192.5 |
| Cask lid thickness (in) | 6.5 |
| Bottom thickness (in) | 10.3 |
| Inner shell thickness (in) | 2.0 |
| Outer shell thickness (in) | 2.75 |
| Gamma shield thickness (in) | 2.75 |
| Radial neutron shield thickness (in) | 4.50 |

Transportable Storage Canister

| | |
|---|---|
| Shell thickness (in) | 0.625 |
| Shell bottom (in) | 1.75 |
| Shield lid thickness (in) | 7 |
| Structural lid thickness (in) | 3 |
| Outer diameter (in) | 67 |
| Internal cavity diameter (in) | 65.8 |
| Internal fuel cavity length (in), depending on class | 163-180 |
| Overall length (in), depending on class | 175-192 |

Fuel Basket

| | |
|---|---|
| Basket assembly length (in), depending on class | 162-180 |
| Basket assembly diameter (in) | 65.5 |
| Number of support disks, depending on class | 30-41 |
| Number of heat transfer disks, depending on class | 17-33 |

Total weight (pounds) including cask, basket, impact limiters, fuel, canister with lids, cask lid, and spacers for each fuel class is approximately:

| | |
|---|---|
| Class 1 (PWR) | 251,000 |
| Class 2 (PWR) | 252,000 |
| Class 3 (PWR) | 249,000 |
| Class 4 (BWR) | 256,000 |
| Class 5 (BWR) | 255,000 |

5.(a)(3) Drawings

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

| | | | |
|---|---|---|---|
| 790-209, Rev. 1 | 790-210, Rev. 1 | 790-500, Rev. 2 | 790-501, Rev. 3 |
| 790-502, Rev. 6 | 790-503, Rev. 2 | 790-504, Rev. 2 | 790-505, Rev. 1 |
| 790-508, Rev. 2 | 790-509, Rev. 2 | 790-516, Rev. 2 | 790-519, Rev. 1 |
| 790-520, Rev. 2 | 790-570, Rev. 4 | 790-571, Rev. 2 | 790-572, Rev. 4 |
| 790-573, Rev. 7 | 790-574, Rev. 3 | 790-575, Rev. 7 | 790-581, Rev. 5 |
| 790-582, Rev. 7 | 790-583, Rev. 4 | 790-584, Rev. 11 | 790-585, Rev. 8 |
| 790-591, Rev. 2 | 790-592, Rev. 5 | 790-593, Rev. 4 | 790-594, Rev. 2 |
| 790-595, Rev. 5 | 790-605, Rev. 8 | 790-611, Rev. 3 | 790-612, Rev. 3 |
| 412-501, Rev. 2 | 412-502, Rev. 2 | | |

5.(b) Contents

(1) Type and Form of Material

The package is designed to transport four types of contents as listed below:

i. 24 intact irradiated PWR fuel assemblies within a TSC;
ii. 56 intact irradiated BWR fuel assemblies within a TSC;
iii. 24 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee within a TSC; or
iv. GTCC waste from Maine Yankee within a TSC.

Each type of package contents is described in detail below.

(i) Intact PWR assemblies

The package is designed to transport 24 irradiated intact PWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. An empty fuel rod position must be filled with a solid filler rod, fabricated from either Zircalloy or Type 304 stainless steel, which displaces an equal or greater volume than that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with Zircaloy cladding. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications of Table 5.(b)(1)(i)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(i)-2 below. PWR fuel assemblies may include standard inserts such as guide tube thimble plugs and burnable poison rods.

The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% $^{235}U$ and 4.2 wt% $^{235}U$ respectively. Unenriched fuel assemblies are not authorized for loading into the TSC. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is 5 years. The maximum weight of $UO_2$ is 11.53 MTU per cask.

Table 5.(b)(1)(i)-1, Intact PWR Fuel Assembly Characteristics

| TSC Class[1] | Vendor[2] | Array | Max. Length (in) | Max. Width (in) | Max. Assembly Weight | Max MTU | No of Fuel Rods | Max Pitch (in) | Min Rod Dia (in) | Min Clad Thick (in) | Max Pellet Dia (in) | Max Active Length (in) | Min Guide Tube Thickness (in) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | CE | 14x14 | 157.3 | 8.11 | 1292 | 0.404 | 176[4] | 0.590 | 0.438 | 0.024 | 0.380 | 137.0 | 0.040 |
| 1 | Ex/ANF | 14x14 | 160.2 | 7.76 | 1271 | 0.369 | 179 | 0.556 | 0.424 | 0.030 | 0.351 | 142.0 | 0.034 |
| 1 | WE | 14x14 | 159.8 | 7.76 | 1177 | 0.362 | 179 | 0.556 | 0.400 | 0.024 | 0.345 | 144.0 | 0.034 |
| 1 | WE | 14x14 | 159.8 | 7.76 | 1302 | 0.415 | 179 | 0.556 | 0.422 | 0.022 | 0.368 | 145.2 | 0.034 |
| 1 | WE, Ex/ANF | 15x15 | 159.8 | 8.43 | 1472 | 0.465 | 204 | 0.563 | 0.422 | 0.024 | 0.366 | 144.0 | 0.015 |
| 1 | Ex/ANF | 17x17 | 159.8 | 8.43 | 1348 | 0.413 | 264 | 0.496 | 0.360 | 0.025 | 0.303 | 144.0 | 0.016 |
| 1 | WE | 17x17 | 159.8 | 8.43 | 1482 | 0.468 | 264 | 0.496 | 0.374 | 0.022 | 0.323 | 144.0 | 0.016 |
| 1 | WE | 17x17 | 160.1 | 8.43 | 1373 | 0.429 | 264 | 0.496 | 0.360 | 0.022 | 0.309 | 144.0 | 0.016 |
| 2 | B&W | 15x15 | 165.7 | 8.54 | 1515 | 0.481 | 208 | 0.568 | 0.430 | 0.026 | 0.369 | 144.0 | 0.016 |
| 2 | B&W | 17x17 | 165.8 | 8.54 | 1505 | 0.466 | 264 | 0.502 | 0.379 | 0.024 | 0.324 | 143.0 | 0.017 |
| 3 | CE | 16x16 | 178.3 | 8.10 | 1430 | 0.442 | 236[4] | 0.506 | 0.382 | 0.023 | 0.3255 | 150.0 | 0.035 |
| 1 | Ex/ANF[3] | 14x14 | 160.2 | 7.76 | 1215 | 0.375 | 179 | 0.556 | 0.417 | 0.030 | 0.351 | 144.0 | 0.036 |
| 1 | CE[3] | 15x15 | 147.5 | 8.20 | 1360 | 0.432 | 216 | 0.550 | 0.418 | 0.026 | 0.358 | 132.0 | --- |
| 1 | Ex/ANF[3] | 15x15 | 148.9 | 8.25 | 1339 | 0.431 | 216 | 0.550 | 0.417 | 0.030 | 0.358 | 131.8 | --- |
| 1 | CE[3] | 16x16 | 158.2 | 8.10 | 1300 | 0.403 | 236[4] | 0.506 | 0.382 | 0.023 | 0.3255 | 136.7 | 0.035 |

[1] Minimum and Maximum initial Enrichments are 1.9 wt% $^{235}U$ and 4.2 wt% $^{235}U$, respectively. All fuel rods are Zircaloy clad.

[2] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[3] 14x14, 15x15, and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively, These are not generic fuel assemblies provided to multiple reactors.

[4] Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 5.(b)(1)(i)-2, Loading Table for Intact PWR Fuel

| Minimum Initial Enrichment wt% $^{235}U$ (E) | Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years) | | | | | 30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years) | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | CE 14x14 | 14x14 | 15x15 | 16x16 | 17x17 | CE 14x14 | 14x14 | 15x15 | 16x16 | 17x17 |
| 1.9 ≤ E < 2.1 | 6 | 8 | 8 | 7 | 8 | 8 | 10 | 11 | 9 | 10 |
| 2.1≤ E < 2.3 | 6 | 7 | 8 | 6 | 7 | 7 | 10 | 10 | 8 | 10 |
| 2.3≤ E < 2.5 | 6 | 7 | 7 | 6 | 7 | 7 | 9 | 10 | 8 | 9 |
| 2.5 ≤ E < 2.7 | 6 | 7 | 7 | 6 | 7 | 7 | 9 | 9 | 7 | 8 |
| 2.7≤ E < 2.9 | 6 | 7 | 7 | 6 | 7 | 6 | 8 | 9 | 7 | 8 |
| 2.9≤ E < 3.1 | 5 | 7 | 7 | 6 | 6 | 6 | 8 | 8 | 7 | 8 |
| 3.1≤ E < 3.3 | 5 | 6 | 7 | 6 | 6 | 6 | 8 | 8 | 7 | 7 |
| 3.3 ≤ E < 3.5 | 5 | 6 | 6 | 6 | 6 | 6 | 7 | 8 | 6 | 7 |
| 3.5 ≤ E < 3.7 | 5 | 6 | 6 | 6 | 6 | 6 | 7 | 7 | 6 | 7 |
| 3.7 ≤ E≤ 4.2 | 5 | 6 | 6 | 6 | 6 | 6 | 7 | 7 | 6 | 7 |
| Minimum Initial Enrichment wt% $^{235}U$ (E) | 35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years) | | | | | 40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years) | | | | |
| | CE 14x14 | 14x14 | 15x15 | 16x16 | 17x17 | CE 14x14 | 14x14 | 15x15 | 16x16 | 17x17 |
| 1.9 ≤ E < 2.1 | 11 | 15 | 15 | 13 | 15 | 18 | 20 | 21 | 20 | 20 |
| 2.1≤ E < 2.3 | 10 | 13 | 14 | 12 | 13 | 15 | 19 | 19 | 18 | 19 |
| 2.3≤ E < 2.5 | 9 | 12 | 13 | 11 | 12 | 14 | 17 | 19 | 17 | 17 |
| 2.5 ≤ E < 2.7 | 9 | 12 | 12 | 10 | 11 | 12 | 16 | 18 | 15 | 17 |
| 2.7≤ E < 2.9 | 8 | 11 | 11 | 9 | 11 | 11 | 15 | 18 | 14 | 17 |
| 2.9≤ E < 3.1 | 8 | 10 | 10 | 9 | 10 | 10 | 14 | 18 | 13 | 15 |
| 3.1≤ E < 3.3 | 7 | 10 | 10 | 9 | 10 | 10 | 13 | 17 | 13 | 15 |
| 3.3 ≤ E < 3.5 | 7 | 9 | 10 | 8 | 9 | 9 | 12 | 17 | 13 | 15 |
| 3.5 ≤ E < 3.7 | 7 | 9 | 10 | 8 | 9 | 8 | 11 | 17 | 12 | 15 |
| 3.7 ≤ E≤ 4.2 | 7 | 8 | 10 | 8 | 8 | 8 | 11 | 15 | 12 | 14 |

5.(b)(1)(ii) Intact BWR assemblies

The package is designed to transport 56 irradiated intact BWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks.

For BWR fuel, the initial enrichment limit (the enrichment of the as-delivered fresh fuel assembly) represents the maximum peak planar-average enrichment allowed for loading into the TSC. The peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly.

The fuel assemblies consist of uranium dioxide pellets with Zircaloy cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 5.(b)(1)(ii)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(ii)-2.

BWR intact fuel assemblies are authorized with or without channels based on a maximum channel width of 120 mils. The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% $^{235}U$ and 4.0 wt% $^{235}U$ respectively. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is six years. The maximum weight of $UO_2$ is 11.08 MTU per cask. Unenriched fuel assemblies are not authorized for loading into the TSC. BWR fuel assemblies with unenriched axial blankets must have an enriched central fuel region and are acceptable for loading into a TSC if the minimum fuel enrichment of the central region is 1.9 wt% $^{235}U$. Any empty fuel position must be filled with a solid filler rod fabricated from either Zircalloy or Type 304 stainless steel.

Table 5.(b)(1)(ii)-1, Intact BWR Fuel Assembly Characteristics

| Canister Class [1,5] | Vendor[4] | Array | Max. Length (in) | Max. Assembly Width (in)[5] | Max. Assembly Weight (lb)[6] | Max MTU | No of Fuel Rods | Max Pitch (in) | Min Rod Dia (in) | Min Clad Thick (in) | Max Pellet Dia (in) | Max Active Length (in)[2] |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 4 | Ex/ANF | 7x7 | 171.3 | 5.51 | 620 | 0.196 | 48 | 0.738 | 0.570 | 0.036 | 0.490 | 144 |
| 4 | Ex/ANF | 8x8 | 171.3 | 5.51 | 563 | 0.177 | 63 | 0.641 | 0.484 | 0.036 | 0.405 | 145.2 |
| 4 | Ex/ANF | 9x9 | 171.3 | 5.51 | 557 | 0.173 | 79 | 0.572 | 0.424 | 0.030 | 0.357 | 145.2 |
| 4 | GE | 7x7 | 171.1 | 5.51 | 681 | 0.199 | 49 | 0.738 | 0.570 | 0.036 | 0.488 | 144.0 |
| 4 | GE | 7x7 | 171.2 | 5.51 | 681 | 0.198 | 49 | 0.738 | 0.563 | 0.032 | 0.487 | 144.0 |
| 4 | GE | 8x8 | 171.1 | 5.51 | 639 | 0.173 | 60 | 0.640 | 0.484 | 0.032 | 0.410 | 145.2 |
| 4 | GE | 8x8 | 171.1 | 5.51 | 681 | 0.179 | 62 | 0.640 | 0.483 | 0.032 | 0.410 | 145.2 |
| 4 | GE | 8x8 | 171.1 | 5.51 | 681 | 0.186 | 63 | 0.640 | 0.493 | 0.034 | 0.416 | 144.0 |
| 5 | Ex/ANF | 8x8 | 176.1 | 5.51 | 588 | 0180 | 62 | 0.641 | 0.484 | 0.036 | 0.405 | 150.0 |
| 5 | Ex/ANF | 9x9 | 176.1 | 5.51 | 576 | 0.167 | 74[3] | 0.572 | 0.424 | 0.030 | 0.357 | 150.0 |
| 5[5] | Ex/ANF | 9x9 | 176.1 | 5.51 | 576 | 0.178 | 79[3] | 0.572 | 0.424 | 0.030 | 0.357 | 150.0 |
| 5 | GE | 7x7 | 175.9 | 5.51 | 683 | 0.198 | 49 | 0.738 | 0.563 | 0.032 | 0.487 | 144.0 |
| 5 | GE | 8x8 | 176.1 | 5.51 | 665 | 0.179 | 60 | 0.640 | 0.484 | 0.032 | 0.410 | 150.0 |
| 5 | GE | 8x8 | 175.9 | 5.51 | 681 | 0.185 | 62 | 0.640 | 0.483 | 0.032 | 0.410 | 150.0 |
| 5 | GE | 8x8 | 175.9 | 5.51 | 681 | 0.188 | 63 | 0.640 | 0.493 | 0.034 | 0.416 | 146.0 |
| 5 | GE | 9x9 | 176.1 | 5.51 | 646 | 0.186 | 74[3] | 0.566 | 0.441 | 0.028 | 0.376 | 150.0 |
| 5 | GE | 9x9 | 176.1 | 5.51 | 646 | 0.198 | 79[3] | 0.566 | 0.441 | 0.028 | 0.376 | 150.0 |

[1] Maximum Peak Planar Average Enrichment 4.0 wt%$^{235}$U. Minimum enrichment is 1.9 wt%$^{235}$U. All fuel rods are Zircalloy clad.

[2] 150 inch active fuel length assemblies contain 6 inch natural uranium blankets on top and bottom.

[3] Shortened active fuel length in some rods.

[4] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[5] Assembly width including channel. Unchanneled or channeled may be loaded based on a maximum channel thickness of 120 mils.

[6] Exxon/ANF assembly weights are listed without channel.

Table 5.(b)(1)(ii)-2, Loading Table for Intact BWR Fuel

| Minimum Initial Enrichment wt% $^{235}U$ (E) | Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years) | | | 30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years) | | |
|---|---|---|---|---|---|---|
| | 9x9 | 8x8 | 7x7 | 9x9 | 8x8 | 7x7 |
| 1.9 ≤ E < 2.1 | 8 | 8 | 8 | 14 | 13 | 15 |
| 2.1≤ E < 2.3 | 7 | 7 | 8 | 12 | 12 | 13 |
| 2.3≤ E < 2.5 | 7 | 7 | 7 | 11 | 10 | 11 |
| 2.5 ≤ E < 2.7 | 7 | 6 | 7 | 9 | 9 | 10 |
| 2.7≤ E < 2.9 | 6 | 6 | 6 | 9 | 8 | 9 |
| 2.9≤ E < 3.1 | 6 | 6 | 6 | 8 | 8 | 8 |
| 3.1≤ E < 3.3 | 6 | 6 | 6 | 7 | 7 | 8 |
| 3.3 ≤ E < 3.5 | 6 | 6 | 6 | 7 | 7 | 7 |
| 3.5 ≤ E < 3.7 | 6 | 6 | 6 | 7 | 7 | 7 |
| 3.7 ≤ E≤ 4.0 | 6 | 6 | 6 | 7 | 7 | 7 |
| Minimum Initial Enrichment wt% $^{235}U$ (E) | 35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years) | | | 40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years) | | |
| | 9x9 | 8x8 | 7x7 | 9x9 | 8x8 | 7x7 |
| 1.9 ≤ E < 2.1 | 24 | 23 | 25 | 34 | 33 | 35 |
| 2.1≤ E < 2.3 | 21 | 20 | 22 | 31 | 30 | 32 |
| 2.3≤ E < 2.5 | 19 | 18 | 20 | 29 | 28 | 29 |
| 2.5 ≤ E < 2.7 | 17 | 16 | 17 | 26 | 25 | 27 |
| 2.7≤ E < 2.9 | 14 | 14 | 15 | 24 | 23 | 24 |
| 2.9≤ E < 3.1 | 13 | 12 | 13 | 21 | 20 | 22 |
| 3.1≤ E < 3.3 | 11 | 11 | 12 | 19 | 18 | 20 |
| 3.3 ≤ E < 3.5 | 10 | 10 | 11 | 17 | 16 | 18 |
| 3.5 ≤ E < 3.7 | 10 | 9 | 10 | 15 | 14 | 16 |
| 3.7 ≤ E≤ 4.0 | 10 | 9 | 10 | 14 | 13 | 15 |

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The package is designed to transport 24 irradiated intact or damaged PWR fuel assemblies, canistered fuel debris, and GTCC waste within the TSC from the Maine Yankee Reactor. The standard Maine Yankee fuel assembly is the intact PWR CE 14x14 (see section 5.(b)(1)(i)).

In the course of reactor operations, some of the 14x14 assemblies were modified to change the standard configuration. These modifications included a) the removal of fuel rods without replacement; b) the replacement of removed fuel rods or burnable poison rods with rods of a different material, such as stainless steel, or with fuel rods of a different enrichment; and c) the insertion of control elements, or instruments or plug thimbles, in guide tube positions. In addition to the modified fuel assemblies, there are fuel assemblies that were designed with variable enrichment and axial blankets. These fuel assemblies are not modified, but differ from the cask design basis fuel assemblies.

Stainless steel spacers may be used in canisters to axially position PWR intact fuel assemblies that are shorter than the available cavity length. The minimum length of the PWR intact fuel assembly internal structure and bottom end fitting and/or spacers will ensure that the minimum distance to the fuel region for the base of the canister is 3.2 inches.

Unenriched fuel assemblies are not authorized for loading.

The following are the allowable Maine Yankee site specific contents:

5.(b)(1)(iii)(A) Maine Yankees' site specific contents not requiring preferential loading patterns:

(1) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly

(4) PWR intact fuel assemblies with fuel rods having variable enrichments with a maximum rod enrichment up to 4.21 wt% $^{235}U$ and that also has a maximum planar average enrichment up to 3.99 wt% $^{235}U$. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(5) PWR intact fuel assemblies with annular axial end blanket enrichments up to 2.6 wt% $^{235}U$. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(6) PWR intact fuel assemblies with burnable poison rods or solid filler rods may occupy up to 16 of 176 fuel rod positions. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(7) PWR intact fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

5.(b)(1)(iii)(B) Maine Yankee site-specific allowable contents requiring preferential loading based on shielding, criticality, or thermal constraints (Maine Yankee CE 14 x 14 intact PWR fuel assemblies). A PWR basket fuel diagram can be found on Figure 5.(b)(1)(iii)(B)-1.

(1) Maine Yankee CE 14 x 14 PWR intact fuel assemblies with a burn up between 45,000 and 50,000 MWD/MTU meeting the following requirements for verification of the oxide layer thickness and high burn up fuel requiring preferential loading in the peripheral PWR fuel basket positions:

A verification program is required to determine the oxide layer thickness on high burn up fuel by measurement or by statistical analysis. A fuel assembly having a burn up between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burn up. The verification program shall be capable of classifying high burn up fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

I. A HIGH BURN UP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

II. A HIGH BURN UP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) PWR intact fuel assemblies with up to 176 fuel rods missing from the fuel assembly lattice. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner loading position in the PWR fuel basket.

(3) PWR intact fuel assemblies with burnable poison rods replaced by hollow Zircaloy rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(4) Intact fuel assemblies with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 canister. Only one start-up source may be loaded in any fuel assembly or any canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(5) PWR intact fuel assemblies with CEA ends (fingertips) and/or an ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a PWR fuel basket corner position and must be loaded in a Class 2 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. CEA fingertips are not considered as CEA's for determination of minimum cool times.

5.(b)(1)(iii)(C) Maine Yankee CE 14 x 14 PWR fuel enclosed in a Maine Yankee Fuel Can (MYFC).

All Maine Yankee CE 14 x 14 PWR fuel enclosed in an MYFC must be loaded in a Class 1 fuel canister in a corner position of the PWR fuel basket. Up to 4 MYFC may be loaded into a TSC. Intact Maine Yankee CE 14 x 14 PWR fuel may be loaded into a Maine Yankee fuel can. The contents that must be loaded in the Maine Yankee fuel can are:

(1) PWR fuel assemblies with up to two intact or damaged fuel rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(2) A damaged fuel assembly with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A damaged fuel assembly cannot have an inserted CEA or other non-fuel component. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(3) Individual intact or damaged fuel rods in a rod type structure, which may be a guide tube, to maintain configuration control. The combined maximum average

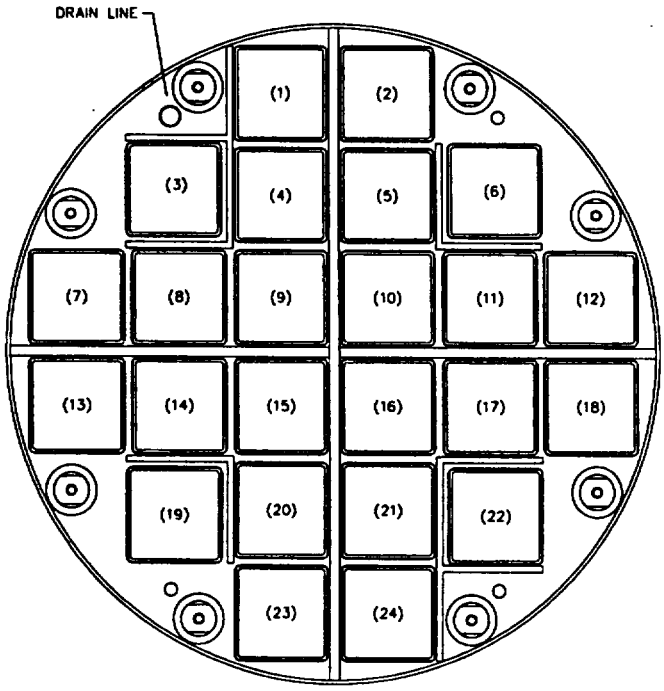
burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(4) Fuel debris consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(5) Consolidated Fuel lattice and structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight less than or equal to 2,100 pounds. A consolidated fuel lattice cannot have an inserted CEA or other non-fuel component. Only one consolidated fuel lattice may be stored in any TSC. Fuel must be cooled a minimum of 24 years.

(6) High burn up fuel assemblies not meeting the oxide layer thickness criteria previously defined in Section 5.(b)(1)(iii)(B)(1). The combined maximum average burn up, minimum cool time and maximum and minimum initial $^{235}U$ enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

**PWR Basket Fuel Loading Position Diagram, Figure 5.(b)(1)(iii)(B)-1**




1. Basket corner positions are positions 3, 6, 19, and 22. Corner positions are also periphery positions.
2. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24. Periphery positions include the corner positions.

