APPLICATION FOR USE OF THE MODEL B SHIPPING CONTAINER FOR TRANSPORT OF RADIOACTIVE MATERIALS

B&W FUEL COMPANY

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1.0 This section provides the safety demonstration of shipping design, construction, and contents as required by 10CFR71.

1.1 <u>Package Description</u> (71.33)

1.1.1 Gross weight loaded with two fuel assemblies and components will be 7600 lbs. maximum.

1.1.2 Model B.

1.1.3 The shipping container is constructed primarily of carbon steel as described in the drawings listed in Exhibit A.

1.1.3.1 The zircaloy or stainless steel cladding of the fuel rods is the containment vessel. The loaded fuel rods are arranged in a rigid configuration and having a volume of water to UO_2 ratio of not more than 2.0.

1.1.3.2 Two 3/16 inch thick full length borated stainless steel plates containing at least 1.5% by weight natural Boron are located between the two fuel assemblies as non-fissile neutron absorbers.

1.1.3.3 The shell of the container is a cylindrical structure constructed of 0.089" thick carbon steel sheet with end domes of 0.125" thick carbon steel. Additional items to stiffen the outer shell to provide support to its basic structure are as follows:

* A series of two 90^o angles which are rolled & welded circumferentially to the shell.

* The parting flanges on both the upper and lower sections of the container shell.

* The base structure of the container consists primarily of

two full length angles welded to the lower half of the container shell.

1.1.3.4 Pressure relief and filler valves, lifting and tiedown devices, humidity indicators and accelerometer viewing ports are on the referenced drawings.

1.1.3.5 Heat dissipation not applicable.

1.1.4 Coolants not applicable.

1.2 <u>Package Contents</u>

1.2.1 Radioactive contents are fresh fuel assemblies of varying designs. The designs are denoted as Design 1-6 and their respective parameters are provided in Table 1.2. Designs 1,2,4-6 are enriched to a maximum of 5.05% in the U-235 isotope and 4.98% for fuel Design 3.

1.2.2 The maximum U-235 loading per fuel assembly is 25.7 kg U-235 (Design 2).

1.2.3 UO_2 sintered pellets enscapsulated in fuel rods.

1.2.4 Not applicable; nuclear safety analysis presumes optimum conditions.

1.2.5 Maximum weight of the contents is 3360 lbs.

1.2.6 Decay heat not applicable.

1.2.7 A nylon shim may be shipped between the contact surfaces of each control component assembly (CCA) and the fuel assembly, at times, to help preserve the quality of these core components. This shim is removed as part of the packaging unloading process.

Because some CCA designs have increased in length at the coupling area, the container end gate must be spaced further from the strong back structure to accommodate the longer CCA couplings. This extra end gate spacing will be achieved by adding steel block adapters between the hinged end gate and strong back frame.

TABLE 1.2

Fuel Assembly Specifications

Assembly Parameter	Design 1	Design 2	Design 3	Design 4
Rod Matrix	15x15	15x15	15x15	17x17
No. of fuel rods	208	208	208	264
No. of non-fuel tubes	17	17	17	25
Fuel rod pitch (in.)	0.568	0.568	0.568	0.496
Max. pellet OD, in.	0.3707	0.3742	0.3622	0.3232
Min. rod clad OD, in.	0.428	0.428	0.414	0.372
Tube material	Zr-4	Zr-4	Zr-4	Zr-4
Max. active fuel length, in.	144	144	144	144
Max. U^{235} enrichment	5.05	5.05	4.98	5.05
Max. U^{235} loading, Kg	25.1978	25.6758	23.7220	24.3108

Fuel Assembly Specifications (Con't.)

Assembly Parameter	Design 5	Design 6		
Rod Matrix	17x17	15x15		
No. of fuel rods	264	204		
No. of non-fuel tubes	25	21		
Fuel rod pitch (in.)	0.502	0.563		
Max. pellet OD, in.	0.3252	0.3671		
Min. rod clad OD, in.	0.377	0.420		
Tube material	Zr-4	Zr-4		
Max. active fuel length, in.	144	144		
Max. U^{235} enrichment	5.05	5.05		
Max. U^{235} loading, Kg	24.6126	24.2355		

2.0 <u>General Standards For All Packaging</u> (71.43)

2.1 There will be no significant chemical, galvanic or other reaction among the container components, or between the container and the fuel assemblies.