Table 5.(b)(1)(iii)(A)-1, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

| Burnup 30 GWD/MTU | | Minimum Cool Time (Years) for | | | | |
|---|---|---|---|---|---|---|
| Enrichment | No CEA (Class 1) | No CEA (Class 2) | 5 Yr CEA | 10 Yr CEA | 15 Yr. CEA | 20 Yr. CEA |
| 1.9 ≤ E < 2.1 | 6 | 6 | 7 | 6 | 6 | 6 |
| 2.1≤ E < 2.3 | 6 | 6 | 7 | 6 | 6 | 6 |
| 2.3≤ E < 2.5 | 6 | 6 | 6 | 6 | 6 | 6 |
| 2.5 ≤ E < 2.7 | 6 | 6 | 6 | 6 | 6 | 6 |
| 2.7≤ E < 2.9 | 6 | 6 | 6 | 6 | 6 | 6 |
| 2.9≤ E < 3.1 | 5 | 6 | 6 | 6 | 6 | 6 |
| 3.1≤ E < 3.3 | 5 | 5 | 6 | 6 | 6 | 5 |
| 3.3 ≤ E < 3.5 | 5 | 5 | 6 | 6 | 5 | 5 |
| 3.5 ≤ E < 3.7 | 5 | 5 | 6 | 5 | 5 | 5 |
| 3.7 ≤ E≤ 4.2 | 5 | 6 | 5 | 5 | 5 | 5 |

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

| Burnup 35 GWD/MTU | | Minimum Cool Time (Years) for | | | | |
|---|---|---|---|---|---|---|
| Enrichment | No CEA (Class 1) | No CEA (Class 2) | 5 Yr CEA | 10 Yr CEA | 15 Yr. CEA | 20 Yr. CEA |
| 1.9 ≤ E < 2.1 | 8 | 8 | 9 | 8 | 8 | 8 |
| 2.1≤ E < 2.3 | 7 | 7 | 9 | 8 | 8 | 8 |
| 2.3≤ E < 2.5 | 7 | 7 | 8 | 7 | 7 | 7 |
| 2.5 ≤ E < 2.7 | 7 | 7 | 8 | 7 | 7 | 7 |
| 2.7≤ E < 2.9 | 6 | 7 | 7 | 7 | 7 | 7 |
| 2.9≤ E < 3.1 | 6 | 6 | 7 | 7 | 6 | 6 |
| 3.1≤ E < 3.3 | 6 | 6 | 7 | 6 | 6 | 6 |
| 3.3 ≤ E < 3.5 | 6 | 6 | 7 | 6 | 6 | 6 |
| 3.5 ≤ E < 3.7 | 6 | 6 | 6 | 6 | 6 | 6 |
| 3.7 ≤ E≤ 4.2 | 6 | 6 | 6 | 6 | 6 | 6 |

Table 5.(b)(1)(iii)(A)-1, continued , Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

| Burnup 40 GWD/MTU | | Minimum Cool Time (Years) for | | | | |
|---|---|---|---|---|---|---|
| Enrichment | No CEA (Class 1) | No CEA (Class 2) | 5 Yr CEA | 10 Yr CEA | 15 Yr. CEA | 20 Yr. CEA |
| 1.9 ≤ E < 2.1 | 11 | 12 | 14 | 13 | 12 | 12 |
| 2.1≤ E < 2.3 | 10 | 10 | 13 | 11 | 11 | 11 |
| 2.3≤ E < 2.5 | 9 | 9 | 12 | 10 | 10 | 10 |
| 2.5 ≤ E < 2.7 | 9 | 9 | 10 | 9 | 9 | 9 |
| 2.7≤ E < 2.9 | 8 | 8 | 10 | 9 | 8 | 8 |
| 2.9≤ E < 3.1 | 8 | 8 | 9 | 8 | 8 | 8 |
| 3.1≤ E < 3.3 | 7 | 7 | 8 | 8 | 8 | 8 |
| 3.3 ≤ E < 3.5 | 7 | 7 | 8 | 7 | 7 | 7 |
| 3.5 ≤ E < 3.7 | 7 | 7 | 8 | 7 | 7 | 7 |
| 3.7 ≤ E≤ 4.2 | 7 | 7 | 7 | 7 | 7 | 7 |

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

| Burnup 45 GWD/MTU | | Minimum Cool Time (Years) for | | | | |
|---|---|---|---|---|---|---|
| Enrichment | No CEA (Class 1) | No CEA (Class 2) | 5 Yr CEA | 10 Yr CEA | 15 Yr. CEA | 20 Yr. CEA |
| 1.9 ≤ E < 2.1 | 18 | 18 | 21 | 19 | 18 | 18 |
| 2.1≤ E < 2.3 | 15 | 16 | 19 | 17 | 17 | 16 |
| 2.3≤ E < 2.5 | 14 | 14 | 18 | 16 | 15 | 15 |
| 2.5 ≤ E < 2.7 | 12 | 13 | 16 | 14 | 14 | 13 |
| 2.7≤ E < 2.9 | 11 | 12 | 14 | 13 | 12 | 12 |
| 2.9≤ E < 3.1 | 10 | 11 | 13 | 12 | 11 | 11 |
| 3.1≤ E < 3.3 | 10 | 10 | 12 | 11 | 10 | 10 |
| 3.3 ≤ E < 3.5 | 9 | 9 | 11 | 10 | 10 | 10 |
| 3.5 ≤ E < 3.7 | 9 | 9 | 10 | 10 | 10 | 10 |
| 3.7 ≤ E≤ 4.2 | 9 | 9 | 10 | 10 | 10 | 10 |

Table 5.(b)(1)(iii)(A)-1, continued, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

| Burnup 50 GWD/MTU | | Minimum Cool Time (Years) for | | | | |
|---|---|---|---|---|---|---|
| Enrichment | No CEA (Class 1) | No CEA (Class 2) | 5 Yr CEA | 10 Yr CEA | 15 Yr. CEA | 20 Yr. CEA |
| 1.9 ≤ E < 2.1 | 27 | 27 | 29 | 27 | 27 | 27 |
| 2.1≤ E < 2.3 | 24 | 24 | 27 | 25 | 24 | 24 |
| 2.3≤ E < 2.5 | 22 | 22 | 25 | 23 | 22 | 22 |
| 2.5 ≤ E < 2.7 | 19 | 19 | 23 | 21 | 20 | 20 |
| 2.7≤ E < 2.9 | 17 | 17 | 21 | 19 | 18 | 18 |
| 2.9≤ E < 3.1 | 15 | 16 | 19 | 18 | 18 | 18 |
| 3.1≤ E < 3.3 | 15 | 15 | 18 | 17 | 17 | 17 |
| 3.3 ≤ E < 3.5 | 15 | 15 | 17 | 17 | 17 | 17 |
| 3.5 ≤ E < 3.7 | 14 | 14 | 15 | 15 | 15 | 15 |
| 3.7 ≤ E≤ 4.2 | 14 | 14 | 15 | 15 | 15 | 15 |

Table 5.(b)(1)(iii)(A)-2, Loading Table (Years) for Maine Yankee CE 14x14 fuel containing ICI Thimbles

| Minimum Initial Enrichment wt% $^{235}U$ (E) | Burnup ≤ 30 GWD/MTU | 30 < Burnup ≤ 35 GWD/MTU | 35 < Burnup ≤ 40 GWD/MTU | 40 < Burnup ≤ 45 GWD/MTU | 45 < Burnup ≤ 50 GWD/MTU |
|---|---|---|---|---|---|
| 1.9 ≤ E < 2.1 | 6 | 8 | 11 | 18 | 27 |
| 2.1≤ E < 2.3 | 6 | 7 | 10 | 16 | 24 |
| 2.3≤ E < 2.5 | 6 | 7 | 9 | 14 | 22 |
| 2.5 ≤ E < 2.7 | 6 | 7 | 9 | 13 | 19 |
| 2.7≤ E < 2.9 | 6 | 6 | 8 | 11 | 17 |
| 2.9≤ E < 3.1 | 5 | 6 | 8 | 10 | 15 |
| 3.1≤ E < 3.3 | 5 | 6 | 7 | 10 | 15 |
| 3.3 ≤ E < 3.5 | 5 | 6 | 7 | 9 | 15 |
| 3.5 ≤ E < 3.7 | 5 | 6 | 7 | 9 | 14 |
| 3.7 ≤ E≤ 4.2 | 5 | 6 | 7 | 9 | 14 |

Table 5.(b)(1)(iii)(A)-3, Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

| Assy Number | Burnup (GWD/MTU) | Enrichment (wt %) | SSR Source (g/s/assy) | Cool Time (years) | Earliest Transportable |
|---|---|---|---|---|---|
| N420 | 45 | 3.3 | 2.1602E+13 | 10 | Jan 2001 |
| N842 | 35 | 3.3 | 3.1396E+12 | 6 | Jan 2001 |
| N868 | 40 | 3.3 | 5.2444E+12 | 7 | Jan 2001 |
| R032 | 45 | 3.5 | 1.4550E+13 | 9 | Jan 2005 |
| R439 | 50 | 3.5 | 1.3998E+13 | 14 | Jan 2010 |
| R444 | 50 | 3.5 | 5.5993E+13 | 19 | Jan 2015 |

Table 5.(b)(1)(iii)(A)-4, Cool time (years) for Maine Yankee CE 14x14 damaged fuel

| Minimum Initial Enrichment wt% $^{235}U$ (E) | Burnup ≤ 30 GWD/MTU | 30 < Burnup ≤ 35 GWD/MTU | 35 < Burnup ≤ 40 GWD/MTU | 40 < Burnup ≤ 45 GWD/MTU | 45 < Burnup ≤ 50 GWD/MTU |
|---|---|---|---|---|---|
| 1.9 ≤ E < 2.1 | 7 | 11 | 19 | 28 | 37 |
| 2.1≤ E < 2.3 | 6 | 9 | 16 | 26 | 34 |
| 2.3≤ E < 2.5 | 6 | 8 | 14 | 23 | 32 |
| 2.5 ≤ E < 2.7 | 6 | 8 | 12 | 21 | 30 |
| 2.7≤ E < 2.9 | 6 | 7 | 11 | 19 | 27 |
| 2.9≤ E < 3.1 | 6 | 7 | 10 | 17 | 25 |
| 3.1≤ E < 3.3 | 5 | 7 | 9 | 15 | 23 |
| 3.3 ≤ E < 3.5 | 5 | 6 | 8 | 13 | 21 |
| 3.5 ≤ E < 3.7 | 5 | 6 | 8 | 12 | 19 |
| 3.7 ≤ E≤ 4.2 | 5 | 6 | 7 | 11 | 17 |

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee

The package is designed to transport Maine Yankee Greater Than Class C Waste within a TSC. Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53. The maximum curie inventory shall not exceed the values shown in Table 5.(b)(1)(iv)-1.

Table 5.(b)(1)(iv)-1, Maine Yankee GTCC Curie Inventory Limits per TSC

| Radionuclide | Curie Inventory (Ci)/ TSC |
|---|---|
| H-3 | 3.00E+02 |
| C-14 | 1.50E+02 |
| Mn-54 | 3.50E+02 |
| Fe-55 | 2.00E+05 |
| Co-58 | 1.00E+01 |
| Co-60 | 2.90E+05 |
| Ni-59 | 8.20E+02 |
| Ni-63 | 9.00E+04 |
| Nb-94 | 1.00E+01 |
| Tc-99 | 1.00E+01 |

5.(b)(2)Maximum quantity of material per package

The maximum content's weight shall not exceed 77,500 pounds.

(i) For the contents described in 5.(b)(1)(i) and 5.(b)(1)(iii): 24 PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, with a maximum weight of 38,500 pounds and a maximum decay heat limit per package not to exceed the values in Table 5.(b)(2)-1. The individual PWR assembly decay heat is limited to 0.83 kW.

| Table 5.(b)(2)-1, PWR Decay Heat Limits | | | | |
|---|---|---|---|---|
| Cool Time (Years) | PWR Decay Heat Limit (kW) Burnup (MWD/MTU) | | | |
| | 35,000 | 40,000 | 45,000 | 50,000[1] |
| 5 | 20.0 | 20.0 | 19.9 | 19.3 |
| 6 | 19.5 | 19.3 | 19.2 | 18.7 |
| 7 | 17.8 | 17.8 | 17.7 | 17.2 |
| 10 | 17.4 | 17.3 | 17.2 | 16.8 |
| 15 | 16.8 | 16.8 | 16.7 | 16.5 |

[1]Maine Yankee PWR fuel assemblies

(ii) For the contents described in 5.(b)(1)(ii): 56 BWR assemblies with a maximum weight of 39,000 pounds and a maximum decay heat limit per package of 16 kW. The individual BWR assembly decay heat is limited to 0.29 kW.

(iii) For the contents described in 5.(b)(1)(iv): GTCC waste with a maximum weight per package of 20,000 pounds in total or 10,000 pounds per compartment. The maximum decay heat for the GTCC is 4.5 kW per package.

5.(c) Transport Index for Criticality Control (Criticality Safety Index)

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

6. The package must be transported as exclusive use in a closed transport vehicle as per 10 CFR 71.47(b).

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

9. Expiration date: October 31, 2007.

REFERENCES

NAC International, Inc., Application dated April 30, 1997.

NAC International, Inc., Supplements dated June 18, 1999, May 31, June 29, August 8, and September 20, 2000, February 28, March 14, March 31, June 1, and November 16, 2001, and January 31, March 13, August 12, September 27, and October 21, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 31, 2002

| NRC FORM 618 (6-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9272 | 6 | 71-9272 | USA/9272/AF-85 | 1 OF | 4 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company LLC application dated January 4, 2002, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: CE-B1

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

The metal inner container is approximately 11-1/4 inches by 18-1/8 inches by 182 inches long. There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 33-1/2 inches by 34-3/4 inches by 208-1/2 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on a series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,964 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Combustion Engineering Drawing Nos.:

L-9272-01, Sheets 1 and 2, Rev. 1, and
L-9272-02, Rev. 1.

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of $UO_2$ fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches. Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(i) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly; Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36; Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52; Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b)(1) Type and form of material (continued)

(iii) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80; Maximum U-235 enrichment is 4.0 weight percent for all corner rods; Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.43 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b) (2) Maximum quantity of material per package

Two fuel assemblies. The total weight of contents not to exceed 1,184 pounds.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into, or out of, the sheathed fuel assembly.

7. Polyethylene inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated.

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

9. Only ABB/Combustion Engineering packagings with Serial Nos. CE-B1/001 through CE-B1/039, inclusive, are authorized for use.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration date: January 31, 2007.

## REFERENCES

Westinghouse Electric Company LLC application dated January 4, 2002.

Supplement dated: January 9, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date January 25, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9274 | 6 | 71-9274 | USA/9274/AF | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Combustion Engineering, Inc. application dated April 9, 1997, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: ABB-2901

(2) Description

A shipping container for low-enriched uranium oxide pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner container is 10.75 ± 1/4 inches square and approximately 30 inches long, constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has a 16-gauge closure head and 12-gage closure ring with drop forged lugs and a 5/8-inch diameter bolt. The drum has approximate dimensions of 22.5-inch ID by 36-inch height. The uranium oxide pellets are packaged in boxes positioned within a steel insert. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with ABB Combustion Engineering Nuclear Systems Drawing No. L-9274-01, Rev. 0.

(b) Contents

(1) Type and form of material

Sintered uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope. The maximum pellet diameter is 0.969 cm, and the minimum pellet diameter is 0.818 cm.

5.(b)(2) Maximum quantity of material per package

227 pounds of pellets, with the U-235 content not to exceed 4.54 kg. The pellets must be packaged on corrugated stainless steel trays, within shipping container boxes and a shipping container insert in accordance with ABB Combustion Engineering Nuclear Systems Drawing Nos. L-9274-02, Sheets 1 and 2, Rev. 0, and L-9274-03, Rev. 0.

Maximum weight of contents within the inner container is 427 pounds, including radioactive material, secondary containers, and other packaging material.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 0.5

6. Corrugated stainless steel trays must be positioned between each layer of pellets, and on the top and bottom of the pellet stack. Spacers must be inserted in partially filled pellet shipping boxes to provide a snug fit.

7. The package may also contain stainless steel pellets, depleted uranium pellets, and neutron poisons such as gadolinia, erbium, and boron carbide.

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

   (a) Prior to each shipment the insert (containment vessel) gasket shall be inspected. This gasket shall be replaced if inspection shows any defects.

   (b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 and the Maintenance Program of Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration date: September 30, 2007.

## REFERENCES

Combustion Engineering, Inc. application dated April 9, 1997.

Supplement dated: May 20, 1997.

ABB Combustion Engineering Nuclear Power, Inc. supplements dated: June 10 and 21, and July 6, 1999.

ABB C-E Nuclear Power, Inc. supplements dated: March 28, and April 4 and 12, 2000.

CE Nuclear Power, LLC supplement dated: September 14, 2000.

Westinghouse Electric Company LLC supplements dated: September 18, 2000, and June 11, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 20, 2002

| NRC FORM 618 (6-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9276 | 0 | 71-9276 | USA/9276/B(U)F-85 | 1 | OF | 26 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

BNFL Fuel Solutions
3600 Glen Canyon Road
Scotts Valley, CA 95066

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BNFL Fuel Solutions application dated April 20, 2001, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. FuelSolutions™ TS125 Transportation Package

(2) Description

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 $1 \times 10^{-7}$ 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 and 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.

(3) Drawings

The FuelSolutions™ TS125 Transportation Package is constructed and assembled in accordance with the following drawings:

FS-200, Revision 1, Sheets 1 through 3
FS-205, Revision 2, Sheets 1 through 3
FS-210, Revision 2, Sheets 1 through 9
FS-220, Revision 1, Sheets 1 through 7
FS-230, Revision 1, Sheets 1 and 2
W21-110, Revision 4, Sheets 1 through 9

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

(b) Contents

(1) 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.

(i) W21 Canister

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.

(ii) 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 locations. Fuel assemblies with [illegible]

## 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) |

W74-2 Notes:

(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 235U 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)<br>≤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 | ≤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.<br>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.<br>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.<br>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.<br>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.<br>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.

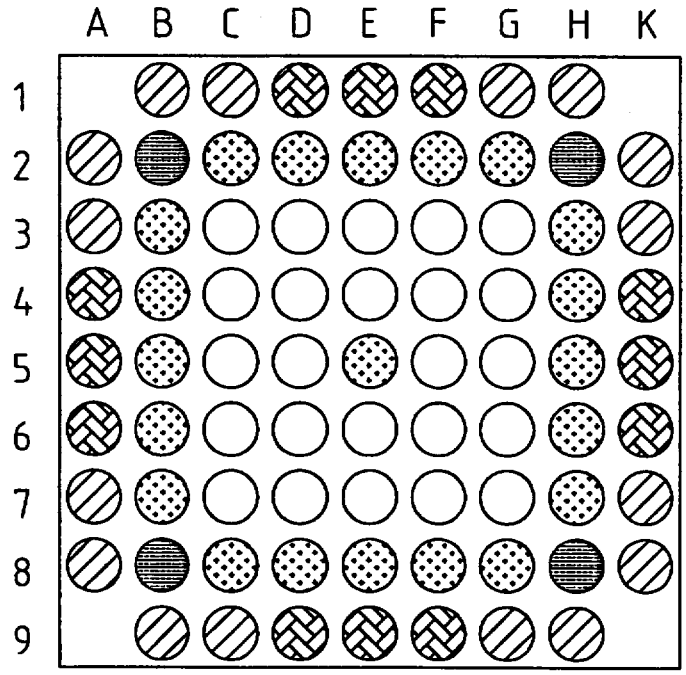
**Table 11 - W74 Canister Assembly Specific Requirements for Big Rock Point MOX Fuel**

| 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:

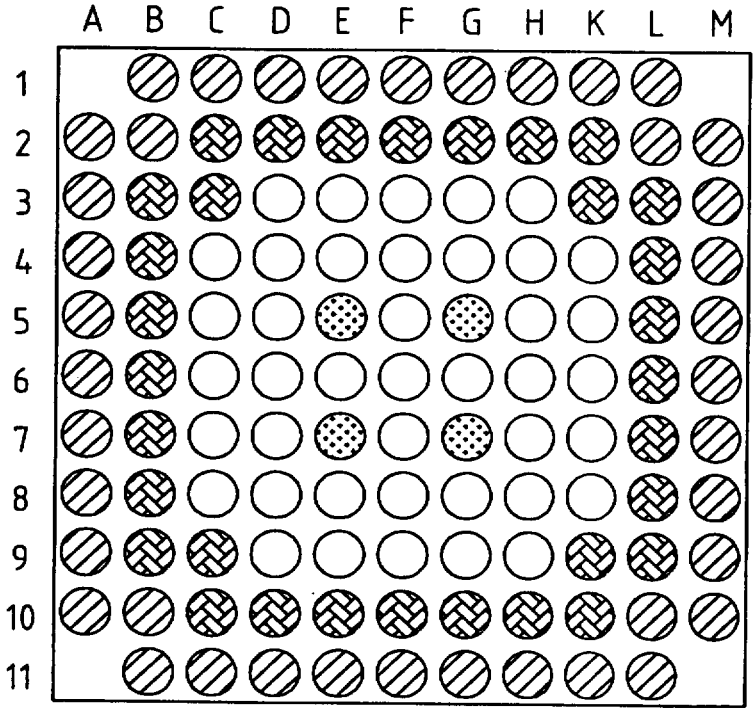
(1) The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.




Fuel Pin Compositions

- 2.55 Wt% U-235
- 3.30 Wt% U-235
- 4.50 Wt% U-235
- 3.30 Wt% U-235 and 1.00 % Gd2O3 in UO2
- 0.711 Wt% U-235 3.65 % PuO2

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

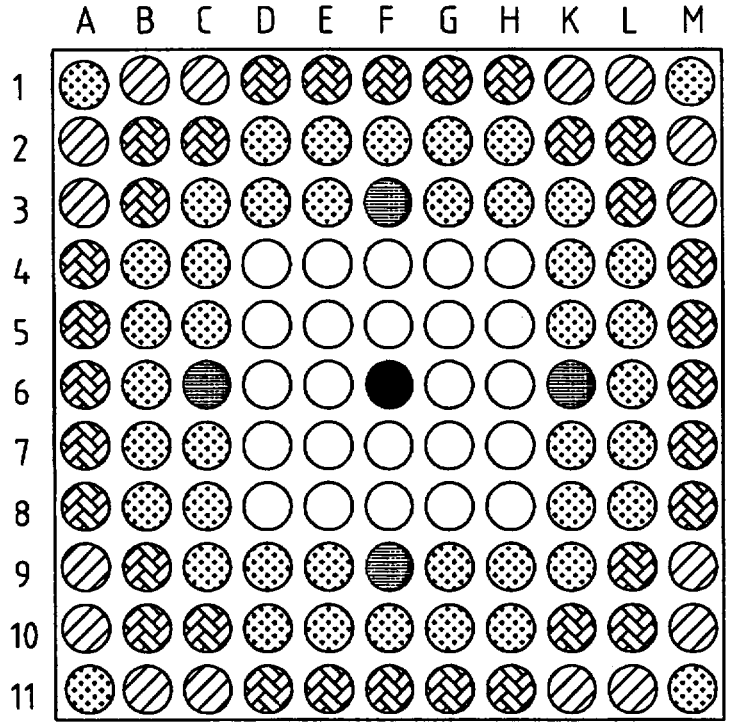



Fuel Pin Compositions

- 2.40 Wt% U-235
- 1.56 Wt% U-235
- 1.03 Wt% PuO2
- 2.40 Wt% U-235
- 2.45 Wt% PuO2
- Water Rods

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**

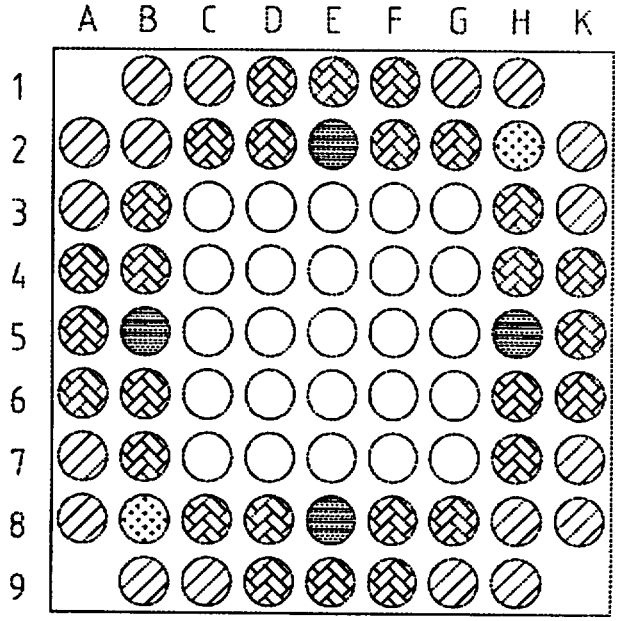



Fuel Pin Compositions

- 2.30 Wt% U-235
- 3.20 Wt% U-235
- 4.60 Wt% U-235
- Solid Zirc Rod
- 4.60 Wt% U-235 1.20 Wt% Gd2O3
- 0.711 Wt% U-235 5.45 Wt% PuO2

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

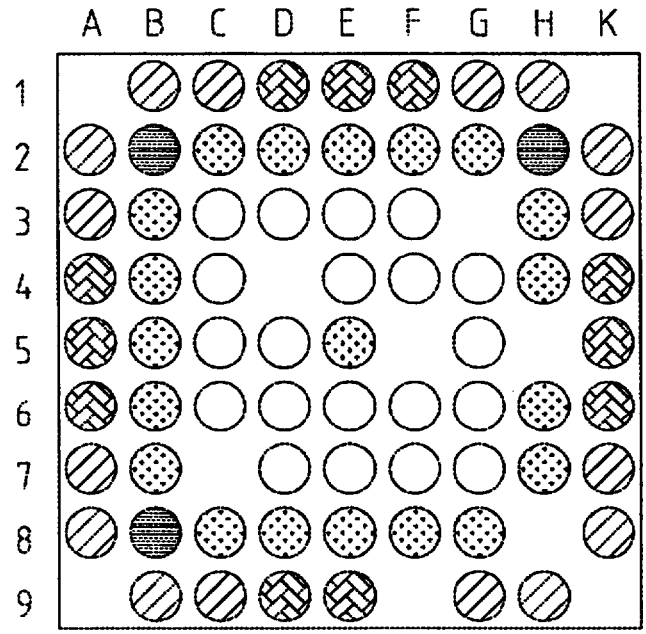
**Figure 3 - G-Pu (11x11) BRP MOX Assembly Array**




Fuel Pin Compositions

- 2.50 Wt% U-235
- 3.40 Wt% U-235
- 2.50 Wt% U-235 25.4 g/rod Pu
- 3.40 Wt% U-235 2.00 Wt% Gd2O3 in UO2
- 4.5% Wt% U-235

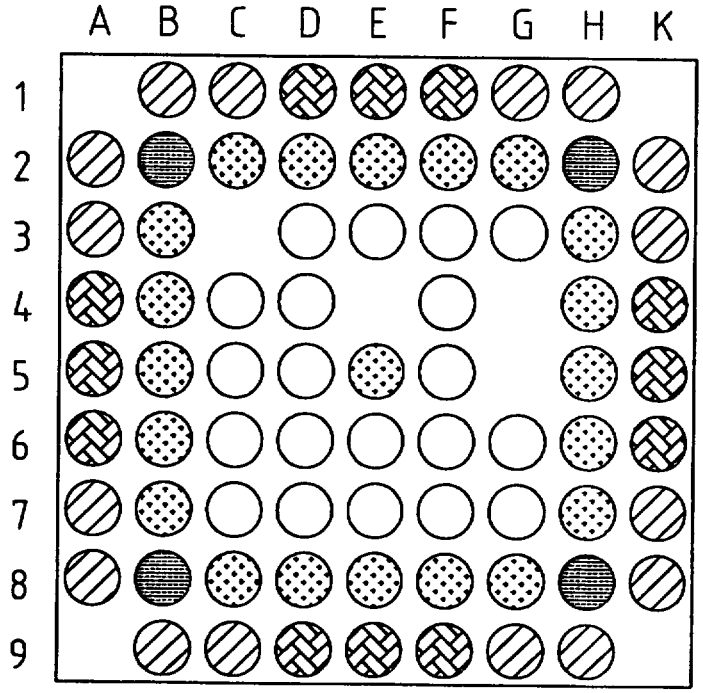
**Figure 4 - UO2 9x9 BRP Assembly with Two Inserted MOX Rods**




Fuel Pin Compositions

- 2.55 Wt% U-235
- 3.30 Wt% U-235
- 4.50 Wt% U-235
- 3.30 Wt% U-235 and 1.00% Gd2O3 in UO2
- 0.711% U-235 3.65% PuO2

**Figure 5 - J2 Partial MOX Assembly Array #1**




Fuel Pin Compositions

- 2.55 Wt% U-235
- 3.30 Wt% U-235
- 4.50 Wt% U-235
- 3.30 Wt% U-235 and 1.00 % Gd2O3 in UO2
- 0.711 Wt% U-235 3.65 % PuO2

**Figure 6 - J2 Partial MOX Assembly Array #2**

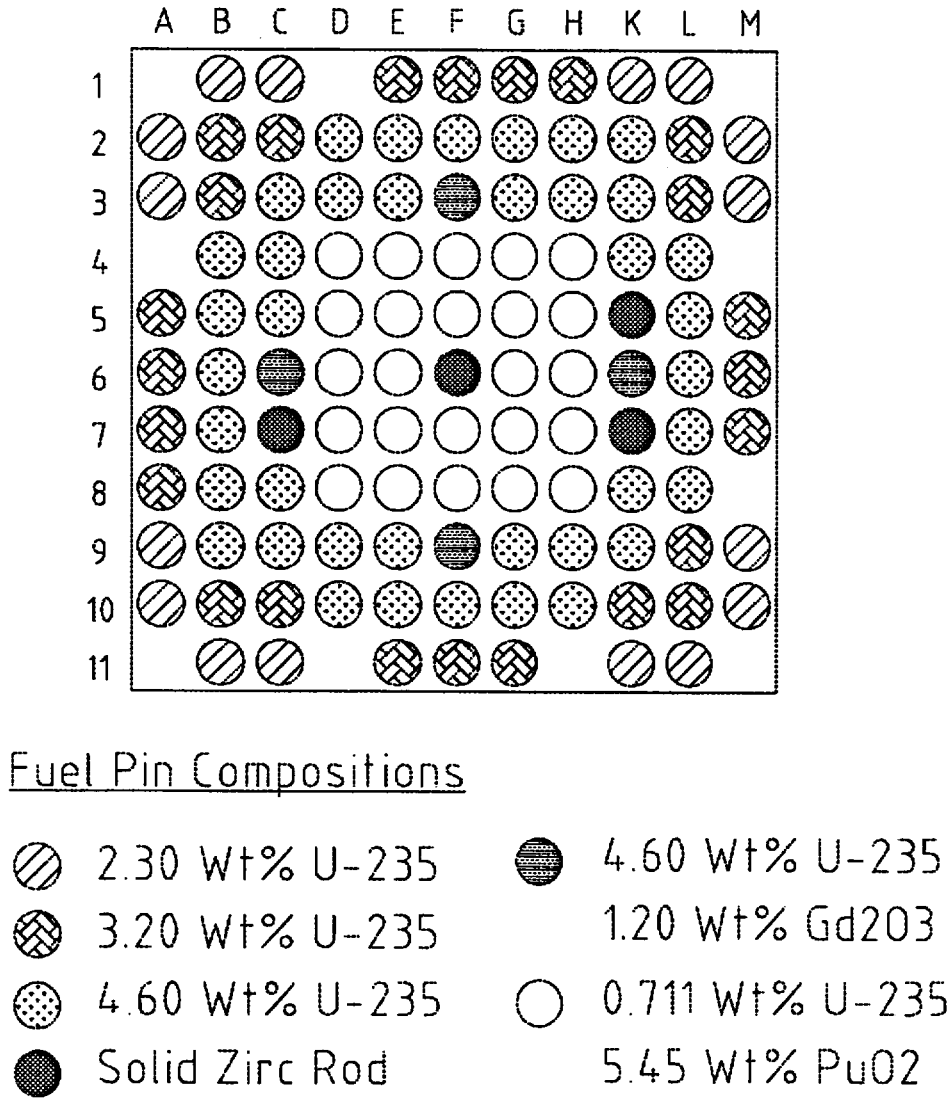
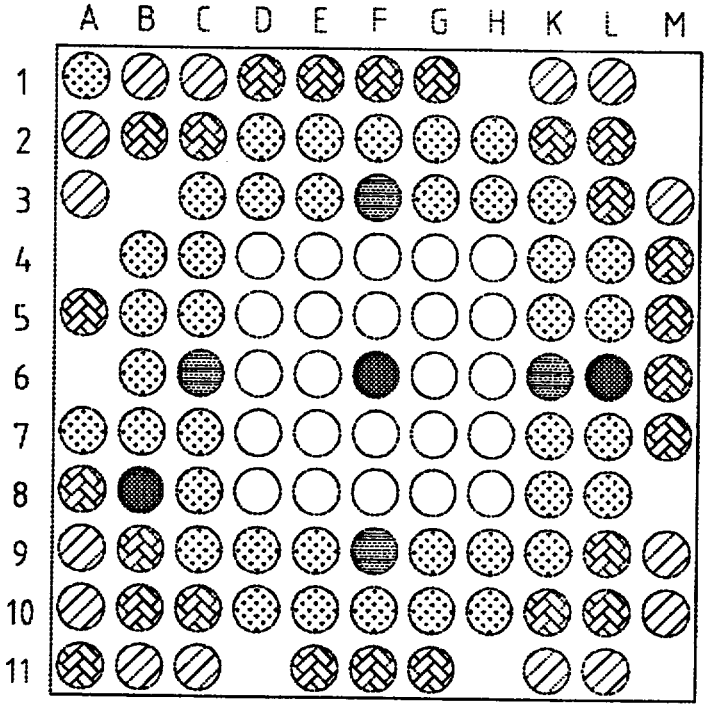



**Figure 7 - G-Pu Partial MOX Assembly Array #1**




Fuel Pin Compositions

- 2.30 Wt% U-235
- 3.20 Wt% U-235
- 4.60 Wt% U-235
- Solid Zirc Rod
- 4.60 Wt% U-235 / 1.20 Wt% Gd2O3
- 0.711 Wt% U-235 / 5.45 Wt% PuO2

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

(2) Maximum Quantity of Material Per Package

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.

(3) 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, or

b) Q ≤1.05 kW, and
Q ≤ 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 ≤ 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) ≤ 0.275 kW.

(c) Transport Index for Criticality Control

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

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

(1) The package shall meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

(2) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR Part 71.12.

8. Expiration date: September 30, 2007

## REFERENCES

BNFL Fuel Solutions Corporation, application dated April 20, 2001.

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
And Safeguards

Date: August 28, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9277 | 1 | 71-9277 | USA/9277/B( )F | 1 OF | 5 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

General Atomics
3550 General Atomics Court
San Diego, CA 92121

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Public Service Company of Colorado
application dated March 28, 1996, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: FSV-1 Unit 3

(2) Description

The FSV-1 Unit 3 is a stainless steel-encased, depleted uranium-shielded cask. The cask body is a cylinder 208-inches long and 28-inches in diameter, except for the top flange area, which is 31-inches in diameter. The cavity is approximately 17.7-inches in diameter and 187.6-inches long.

The cask may be used in one of seven configurations (A through G) depending on contents. Configurations A, B, C, and D are used to ship solid, non-fissile irradiated hardware. These configurations use an outer lid consisting of a 3.75-inch thick stainless steel plate and a 2.25-inch thick depleted uranium shield. The lid is bolted to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the outer lid and cask body. Configuration B does not require an inner container. Configuration C uses a supplemental stainless steel shield ring and cover plate. Configuration D uses a supplemental carbon steel shield ring and cover plate.

Configuration E is used to ship Fort St. Vrain (FSV) high temperature gas reactor (HTGR) fuel elements. This configuration uses the stainless steel inner container (as shown in General Atomic Drawing Nos. GADR 55-2-1, Rev. C, and GADR 55-2-2, Rev. A) as the containment vessel. The inner container lid is a stainless steel shell containing depleted uranium 4.15-inches thick. The inner lid is secured to the inner container body by 12 0.5-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the inner lid and inner container body. Configuration E is equipped with an impact limiter on the upper end.

Configuration F and G are used to shop solid non-fissile irradiated and contaminated hardware from the FSV HGTR. These configurations use a 4.75-inch thick steel outer lid. The lid is secured to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a molded silicone elastomeric seal ring between the outer lid and the cask body. Configurations F and G both use an impact limiter on the upper end. Configurations F and G also use a burial canister with a 12-inch thick carbon steel plug. The shielded spacer in the burial canister is used only in Configuration G.

The overall weight for the FSV-1 Unit 3 package is 46,025 pounds for Configurations A, B, C, and D and 47,600 pounds for Configurations E, F, and G.

(3) Drawings

The FSV-1 Unit 3 package is constructed in accordance with the following drawings:

Configuration A

National Lead Company Drawing Nos. : 70086F, Rev. 7; 70296F, Rev. 2; and General Atomics Drawing No. 1501-003, Rev. C

Configuration B

Same as for Configuration A except that an inner container is not required.

Configuration C and D

In addition to the drawings for Configuration A, General Atomics Drawing Nos. GADR 55-2-10, Issue D, and GADR 55-2-14, Issue N/C (optional). Configuration C uses a supplemental stainless steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue B. Configuration D uses a supplemental carbon steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue A.

Configuration E

In addition to the drawings for Configuration A, General Atomics Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; and GADR 55-2-3, Issue B.

Configuration F and G

In addition to the drawings for Configuration A, General Atomic Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; GADR 55-2-12, Issue C; and GADR 55-2-13, Issue A.

5. (b) Contents

(1) Type and form of material

(i) Irradiated fuel elements consisting of graphite body, hexagonal in horizontal cross section, approximately 31.2-inches high and 14.2-inches across the flats. Prior to irradiation, each fuel element contains thorium and uranium enriched to a maximum of 93.5 w/o in the U-235 isotope, or

(ii) Solid, irradiated, and contaminated hardware, which may include fissile material, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53 and neutron source components, or

(iii) Solid, nonfissile, irradiated and contaminated hardware which has been removed from the Fort St. Vrain High Temperature Gas Cooled Reactor and the surface contamination does not exceed 51 millicuries per package.

(2) Maximum quantity of material per package

Decay heat not to exceed 4.1 kw and:

(i) Item 5(b)(1)(i) above:

Six fuel elements each containing a maximum of 1.4 kg of enriched uranium, having a thorium/uranium ratio greater than 8.1:1 and weighing approximately 300 pounds. The gross weight of the cask cavity contents, including the component spacers, inner container, and irradiated fuel elements shall not exceed 4,430 pounds. Contents must be shipped in Configuration E.

(ii) Item 5(b)(1)(ii) above:

The gross weight of the cask cavity contents, including appropriate component spacers, liners, inner containers, shield rings and solid, non-fissile, irradiated and contaminated hardware shall not exceed 3,720 pounds. Contents must be shipped in Configurations A, B, C, or D.

(iii) Item 5(b)(1)(iii) above:

The gross weight of all of the cask cavity contents, including burial canister and spacers, with or without supplemental shielding shall not exceed 4,430 pounds. Contents must be shipped in Configurations F or G.

5. (c) Transport Index for Criticality Control

   Minimum transport index to be shown on label for nuclear criticality control: 100

6. As needed, appropriate component spacers must be used in the cask cavity when shipping the contents described in paragraph 5(b) to limit movement of contents during shipment.

7. For transport of the contents of Item 5(b)(1)(ii) in Configuration D, the dose rate measured on the surface of the package must not exceed 200 mr/hr. For the purpose of this requirement, the surface of any personnel barrier may not be considered the surface of the package.

8. The Model No. FSV-1 Unit 3 cask may be wrapped with reinforced plastic when shipping the contents described in Item 5(b)(1)(ii) or (iii) provided the heat generation rate does not exceed 500 watts. The applicable requirements of 10 CFR 71.87 must be satisfied prior to wrapping the cask.

9. Use of packaging fabricated after August 31, 1986, is not authorized.

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

    (a) Configurations A, B, C, and D of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume I, of the application, as supplemented. The package shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume I, of the application, as supplemented.

    (b) Configurations E, F, and G of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume II, of the application, as supplemented. The packages shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume II, of the application, as supplemented.

    (c) The main flange seals must be replaced within twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

    (d) The silicone O-ring on the inner container primary plug in Configuration E must be replaced within the twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: July 31, 2006

| NRC FORM 618 (5-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9277 | 1 | 71-9277 | USA/9277/B( )F | 5 OF | 5 |

## REFERENCE

Public Service Company of Colorado application dated March 28, 1996, as supplemented by Chem-Nuclear Systems, L.L.C., letter dated May 19, 1997, and General Atomics letter dated June 6, 1997, as supplemented April 11, 2001, and June 7, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 9, 2001

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9279 | 1 | 71-9279 | USA/9279/B(U)F-85 | 1 OF | 4 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
Department of Energy
Washington, DC 20586

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Westinghouse Electric Corporation application dated August 27, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HalfPACT Waste Shipping Container

(2) Description

A stainless steel and polyurethane foam insulated shipping container designed to provide double containment for shipment of contact-handled transuranic waste. The packaging consists of an unvented, 1/4-inch thick stainless steel inner containment vessel (ICV), positioned within an outer containment assembly (OCA) consisting of an unvented 1/4-inch thick stainless steel outer containment vessel (OCV), an approximate 8-inch thick layer of polyurethane foam, a 1/4-inch thick layer of ceramic fiber paper and a 1/4 to 3/8-inch thick outer stainless steel shell. The package is a right circular cylinder with outside dimensions of approximately 94 inches diameter and 70 inches height. The package weighs not more than 18,100 pounds when loaded with the maximum allowable contents of 7,600 pounds.

The OCA has a domed lid which is secured to the OCA body with a locking ring. The OCV containment seal is provided by a butyl rubber O-ring. The OCV is equipped with a seal test port and a vent port.

The ICV is a right circular cylinder with domed ends. The outside dimensions of the ICV are approximately 74 inches diameter and 69 inches height. The ICV lid is secured to the ICV body with a locking ring. The ICV containment seal is provided by a butyl rubber O-ring. The ICV is equipped with a seal test port and vent port. Aluminum spacers are placed in the top and bottom domed ends of the ICV during shipping. The cavity available for the contents is a cylinder of approximately 73 inches diameter and 69 inches height.

5.(a)(3) Drawings

The package is constructed and assembled in accordance with Portemus Engineering, Inc. Drawings 707-SAR sheets 1-12, Revision 4 and 163-001, sheets 1-3, Revision 2.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, standard waste box (SWB), 85-gallon drum or pipe overpack. The payload containers are described in Appendix 2.1, "Specification of Authorized Payload Containers" of the CH-TRAMPAC, Rev. 0. Explosives, corrosives (pH less than 2 or greater than 12.5), nonradioactive pyrophorics, and compressed gases are prohibited. Within a payload container radioactive pyrophorics must not exceed 1 weight percent by weight and free liquids must not exceed 1 percent by volume. Flammable volatile organic compounds are limited to 500 ppm in the headspace of any payload container.

(2) Maximum quantity of material per package

The package contents are limited to 7,600 pounds, including the weight of the payload containers and any other components of the payload assembly. The maximum gross weight for a payload container not to exceed the following:

(i) 328 pounds per pipe overpack (6 inch)
(ii) 547 pounds per pipe overpack (12 inch)
(iii) 1,000 pounds per 55-gallon drum
(iv) 1,000 pounds per 85-gallon drum
(v) 4,000 pounds per SWB

Maximum number of payload containers per package and authorized packaging configurations as follows:

(i) 7 55-gallon drums
(ii) 1 SWB
(iii) 1 SWB, containing up to four 55-gallon drums
(iv) 1 SWB, containing one bin
(v) 4 85-gallon drums, each overpacking one 55-gallon drum

Fissile material not to exceed the following:

| Payload Container Type | Pu-239 Equivalent Per Payload Container | Pu-239 Equivalent Per Package |
|---|---|---|
| 55-/85-gallon drum | 200 grams | 325 grams |
| Pipe overpack | 200 grams | 1,400 grams |
| SWB | 325 grams | 325 grams |

Pu-239 equivalent must be determined in accordance with Section 3.1, "Nuclear Criticality" in CH-TRAMPAC, Rev. 0

Maximum decay heat per package not to exceed 30 watts. Decay heat per payload container not to exceed the values in Table 5-5 of CH-TRAMPAC, Rev. 0, "List of Approved Shipping Categories, Maximum Allowable Hydrogen Gas Generation Rates, and Maximum Allowable Wattages."

(c) Transport Index for Criticality Control

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

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight and center of gravity; and radiation dose rate must be determined and limited in accordance with CH-TRAMPAC, Rev. 0.

7. Each payload container must be assigned to a shipping category in accordance with Section 5.1, "Payload Shipping Category" of CH-TRAMPAC, Rev. 0. Each payload container and payload assembly must not exceed the allowable wattage in accordance with Section 5.2, "Decay Heat," or must be tested for gas generation in accordance with Section 5.5, "Test Category Requirements" of CH-TRAMPAC, Rev. 0

8. Each payload container must be labeled to indicate its shipping category. Payload containers within a package shall be selected in accordance with Section 6.0, "Payload Assembly Requirements" of CH-TRAMPAC, Rev. 0.

9. Each payload container must be equipped with filtered vents meeting the minimum requirements of Section 2.5, "Specification for Filter Vents" of CH-TRAMPAC, Rev. 0. Drums which were not equipped with filtered vents during storage must be aspirated in accordance with Section 5.4, "Venting and Aspiration" of CH-TRAMPAC, Rev. 0

10 Only waste defined by an approved content code in CH-TRUCON, Rev. 0 is authorized for transport.

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

    (a) 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.

    (b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

    (c) All free standing water must be removed from the inner containment vessel cavity and the outer containment vessel cavity before shipment.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

13. Expiration date: October 31, 2005.

REFERENCES

Westinghouse Electric Corporation application dated August 27, 1998.

Supplements dated: July 27 and September 21, 2000, and July 26, 2002.

Contact-Handled Transuranic Waste Authorized Methods for Payload Control (CH-TRAMPAC), Rev. 0, August 1998

CH-TRU Waste Content Codes (CH-TRUCON), Rev. 0, dated August 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles L. Miller for
E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 16, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9280 | 1 | USA/9280/AF-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Babcock & Wilcox Company
P. O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

BWX Technologies, Inc. application dated August 20, 1997, as supplemented.

c. DOCKET NUMBER 71-9280

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UBE-1

(2) Description

A steel drum for the transport of solid uranium and uranium-beryllium waste materials. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is approximately 600 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Babcock & Wilcox Company Drawing. No. LP3023C, Rev. 4.

(b) Contents

(1) Type and form of material

Uranium and uranium-beryllium mixtures in the form of solids, and solid waste materials.

(2) Maximum quantity of material per package

550 pounds. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than $1 \times 10^{-3}$ $A_2$ per package.

Page 2 - Certificate No. 9280 - Revision No. 1 - Docket No. 71-9280

5.(c) Transport Index for Criticality Control

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

| Maximum Fissile Mass Per Package (grams U-235 per package) | Minimum Transport Index |
|---|---|
| 2.0 | 0.5 |
| 5.0 | 1.0 |
| 6.0 | 1.2 |
| 10.0 | 2.0 |
| 20.0 | 4.0 |
| 25.0 | 5.0 |
| 50.0 | 10.0 |
| 100.0 | 20.0 |

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application.

   (b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: December 31, 2002.

REFERENCES

BWX Technologies, Inc. application dated August 20, 1997.

Supplements dated: October 6 and December 8, 1997; and February 6, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date February 10, 1998

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9281 | 2 | USA/9281/AF-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

BWX Technologies, Inc.
Naval Nuclear Fuel Division
P. O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

BWX Technologies, Inc., application dated December 9, 1997, as supplemented.

c. DOCKET NUMBER 71-9281

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: UBE-2

(2) Description

A steel drum for the transport of uranium and uranium-beryllium waste materials of solid form. The waste may be contained within compacted 55-gallon drums. The packaging is a 70-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 18-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is approximately 1000 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Babcock & Wilcox Company Drawing. No. LP3024C, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium and uranium-beryllium waste of solid form. The waste may be contained within compacted 55-gallon drums.

(2) Maximum quantity of material per package

950 pounds, including compacted secondary containers. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than $1 \times 10^{-3}$ $A_2$ per package.

Page 2 - Certificate No. 9281 - Revision No. 2 - Docket No. 71-9281

5.(c) Transport Index for Criticality Control

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

| Maximum Fissile Mass Per Package (grams U-235 per package) | Minimum Transport Index |
|---|---|
| 2.0 | 0.5 |
| 5.0 | 1.0 |
| 6.0 | 1.2 |
| 10.0 | 2.0 |
| 20.0 | 4.0 |
| 25.0 | 5.0 |
| 50.0 | 10.0 |
| 100.0 | 20.0 |

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application.

   (b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: May 31, 2003.