The shipping container is made primarily of carbon steel and the exposed material of the fuel assemblies is primarily zircaloy and stainless steel. Packing media may include polyethylene and fibre panels outside the fuel regions.

2.2 The self-contained closure hardware must be deliberately unfastened.

2.3 Lifting Devices

2.3.1 There are 8 lifting eyes on the lid of the container, 4 of which (2 on opposite sides of each end) are used to lift the loaded container. This was shown by lifting the loaded container free of the floor by each of its lifting eyes and holding to illustrate no yielding in the lifting eye. The system of 4 lifting eyes consequently is capable of supporting three times the weight of the loaded container without generating stress in excess of the yield strength.

2.3.2 Covered by 2.3.1 above.

2.3.3 There are no other structural parts of the package which could conceivably be used to lift the package. Further assurance of the use of only designated lift points is provided by adequate identification of the proper lift points on the container, and the fact that the container will be part of a Fissile Class II or III Shipment. As such the containers shall be transported with a vehicle for the sole use of BWFC. The controls imposed by BWFC for loading and unloading the containers will assure that only the designated system of lifting devices is used, and that only one loaded container is lifted at a time.

2.3.4 Failure of the lifting devices under load will not impair the containment or shielding properties of the package. Such failure if it occurred would only damage a portion of the container cover which is not considered as shielding and is not part of the structural members retaining the assemblies in the container. Detailed evaluation of the mode of failure involved is included in Exhibit C.

2.3.4.1 Tie-down devices

There is no system of tie-down devices which is a structural part of the container. The container is secured to the vehicle by binder chains passed over the container and fastened to the truck bed. In addition the containers will be chocked on the truck bed.

2.3.4.2 Since this container will be part of a Fissile Class II or III Shipment, it will be transported by an exclusive use vehicle, with specific instruction in the special arrangements providing for sole use by the BWFC. BWFC will supervise the loading of the vehicle to assure that the containers are tied down as described above. This administrative control by BWFC is adequate to assure that no structural part of the container is used as a tie down device. Detailed engineering evaluation of the possible tie-down points is included in Exhibit C.

2.3.4.3 There is no tie-down device which is a structural part of the container.

3.0 <u>Criticality Standards for Fissile Material Packages</u> (71.33)

3.1 The damaged container nuclear safety analysis demonstrates that an array of damaged containers is subcritical under varying conditions of moderation and full reflection. Consequently, one container is likewise subcritical under the criteria of this paragraph.

3.2 Not applicable; there will be no liquid contents during normal transport.

3.3 Not applicable; there will be no liquid contents during normal transport.

4.0 <u>Standards for Normal Conditions of Transport</u> (71.35)

4.1 A prototype container was tested under normal conditions of transport. The report of the test is included in the attachments to this section, as Exhibit B.

The materials of the containers and contents are such that their effectiveness cannot be substantially affected by either temperature extreme of 130°F, or -40°F.

Pressure relief valves will maintain the container shell pressure differential to less than 4.5 psi.

Water spray test is not applicable because the container shell and structural components are steel.

The free drop tests performed resulted in no significant damage to the container or contents.

The corner drop test is not required by virtue of the materials of construction.

The penetration test was not performed in that it is not credible that the test could puncture the container shell and result in the release of radioactive material.

The compression test performed was limited to demonstrating that two fully loaded containers could be stacked on top of one container. This test is adequate to assure safety in that the shipments will be made by exclusive use vehicle as Fissile Class II or III, and BWFC administrative controls will limit the stacking height.

In view of the above testing and assessment, it is concluded that:

4.1.1 There will be no release of radioactive material from the containment vessel.

4.1.2 The effectiveness of the packaging will not be substantially reduced.

4.1.3 There will be no mixture of gasses or vapors in the package which could through any credible increase of pressure or explosion, significantly reduce the effectiveness of the package.