REFERENCES

BWX Technologies, Inc., application dated December 9, 1997.
Supplement dated: March 6, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 12, 2000

NRC FORM 618
(3-88)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9282 | 0 | USA/9282/B(U)-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Source Production
and Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Source Production and Equipment Company, Inc. application dated June 28, 1999, as supplemented

c. DOCKET NUMBER 71-9282

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SPEC-300

(2) Description

The SPEC-300 is a radiographic device that consists of a source assembly, a depleted uranium shield, and a stainless steel enclosure. The radioactive source assembly is housed in a zircaloy or titanium "S" tube that is surrounded by the depleted uranium shield. The depleted uranium shield is secured in the stainless steel enclosure. The void space between the depleted uranium shield and the enclosure is filled with high density polyurethane foam. The package is approximately 26 inches long, 14 inches wide, and 15 inches high. The maximum gross weight of the package is 780 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Co., Inc., General Arrangement drawings: 198000 sheets 1-8, Rev. 4 and B190700 sheet 1, Rev. 3.

(b) Contents

(1) Type and form of material

Cobalt-60 sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

300 Curies (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly must be fabricated of materials capable of resisting a 1475 °F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

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

   (a) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented; and

   (b) The package must meet the Acceptance Test and Maintenance Program of Chapter 8.0 of the application, as supplemented.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration date: April 30, 2005.

REFERENCES

Source Production and Equipment Company, Inc., application dated June 28, 1999.

Supplements dated: October 6, November 4, November 22, and December 15, 1999; and February 29 and March 27, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9283 | 0 | USA/9283/B(U)-85 | 1 | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

AEA Technology/QSA Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

AEA Technology/QSA Inc. application dated May 21, 1998, as supplemented.

c. DOCKET NUMBER 71-9283

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: OPL-660 and OP-660

(2) Description

The Model Nos. OPL-660 and OP-660 consist of a radiography camera within a protective container. The protective container is a 20 mm Cartridge Shipping and Storage box fabricated according to military specification MIL-S-23389B. The protective container is of welded steel construction and is approximately 18½ inches long, 14½ inches high, and 8¼ inches wide. The protective container is fitted with foam and wood inserts and a lid that is secured by latches. The Model 660 series projector fits snugly in the center of the foam inserts within the protective container. The Model No. OPL-660 container has thin lead sheets to provide extra shielding at the ends and bottom. The maximum weight of the package is 88 pounds.

The Model 660 series projector is a radiography device. The projector's overall dimensions are approximately 12⅞ inches long, 5¼ inches wide, and 9⅝ inches high. The projector weighs a maximum of 56 pounds. The principal components of the 660 series projectors include an outer steel shell, polyurethane foam, a depleted uranium shield, an "S" tube, and end plugs. The sealed source contents are securely positioned in the "S" tube by a source cable locking device and shipping plug.

(3) Drawings

The packaging is constructed in accordance with the following AEA Technology QSA, Inc., Drawings:

R66050, Rev. C, Sheets 1 & 2, and R66060, Rev. A, Sheets 1-3.

(b) Contents

(1) Type and form of material

Iridium-192 sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

(I) 140 Curies (output) for the Model No. 660B or 660BE projectors.

(ii) 120 Curies (output) for the Model No. 660, 660E, 660A or 660AE projectors.

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly must be fabricated of materials capable of resisting a 1475 °F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

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

(a) The package must meet the Acceptance Test and Maintenance Program of Chapter 8.0 of the application, as supplemented; and

(b) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented.

9. The package authorized by this certificate is hereby approved for use under general license provisions of 10 CFR §71.12.

10. Expiration date: June 30, 2003.

REFERENCES

AEA Technology QSA, Inc., application dated May 21, 1998.

Supplement dated: June 15, 1998

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 6/19/98

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9284 | 2 | 71-9284 | USA/9284/B(U)F-85 | 1 | OF | 4 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Columbiana Hi Tech Front End, LLC
200 West Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Eco-Pak Specialty Packaging application dated June 19, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: ESP-30X Protective Shipping Package for 30-inch $UF_6$ Cylinders

(2) Description

An overpack for the transport of 30-inch enriched uranium hexafluoride ($UF_6$) cylinders. The shape of the overpack is a right circular cylinder constructed of two 11 gauge carbon steel shells. The area between the shells is filled with fire retardant, phenolic foam per ESP specification ESP-PF-1. The volume between the 1/2" inch thick end plates of the two shells is also filled with phenolic foam. A stepped horizontal joint permits the top half of the overpack to be removed from the base. The horizontal joint of each half of the overpack is covered with steel and a 5/8" thick silicone gasket seals the joint. The overpack halves are secured with ten 3/4" diameter steel bolts and nuts.

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Outer shell inside diameter | 43" |
| Outer shell length | 96" |
| Inner shell inside diameter | 30 7/8" |
| Inner shell length | 82 5/8" |
| Overpack weight | 2,955 pounds |
| 30B Cylinder weight | 1,390 pounds |
| $UF_6$ maximum load | 5,020 pounds |
| Maximum package gross weight (including contents) | 9,365 pounds |

(3) Drawings

The packaging is constructed and assembled in accordance with ESP Drawing Nos.:

30X-1 SAR, Rev. 2, Sheets 1-4

5.(b) Contents

(1) Type and form of material

The $UF_6$ must be packaged in Model 30B $UF_6$ cylinders which have been fabricated, inspected, tested and maintained in accordance with the requirements of ANSI N14.1. The $UF_6$, which may contain either virgin or recycled uranium, must not contain more than the following maximum quantities of radionuclides and impurities:

| | |
|---|---|
| $U^{232}$ | 5.0E-09 g/gU |
| $U^{234}$ | 2.0E-03 g/gU |
| $U^{235}$ | 5.0E-02 g/gU |
| $U^{236}$ | 2.5E-02 g/gU |
| $U^{238}$ | balance of total uranium content |
| Pu and Np | Alpha activity not exceed 3.3 Bq/gU |
| $Tc^{99}$ | 5.0E-06 g/gU |
| $Th^{228}$ | 1.17E-09 g/gU |

Fission Products — $4.4 \times 10^5$ Mev Bq/d kgU (total contribution from gamma emitting fission products); this results in the following individual maximum activities:

| | |
|---|---|
| $Ru^{106}/Rh^{106}$ | 2095 Bq/gU |
| $Ru^{103}/Rh^{103}$ | 885 Bq/gU |
| $Ce^{144}/Pr^{144}/Pr^{144}$ | 8349 Bq/gU |
| $Sb^{125}$ | 1030 Bq/gU |
| $Cs^{134}$ | 283 Bq/gU |
| $Cs^{137}/Ba^{137}$ | 778 Bq/gU |
| $Zr^{95}$ | 598 Bq/gU |
| $Nb^{95}$ | 574 Bq/gU |

The total concentration of elements that form non-volatile fluorides (including Al, Ba, Bi, Cd, Co, Cr, Cu, Fe, Pb, Li, Mg, Mn, Ni, K, Ag, Na, Sr, Th, Sn, Zn, and Zr) must not exceed 3.0E-03 g/gU.

The contents of other elements must not exceed the following concentrations in g/gU.

| | | | | |
|---|---|---|---|---|
| Sb<1 | As<3 | B<1 | Bi<5 | Cl<100 |
| Cr<10 | Nb<1 | P<50 | Ru<1 | Si<100 |
| Ta<1 | Ti<1 | Mo<1.4 | W<1.4 | V<1.4 |

Additionally, for reprocessed $UF_6$, the maximum total activity present in the package is limited to 957 mixture $A_2$ values.

(2) Maximum quantity of material per package

The package contents are limited to a maximum of 5,020 pounds of $UF_6$ enriched to not more than 5 wt%$U^{235}$. The maximum H/U atomic ratio for the $UF_6$ is 0.088.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 5.0

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

(2) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(3) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

7. The 30-inch diameter $UF_6$ cylinder must be fabricated, inspected, tested and maintained in accordance with American National Standard N14.1-1995 or an earlier version of ANSI N14.1 in effect at the time of fabrication. Cylinders must be fabricated in accordance with Section VIII, Division I, of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code and be ASME Code stamped.

8. The 30-inch diameter $UF_6$ cylinder valve stem and plug may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

9. The leak tightness of the 30B $UF_6$ cylinder shall be verified using a test having a sensitivity of at least $1 \times 10^{-3}$ std-cc/sec per ANSI Standard N14.5-1997 prior to loading into the ESP-30X overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration date: May 31, 2005.

## REFERENCES

ESP application dated June 19, 1998.

Supplements dated: August 27, 1999; March 22, May 12, and May 18, 2000; and April 11, 2002

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date July 16, 2002

NRC FORM 618
(3-98)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9285 | 1 | USA/9285/AF-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Global Nuclear Fuel - Americas, L.L.C.
P. O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

General Electric Company application dated August 4, 1998, as supplemented.

c. DOCKET NUMBER 71-9285

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: SRP-1

(2) Description

A steel drum for the transport of solid uranium contaminated residues. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is 825 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with General Electric Company Drawing No. 0025E98, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium-contaminated solid residues.

(2) Maximum quantity of material per package

775 pounds. The maximum uranium enrichment is 5.0 weight percent U-235. The maximum fissile mass is 104 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. In addition, the uranium may not exceed 0.05 weight percent U-234 and 0.025 weight percent U-236.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.6

Page 2 - Certificate No. 9285 - Revision No. 1 - Docket No. 71-9285

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application.

   (b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: October 31, 2003.

REFERENCES

General Electric Company application dated August 4, 1998.

Supplements dated: October 2, 1998; and October 14, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9287 | 0 | USA/9287/B(U)-85 | 1 | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Packaging Technology, Inc.,
4507-D Pacific Highway East
Tacoma, Washington 98424-2633

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Packaging Technology, Inc., application dated November 18, 1998, as supplemented.

c. DOCKET NUMBER 71-9287

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: SteriGenics Eagle

(2) Description

A stainless steel, lead shielded shipping cask for special form cobalt-60 sealed sources. The package consists of a cylindrical cask body with closure lid, and removable toroidal impact limiters, and a basket that carries and positions the cobalt-60 sealed source capsules. The packaging is constructed primarily of ASTM Type 304 stainless steel. The package is designed to transport up to 330,000 curies of cobalt-60.

The outside diameter of the cask body is approximately 37-11/16 inches. The diameter of the inner cavity is approximately 10-3/4 inches. The stainless steel inner shell has a minimum thickness of 1 inch and the stainless steel outer shell is 1 inch thick. The region between the two shells is filled with lead shielding. The closure lid and cask bottom end each consist of two stainless steel plates with lead between the two plates. The lead shielding thickness is approximately 10-3/8 inches on the side, 14-3/8 inches in the closure lid, and 11-7/8 inches on the cask bottom. The closure lid is secured by 12, 3/4-inch bolts. The closure lid is equipped with a Viton O-ring seal. The lid has a drain port and a vent port, and the cask body has a drain port. Each port is closed by a plug.

A double stainless steel thermal radiation shield is provided on the outside of the cask body in the region between the two impact limiters. The inner thermal shield is about 3/4-inches thick and is radially separated from the cask outer shell by 12 gauge spacers at each end. The outer shield is a sheet of 10 gauge material separated from the inner shield by a spiral wrap of 12 gauge wire.

The top and bottom impact limiters are toroidal stainless steel shells. They are attached to either end of the cask body using 12, 1-inch diameter ball-lock pins orientated radially around the cask body. One pin on each limiter is installed with a lockwire to provide a tamper-indicating device.

The cask lifting attachments thread into the upper cask body. The cask lid is also equipped with removable lid-lifting attachments. The cask rests on a steel pallet and is held down to the pallet by means of a steel frame placed on the top impact limiter. This steel frame is used to tie the cask to the conveyance. The maximum weight of the package, including contents is 20,000 lbs.

5(a)(2) cont.

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Cask Body Outer Diameter | 37-11/16 inches |
| Cask Body Height | 49-7/8 inches |
| Cask Cavity Inner Diameter | 10-3/4 inches |
| Cask Cavity Inner Height | 19 inches |
| Lead Shield Sidewall Thickness | 10-3/8 inches |
| Overall Package Dimension | |
| Diameter at Impact Limiters | 60 inches |
| Diameter at Body | 37-11/16 inches |
| Height with Impact Limiters | 76 inches |
| Maximum Contents Weight | 50 pounds |
| Maximum Package Weight (Including Contents) | 20,000 pounds |

(3) Drawings

The packaging is constructed and assembled in accordance with Packaging Technology, Incorporated, Drawing No. 98003-SAR, Rev.1, Sheets 1 through 8.

(b) Contents

(1) Type and form of material

Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package:

330,000 curies. Not to exceed 18,400 curies per special form source.

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

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented.

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: December 31, 2004.

REFERENCES

Packaging Technology, Inc., application dated November 18, 1998.

Supplement dated: August 20, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 2/2/00

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9288 | 4 | 71-9288 | USA/9288/AF-85 | 1 | OF | 4 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Columbiana Hi Tech Front End, LLC
200 West Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Eco-Pak Specialty Packaging application dated August 9, 1998, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Eco-Pak OP Uranium Oxide Transport Unit Package

(2) Description

A shipping container for uranium oxide pellets, powder, and uranium bearing materials. The package is roughly cubical and is approximately 45-inches x 45-inches x 62-inches high. The package has four internal cylinders in which storage vessels are inserted.

The outer shell of the package is constructed of 11 gauge mild steel and the space between the outer shell and the internal cylinders are filled with fire-retardant, closed cell, phenolic foam.

The internal cylinders are constructed of 11 gauge mild steel with an inner diameter of 10 1/4-inches. The cylinders are closed with twelve 1/2-inch bolts on a 1/2-inch thick 14-inch diameter inner silicone-gasketed carbon steel lid and a 1/2-inch thick 16 5/8-inch diameter outer silicone-gasketed carbon steel lid with the same number of bolts.

The inner storage vessel is a double walled container constructed of A-240, series 300, stainless steel with the annulus filled with fire retardant, closed-cell phenolic foam. The diameter of the outer annulus is 10-inches. The storage vessel has an inner diameter of either 7-inches or 7 3/4-inches. The storage vessel is closed by twelve 1/2-inch diameter bolts on a 5/8-inch thick stainless steel lid with a silicone gasket.

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Containment vessel inside diameter | 10 1/4-inches |
| Storage vessel outside diameter | 10-inches |
| Storage vessel inside diameter | 7-inches or 7 3/4-inches |
| Overall package dimensions | |
| width | 45-inches |
| length | 45-inches |
| height | 62-inches |
| Maximum contents weight ($UO_2$) | 1423 pounds |
| OP Transport unit | 1648 pounds |
| OP Storage vessels (4) | 686 pounds |
| Maximum package weight (including contents) | 3757 pounds |

(3) Drawings

The packaging is constructed and assembled in accordance with ESP Drawing Nos.:

OP-TU-SAR, Rev. 7, Sheet 1, OP-TU-SAR, Rev. 4, Sheets 2-4

5.(b) Contents

(1) Type and form of material

Uranium oxide pellets and powder. The contents may include up to 1000 grams of polyethylene per storage vessel (4000 grams per package), or other plastics, provided that the total water equivalent of the plastic is less than 1307 grams per storage vessel (5228 grams per package). In addition, the contents are limited to:

A. Uranium oxide powder enriched to no more than 4.5 weight percent in the U-235 isotope with a maximum of 356 pounds per 7-3/4-inch diameter storage vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 4, Rev. No. 4) and a maximum load of 1423 pounds per package .

B. Uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope with a maximum load of 356 pounds per 7-inch diameter storage vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 4) and a maximum load of 1423 pounds per package.

C. Uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope with a maximum load of 356 pounds per 7-inch diameter oxide vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 4) and a maximum load of 1423 pounds per package .

5.(b)(1)D. Uranium-bearing materials in the form of solids, or solidified or dewatered materials. Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm$^3$ (density of $UO_2$). Material such as U-metal, U-metal alloys, or uranium hydrites (e.g., UH) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Pellets or previously pelletized materials are not allowed for transport in the 7-3/4-inch diameter storage vessel. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of oils, deuterium, tritium and beryllium. Materials with a hydrogen density greater than water must be excluded. Uranium enriched to 5.0 weight percent in the U-235 isotope with a maximum load of 356 pounds per storage vessel and a maximum load of 1423 pounds per package can only be loaded into the 7-inch diameter oxide vessel shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 4. Uranium enriched to 4.5 weight percent in the U-235 isotope with a maximum load of 356 pounds per storage vessel and a maximum load of 1423 pounds per package can only be loaded into the 7-3/4-inch diameter oxide vessel shown in ESP Drawing No. OP-TU-SAR, Sheet 4, Rev. No. 4.

5.(b)(2) Maximum quantity of material per package

1423 pounds of $UO_2$ material.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 2.0

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: March 31, 2005.

REFERENCES

ESP application dated August 9, 1998.

Supplements dated: September 8 and October 29, 1999; and January 12, February 9, March 2, March 17, June 20, June 29, July 12, July 18, September 20, and November 9, 2000; and April 11, 2002

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date July 16, 2002

| NRC FORM 618 (8-2000) 10 CFR 71 | | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| **CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES** | | | | | |
| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9289 | 1 | 71-9289 | USA/9289/B(U)F-85 | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Framatome ANP, Inc.
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome Cogema Fuels application
dated January 8, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: WE-1

(2) Description

A fresh fuel assembly shipping container. The package consists of a cylindrical outer container and a rectangular inner container bolted to a strongback. The outer container is constructed of 11 gauge carbon steel and opens into two semi-cylindrical halves. The inner container is comprised of 1-inch thick carbon steel plates that are bolted together. The fuel assembly is surrounded by thermal insulation and secured inside the inner container with nine integral clamp frames. The inner container is secured to the strongback by bolts and clamp arms. Wood blocks surround the region between the inner container and the strongback. The strongback is supported by 14 shock mounts attached to the outer container. The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Inner container length | 165 inches |
| Inner container width (sq) | 16 ½ inches |
| Outer container length | 216 inches |
| Outer container diameter | 44 inches |
| Maximum content weight | 1610 pounds |
| Maximum package weight (including contents) | 9090 pounds |

5. (a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with the following Framatome Cogema Fuels Drawing Nos.:

1273964, Rev. 0
1273965, Rev. 1
1273966, Rev. 0
1273967, Rev. 0
1273968, Rev. 0

(b) Contents

(1) Type and form of material

A fuel assembly composed of uranium dioxide pellets within zircalloy cladding. The fuel assembly has the following specifications:

| | |
|---|---|
| Assembly type | BW 17x17 |
| No. fuel rods | 264 |
| No. non-fuel tubes | 25 |
| Nominal fuel rod pitch, in. | 0.496 |
| Maximum fuel pellet OD, in. | 0.3232 |
| Nominal clad OD, in. | 0.374 |
| Nominal clad thickness, in. | 0.022 |
| Nominal guide and instrument tube OD, in. | 0.48 |
| Nominal guide and instrument tube ID, in. | 0.452 |
| Nominal active fuel length, in. | 144 |
| Maximum uranium enrichment, weight percent U-235 | 4.6 |
| Maximum U-235 mass, kg | 22.14 |

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9289 | 1 | 71-9289 | USA/9289/B(U)F-85 | 3 OF | 3 |

5. (b) Contents (continued)

(2) Maximum quantity of material per package

One fuel assembly weighing no more than 9,090 pounds. The radioactive material may not exceed any of the following limits:

| | |
|---|---|
| U-232 | 0.01 microgram per gram of uranium |
| U-234 | 0.001 gram per gram of uranium |
| U-236 | 0.013 gram per gram of uranium |
| Tc-99 | 5 micrograms per gram of uranium |
| Fission Products | $4.4 \times 10^5$ MeV-Becquerel per kilogram of uranium |
| Np and Pu | 35 Becquerels per gram of uranium |

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 100

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

(a) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: February 29, 2004.

REFERENCES

Framatome Cogema Fuels application dated: January 8, 1999.

Framatome Cogema Fuels supplements dated: February 9, 11, and 25, 1999; November 13, 2000; and February 9, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: February 14, 2001

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9290 | 0 | 71-9290 | USA/9290/B(U)-85 | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

MDS Nordion
447 March Road
Kanata, Ontario
Canada K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated March 9, 2001, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. F-430/GC-40 Transport Package

(2) Description

The Model No. F-430/GC-40 Transport package is designed to transport MDS Nordion's Gammacell-40 (GC-40) irradiator containing cesium-137 sealed sources in special form. The F-430 overpack provides impact and thermal protection for the radioactive contents. Containment is provided by the special form sealed source and shielding is provided by the GC-40 irradiator body.

The F-430 is stainless steel cylindrical package with a 50" diameter and a height of 50" that is placed on a removable mild steel skid. The maximum weight of the package is 7000 pounds. The maximum weight of the GC-40 contents is 4290 pounds.

The overpack consists of three layers of 12 gauge stainless steel with polyurethane foam between them. The cavity between the outer and middle walls is filled with six inches of 8 lb/ft$^3$ foam for impact and thermal protection. Additionally, the cavity between the inner wall and the middle wall is filled with 40 lb/ft$^3$ foam. This foam is the fire shield that is designed to prevent direct flame exposure to the contents from the 30-minute thermal test. Plugged vent holes are provided between the walls to prevent pressure buildup and allow a pathway for escape of gases from foam during the thermal test.

The package contents consists of a Cesium-137 sealed source contained within an MDS Nordion GC-40 irradiator (upper or lower heads). The GC-40 is a research irradiator with lead shielding and a lead filled source drawer.

5.(a)(2) continued

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Package outside diameter | 50 inches |
| Package height | 50 inches |
| Cavity diameter | 36 inches |
| Cavity height | 35.25 inches |
| Removable skid | 50 inches x 50 inches x 8 inches (height) |
| Overpack weight | 2680 pounds |
| Contents weight | 4290 pounds |
| Maximum package weight | 7000 pounds |

(3) Drawings

The packaging is constructed in accordance with the MDS Nordion drawings F643001-001, Rev. B, Sheets 1-2.

(b) Contents

(1) Type and form of material

Cesium-137 as a sealed source which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

2,000 Curies.

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9290 | 0 | 71-9290 | USA/9290/B(U)-85 | 3 OF | 3 |

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: February 28, 2007.

REFERENCES

MDS Nordion application dated March 9, 2001.

Supplements dated: May 3 and October 18, 2001

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles J. Miller for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 3/5/01

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9291 | 3 | 71-9291 | USA/9291/B(U)F-85 | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Columbiana Hi Tech Front End, LLC
200 West Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Eco-Pak Specialty Packaging application dated November 30, 2000, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: Eco-Pak Liqui-Rad (LR) Transport Unit Package

(2) Description

The LR Package is designed to transport Type B quantities of fissile uranyl nitrate solutions. The package uses thermal and impact limiting systems to protect the containment vessel and prevent the contents from being released. The primary structural components of the LR packaging consist of a stainless steel containment vessel, a carbon steel outer vessel and a carbon steel framing system. The containment vessel is built in accordance with ASME Pressure Vessel Code (Section VII, Division 1) but does not require an ASME stamp. Double O-ring seals on the containment vessel's primary and secondary lids provide a leak tight seal which is leak testable. A closed-cell phenolic foam surrounds the top and bottom head area of the containment vessel and ceramic fiber blanket and board insulation are used in the sidewalls and outer lid for thermal insulation and impact absorption. The maximum volume of the contents is limited to 230 gallons which maintains a minimum ullage of 33 gallons.

The LR is a cylindrical package set in a rectangular angle frame. The dimensions of the package are approximately 56"(l) x 56"(w) x 73"(h). The maximum weight of the package is 5692 pounds. The outer vessel is constructed of 10 gauge carbon steel. The containment vessel is constructed of 1/4 inch stainless steel with 1/4 inch thick flanged and dished heads. The containment vessel is rated at 50 psig pressure. Closed-cell phenolic or polyurethane foam and ceramic fiber insulation are sandwiched between the containment vessel and the package's outer shell.

The package is designed to be leak-tight (maximum allowable leakrate of $1 \times 10^{-7}$ ref-cm$^3$/sec). The containment vessel is closed using a double O-ring and is secured by sixteen 5/8 inch stainless steel studs. The outer lid is closed with four 5/8 inch

5.(a)(2) Continued

stainless steel bolts and nuts. The package is also equipped with plastic plugs to vent any gases that may be generated by the insulation during a fire event. All valves and fittings are provided within sealed enclosures to contain any leakage during valve failure.

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos.:

LR-SAR, Sheet 1 of 4, Rev. 2, Sheet 2 of 4, Rev. 3, Sheet 3 of 4, Rev. 5 and Sheet 4 of 4, Rev. 6.

5.(b) Contents

(1) Type and form of material

Low enriched Uranyl Nitrate solutions with the specifications shown in Table 1 below. The uranium concentration must be less than or equal to 125 gU/liter with an enrichment less than or equal to 5.0 wt% U-235. Non-fissile chemical impurities may be present up to the chemical impurity specification in Table 1. Additionally, fissile isotopes are also limited to the quantities in Table 1.

(2) Maximum quantity of material per package

230 gallons of Uranyl Nitrate solution with limits as shown in table 1.

Table 1

| ITEM | SPECIFICATION |
|---|---|
| Solution Density | ≤1.17 g/cc |
| Chemical Impurities | ≤1500 µg/gU |
| Nitric Acid Normality | 0.1 - 0.7 |
| Uranium Concentration | ≤125 gU/l |
| U-232 | ≤2.0E-03 µg/gU |
| U-234 | ≤2.0E+03 µg/gU |
| U-235 | ≤0.05 g/gU (12 pounds maximum quantity of U-235 per LR) |
| U-236 | ≤2.5E+04 µg/gU |
| U-238 | remainder of uranium |
| Pu/Np Alpha Activity | ≤93 Bq/gU |
| Gamma Emitters | 0.515E-01 Ci |

5. (c) Transport Index for Criticality Control (Criticality Safety Index)

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

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: October 31, 2006.

REFERENCES

ESP application dated November 30, 2000.

Supplements dated: June 28, and August 30, 2001, and January 18, April 11 and June 19, 2002.

Columbiana Hi Tech Front End, LLC supplements dated: August 21, September 4, and November 8, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 12/04/02

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9292 | 2 | 71-9292 | USA/9292/AF-85 | 1 | OF | 4 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Combustion Engineering, Inc. application dated August 23, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PATRIOT

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

The metal inner container is approximately 11-1/4 inches high by 18-1/8 inches wide by 179-3/4 inches lon There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 30-1/4 inches wic by 31-3/4 inches high by 207-1/2 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,988 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with ABB CE Nuclear Power, Inc. Drawing N

L-9292-01, Sheets 1 and 2, Rev. 2, and
L-9292-02, Rev. 3.

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of $UO_2$ fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches. Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(i) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly; Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36; Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52; Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(iii) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80; Maximum U-235 enrichment is 4.0 weight percent for all corner rods; Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.4 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|---|---|---|
| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE / PAGES |
| 9292 | 2 | 71-9292 | USA/9292/AF-85 | 3 OF 4 |

5.(b)(2) Maximum quantity of material per package

Two fuel assemblies. The total weight of contents not to exceed 1,320 pounds.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into, or out of, the sheathed fuel assembly.

7. Polyethylene inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated.

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 1 CFR 71.12.

10. Expiration date: January 31, 2005.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9292 | 2 | 71-9292 | USA/9292/AF-85 | 4 OF | 4 |

## REFERENCES

ABB C-E Nuclear Power, Inc. application dated August 23, 1999.

Supplements dated: October 19 and December 10, 1999; and January 10, March 28, and April 4 and 12, 2000.

CE Nuclear Power, LLC supplement dated: September 14, 2000.

Westinghouse Electric Company LLC supplement dated: September 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date October 10, 2000

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9293 | 1 | 71-9293 | USA/9293/B(U)F-85 | 1 OF | 9 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
Four Skyline Drive
Hawthorne, New York 10532

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc., application dated May 19, 1999, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. TN-68 Transport Package

(2) Description

The TN-68 is predominantly a steel package that is used to transport up to 68 intact BWR fuel assemblies with or without channels. The overall dimensions of the package are 271 inches long and 144 inches in diameter with the impact limiters installed.

The package generally consists of four components, the fuel basket assembly, a containment vessel within a forged steel cask body, a radial neutron shield, and impact limiters.

The basket assembly locates and supports the fuel assemblies, transfers heat to the cask body wall and provides neutron absorption to satisfy sub-criticality requirements. The basket structure consists of an assembly of stainless steel cells, joined by fusion welding of 1.75 inch wide stainless steel plates. Above and below the plates are slotted borated aluminum (or boron carbide/aluminum) metal matrix composite neutron poison plates which form an egg-crate structure. This construction forms a honey-comb like structure of cell liners which provides compartments for 68 fuel assemblies. The nominal dimensions of each cell is 6.0 inches x 6.0 inches.

A thick-walled (6.0 inch), forged steel cask body for gamma shielding surrounds the containment vessel, by an independent shell and bottom plate of carbon steel. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The thickness of the bottom of the cask body is 8.25 inches. A 4.5 inch thick steel gamma shield is also welded to the inside of the containment lid.

5.(a)(2) continued

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plates, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment vessel length is approximately 189 inches with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 69.5 inches and a length of 178 inches. The containment lid is 5 inches thick and is fastened to the cask body with 48 bolts. Double metallic O-ring seals are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure. There are two penetrations through the containment vessel which are located in the lid. These penetrations are for draining and venting. Double metallic seals are also used on these two lid penetrations. The OP port provides access to the interspace lid seals for leak testing purposes. The OP transport cover is not part of the containment boundary.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

The package has impact limiters at each end of the cask body. The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells that maintain the wood in a dry atmosphere and provide wood confinement when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters are attaching to each other using 13 tie rods and to the cask by eight bolts attaching to brackets welded to the outer shell in eight locations (four bolting locations per impact limiter).

The approximate dimensions and weights of the package are as follows:

| | |
|---|---|
| Overall length (with impact limiters, in) | 271 |
| Overall length (without impact limiters, in) | 197 |
| Impact Limiter Outside diameter, (in) | 144 |
| Outside diameter (without impact limiters, in) | 98 |
| Cavity diameter (in) | 69.5 |
| Cavity length (in) | 178 |
| Containment shell thickness (in) | 1.5 |
| Containment vessel length (in) | 184 |
| Body wall thickness (in) | 7.5 |
| Containment lid thickness (in) | 5 |
| Overall lid thickness (in) | 9.5 |
| Bottom thickness (in) | 9.75 |
| Resin and aluminum box thickness (in) | 6 |
| Outer shell thickness (in) | 0.75 |
| Overall basket length (in) | 164 |
| Maximum weight of package (pounds) | 272,000 |
| Maximum weight of BWR fuel contents (pounds) | 47,900 |
| Maximum weight of impact limiters and attachments (pounds) | 32,000 |

5.(a)(3) Drawings

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

972-71-1, Revision 1
972-71-2, Revision 2
972-71-3, Revision 4
972-71-4, Revision 2
972-71-5, Revision 1
972-71-6, Revision 1
972-71-7, Revision 3
972-71-8, Revision 2
972-71-9, Revision 2
972-71-10, Revision 1
972-71-11, Revision 1
972-71-12, Revision 0
972-71-13, Revision 0
972-71-14, Revision 1



5.(b) Contents

(1) Type and form of material

Contents are limited to 68 unconsolidated intact irradiated GE BWR fuel assemblies with zircalloy cladding. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. Partial fuel assemblies (i.e. spent fuel assemblies from which fuel rods are missing), shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water equal to that displaced by the original rod(s).

Spent nuclear fuel may be transported with or without channels. Any fuel channel thickness up to 0.120 is acceptable on any of the fuel designs shown below. The maximum initial rod pressurization is 155 psig. The maximum fuel assembly length is 176.2 inches and the maximum fuel assembly width is 5.44 inches.

Permissible fuel assemblies are limited as stated in table 1 (fuel types may be C, D, or S lattice):

5.(b)(1) continued

**Table 1, Fuel characteristics**

| GE fuel generation | model | array | rod pitch | fuel rods | rod od | clad thick | pellet dia. | water rods | water rod od | water rod id | U content (MTU/ Assembly) | Max active fuel length |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 2A | 2a | 7x7 | 0.738 | 49 | 0.570 | 0.036 | 0.488 | 0 | x | x | 0.1977 | 144 |
| 2, 2B | 2 | 7x7 | 0.738 | 49 | 0.563 | 0.032 | 0.487 | 0 | x | x | 0.1977 | 144 |
| 3, 3A, 3B | 3 | 7x7 | 0.738 | 49 | 0.563 | 0.037 | 0.477 | 0 | x | x | 0.1896 | 144 |
| 4, 4A,4B | 4 | 8x8 | 0.640 | 63 | 0.493 | 0.034 | 0.416 | 1 | 0.493 | 0.425 | 0.1880 | 146 |
| 5 | 5 | 8x8 | 0.640 | 62 | 0.483 | 0.032 | 0.410 | 2 | 0.591 | 0.531 | 0.1876 | 150 |
| 6, 6B | 5 | 8x8 | 0.640 | 62 | 0.483 | 0.032 | 0.410 | 2 | 0.591 | 0.531 | 0.1876 | 150 |
| 7, 7B | 5 | 8x8 | 0.640 | 62 | 0.483 | 0.032 | 0.410 | 2 | 0.591 | 0.531 | 0.1876 | 150 |
| 8, 8B -2w | 82 | 8x8 | 0.640 | 62 | 0.483 | 0.032 | 0.411 | 2 | 0.591 | 0.531 | 0.1885 | 150 |
| 8, 8B-4W* | 84 | 8x8 | 0.640 | 60 | 0.483 | 0.032 | 0.411 | 4 | 0.591 | 0.531 | 0.1824 | 150 |
| 8, 8B-4W** | 84 | 8x8 | 0.640 | 60 | 0.483 | 0.032 | 0.411 | 4 | 0.483 | 0.431 | 0.1824 | 150 |
| 9, 9B | 9 | 8x8 | 0.640 | 60 | 0.483 | 0.032 | 0.411 | 1 | 1.34 | 1.26 | 0.1824 | 150 |
| 10 | 9 | 8x8 | 0.640 | 60 | 0.483 | 0.032 | 0.411 | 1 | 1.34 | 1.26 | 0.1824 | 150 |
| 11 | 11 | 9x9 | 0.566 | 74 | 0.440 | 0.028 | 0.376 | 2 | 0.98 | 0.92 | 0.1757 | 146 full, 90 partial |
| 13 | 11 | 9x9 | 0.566 | 74 | 0.440 | 0.028 | 0.376 | 2 | 0.98 | 0.92 | 0.1757 | 146 full, 90 partial |
| 12 | 12 | 10x10 | 0.510 | 92 | 0.404 | 0.026 | 0.345 | 2 | 0.98 | 0.92 | 0.1857 | 150 full, 93 partial |

*2 large water rods
**2 small water rods

Notes on table 1:

1. All dimensions in inches.
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inches thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt% U-235 average enrichment.
4. The fuel pitch is for C and D lattice designs. The S lattice fuels have a smaller pitch, which is less reactive.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

Provided all of the requirements of this section are met, the bounding fuel characteristics are: a) maximum initial lattice-average enrichment is 3.7%; b) the mnimum initial bundle average enrichment is 3.3%; c) the maximum assembly average burnup is 40,000

5.(b)(1) continued

MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies. The maximum burnup, minimum initial enrichments and cooling times for each of the three fuel assembly types is contained in the tables below.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment.

In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

**Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)**
**TYPE I BWR Fuel**

| | Burnup (GWd/MTU) | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Initial Enrichment (bundle ave %w) | 15 | 20 | 30 | 32 | 33 | 34 | 35 | 36 | 37 | 38 | 39 | 40 |
| 1.0 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.1 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.2 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.3 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.4 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.5 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | X | X | X | X | X |
| 1.6 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | X | X | X | X |
| 1.7 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | X | X | X |
| 1.8 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 11 | 12 | X | X |
| 1.9 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 11 | 12 | X | X |
| 2.0 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 | X |
| 2.1 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 | 12 |
| 2.2 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 | 12 |
| 2.3 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 11 | 12 | 12 |
| 2.4 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 |
| 2.5 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 |
| 2.6 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 | 12 |
| 2.7 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 11 | 12 |
| 2.8 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 12 |
| 2.9 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 11 | 12 |
| 3.0 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 12 |
| 3.1 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 | 12 |
| 3.2 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 11 |
| 3.3 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 |
| 3.4 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 |
| 3.5 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 |
| 3.6 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 |
| 3.7 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 | 10 |

**Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)**
**TYPE II BWR Fuel**

Burnup (GWd/MTU)

| Initial Enrichment (bundle ave %w) | 15 | 20 | 30 | 32 | 33 | 34 | 35 | 36 | 37 | 38 | 39 | 40 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1.0 | 18 | 21 | X | X | X | X | X | X | X | X | X | X |
| 1.1 | 17 | 20 | X | X | X | X | X | X | X | X | X | X |
| 1.2 | 17 | 20 | X | X | X | X | X | X | X | X | X | X |
| 1.3 | 17 | 20 | X | X | X | X | X | X | X | X | X | X |
| 1.4 | 17 | 20 | X | X | X | X | X | X | X | X | X | X |
| 1.5 | 16 | 19 | 25 | 26 | 26 | X | X | X | X | X | X | X |
| 1.6 | 16 | 19 | 25 | 26 | 26 | X | X | X | X | X | X | X |
| 1.7 | 16 | 19 | 25 | 25 | 26 | 26 | 27 | X | X | X | X | X |
| 1.8 | 16 | 19 | 24 | 25 | 26 | 26 | 27 | 27 | X | X | X | X |
| 1.9 | 16 | 19 | 24 | 25 | 25 | 26 | 27 | 27 | X | X | X | X |
| 2.0 | 16 | 18 | 24 | 25 | 25 | 26 | 26 | 27 | 28 | X | X | X |
| 2.1 | 15 | 18 | 23 | 25 | 25 | 26 | 26 | 27 | 27 | X | X | X |
| 2.2 | 15 | 18 | 23 | 25 | 25 | 25 | 26 | 27 | 27 | X | X | X |
| 2.3 | 15 | 18 | 23 | 24 | 25 | 25 | 26 | 26 | 27 | 27 | X | X |
| 2.4 | 15 | 18 | 22 | 24 | 24 | 25 | 26 | 26 | 27 | 27 | X | X |
| 2.5 | 15 | 17 | 22 | 24 | 24 | 25 | 25 | 26 | 26 | 27 | X | X |
| 2.6 | 15 | 17 | 22 | 24 | 24 | 24 | 25 | 26 | 26 | 27 | X | X |
| 2.7 | 15 | 17 | 22 | 24 | 24 | 24 | 25 | 26 | 26 | 26 | 27 | 27 |
| 2.8 | 14 | 17 | 22 | 23 | 24 | 24 | 25 | 25 | 26 | 26 | 27 | 27 |
| 2.9 | 14 | 17 | 22 | 23 | 23 | 24 | 24 | 25 | 26 | 26 | 27 | 27 |
| 3.0 | 14 | 17 | 21 | 23 | 23 | 23 | 24 | 25 | 25 | 26 | 27 | 27 |
| 3.1 | 14 | 17 | 21 | 23 | 23 | 23 | 24 | 25 | 25 | 26 | 27 | 27 |
| 3.2 | 13 | 16 | 21 | 23 | 23 | 23 | 24 | 24 | 25 | 25 | 26 | 27 |
| 3.3 | 13 | 16 | 21 | 23 | 22 | 23 | 23 | 24 | 25 | 25 | 26 | 26 |
| 3.4 | 13 | 16 | 21 | 23 | 22 | 23 | 23 | 24 | 25 | 25 | 26 | 26 |
| 3.5 | 13 | 16 | 21 | 22 | 22 | 23 | 23 | 24 | 25 | 25 | 26 | 26 |
| 3.6 | 13 | 16 | 21 | 21 | 22 | 22 | 23 | 24 | 25 | 25 | 26 | 26 |
| 3.7 | 12 | 15 | 20 | 21 | 22 | 22 | 23 | 24 | 25 | 25 | 25 | 26 |

**Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)**
**TYPE III BWR Fuel**

| | Burnup (GWd/MTU) | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Initial Enrichment (bundle ave %w) | 15 | 20 | 30 | 32 | 33 | 34 | 35 | 36 | 37 | 38 | 39 | 40 |
| 1.0 | 10 | 11 | X | X | X | X | X | X | X | X | X | X |
| 1.1 | 10 | 11 | X | X | X | X | X | X | X | X | X | X |
| 1.2 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.3 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.4 | 10 | 10 | X | X | X | X | X | X | X | X | X | X |
| 1.5 | 10 | 10 | 15 | 16 | 16 | 17 | 17 | X | X | X | X | X |
| 1.6 | 10 | 10 | 14 | 16 | 16 | 17 | 17 | 17 | X | X | X | X |
| 1.7 | 10 | 10 | 14 | 15 | 16 | 16 | 17 | 17 | 17 | X | X | X |
| 1.8 | 10 | 10 | 14 | 15 | 15 | 16 | 16 | 17 | 17 | 18 | X | X |
| 1.9 | 10 | 10 | 14 | 15 | 15 | 16 | 16 | 17 | 17 | 18 | X | X |
| 2.0 | 10 | 10 | 14 | 15 | 15 | 16 | 16 | 16 | 17 | 17 | 18 | X |
| 2.1 | 10 | 10 | 14 | 15 | 15 | 15 | 16 | 16 | 16 | 17 | 18 | 18 |
| 2.2 | 10 | 10 | 13 | 14 | 15 | 15 | 16 | 16 | 16 | 17 | 17 | 18 |
| 2.3 | 10 | 10 | 13 | 14 | 15 | 15 | 16 | 16 | 16 | 17 | 17 | 18 |
| 2.4 | 10 | 10 | 13 | 14 | 15 | 15 | 15 | 16 | 16 | 17 | 17 | 18 |
| 2.5 | 10 | 10 | 13 | 14 | 14 | 15 | 15 | 16 | 16 | 16 | 17 | 18 |
| 2.6 | 10 | 10 | 13 | 14 | 14 | 15 | 15 | 16 | 16 | 16 | 17 | 17 |
| 2.7 | 10 | 10 | 13 | 14 | 14 | 15 | 15 | 15 | 16 | 16 | 17 | 17 |
| 2.8 | 10 | 10 | 13 | 13 | 14 | 14 | 15 | 15 | 16 | 16 | 17 | 17 |
| 2.9 | 10 | 10 | 13 | 13 | 14 | 14 | 15 | 15 | 15 | 16 | 16 | 17 |
| 3.0 | 10 | 10 | 12 | 13 | 14 | 14 | 14 | 15 | 15 | 16 | 16 | 17 |
| 3.1 | 10 | 10 | 12 | 13 | 14 | 14 | 14 | 15 | 15 | 15 | 16 | 16 |
| 3.2 | 10 | 10 | 12 | 13 | 14 | 14 | 14 | 15 | 15 | 15 | 16 | 16 |
| 3.3 | 10 | 10 | 12 | 13 | 13 | 14 | 14 | 14 | 15 | 15 | 16 | 16 |
| 3.4 | 10 | 10 | 12 | 13 | 13 | 13 | 14 | 14 | 15 | 15 | 16 | 16 |
| 3.5 | 10 | 10 | 12 | 13 | 13 | 13 | 14 | 14 | 14 | 15 | 15 | 16 |
| 3.6 | 10 | 10 | 12 | 12 | 13 | 13 | 14 | 14 | 14 | 15 | 15 | 15 |
| 3.7 | 10 | 10 | 12 | 12 | 13 | 13 | 14 | 14 | 14 | 15 | 15 | 15 |

5.(b) continued

(2) Maximum quantity of material per package

The maximum contents weight is 75,600 pounds. The maximum weight of the irradiated fuel contents is 47,900 pounds.

(3) Decay Heat Limit

Maximum decay heat per package not to exceed 21.2kW. The maximum heat load per assembly is 0.313 kW/assembly.

(c) Transport Index for Criticality Control

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

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

(a) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

7. Known or suspected fuel assemblies with cladding defects greater than pin hole leaks and or hairline cracks are not authorized.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

9. Expiration date: February 28, 2006.

★ ★ ★ ★ ★

REFERENCES

Transnuclear, Inc., application dated May 19, 1999.

Supplements dated March 2, October 18, and November 13, 2000 and January 12, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 14, 2001

| NRC FORM 618 (6-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9294 | 2 | 71-9294 | USA/9294/AF-85 | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Global Nuclear Fuel - Americas, LLC
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NPC

(2) Description

A cubic stainless steel and foam outer packaging with nine cylindrical containment vessels for the transport of unirradiated, low-enriched uranium oxide powder. The overall package dimensions are approximately 45 inches wide, 45 inches deep, and 44 inches high.

The outer packaging consists of a 10-gage stainless steel outer shell with a ceramic fiber board liner and rigid polyurethane foam filler. The foam filler has a three-by-three array of vertical cylindrical cutouts that accommodate stainless steel sleeves for placement of the containment vessels. The outer packaging is equipped with a top cover that is secured to the outer packaging body by a combination of 16 closure cap screws and four closure strips secured by 24 bolts.

The containment vessel is a maximum 8.515 inches in inner diameter and approximately 32 inches in overall length. The containment vessel is constructed of 18-gage stainless steel, surrounded by a cadmium sheet and polyethylene wrap within a 24-gage stainless steel jacket. The containment vessel is closed by a 16-gage closure lid, a silicone rubber gasket, and a band clamp assembly, which is composed of a 0.063-inch thick strap and retainer, a T-bolt, and a nut.

The gross weight of the package (packaging and contents) is 1,293 kg (2,850 pounds). The maximum weight of the contents (uranium oxide powder) is 540 kg (1,190 pounds).

5.(a) (3) Drawings

The packaging is fabricated and assembled in accordance with the following Global Nuclear Fuel - Americas, LLC, Drawing Nos.:

0019D0001, Rev. 5
0019D0002, Rev. 5
0019D0003, Rev. 3
0019D0004, Rev. 5
0019D0005, Rev. 2
0019D0006, Rev. 3
0019D0007, Rev. 5
0019D0008, Rev. 2

(b) Contents

(1) Type and form of material

Uranium oxide powder. Uranium may be enriched to a maximum 5 weight percent in the U-235 isotope. The maximum moisture content of the uranium oxide powder is 5 weight percent.

(2) Maximum quantity of material per package

540 kg (1,190 pounds), with a maximum of 60 kg (132.2 pounds) per containment vessel.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 0.7

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: February 28, 2006.

## REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

Supplements dated: August 1, 23 and 27, 2001; and March 4, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges

M. Wayne Hodges Acting for
E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date March 8, 2002

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9296 | 2 | 71-9296 | USA/9296/B(U)-85 | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*
AEA Technology/QSA Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
AEA Technology/QSA Inc. application dated August 16, 2000, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No. 880 Series Packages

(2) Description

The Model No. 880 series packages are designed for use as a radiography exposure device and a transport package for Type B quantities of radioactive material in special form. The Model No. 880 has three versions called the 880 Delta, 880 Sigma and the 880 Elite. The 880 Delta has a maximum capacity of 150 Curies of Iridium-192, the 880 Sigma has a maximum capacity of 130 Curies of Iridium-192, and the 880 Elite has a maximum capacity of 50 Curies of Iridium-192. The Delta and Sigma versions are identical and the Elite has a lighter weight depleted uranium shield. An optional jacket can be placed on the packages when they are used as an industrial radiography exposure device or a transport package.

All versions of the package, without the jacket, are cylindrical in shape with a diameter of 5 inches and a length of 13 5/16 inches. With the jacket, the shape of the packages is an extruded triangle 9 inches high, 7 1/2 inches wide, and 13 5/16 inches long. The weight of the Delta and Sigma versions is 45 pounds (49 pounds with the jacket) and the elite version is 36 pounds (40 pounds with the jacket).

The major components of the packages consist of a welded stainless steel cylindrical body, a depleted uranium shield, a stainless steel rear plate with a locking assembly, a stainless steel front plate with a shielded port, and an optional jacket.

The welded cylindrical body consists of a five inch diameter, 0.06 inch wall tube shell with 0.12 inch end-plates. A U-bracket is welded to each end-plate and is located on the inside cavity of the shell tube. The depleted uranium shield is centrally located within the welded body between the end-plate and is fastened to each U-bracket by a 0.37 inch diameter titanium shield pin. A U-shaped copper spacer fills the gap

between the shield and the U-bracket. An S-shaped titanium source tube is casted into the center of the shield to provide a cavity for the source wire assembly to travel through during use.

The front and rear plates are attached to the welded body with four tamperproof screws through rivnuts assembled into end-plates. The rear plate assembly consists of a source locking mechanism fastened to the rear plate. The front plate assembly consists of a shielded port mechanism contained within the front plate.

An optional polyurethane jacket covers the package cylinder and provides a handle and a stable base. The jacket handle contains a wire molded in for additional reinforcement.

(1) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA, Inc., drawings R88000, Rev. D, Sheets 1-5.

(b) Contents

(1) Type and form of material

Iridium-192 as a sealed source which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

150 Curies (output) for the Model No. 880 Delta.

130 Curies (output) for the Model No. 880 Sigma.

50 Curies (output) for the Model No. 880 Elite.

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and source assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

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

   (a) The package must meet the Acceptance Test and Maintenance Program of Chapter 8.0 of the application, as supplemented; and

   (b) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration date: March 31, 2006.

REFERENCES

AEA Technology/QSA Inc., application dated August 16, 2000.

Supplements dated: December 6, 2000, January 10, 2001 and August 9, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 20, 2002

NRC FORM 618
(6-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9299 | 0 | 71-9299 | USA/9299/B(U)-85 | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

MDS Nordion
447 March Road
Kanata, Ontario, Canada K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated June 28, 2000, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: F-423

(2) Description

A double-walled welded stainless steel overpack for shipping sealed sources within the Gammacell 220 (GC220) gamma irradiator. The packaging consists of concentric box-like stainless steel shells separated by an annulus of rigid polyurethane foam. The overall overpack wall thickness is eight inches on the sides, twelve inches on the front and rear, and four inches on the base. The overpack lid is constructed of a sheet of 1/2-inch thick stainless steel on top, a sheet of 1/4-inch thick cold-rolled steel on the bottom, and 4-inches of polyurethane foam in between. The package is closed by bolting the lid to the body with 40 one-inch diameter bolts.

The GC220 irradiator is positioned inside the cavity formed by the inner stainless steel shell, along with an inner steel frame and a rigid polyurethane foam bonnet and lower crush pad. Shielding is provided by the GC220 irradiator, which is a welded steel lead-filled device. The GC220 is a lead-filled shielding head mounted on a steel stand. The GC220 shielding head consists of inner and outer steel shells with lead in between. The nominal lead thickness is 10 inches. The GC220 has an irregular shape, however, the base is 60-inches long by 40-inches wide. In its shipping configuration, the GC220 is 58-inches high. The GC220 shielding plug is welded from 304 stainless steel and lead filled. The GC220 drawer is welded from 304 stainless steel and is lead filled.

5(a) (2) (continued)

The maximum package weight (including contents) is 21,000 lbs (9,530 kgs). The approximate package component dimensions and weights are as follows:

| Component | Weight (lbs / kg) | Nominal Dimensions (L x W x H inches) |
|---|---|---|
| Overpack Lid | 1,046 / 474 | 67.50 x 55.00 x 4.75 |
| Inner Frame | 1,038 / 470 | 60.50 x 48.00 x 54.13 |
| Bonnet | 846 / 384 | 52.00 x 41.50 x 36.75 |
| GC220 | 8,575 / 3,890 | 60.00 x 40.00 x 58.00 |
| Overpack Body | 8,750 / 3,969 | 86.50 x 66.00 x 80.37 |
| Lower Crush Pad | 354 / 160 | 47.00 x 31.00 x 7.00 |

(3) Drawings

The packaging is constructed in accordance with MDS Nordion Drawing No. F642301-001:
Sheet 1, Revision D
Sheet 2, Revision C
Sheet 3, Revision B
Sheet 4, Revision A

(b) Contents

(1) Type and form of material

i. Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.
ii. Cobalt-60 as sealed sources described in Condition No. 6 below.

(2) Maximum quantity of material per package

26,000 curies, a maximum of 48 sources per package, and a maximum of 5,000 curies per source.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | | U.S. NUCLEAR REGULATORY COMMISSION | |
|---|---|---|---|---|---|
| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9299 | 0 | 71-9299 | USA/9299/B(U)-85 | 3 OF | 3 |

6. Sealed sources limited to MDS Nordion sealed source capsules manufactured before February 19, 1973: C-166, C-167, and C-185. In addition, these sources must meet the following:

   (a) Sources must conform to the specifications identified in the application in Figure 4.2 for the C-166 source, Figure 4.3 for the C-167 source, and Figure 4.4 for the C-185 source;

   (b) Sources must be shown to not be leaking within six months prior to shipment; and

   (c) Sources must not have been damaged during their service life.

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

9. Expiration date: August 31, 2006.

REFERENCES

MDS Nordion application dated June 28, 2000.

Supplements dated: May 11 and June 28, 2001

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date August 23, 2001

| NRC FORM 618 (6-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9300 | 0 | 71-9300 | USA/9300/B(U)-85 | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

BNFL, Inc.
10269 US 31 North
Charlevoix, MI 49720-9436

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BNFL, Inc., application dated June 19, 2001, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: BRP RVP SAR-5339

(2) Description

A reactor pressure vessel and internal components contained within a cylindrical steel outer packaging which is filled with low density concrete and welded closed. The reactor vessel is an open, cylindrical, 5-1/4-inch thick carbon steel vessel with a hemispherical bottom head. The inner diameter of the reactor vessel is approximately 8 feet, 10-7/8 inches, and the length of the reactor vessel is approximately 24 feet. The reactor vessel contains internal components, including the top guide plate, grid bar end pieces, steam baffle, emergency cooling sparger, seal housing, thermal shield, thermal shield retainer, seal weights, core support plate, inlet diffuser and inlet baffle. The internal components are constructed of stainless steel, and the internal surfaces of the reactor vessel are clad with stainless steel. The reactor vessel is partially surrounded with 3-inch thick stainless steel mirror insulation.

The outer packaging is composed of a 3-inch thick cylindrical shell welded to a 4-inch thick bottom plate, and is constructed of ASME SA-516, Grade 70, carbon steel. Supplemental steel shielding approximately 4 inches thick is welded to the inside of the cylindrical shell at the core region. The 4-inch thick top plate is attached to the reactor vessel flange by 14 studs and nuts, and is welded to the packaging outer shell. The reactor vessel cavity and the region between the reactor vessel and the outer packaging are filled with low density concrete. There are no lifting or tie-down devices that are a structural part of the package. The overall dimensions of the package are approximately 25 feet in length and 13 feet in diameter. The package weighs approximately 565,000 pounds.

5. (a) (3) Drawings

The package is constructed and assembled in accordance with BNFL, Inc., Figure 2-1, Sheets 1 through 4, Rev. 0; Sheet 5, Rev. 1; and Sheets 6 and 7, Rev. 0; included in the application dated June 19, 2001, as supplemented December 19, 2001.

(b) Contents

(1) Type and form of material

Irradiated steel reactor pressure vessel, mirror insulation, and internal components.

(2) Maximum quantity of material per package

Greater than a Type A quantity of radioactive material contained in an irradiated reactor vessel, mirror insulation, and internal components and associated surface contamination. Fissile material may be present provided the fissile material meets the exemption standards in 10 CFR 71.53.

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

(a) The package must be prepared for shipment and transported in accordance with Chapter 7 of the application, as supplemented.

(b) The package must be acceptance tested in accordance with Chapter 8 of the application, as supplemented.

7. Transport of the package may only be initiated if (1) the ambient temperature at the Big Rock Point plant is greater than 0°F, and (2) the minimum predicted temperature along the transport route for the anticipated transport period is greater than 0°F.

8. The package authorized by this certificate must be transported on a motor vehicle and on a railroad car assigned for the sole use of the shipper.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration date: March 31, 2007.

## REFERENCES

BNFL, Inc., application dated June 19, 2001.

Supplements dated: August 21 and December 19, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date April 8, 2002

| NRC FORM 618 (6-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9302 | 0 | 71-9302 | USA/9302/B(U)F-85 | 1 OF | 8 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear Inc.
Four Skyline Drive
Hawthorne, New York 10532

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear Inc., application dated May 2, 2001.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a. Packaging:

(1) Model No.: NUHOMS®-MP197

(2) Description:

The NUHOMS®-MP197 package consists of an outer cask, into which a NUHOMS®-61BT transportable dry shielded canister (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the cask body.

Cask

The NUHOMS®-MP197 transport cask is fabricated primarily of stainless steel. Non-stainless steel members include the cask lead shielding between the containment boundary inner shell and the structural shell, the o-ring seals, the neutron shield, and carbon steel closure bolts. The body of the cask consists of a 1.25 inch thick, 68 inch inside diameter, stainless steel inner (containment) shell and a 2.5 inch thick, 82 inch outside diameter stainless steel structural shell, without impact limiters, which sandwich the 3.25 inch thick cast lead shielding. The overall external dimensions of the cask are 208 inches long and 91.5 inches in outer diameter. The weight of cask body is 148,840 pounds, including about 10,000 pounds of neutron shield and 60,000 pounds of cast lead.

The containment system of the NUHOMS®-MP197 transportation cask consists of the inner shell, a 6.50 inch thick bottom plate, 2.5 inch thick RAM access closure with a diameter of approximately 24 inches, a top closure flange, a 4.5 inch thick top closure lid with closure bolts, drain port closures and bolts, and double o-ring seals for each penetration. The containment vessel prevents leakage of radioactive material from the cask cavity. The cask cavity is pressurized to above atmospheric pressure with an inert gas (helium). Helium assists in the heat removal. Shielding is provided by about 4 inches of stainless steel, 3.25 inches of lead, and about 4.5 inches of neutron shielding. Four removable trunnions are provided for handling and lifting of the cask.

Dry Shielded Canister (DSC)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The shell has an outside diameter of about 67 inches and an external length of about 200 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is given to the DSC as a containment boundary. The basket is designed to accommodate 61 intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The poison plates are constructed from borated aluminum, and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell is a laminate of balsa wood and redwood. Each impact limiter is attached to the cask top (front) and bottom (rear) by 12 bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two hoist rings for handling. The hoist rings are threaded into the impact limiter shell. During transportation, the impact limiter hoist rings are removed. An aluminum thermal shield is added to the bottom impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts, is approximately 28,000 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear Inc., drawing numbers:

1093-71-1, Revision 0,
NUHOMS®-MP197 Packaging
Transport Configuration

1093-71-2, Revision 1,
NUHOMS®-MP197 Packaging
General Arrangement

1093-71-3, Revision 1,
NUHOMS®-MP197 Packaging
Parts List

1093-71-4, Revision 1,
NUHOMS®-MP197 Packaging
Cask Body Assembly

1093-71-5, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-6, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-7, Revision 0,
NUHOMS®-MP197 Packaging
Lid Assembly & Details

1093-71-8, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Assembly

1093-71-9, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Details

1093-71-10, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Assembly

1093-71-11, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-12, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-13, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-14, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-15, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-16, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-17, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-18, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-20, Revision 0,
NUHOMS®-MP197 Packaging
Regulatory Plate

1093-71-21, Revision 0,
NUHOMS®-MP197 Packaging
on Transport Skids

5.b Contents of Packaging

(1) Type and Form of Material:

(a) Intact irradiated BWR fuel assemblies, with or without fuel channels, with uranium oxide pellets and zircaloy cladding. Channel thickness is limited to 0.065 to 0.120 inches. Prior to irradiation, the fuel assemblies must meet the dimensions and specifications of Table 1. Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the NUHOMS®-61BT DSC.

(b) The maximum burn-up and minimum cooling times for the individual assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(c). The maximum total allowable cask heat load is 15.86 kW.

(c) The maximum assembly decay heat of an individual assembly is 260 watts.

(d) BWR fuel assembly poison material shall meet the design requirements of Table 3.

**TABLE 1[1]**

| Assembly Type | 7x7 49/0 | 8x8 63/1 | 8x8 62/2 | 8x8 60/4 | 8x8 60/1 | 9x9 74/2 | 10x10 92/2 |
|---|---|---|---|---|---|---|---|
| Maximum Initial Enrichment (wt% $^{235}U$) | See Table 3 | See Table 3 | See Table 3 | See Table 3 | See Table 3 | See Table 3 | See Table 3 |
| Rod Pitch (in) | 0.738 | 0.640 | 0.640 | 0.640 | 0.640 | 0.566 | 0.510 |
| Number of Fuel Rods per Assembly | 49 | 63 | 62 | 60 | 60 | 66-full 8-partial | 78-full 14-partial |
| Fuel Rod OD (in) | 0.563 | 0.493 | 0.483 | 0.483 | 0.483 | 0.440 | 0.404 |
| Minimum Cladding Thickness (in) | 0.032 | 0.034 | 0.032 | 0.032 | 0.032 | 0.028 | 0.026 |
| Pellet Diameter | 0.487 | 0.416 | 0.410 | 0.410 | 0.411 | 0.376 | 0.345 |
| Maximum Active Fuel Length (in) | 144 | 146 | 150 | 150 | 150 | 146-full 90-partial | 150-full 93-partial |

Footnote to Table 1

1) Maximum Co-59 content in the Top End Fitting region is 4.5 gm per assembly
Maximum Co-59 content in the Plenum region is 0.9 gm per assembly
Maximum Co-59 content in the In-Core region (including the whole fuel channel) is 4.5 gm/assm
Maximum Co-59 content in the Bottom region is 4.1 gm per assembly

## TABLE 2

| Intact BWR Fuel Assembly Characteristics | |
|---|---|
| **Physical Parameters:** | |
| Fuel Design | 7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel |
| Cladding Material | Zircaloy |
| Fuel Damage | Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR fuel" |
| Channels | Fuel may be stored with or without fuel channels |
| Maximum assembly weight | 705lbs |
| **Radiological Parameters:** | |
| Group 1: | |
| Maximum Burnup: | 27,000 MWd/MTU |
| Minimum Cooling Time: | 6-Years |
| Maximum Initial Enrichment: | See Table 3 |
| Minimum Initial Bundle Average Enrichment: | 2.0 wt. % U-235 |
| Maximum Initial Uranium Content: | 198 kg/assembly |
| Maximum Decay Heat: | 260 W/assembly |
| Group 2: | |
| Maximum Burnup: | 35,000 MWd/MTU |
| Minimum Cooling Time: | 12-Years |
| Maximum Initial Enrichment: | See Table 3 |
| Minimum Initial Bundle Average Enrichment: | 2.65 wt. % U-235 |
| Maximum Initial Uranium Content: | 198 kg/assembly |
| Maximum Decay Heat: | 260 W/assembly |

| Intact BWR Fuel Assembly Characteristics | |
|---|---|
| **Radiological Parameters:** | |
| Group 3: | |
| Maximum Burnup: | 37,200 MWd/MTU |
| Minimum Cooling Time: | 12-Years |
| Maximum Initial Enrichment: | See Table 3 |
| Minimum Initial Bundle Average Enrichment: | 3.38 wt. % U-235 |
| Maximum Initial Uranium Content: | 198 kg/assembly |
| Maximum Decay Heat: | 260 W/assembly |
| Group 4: | |
| Maximum Burnup: | 40,000 MWd/MTU |
| Minimum Cooling Time: | 15-Years |
| Maximum Initial Enrichment: | See Table 3 |
| Minimum Initial Bundle Average Enrichment: | 3.4 wt. % U-235 |
| Maximum Initial Uranium Content: | 198 kg/assembly |
| Maximum Decay Heat: | 260 W/assembly |

**TABLE 3**

Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment

| NUHOMS® -61BT DSC Basket Type | Maximum Fuel Assembly Lattice Average Enrichment(wt.% U-235) | Minimum Boron-10 Areal Density for Boral® ($g/cm^2$) | Minimum Boron-10 Areal Density for Borated Aluminum, Metamic®, and Boralyn® ($g/cm^2$) | Areal Density Used in the Criticality Evaluation [75% Credit for Boral®] ($g/cm^2$) |
|---|---|---|---|---|
| | Intact Fuel Assemblies | | | |
| A | 3.7 | 0.025 | 0.021 | 0.019 |
| B | 4.1 | 0.038 | 0.032 | 0.029 |
| C | 4.4 | 0.048 | 0.040 | 0.036 |

(2) Maximum quantity of material per package:

(a) The quantity of material authorized for transport is 61 intact standard BWR fuel assemblies with or without fuel channels. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For material described in 5.b(1) the approximate maximum payload is 21,500 lbs.

5.c. Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: "0"

6. Fuel assemblies with missing fuel rods shall not be shipped unless the missing fuel rods are replaced by dummy rods that displace an equal or greater amount of water.

7. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

a. Each package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented. In addition this will include:

(1) verification of the basket type A, B, or C, by inspection of the last digit of the serial number on the grapple ring at the bottom of the DSC,

(2) verification that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Tables 2 and 3. The enrichment limit must correspond to the basket type determined in 7.a.(1) above.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented. In addition this will include replacement of the cask lid bolts after 85, or fewer, round trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during normal conditions of transport.

8. This package is approved for exclusive use by rail, truck, or marine transport.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration Date: July 31, 2007.

REFERENCES


Transnuclear Inc., Safety Analysis Report for the NUHOMS®-MP197 Transport Packaging, dated May 2, 2001.

Transnuclear Inc., letters dated November 20, 2001, January 29, 2002, January 31, 2002, March 1, 2002, March 20, 2002, April 29, 2002, and May 16, 2002.


FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9511 | 3 | 71-9511 | USA/9511/B(U) | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20545

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

U. S. Department of Energy
application dated February 26, 1991,
as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No: BUSS R-1

(2) Description

The packaging is a cylindrical, forged stainless steel cask. The cask body is a one-piece forging, 54.25 inches OD by 49 inches high. The cask cavity is 20.25 inches in diameter by 23 inches high. A solid, stainless steel basket, 19.95 inches in diameter by 22.83 inches high, sits in the cask cavity. The basket has either four, six, twelve, or sixteen 2.875-inch diameter holes that serve as receptacles for the source capsules. Eleven 4-inch high, circumferential fins surround the cask body exterior. A covered vent port and a covered drain port are located on the side of the cask body. The cask lid is a one-piece forging, 28.78 inches in diameter by 12.84 inches thick. Twelve 1.5-inch diameter bolts fasten the cask lid to the cask body through a 3.8-inch thick flange. The cask lid and port covers each have concentric, double O-rings. The inner O-ring is metallic and retains the helium coolant which fills the cask cavity. The outer O-ring is elastomeric and provides an annular test volume for leak testing the metallic O-ring. The cask has an impact limiter on each end. The impact limiter is polyurethane foam in a stainless steel shroud.

The overall dimensions of the packaging with impact limiters are 84.7 inches in diameter by 107 inches high. The maximum total weight of the contents is 400 pounds. The maximum weight of the package, including contents, is 30,000 pounds. The shipping skid and personnel barrier, which are not part of the package, weigh an additional 3,700 pounds.

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with the following drawings:

| Drawing No. | Title |
|---|---|
| S54773, Sht. 1, Rev. B | Cask with Impact Limiters |
| S48981, Sht. 1, Rev. H | Cask Assembly |
| T73684, Sht. 1, Rev. N, and Sht. 2, Rev. M | Body, Cask, 304 (BUSS) |
| R44382, Sht. 1, Rev. B | Alternate Detail N for Upper Port of Unit 1, Heat No. 82V65-1-1 |
| T73693, Sht. 1, Rev. M | Cask Lid (BUSS) 304 SST |
| S66574, Sht. 1, Rev. B | Bolt, Tension, 12 Point External Wrenching , Flanged |
| T99946, Sht. 1, Rev. E | Seal, Helicoflex, Cask Lid (BUSS) |
| T73685, Sht. 1, Rev. E | Plug, Drain (BUSS) |
| T99945, Sht. 1, Rev. D | Seal, Helicoflex, Drain Plug (BUSS) |
| R44676, Sht. 1, Rev. A | Bore Plug Assembly, Cask Body |
| R43728, Sht. 1, Rev. A | Bore Plug, Cask Body |
| S48929, Shts. 1 and 2, Rev. G, Sht. 3, Rev. E | Impact Limiter BUSS |
| R44381, Sht. 1, Rev. B, and Sht. 2, Rev. A | Impact Limiter BUSS, Non Conformance |
| S50032, Sht. 1, Rev. D | Cradle, BUSS Cask |
| S52606, Shts. 1 and 2, Rev. C | Pallet |
| S52608, Sht. 1, Rev. C | Block, Mounting |
| S50052, Sht. 1, Rev. F | Basket, Cask Body, 4 Hole (BUSS) |
| S50053, Sht. 1, Rev. E | Basket, Cask Body, 6 Hole (BUSS) |
| S50054, Sht. 1, Rev. D | Basket, Cask Body, 12 Hole (BUSS) |
| S50055, Sht. 1, Rev. E | Basket, Cask Body, 16 Hole (BUSS) |

5.(b) Contents

(1) Type and form material.

Melt-cast cesium chloride (CsCl) or pressed-filled strontium fluoride ($SrF_2$) capsules meeting the requirements of special form radioactive material. The capsules are as described in supplement dated February 28, 1992.

5.(b) Contents (continued)

(2) Maximum quantity of material per package.

| Basket Type | Capsule Type | Maximum Capsule Thermal Power (Watts) | Maximum Cask Thermal Power (Watts) | Maximum Cask Activity (million Ci) |
|---|---|---|---|---|
| 16-Hole | CsCl | 250 | 4000 | 0.85 |
| 12-Hole | CsCl | 333 | 4000 | 0.85 |
| 6-Hole | $SrF_2$ | 650 | 3900 | 0.65 |
| 4-Hole | $SrF_2$ | 850 | 3400 | 0.56 |

6. For shipments of CsCl capsules, the shipment period must be completed within thirty (30) days following the placement of the cask lid on the cask body.

7. The lifting lugs must not be used as tie-downs, and the lifting lug holes must be plugged or covered during transit.

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

(a) The package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

(b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application, as supplemented.

9. The package authorized by this certificate must be transported on a conveyance assigned for the exclusive use of this package.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

11. Expiration date: July 31, 2007.

## REFERENCES

U. S. Department of Energy application dated February 26, 1991.

Supplements dated February 28, 1992; May 6, 1994; August 26, 1994; July 29, 1997; and June 7, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material
Safety and Safeguards

Date: 07/26/02

| NRC FORM 618 (8-2000) 10 CFR 71 | | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | | | | |
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9786 | 5 | 71-9786 | USA/9786/B(U) | 1 OF | 5 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

S3G Core Basket Disposal Container
Safety Analysis Report for Packaging
dated June 1980, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a) Packaging

(l) Model No.: S3G Core Basket Disposal Container Assembly

(2) Description

The package consists of either one irradiated S3G, S1C or S7G core basket packaged in an inner, lead-filled container (S5W Core Basket Removal Container (CBRC)) which is placed inside an outer container (S3G Core Basket Disposal Container (CBDC)). The package weighs approximately 172,000 pounds.

The S3G CBDC is a 4-inch thick steel cylinder, 89 inches in outside diameter, 131 inches long, with an 8-inch thick top end plate and a 5-inch thick bottom end plate. Both end plates are welded to the cylinder with full penetration welds.

The S5W CBRC, which will be disposed of along with the outer S3G CBDC and inner core basket, is basically a cylindrical shaped container comprised of lead shielding sandwiched between two 304 stainless steel shells. The 1-inch thick inner shell is 60 inches O.D. and 107.5 inches long. The outer shell is made up of two geometries, a 72.5-inch O.D., 0.5-inch thick cylindrical shell that measures 66 inches long and joins a truncated conical shell which has a 64-inch O.D. at the small end. The two shells are joined by a full thickness penetration weld and a weld backup strap on the inside shell surface. Full penetration welds are also made on both ends of the shells to the top canning and shield ring.

The S5W CBRC will contain either an S3G, S1C or S7G core basket. The irradiated S3G core basket is an Inconel 600 cylindrical shell. Three, 3-inch thick 304 stainless steel plates are positioned in the core basket prior to removal to provide overhead radiation shielding. The lower plate is 46.2 inches in diameter. The upper plates have the same diameter but contain six extensions that fit inside recessed cutouts within the core basket. The total core basket weight is approximately 9,650 pounds.

5.(a)(2) Description (continued)

The S1C core basket is a 304 stainless steel cylindrical shell positioned inside a 304 stainless steel thermal shield. The overhead shielding consists of a set of 2-inch thick 304 stainless steel plates attached to the S1C core basket to provide radiation shielding during handling. The core basket weight is approximately 8,523 pounds.

The S7G core basket is an Inconel 600 cylindrical shell. A 304 stainless steel laminated plate (8-inches thick) with lifting attachments is attached to the top of the S7G core basket to provide radiation shielding during handling. The core basket weight is approximately 8,873 pounds.

The package may alternatively consist of S8G irradiated components positioned within an irradiated components discharge rack (ICDR) which is placed in an S3G CBDC. The ICDR is a steel rack approximately 128 inches high and 80 inches in diameter, and is designed to fit inside the S3G CBDC. The ICDR consists of a center cylinder assembly surrounded by 23 storage tubes, a top plate and a cylinder support base. The center cylinder is HY-80 steel, has a 36-inch outer diameter and a 4.5-inch wall thickness, and is 117 inches high. There are 9 storage tubes positioned inside the center cylinder. The total weight of the irradiated components, the ICDR, and the S3G CBDC is approximately 125,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Bettis Drawing No. 1527E40 for the S3G Core Basket Assembly and KAPL Drawing No. 152D7009 for the S1C Core Basket Assembly and KAPL Drawing No. 232B4874 for the S7G Core Basket Assembly and KAPL Drawing No. 978E644 for the S8G Irradiated Components.

(b) Contents

(1) Type and form of material

(i) An irradiated core basket either the S3G, S1C or S7G and S5W CBRC. The shipment may include surface contamination in the form of activated corrosion products and for the S3G core basket approximately 8 gallons of residual water.

(ii) S8G irradiated components within an ICDR. The shipment may include surface contamination in the form of activated corrosion products.

(2) Quantity of material per package

(i) Item 5(b)(1)(i) above:

One irradiated core basket and S5W CBRC as described in 5(b)(1). Surface contamination not to exceed 20.6 curies for the S3G core basket, 7.45 curies for the S1C core basket or 1.2 curies for the S7G core basket. The activation level of the irradiated S3G core basket is not to exceed 131,000 curies; the irradiated S1C core basket not to exceed 20,000 curies; and the activation level of the irradiated S7G core basket is not to exceed 140,000 curies.

(ii) Item 5(b)(1)(ii) above:

Irradiated components, including 141 instrument lines, 18 lower control drive mechanism assemblies, 4 filled sleeves, and 1 instrumentation stalk. Surface contamination not to exceed 65.5 curies. Activation level of the irradiated components not to exceed 2,440 curies.

6. Shipment of an irradiated S3G core basket must be made no earlier than 75 days after reactor shutdown.

7. Shipment of an irradiated S1C core basket must be made no earlier than 60 days after reactor shutdown.

8. Shipment of an irradiated S7G core basket must be made no earlier than 180 days after reactor shutdown.

9. Shipment of S8G irradiated components must be made no earlier than 100 days after reactor shutdown.

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

(a) Each packaging must meet the following Acceptance Tests and Maintenance Program:

S3G Core Basket

Section 8.0 of application dated June 1980

S1C Core Basket

Section 8.0 of application dated August 1983

S7G Core Basket

Section 8.0 of application dated May 1987

S8G Irradiated Components

Section 8.0 of application dated September 1991

(b) The package shall be prepared for shipment and operated in accordance with the following operating procedures:

S3G Core Basket

Section 7.0 of application dated June 1980

S1C Core Basket

Section 7.0 of application dated August 1983

S7G Core Basket

Section 7.0 of application dated May 1987

S8G Irradiated Components

Section 7.0 of application dated September 1991

11. Expiration date: August 31, 2006.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9786 | 5 | 71-9786 | USA/9786/B(U) | 5 OF | 5 |

## REFERENCES

S3G Core Basket Disposal Container Safety Analysis Report for Packaging, WAPD-REO(C)-122, dated June 1980, as revised (Revision 2, dated May 5, 1986).

Safety Analysis Report for Packaging an S1C Core Basket-Thermal Shield Assembly in the S3G Core Basket Disposal Container, S1C CB-TS, dated August 1983.

S7G Core Basket in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, dated May 1987.

S8G Irradiated Components in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, Revision 2, dated September 1991.

DOE memorandums G#7627 dated November 16, 1983; G#C86-3736 dated May 24, 1986; G#C86-3750 dated July 15, 1986; G#87-5663 dated July 7, 1987; G#91-10937 dated July 31, 1991; G#C91-11007 dated September 18, 1991; G#96-03335 dated February 16, 1996; and G#01-03414 dated January 31, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material
Safety and Safeguards

Date: March 29, 2001

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9787 | 4 | USA/9787/B(U)F | 1 | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

A1W-3 Power Unit Shipping Container
Safety Analysis Report for Packaging
dated August 1980, as supplemented.

c. DOCKET NUMBER 71-9787

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: A1W-3 Power Unit Shipping Container (PUSC)

(2) Description

The package is a right circular cylindrical steel weldment. A module support device, a puncture protection cover, and an energy absorber are attached to the top end of the package. The module support device is secured to the power unit adapter flange with two 3-inch diameter shipping studs and is bolted to the upper clamp and cradle assembly with forty 2.25-inch diameter assembly studs. Module support cylinders (with locking nuts) and control rod holddown devices prevent fuel module or control rod motion. The top puncture protection cover, with 4-inch thick steel side walls and a 5-inch thick steel top plate, fits down over the eggcrate assembly and attaches to the bottom plate of the model support device with forty 2-inch diameter bolts. The top energy absorber, with an 80-inch outer diameter, a 54-inch inner diameter, and a height of 25 inches is welded to the top puncture protection cover.

At the bottom end of the A1W-3 PUSC, a 4-inch thick steel puncture protection plate, with a diameter of 50.5 inches, is attached to the bottom of the power unit thermal shield by eight sets of U-bolts. A bottom energy absorber, with an 89.25-inch outer diameter, a 52.5-inch inner diameter, and a height of 36 inches covers the bottom puncture protection plate and is attached to the thermal shield of the power unit with twelve 0.875-inch diameter bolts. The support barrel, which is part of the power unit assembly, and a stainless steel plug/band assembly, which extends around the outside of the support barrel, provide puncture protection for the sides of the power unit.

5. (a)(2) Description (continued)

For shipment, the loaded A1W-3 PUSC is secured horizontally in a shipping pedestal. The shipping pedestal consists of a base on which two support beams are mounted horizontally with rubber shock mounts. The base is bolted to the deck of a 300-ton railcar and to four stop blocks which are welded to the railcar deck. The A1W-3 PUSC is secured to the shipping pedestal using the upper clamp and cradle assembly and the center and lower yoke/saddle assemblies. The yokes span across the upper side of the A1W-3 PUSC and are attached to the support beams while the saddles suspend from the yokes and support the weight of the loaded A1W-3 PUSC. A 0.25-inch thick railcar cover is used to enclose the entire A1W-3 PUSC for shipment.

The loaded A1W-3 PUSC (excluding the shipping pedestal) is 349 inches long, has a maximum diameter of 134 inches, and weighs approximately 297,000 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application documents (see References, below).

(b) Contents

One unirradiated A1W-3 power unit assembly.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: 100

6. Control rods and control rod holddown devices must be installed in the power unit as described in the application.

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

8. Expiration date: January 31, 2005

Page 3 - Certificate No. 9787 - Revision No. 4 - Docket No. 71-9787

## REFERENCES

A1W-3 Power Unit Shipping Container Safety Analysis Report for Packaging, and WAPD-REO(c)-118, dated August 1980.

Supplements: A1W-3 Power Unit Shipping Container Modified Top Energy Absorber Revised Safety Analysis Report for Packaging, Attachment 1 to WAPD-REO(c)-118, dated February 1985, Naval Reactors Memorandum G#94-03572 dated November 4, 1994, and November 10, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 1/19/00

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9788 | 12 | USA/9788/B(U)-85 | 1 | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Deactivated S5W Reactor Compartment Safety Analysis Report for packaging dated July 1981, as supplemented.

c. DOCKET NUMBER 71-9788

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: S5W and S6G Reactor Compartment

(2) Description

The package consists of a deactivated and defueled S5W or S6G Reactor Compartment which has been separated from the remainder of the submarine hull and prepared for shipment by sealing all openings and attaching handling fixtures. For each package model, the reactor compartment itself is between two containment bulkheads which are added to the package before shipping. The ship's hull and the containment bulkheads define the package containment boundaries. The containment bulkheads are either installed at the ends of the package or recessed. The strength of all package boundary closures is at least equivalent to the strength of the bulkheads. The deactivated reactor plant remains in place within the reactor compartment during shipment. The plant is defueled and drained except for small inaccessible pockets of liquid, primarily water. Potentially radioactively contaminated components and piping from other locations in the ship may be placed within the package and secured.

The S5W Reactor Compartment package is between 35 and 45 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The forward containment bulkhead may include existing ship structure which has been sealed to form a watertight bulkhead. The hull is constructed of HY-80 steel. The maximum weight of the S5W package is 2,160,000 pounds for the 598 and 585 classes and is 2,262,400 pounds for all other classes.

The S6G Reactor Compartment package is approximately 43 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may installed at the ends of the package or may be recessed. The hull is constructed of HY-80 steel. The maximum weight of the package is 3,360,000 pounds.

Page 2 - Certificate No. 9788 - Revision No. 12 - Docket No. 71-9788

5.(a) (3) Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application, as supplemented (see References, below).

(b) Contents

Activated structural components associated with the S5W and S6G reactor vessel complex, plant piping, ion exchanger resin, purification filter media which may be solidified (S6G only), residual liquid and other miscellaneous components and materials contaminated with radioactive corrosion products (crud).

Ion exchanger resins with up to 3.1 curies of Co-60 may be shipped in the S5W package. Ion exchanger resins and/or purification filter media with up to a combined total of 4.5 curies of Co-60 may be shipped in the S6G package.

6. Residual liquids contained within plant systems must be removed prior to transport to the maximum extent practical, in accordance with established procedures, methods, and controls, as described in submittal dated April 5, 1996, or in the Safety Analysis Report for the individual submarine class reactor compartment packages. Not more than 660 gallons of residual liquids remain in an S5W package, and not more than 1,200 gallons of residual liquids remain in an S6G package.

7. For packages with recessed containment bulkheads, the aft containment bulkheads and stiffeners, horizontal divider plate, and any structure between the pressure hull and the outer non-pressure hull must be recessed at least 7 inches from the aft end of the S5W package. The forward containment bulkhead and stiffeners, existing tank stiffeners, deck structure, and horizontal girders must be recessed at least 15 inches from the forward end of the S5W package. For S6G package with recessed containment bulkheads, both the aft and forward containment bulkheads, stiffeners and horizontal girders must be recessed at least 15 inches from the end of the package.

8. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the normal daily minimum temperature expected during the shipment of the package as determined on the basis of weather forecasts.

9. Ion exchanger resin with up to 3.1 curies ($1.1 \times 10^{11}$ becquerels) of Co-60 may be shipped in the S5W package. Shipment of the S5W packages shall not occur before 180 days after final reactor shutdown. Ion exchanger resin and purification media with up to a combined total of 4.5 curies ($1.6 \times 10^{11}$ becquerels) of Co-60 may be shipped in the S6G package provided ion exchanger resin does not exceed 3.5 curies ($1.3 \times 10^{11}$ becquerels) of Co-60 and purification media does not exceed 3.3 curies ($1.2 \times 10^{11}$ becquerels) of Co-60. Shipment of the S6G packages shall not occur before 365 days after final reactor shutdown.

10. Additional shielding may be provided on the exterior of the package by steel plates securely welded to the package surface so as to remain in place under the hypothetical accident conditions in 10 CFR Part 71.

Page 3 - Certificate No. 9788 - Revision No. 12 - Docket No. 71-9788

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

    (a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures", of the application.

    (b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program", of the application.

12. Expiration date: September 30, 2003.

REFERENCES

Deactivated S5W Reactor Compartment Safety Analysis Report for Packaging, WAPD-REO(C)-250, dated July 1981.

Supplements: Naval Reactors Memorandum Z#C90-14416 dated March 29, 1990, supplement dated July 6, 1990;
Naval Reactors Memorandum Z#C90-14456 dated August 30, 1990;
Naval Reactors Memorandum Z#C92-14438 dated August 3, 1992;
Naval Reactors Memorandum Z#C93-00069 dated October 14, 1993;
Naval Reactors Memorandum Z#C95-00113 dated March 16, 1995;
Naval Reactors Memorandum Z#96-14430 dated April 5, 1996;
Naval Reactors Memorandum Z#96-14434 dated April 10, 1996;
Naval Reactors Memorandum Z#C95-00191 dated December 14, 1995;
Naval Reactors Memorandum Z#96-14457 dated June 20, 1996;
Naval Reactors Memorandum Z#C96-14520 dated November 22, 1996;
Naval Reactors Memorandum Z#C96-14549 dated December 19, 1996;
Naval Reactors Memorandum Z#C97-14698 dated October 31, 1997; and
Naval Reactors Memorandum Z#C98-00021 dated February 27, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 09/25/98

NRC FORM 618
(R-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9791 | 6 | 71-9791 | USA/9791/B(U)-85 | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging dated January 1982, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: PWR-2 Lower Core Barrel Shipping and Disposal Container

(2) Description

The PWR-2 Lower Core Barrel Shipping and Disposal Container package consists of an inner burial container and a reusable outer container. The inner container is loaded with a D1G prototype pressure vessel assembly. The package weighs approximately 400,000 pounds.

The outer container is a 4-inch thick steel cylinder, 127 inches in outside diameter, 212 inches long, with two 6-inch thick end plates. The bottom end plate is welded to the cylinder with a full penetration weld and the top end plate is bolted with 107, 2-inch diameter fasteners.

The package is equipped with two 2.5-inch thick by 10-inch long circumferential impact limiter rings on the side, two concentric impact limiter rings on the ends, and aluminum honeycomb crush blocks in the top and bottom spaces between the inner and outer containers.

The container is supported horizontally on the railroad car by eight gussets attached to two horizontal plates. Each plate is bolted to the top flange of an I-beam. The bottom flange of the I-beam is bolted to a 300-ton railroad car.

5. (a) Packaging (continued)

The inner disposal container (liner) is of the following design:

The D1G prototype pressure vessel assembly has an inner burial container that consists of two cylinders constructed of HY-80 steel connected by a transition ring that is welded to the two cylinders. The maximum outer diameter of the cylinder is approximately 118 inches at the upper flange. The overall length of the inner container is 184.5 inches. The container wall is 3.12 inches in the upper cylinder and 4 inches in the bottom cylinder. The bottom plate varies in thickness from 6 to 2.4 inches and is attached to the container by 12, 4.5-inch thick gussets. The cover plate is approximately 10 inch thick and is attached to the container by a 3.25-inch thick closure weld. The container is axially positioned within the outer container by aluminum honeycomb energy absorbers.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Drawing Nos. 1575E12, 1574E96, 6236E43, Sheets 1 through 3, Rev. A, and 6236E44, General Electric Drawing Nos. 977E709 and 977E467, and KAPL, Inc. Drawing Nos. 108E6847and 108E6846.

(b) Contents

(1) Type and form of material

An irradiated D1G prototype pressure vessel assembly, including pressure vessel, core barrel, thermal shields, and two surveillance train assemblies. In addition, the contents may include surface contamination in the form of activated corrosion products and 119 gallons of residual water.

(2) Quantity of material in package

One irradiated D1G prototype pressure vessel assembly. Surface contamination not to exceed 4.61 curies. Displaced material from cutting operations not to exceed 10.6 curies. The irradiated components not to exceed 60,000 curies.

6. The package will be operated in accordance with the procedures described in Chapter 7 of the application and in accordance with Naval Reactors letter G#C97-03596 dated August 28, 1997, or G#C98-10723 dated February 13, 1998. The package will be tested and maintained in accordance with the procedures in Chapter 8 of the application and in accordance with Naval Reactors letter G#C97-03596 dated August 28, 1997, or G#C98-10723 dated February 13, 1998.

7. Expiration date: July 31, 2007.

## REFERENCES

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging, WAPD-LP(CES)CS-670 dated January 1982.

Supplements: Naval Reactors letters G#7241 dated December 2, 1982, G#84-452 dated March 28, 1984, G#C92-03331 dated January 29, 1992, G#92-03546 dated June 5, 1992, G#92-03589 dated July 2, 1992, G#97-053513 dated June 11, 1997, G#C97-03596 dated August 28, 1997, G#C98-10723 dated February 13, 1998, G#98-10801 dated May 5, 1998, and G#02-0688 dated January 16, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

ate: 1 May, 2002

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
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# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9792 | 6 | 71-9792 | USA/9792/B(U) | 1 OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Department of Energy application dated April 22, 1991, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

(a) Packaging

(1) Model No.: Model 1 D1G Core Basket-Thermal Shield Shipping and Storage Container

(2) Description

The Model 1 D1G Core Basket-Thermal Shield (CB-TS) Shipping and Storage Container is a right circular cylinder approximately 115 inches in diameter and either 209 inches long (D1G design including impact limiter assembly) or 216 inches long (D2W design including impact limiter assembly). Access for loading is provided by a removable closure head. The container, consisting of the cylindrical side walls and the bottom end, has a three layer construction with a steel inner vessel approximately eight inches thick covered with approximately nine inches of reinforced concrete which is encased by a 3/8-inch thick outer shell. The CB-TS is secured in place inside the container with an 8-inch thick steel preload ring which is bolted to the inner vessel with 72 high strength bolts.

Closure of the containment vessel is provided by the 6-inch thick steel closure head which is fastened to the inner vessel with 72 high strength bolts. A steel closure ring is welded over the bolts and provides containment. A carbon steel inner impact limiter is welded to the top end of the closure ring. A wood outer impact limiter is bolted to the top plate of the container outer shell.

The shipping container is transported with its axis horizontal and is supported by a shipping skid. The loaded container weighs up to 185 tons.

5. (a) (3) Drawings

Packagings for which fabrication was begun before March, 1991 are constructed in accordance with the General Electric Company Drawings contained in Appendix 2.10.4 of the application, and packagings for which fabrication was begun after March, 1991 are constructed in accordance with the KAPL Drawings for the redesign configuration in Appendix 2.10.4 of the application.

(b) Contents

One irradiated D1G core basket-thermal shield assembly, and not more than one core's worth of irradiated D1G support assemblies, D1G lower control rod drive mechanisms, and D1G upper support assemblies; surface contamination in the form of activated corrosion products; and not more than 3.5 gallons of residual water.

6. (a) Preloading of the preload plate and the closure head and sealing the container must be done with a temperature at or above +40 °F.

(b) Shipment of containers S/N 0000001 through 0000007 shall be made only when the average daily temperature is expected to be above +40 °F. Shipment of containers S/N 000008 through 000019 and S/N N00020 through N00031 shall be made only when the average daily temperature is expected to be above +10 °F, subject to the following exception: shipment of any container with the closure Head identified as 04241-171D6617 P5, SER N00031 (Forging S/N BG-7140) shall be made only when the average daily temperature is expected to be above +30 °F.

(c) The D1G CB-TS Shipment shall be made no earlier than 150 days after shutdown of the reactor.

7. The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, and each packaging shall be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8.0 of the application.

| NRC FORM 618 (8-2000) 10 CFR 71 | CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES | | U.S. NUCLEAR REGULATORY COMMISSION | | |
|---|---|---|---|---|---|
| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
| 9792 | 6 | 71-9792 | USA/9792/B(U) | 3 OF | 3 |

8. Expiration Date: September 30, 2007.

REFERENCES

Department of Energy, Division of Naval Reactors, application dated April 22, 1991.

Supplements dated: Naval Reactors Letters G#92-03668, dated August 27, 1992; G#C95-10762, dated April 10, 1995; G#C96-03576, dated November 1, 1996; and G#C02-0751, dated April 5, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

ate: September 20, 2002

| NRC FORM 618 (8-2000) 10 CFR 71 | U.S. NUCLEAR REGULATORY COMMISSION |
|---|---|

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | PAGES |
|---|---|---|---|---|---|
| 9793 | 10 | 71-9793 | USA/9793/B(U)F | 1 OF | 8 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U. S. Department of Energy
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

"Core Independent M-140 Safety Analysis Report for Packaging" transmitted February 27, 1991, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

a) Packaging

(1) Model No.: M-140

(2) Description

The M-140 is a stainless steel cask for transporting spent fuel. The cask is a right circular cylinder and is transported in the upright position. The package's approximate dimensions and weights are as follows:

| | |
|---|---|
| Cavity diameter | 70 inches |
| Cavity height | 146 inches |
| Body outer diameter | 98 inches |
| Body steel wall thickness | 14 inches |
| Package overall outer diameter | 126 inches |
| Package overall height | 194 inches |
| Packaging weight, including standard internals | 315,000 pounds |
| Maximum package weight, including contents | 375,000 pounds |

The cask body is made from 304 stainless steel forgings. The cask walls are 14-inches thick and the bottom plate is 12-inches thick. The cask body flange provides a seating surface for the closure head and its protective dome. The flange contains 36 wedge assemblies located radially around the inside diameter. Retention of the closure head is achieved by engaging the wedges in a tapered groove in the circumferential edge of the closure head. The cask body has 180 external cooling fins welded to the exterior wall. A support ring is welded to the external cooling fins

5.(a)(2) Description (continued)

at a point above the center of gravity. The support ring seats on, and is bolted to, the rail car mounting ring during transport. The cask bottom is equipped with an energy absorber which is composed of five concentric stainless steel rings varying in thickness and height.

The closure head is made from forged 304 stainless steel and is approximately 13-inches thick and 81.7 inches in diameter. The closure head is equipped with an access port, which is approximately 24 inches in diameter, and is offset from the center of the closure head. The access port plug is a stepped design with a maximum diameter of approximately 31 inches and is attached to the closure head by 24 bolts. The closure head and access port are sealed with double ethylene propylene O-ring seals. Seal test ports are provided for the closure head and access port seals. A stainless steel protective dome is positioned over the closure head and is secured to the cask body flange by 12, 1.38-inch diameter, 38.5-inch long studs installed in a vertical direction and 6, 2.5-inch diameter, 9-inch long shear bolts installed in the radial direction.

The containment system is composed of the cask body, the closure head, and the closure head access port plug. There are seven penetrations in the standard containment system: a closure head, a drain port, a vent port, and an access port in the closure head, a thermocouple penetration, a water inlet penetration, and a water outlet penetration in the cask body. Each penetration is sealed with a plug and a double ethylene propylene O-ring seal and is equipped with a leak test port. For the full-core D1G-2, two additional penetrations were added to the closure head: a removable fuel assembly (RFA) access port and another vent penetration.

The spent fuel modules are positioned in an internals assembly. The internals assembly is composed of stacked internal spacer plates which have openings for the spent fuel modules. The internals assembly has a top plate or top plate subassembly

5.(b) Contents

(1) Type and form of material

Spent fuel, limited to the following types, including associated activated corrosion products:

(i) S3G-3 spent fuel.

(ii) S8G spent fuel.

(iii) D1G Core 2 spent fuel.

(iv) D2W spent fuel.

(v) A1G spent fuel.

(vi) S6W spent fuel.

(2) Maximum quantity of material per package

Total package weight, including spent fuel and internals assembly, not to exceed 375,000 pounds; and

(i) For contents described in 5(b)(1)(i):

S3G-3 spent fuel modules, not to exceed 62,300 Btu/hr decay heat per package.

(ii) For contents described in 5(b)(1)(ii):

S8G spent fuel, not to exceed 51,609 Btu/hr decay heat per package (prototype spent fuel modules), or 45,713 Btu/hr decay heat per package (shipboard modules).

(iii) For contents described in 5(b)(1)(iii):

D1G-2 spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve B from Figure 3-5 of the Safety Analysis Report for Packaging (SARP) or by use of a shielding hold time from 8(b) below, whichever hold time is greater.

5.(b)(2) Maximum quantity of material per package (continued)

(iv) For contents described in 5(b)(1)(iv):

D2W spent fuel modules, not to exceed 63,000 Btu/hr decay heat per package for prototype spent fuel, 53,000 Btu/hr decay heat per package for shipboard Type 3 spent fuel modules, or 45,900 Btu/hr decay heat per package for shipboard Type 5 spent fuel modules.

(v) For contents described in 5(b)(1)(v):

A1G spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve C from Figure 3-5 of the SARP or an administrative hold of 50 days, whichever hold time is greater.

(vi) For contents described in 5(b)(1)(vi):

S6W spent fuel modules, not to exceed 46,011 Btu/hr decay heat per package for a shipboard core or 47,160 Btu/hr for a prototype core at the time of container draining.

(c) Transport Index for Criticality Control

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

| Spent fuel module | TI |
|---|---|
| S3G-3 | 100 |
| S8G | 100 |
| D1G Core 2 (with standard internals) | 100 |
| D1G Core 2 (with D1G-2 internals) | 0 |
| D2W | 100 |
| A1G | 0 |
| S6W | 100 |

6. For S3G-3 spent fuel shipments:

   (a) Only a full load is authorized. A minimum of 12 fuel modules must have either control rods or poison shipping rods. All rodded and unrodded modules must be positioned as specified on page 6-11 (Rev. 1) of "S3G-3 Recoverable Irradiated Fuel in the M-140 Safety Analysis Report for Packaging."

   (b) Minimum fuel cooling time is 130 days after shutdown.

   (c) Core age must be at least 4,000 logging corrected full power hours.

   (d) Control rod hold-down devices must be installed on cells which have control rods.

   (e) All cells must have top and bottom energy absorbers.

   (f) The weight of the spent fuel modules must be limited as specified on page 1-23 (Rev. 2) of "S3G-3 Recoverable Irradiated Fuel in the M-140 Safety Analysis Report for Packaging."

   (g) S3G-3 internals assembly must be used for shipment of S3G-3 spent fuel modules.

7. For S8G spent fuel shipments:

   (a) Only a full load is authorized. Full and partial fuel modules may be shipped in any combination. All full and partial fuel modules must have control rods.

   (b) Minimum fuel cooling time is 248 days after shutdown for prototype modules and 157 days after shutdown for shipboard modules.

   (c) All fuel modules must have lower supports and grapple adapters.

   (d) Standard internals assembly must be used for shipment of S8G fuel modules. Full fuel modules must have two full (side) spacers; partial fuel modules must have two full (side) spacers and one partial (inside) spacer.

   (e) The weight of the fuel modules must be limited as specified on page 1.23 (Rev. 4) of "S8G Recoverable Irradiated Fuel in the M-140 Safety Analysis Report For Packaging."

8. For D1G Core 2 spent fuel shipments:

   (a) Either a full core or up to eight fuel modules may be shipped per package. Fuel modules of different types may be shipped in any combination.

   (b) The minimum cooling time shall be the greater of 90 days for rail transport, 105 days for ship transport, or that calculated from Curve B of Figure 3-5 of the SARP.

   (c) All normally rodded fuel modules must have control rods. Control rod hold-down devices must be installed on rodded modules.

   (d) Rodded modules must have top and bottom energy absorbers. Unrodded modules must have top energy absorbers.

   (e) For shipments of up to eight fuel modules, the standard internals assembly must be used. Fuel module cavity spacers must be used for all fuel modules.

   (f) For full core shipments, the full core internals must be used.

9. For D2W spent fuel shipments:

   (a) Up to one spent fuel core shipped per package. Fuel modules of different types may be shipped in any combination. Up to nine fuel modules may be shipped per package, provided that one of the fuel modules is the prototype RFA.

   (b) Minimum fuel cooling time is 180 days after shutdown.

   (c) All normally rodded fuel modules must have control rods. Control rod holddown devices must be installed on all rodded modules. The universal grapple adapters serve as the rod holddown devices.

   (d) The standard internals assembly must be used for shipment of D2W fuel modules. All fuel modules must be shipped with the appropriate cell spacers, as shown in Appendix 1.4 of the application dated October 14, 1994.

10. For A1G spent fuel shipments:

    (a) The A1G fuel modules may be shipped in any combination of fuel modules and cell support housings, and shall not exceed a total of eight items (partial shipments, with less than eight items, are acceptable).

    (b) The A1G fuel modules shall be shipped with grapple adapters, support stands, rod holddown devices, and control rods installed. Fuel modules shall not be shipped with poison (shipping) rods installed.

    (c) Cell support housings shall be shipped with grapple adapters and bottom spacers installed.

    (d) Minimum fuel cooling time shall be the greater of 50 days after shutdown or that calculated using Curve C from Figure 3-5 of the SARP.

11. For S6W spent fuel shipments:

    (a) Up to eight fuel modules from a single core may be shipped.

    (b) The minimum fuel cooling time before container draining shall be 300 days after shutdown for a shipboard core or 450 days after shutdown for a prototype core.

    (c) All fuel modules must be shipped with control rods, control rod restraints, and grapple adapters installed. A lower pedestal must be installed in each module holder port.

    (d) The standard internals assembly and the S6W guide spacers must be used for shipment of S6W fuel modules.

12. The package must contain no more than 6 gallons of residual water, except that shipments of D2W recoverable irradiated fuel may contain up to 11 gallons of residual water.

13. Failed fuel or fuel with defective cladding is not authorized for shipment.

14. Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except:

    All containment seals, including the main closure head seal, must be replaced with new seals within the 12-month period prior to each shipment, or earlier if inspection shows any defect.

15. The package must be prepared for transport and operated in accordance with Chapter 7 of the application, except:

    The containment seals, excluding the main closure head seal, must pass a leak test after final closure prior to each shipment. The leak test must have a sensitivity of at least $1 \times 10^{-3}$ std-cm$^3$/sec.

16. Prior to first use, and within the 12-month period prior to each shipment, all containment seals, including the main closure head seal, must be leak tested to show a leak rate no greater than $1 \times 10^{-4}$ std-cm$^3$/sec. The leak test must have a sensitivity of at least $5 \times 10^{-5}$ std-cm$^3$/sec.

17. Expiration date: October 31, 2006.

REFERENCES

"Core Independent M-140 Safety Analysis Report For Packaging," transmitted February 27, 1991.

Supplements dated: May 23, June 21, and July 17, 1991; February 4 and 7, August 17, and December 2, 1992; October 14, 1994; September 1, and November 16, 1995; May 13, August 7, September 26, and November 26, 1996; February 10, 1997; June 11, 1998; and April 11, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 06/11/01

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9794 | 3 | USA/9794/B(U)-85 | 1 | 3 |

. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANA..YSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Safety Analysis Report for Packaging for CGN Reactor Compartment Disposal, dated July 12, 1994, as supplemented

c. DOCKET NUMBER 71-9794

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: CGN Reactor Compartment Disposal Package

(2) Description

The package consists of one deactivated and defueled CGN 36, 37, 38, 39, or 40 (36-40) Reactor Compartment that has been separated from the remainder of the cruiser hull and prepared for shipment by enclosing the entire reactor compartment within a welded steel container. The package is approximately cylindrical, about 40-feet high and about 32-feet in diameter. The entire package is a sixteen-sided polyhedron with an enlarged base containing support fixtures, which extend approximately 10 feet beyond the diameter of the package and provide lift points for the package. The container is constructed of high strength steel (MIL-S-22698). The reactor compartment decks, inner-bottom tank structure, secondary shield, and primary shield tank provide internal support and are fastened to the container by welding. The reactor compartment components are drained of water, except for small inaccessible pockets. The maximum weight of the CGN 36-40 package is 5,000,000 pounds. Potentially radioactive contaminated components and piping from areas outside the reactor compartment may be secured within the package.

(3) Drawings

The packaging is constructed in accordance with the drawings in Chapter 1 of the application.

5. (b) Contents

(1) Type and form of material

Activated structural components associated with the CGN 36-40 reactor, plant piping, ion exchanger resin, purification filter media (which may be solidified), and other components contaminated with radioactive corrosion products (crud). Residual liquid, primarily water, some of which contains low level radioactivity, may be present in quantities up to 850 gallons in the CGN 36-40 package.

(2) Maximum quantity of material per package

The maximum quantity of radioactive material contents (crud and activation) shall not exceed the quantities specified in Section 1.2.3.1 of the application.

6. (a) The shipment of a CGN 36-37 package shall be no earlier than 639 days after shutdown.
(b) The shipment of a CGN 38-40 package shall be no earlier than 365 days after shutdown.

7. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the daily minimum temperature expected during shipment of the package, as determined on the basis of weather forecasts.

8. (a) For CGN 36-37 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 4.1 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 10.6 curies.

(b) For CGN 38-40 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 5.3 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 9.58 curies.

9. (a) CGN 36-37 reactor vessels shall have been operated for less than 18,683 effective full power hours.
(b) CGN 38-40 reactor vessel shall have been operated for less than 14,300 effective full power hours.

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

(a) The package must be prepared for shipment and operated in accordance with Chapter 7 of the application.

(b) The package must be acceptance tested in accordance with Chapter 8 of the application.

11. Expiration date: September 30, 2005.

## REFERENCES

"Safety Analysis Report for Packaging for CGN Reactor Compartment Disposal," dated July 12, 1994.

Supplements Dated: November 10, 1994; July 14, 1995; November 22, 1996; June 16 and July 17, 1998; and December 22, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan J. Shankman for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 20, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIALS PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. PACKAGE IDENTIFICATION NUMBER | d. PAGE NUMBER | e. TOTAL NUMBER PAGES |
|---|---|---|---|---|
| 9795 | 1 | USA/9795/B(U)-85 | 1 | 2 |

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

A1G Irradiated Component Disposal Container
Safety Analysis Report for Packaging
dated July 10, 1997, as supplemented.

c. DOCKET NUMBER 71-9795

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: A1G Irradiated Component Disposal Container

(2) Description

The Model No. A1G Irradiated Component Disposal Container is stainless steel cask with an impact limiter at the upper end. The cask body is cylindrical in shape with overall dimensions of approximately 134.6 inches long by 122 inches diameter at the container body flange. The cask cavity is approximately 134.6 inches long by 91 inches diameter. The wall of the cask is 304 stainless steel, 10 inches thick at the bottom and 5 inches thick at the top. The bottom of the cask is an 11 inch thick circular steel plate. The cask lid is closed by a full penetration weld. The upper impact limiter is a stainless steel ring attached with 21 studs to the cask body. A centering plate and pedestals, welded to the bottom end plate, are used to position the contents within the package. The maximum weight of the package is 200,000 pounds. The maximum weight of the contents is approximately 36,300 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures and sketches included in the application documents (see References, below).

(b) Contents

The contents of the package are A1G cell support housings and other miscellaneous core components from a spent A1G reactor core. The contents will not exceed 19 A1G cell support housing assemblies or will not exceed 18 A1G cell support housing assemblies and one miscellaneous component cylinder (MCC). The other contents of the package include potential residual water not greater than 6 gallons, diatomaceous earth desiccant to absorb the residual water and a stainless steel pumpdown lance which may be left in the package. The maximum radioactivity of the contents is 5,600 curies. The total radioactivity is based on transport no earlier than 50 days after core shutdown.

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

   (b) The packaging must meet the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

   (c) The package may contain no more than 6 gallons of residual water.

   (d) The ICDC shall be shipped no earlier that 50 days after core shutdown.

   (e) The total number of cluster joint stud remnants loaded into each ICDC must not exceed 25.

   (f) The miscellaneous component cylinder is limited to a maximum of 39 upper lead screw segments (approximately 33 inches in length), 39 upper tie rod segments (approximately 33 inches in length), and a maximum of 25 cluster joint stud remnants. The maximum number of cluster upper joint stud nuts loaded in an ICDC, either inside the MCC or attached to cluster joint stud remnants at the bottom of cell support housings, must not exceed 250.

   (g) The gross weight of the package shall not exceed 200,000 pounds.

7. Expiration date: April 30, 2003

REFERENCES

A1G Irradiated Component Disposal Container Safety Analysis Report for Packaging dated July 10, 1997.

Supplements: U.S. Department of Energy, Division of Naval Reactors Memorandum G#C98-11009 dated December 2, 1998.
U.S. Department of Energy, Division of Naval Reactors Memorandum G#99-03507 dated May 3, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 5/19/99

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: BYPROD. NORM. FORM

| Model | Package ID # | Expiration Date |
|---|---|---|
| CI-20WC-2 | USA/9098/B( ) | 05/31/2004 |
| CI-20WC-2A | USA/9098/B( ) | 05/31/2004 |
| PAS-1 | USA/9184/B(U) | 07/31/2004 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: BYPROD. SPEC. FORM

| Model | Package ID # | Expiration Date |
|---|---|---|
| A-0109 | USA/6280/B( ) | 02/28/2005 |
| BUSS R-1 | USA/9511/B(U) | 07/31/2007 |
| C-1 | USA/9036/B(U)-85 | 10/31/2006 |
| EAGLE | USA/9287/B(U)-85 | 12/31/2004 |
| F-294 | USA/9258/B(U)-85 | 12/31/2003 |
| F-423 | USA/9299/B(U)-85 | 08/31/2006 |
| F-430/GC-40 | USA/9290/B(U)-85 | 02/28/2007 |
| IR-100 | USA/9157/B(U)-85 | 09/30/2004 |
| LCG-25A | USA/4888/B( ) | 01/31/2007 |
| LCG-25B | USA/4888/B( ) | 01/31/2007 |
| LCG-25C | USA/4888/B( ) | 01/31/2007 |
| MW-3000 | USA/9030/B( ) | 10/31/2005 |
| NPI-20WC-6 | USA/9102/B( ) | 10/31/2003 |
| NPI-20WC-6 MKII | USA/9215/B(U) | 10/31/2002 |
| OP-100 | USA/9185/B(U)-85 | 11/30/2003 |
| OP-660 | USA/9283/B(U)-85 | 06/30/2003 |
| OPL-660 | USA/9283/B(U)-85 | 06/30/2003 |
| ORNL TRU CALIF | USA/5740/B( ) | 07/31/2006 |
| SENTINEL-100F | USA/5862/B( ) | 09/30/2005 |
| SENTINEL-25A | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-25B | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-25C | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-25C3 | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-25D | USA/4888/B( ) | 01/31/2007 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: BYPROD. SPEC. FORM

| Model | Package ID # | Expiration Date |
|---|---|---|
| SENTINEL-25E | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-25F | USA/4888/B( ) | 01/31/2007 |
| SENTINEL-8 | USA/9030/B( ) | 10/31/2005 |
| SNAP-21 | USA/5830/B( ) | 11/30/2005 |
| SPEC 2-T | USA/9056/B(U) | 04/30/2005 |
| SPEC-150 | USA/9263/B(U)-85 | 06/30/2005 |
| SPEC-300 | USA/9282/B(U)-85 | 04/30/2005 |
| URIPS-8A | USA/6786/B( )F | 09/30/2003 |
| URIPS-8B | USA/6786/B( )F | 09/30/2003 |
| 181361 | USA/5796/B(U) | 07/31/2002 |
| 181375 | USA/5796/B(U) | 07/31/2002 |
| 4.5 TON CF | USA/6642/B( ) | 02/28/2007 |
| 5979 | USA/5979/B( ) | 09/30/2005 |
| 5984 | USA/5984/B( ) | 08/31/2007 |
| 650L | USA/9269/B(U)-85 | 11/30/2005 |
| 6717-B | USA/6717/B(U) | 11/30/2003 |
| 680-OP | USA/9035/B(U)-85 | 05/31/2005 |
| 702 | USA/6613/B(U) | 06/30/2003 |
| 715 | USA/9039/B(U) | 02/28/2003 |
| 741-OP | USA/9027/B(U)-85 | 02/28/2006 |
| 770 | USA/9148/B(U) | 03/31/2002 |
| 771 | USA/9107/B(U) | 06/30/2003 |
| 855 | USA/9165/B(U) | 12/31/2003 |
| 865 | USA/9187/B(U) | 12/31/2003 |
| 880 SERIES PKG | USA/9296/B(U)-85 | 03/31/2006 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: FISSILE URANIUM

| Model | Package ID # | Expiration Date |
|---|---|---|
| ABB-2901 | USA/9274/AF | 09/30/2007 |
| ANF-250 | USA/9217/AF | 06/30/2005 |
| ATR | USA/9099/B(U)F-85 | 01/31/2004 |
| A1W-3 PUSC | USA/9787/B(U)F | 01/31/2005 |
| BRP RVP SAR5339 | USA/9300/B(U)-85 | 03/31/2007 |
| BW-2901 | USA/9251/AF | 10/31/2007 |
| CE-B1 | USA/9272/AF-85 | 01/31/2007 |
| DHTF | USA/9203/AF | 02/28/2006 |
| D2G POWER UNIT | USA/6441/B( )F | 08/31/2007 |
| ECO-PAK OP TU | USA/9288/AF-85 | 03/31/2005 |
| ESP-30X | USA/9284/B(U)F-85 | 05/31/2005 |
| FL 10-1 | USA/9009/B( )F | 09/30/2004 |
| FSV-3 | USA/6347/AF | 09/30/2007 |
| INNER HFIR UN | USA/5797/B(U)F | 09/30/2007 |
| LIQUI-RAD | USA/9291/B(U)F-85 | 10/31/2006 |
| MCC-3 | USA/9239/AF | 03/31/2007 |
| MCC-4 | USA/9239/AF | 03/31/2007 |
| MCC-5 | USA/9239/AF | 03/31/2007 |
| MO-1 | USA/9069/B( )F | 12/31/2002 |
| MODEL B | USA/6206/AF | 09/30/2005 |
| MODEL 1 S-6213 | USA/9186/B(U)F | 05/31/2007 |
| MODEL 2 S-6213 | USA/9186/B(U)F | 05/31/2007 |
| NCI-21PF-1 | USA/9234/B(U)F | 12/31/2003 |
| NFS-URANYL NIT. | USA/5059/AF | 05/31/2007 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: FISSILE URANIUM

| Model | Package ID # | Expiration Date |
|---|---|---|
| NNFD 5X22 | USA/9250/B(U)F-85 | 01/31/2003 |
| NNFD-10 | USA/6357/AF | 04/30/2006 |
| NONE SPECIFIED | USA/6406/AF | 07/31/2002 |
| NPC | USA/9294/AF-85 | 02/28/2006 |
| OUTER HFIR UN | USA/5797/B(U)F | 09/30/2007 |
| PADUCAH TIGER | USA/6553/AF | 07/31/2004 |
| PATRIOT | USA/9292/AF-85 | 01/31/2005 |
| RA-3 | USA/4986/AF | 03/31/2003 |
| SP-1 | USA/9248/AF | 02/28/2004 |
| SP-2 | USA/9248/AF | 02/28/2004 |
| SP-3 | USA/9248/AF | 02/28/2004 |
| SRP-1 | USA/9285/AF-85 | 10/31/2003 |
| ST | USA/9246/AF | 11/30/2006 |
| S5W POWER UNIT | USA/5580/B( )F | 12/31/2007 |
| TRIGA-I | USA/9034/AF | 12/31/2005 |
| TRIGA-II | USA/9037/AF | 12/31/2005 |
| UBE-1 | USA/9280/AF-85 | 12/31/2002 |
| UBE-2 | USA/9281/AF-85 | 05/31/2003 |
| UNC-2600 | USA/5086/B(U)F | 01/31/2004 |
| UNC-2901 | USA/6294/AF | 03/31/2006 |
| UX-30 | USA/9196/AF-85 | 02/28/2006 |
| WE-1 | USA/9289/B(U)F-85 | 02/29/2004 |
| 235R001 | USA/6386/B(U)F | 04/30/2005 |
| 51032-1 | USA/6581/AF | 05/31/2004 |
| 51032-2 | USA/9252/AF | 09/30/2003 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: FISSILE URANIUM

| Model | Package ID # | Expiration Date |
|---|---|---|
| 814A | USA/5149/B( )F | 06/30/2005 |
| 927A1 | USA/6078/AF | 10/31/2005 |
| 927C1 | USA/6078/AF | 10/31/2005 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: IRRADIATED FUEL

| Model | Package ID # | Expiration Date |
|---|---|---|
| BMI-1 | USA/5957/B( )F | 03/31/2006 |
| CNS 1-13G | USA/9216/B( )F | 12/31/2002 |
| FSV-1 | USA/6346/B( )F | 05/31/2006 |
| FSV-1 UNIT 3 | USA/9277/B( )F | 07/31/2006 |
| GA-4 | USA/9226/B(U)F-85 | 10/31/2003 |
| GE-100 | USA/5926/B( )F | 05/31/2003 |
| HI-STAR 100 | USA/9261/B(U)F-85 | 03/31/2004 |
| IF-300 | USA/9001/B( )F | 09/30/2005 |
| M-130 | USA/6003/B( )F | 09/30/2007 |
| M-140 | USA/9793/B(U)F | 10/31/2006 |
| NAC-LWT | USA/9225/B(U)F-85 | 02/28/2005 |
| NAC-STC | USA/9235/B(U)F-85 | 03/31/2004 |
| NAC-1 | USA/9183/B( )F | 09/30/2004 |
| NLI-1/2 | USA/9010/B( )F | 04/30/2006 |
| NLI-10/24 | USA/9023/B( )F | 07/31/2003 |
| NUHOMS-MP187 | USA/9255/B(U)F-85 | 09/10/2003 |
| NUHOMS-MP197 | USA/9302/B(U)F-85 | 07/31/2007 |
| T-2 | USA/5607/B( )F | 05/31/2003 |
| T-3 | USA/9132/B(M)F | 04/01/2006 |
| TN-BRP | USA/9202/B(U)F | 06/30/2004 |
| TN-FSV | USA/9253/B(U)F-85 | 05/31/2004 |
| TN-REG | USA/9206/B(U)F | 05/31/2005 |
| TN-68 | USA/9293/B(U)F-85 | 02/28/2006 |
| TN-8 | USA/9015/B( )F | 05/31/2006 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: IRRADIATED FUEL

| Model | Package ID # | Expiration Date |
|---|---|---|
| TN-8L | USA/9015/B( )F | 05/31/2006 |
| TN-9 | USA/9016/B( )F | 05/31/2006 |
| TS125 | USA/9276/B(U)F-85 | 09/30/2007 |
| 125-B | USA/9200/B(M)F | 03/31/2006 |
| 2000 | USA/9228/B(U)F-85 | 03/31/2006 |
| UMS UNIVERSAL | USA/9270/B(U)F-85 | 10/31/2007 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: PU AIR

| Model | Package ID # | Expiration Date |
|---|---|---|
| PAT-1 | USA/0361/B(U)F-85 | 09/30/2003 |
| PAT-2 | USA/9150/B(U)-85 | 07/31/2006 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: PU NORM. FORM

| Model | Package ID # | Expiration Date |
|---|---|---|
| B-3 | USA/6058/B( ) | 12/31/2005 |
| NRBK-41 | USA/9221/B( )F | 09/30/2006 |
| TRUPACT-II | USA/9218/B(U)F-85 | 06/30/2004 |
| 6400 | USA/6400/B( )F | 11/30/2007 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: PU SPEC. FORM

| Model | Package ID # | Expiration Date |
|---|---|---|
| BCL-2 | USA/9068/B( )F | 05/31/2007 |
| BCL-3 | USA/9067/B( )F | 09/30/2007 |
| S5W REFUEL.SRCE | USA/5757/B( )F | 03/31/2003 |
| 1500 | USA/5039/B()F | 10/31/2003 |

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
02/05/2003

Type of Packaging: WASTE, B

| Model | Package ID # | Expiration Date |
|---|---|---|
| A1G ICDC | USA/9795/B(U)-85 | 04/30/2003 |
| CGN RCDP | USA/9794/B(U)-85 | 09/30/2005 |
| CNS 1-13C | USA/9081/B( ) | 01/31/2003 |
| CNS 1-13C II | USA/9152/B( )F | 05/31/2004 |
| CNS 10-160B | USA/9204/B(U)-85 | 10/31/2005 |
| CNS 3-55 | USA/5805/B( ) | 03/31/2004 |
| CNS 8-120B | USA/9168/B(U) | 06/30/2005 |
| D1G CB-TS | USA/9792/B(U) | 09/30/2007 |
| HALFPACT | USA/9279/B(U)F-85 | 10/31/2005 |
| N-55 | USA/9070/B(U) | 01/31/2005 |
| PWR-2 CORE BAR. | USA/9791/B(U)-85 | 07/31/2007 |
| RH-TRU 72-B | USA/9212/B(M)F-85 | 02/28/2005 |
| S3G CBDCA | USA/9786/B(U) | 08/31/2006 |
| S5W REC. COMPT. | USA/9788/B(U)-85 | 09/30/2003 |
| S6G REC. COMPT. | USA/9788/B(U)-85 | 09/30/2003 |
| TN-RAM | USA/9233/B(U) | 01/31/2005 |
| 10-135B | USA/9210/B( ) | 01/31/2005 |
| 10-142 | USA/9208/B( ) | 08/31/2007 |
| 3-82B | USA/6574/B( ) | 05/31/2006 |

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

# BIBLIOGRAPHIC DATA SHEET

*(See instructions on the reverse)*

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0383
Volume 2
Revision 24

2. TITLE AND SUBTITLE

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

3. DATE REPORT PUBLISHED

| MONTH | YEAR |
|---|---|
| March | 2003 |

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8 above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The purpose of this directory is to make available a convenient source of information on packaging approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved Quality Assurance programs. The reports include a listing of all users of each package design and approved Quality Assurance programs prior to the publication date of the directory.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Transportation
Packaging
Radioactive Materials

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

*(This Page)*

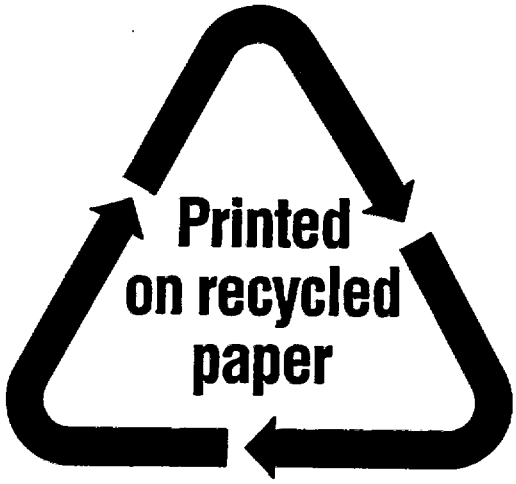
unclassified

*(This Report)*

unclassified

15. NUMBER OF PAGES

16. PRICE




Federal Recycling Program

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

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