4.1.4 Not applicable in that coolants are not involved.

4.1.5 Not applicable in that coolants are not involved.

4.2 The design and construction of the container and contents is such that under normal conditions of transport:

4.2.1 The package will be subcritical, see 71.33 (3.1).

4.2.2 The geometric form of the contents were not substantially altered by normal transport conditions.

4.2.3 Not required in that nuclear safety analysis presumes in-leakage of water.

4.2.4 There will be no substantial reduction of the effectiveness of the packaging including:

4.2.4.1 Reduction by more than 5 percent in the total effective volume of the packaging on which nuclear safety is assessed;

4.2.4.2 Reduction by more than 5 percent in the effective spacing on which nuclear safety is assessed, between the center of the containment vessel and the outer surface of the packaging; or

4.2.4.3 Occurrence of any aperture in the outer surface of the packaging large enough to permit the entry of a 4-inch cube.

5.0 Standards for Hypothetical Accident Conditions For a Single Package (71.73)

5.1 The effects of a hypothetical accident on a loaded container have been assessed as follows:

Free Drop - a prototype container loaded with two dummy fuel assemblies was drop tested. A report of the results of the tests is included with this section as Exhibit B.

Puncture - although this test is not essential in that nuclear safety analysis presumes the container to be flooded, the test was performed and is reported in Exhibit B.

Thermal - all materials of the container and contents significant to safety are such that they can withstand 1475^{O}F for 30 minutes.

Water Immersion - this test is not necessary since nuclear safety analysis presumes the container to be flooded.

The nuclear safety analysis of an array of damaged shipping containers presented in Section 6.0 shows the array to be subcritical. Consequently, a single container from the array is subcritical.

6.0 Specific Standards for a Fissile Class II or III Shipment (71.59 & 71.61)

6.1 The undamaged shipment shall be subcritical with an identical shipment in contact with it and with the two shipments closely reflected on all sides by water if considered as Fissile Class III. If Fissile Class II, five times the allowable number of undamaged packages would be subcritical stacked together in any arrangement and closely reflected on all sides of the stack by water. Undamaged shipping containers present a minimum of twenty-four inches separative distance between assemblies in adjacent containers. Since an infinite array of damaged containers with only seven to eighteen inches separation distance is subcritical, the undamaged containers with their additional spacing are likewise subcritical.

6.2 The shipment must be subcritical if each package were subject to the hypothetical accident conditions of Fissile Class III. Fissile Class II conditions require that twice the allowable number of packages, if each package were subjected to the tests specified in 71.73 would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation. Nuclear safety analyses have been performed for an infinite array of loaded shipping containers presumed damaged in excess of the actual damage experienced in testing and arranged in the most reactive configuration.

Maximum K-effective was determined for Design 3 (Mk-B11) fuel assembly with 4.98 wt% U-235 fuel. Considering various degrees of moderation K-maximum was determined for Design 3 to be 0.94994 with full density moderation providing the optimum moderator condition. For all other assembly types maximum K-effective was less than 0.95 for 5.05 wt% U-235 fuel. Excluding Design 3, the limiting assembly was Design 2 (Mk-B10F) with a K-maximum of 0.94348. The maximum K-effective values include a Δk bias of 0.01159 $\pm$ 0.00347 (1.763σ).

For the purpose of this analysis, the container shell was considered to be crushed to the level of the internal structural members for its full length and entire periphery. The containers were then envisioned to be in an array oriented top to top and bottom to bottom. This arrangement provides for closer approach of assemblies than does the normal top to bottom shipping configuration, and is considered to be the most reactive. Separation distances were then determined allowing only for the spacing provided by the internal structure and ignoring the contribution of such external structures as the skid frame, the stacking brackets and shell strengthening ribs. The smallest separation distances considered credible under these conditions are seven inches between top to top layers, and eighteen inches between bottom to bottom layers. Minimum side to side separation between the nearest assemblies in sets is eight inches. The dimensional relationship between assemblies and the boron-steel poison plates and the steel strong-back plates was retained, and the assemblies were contained within the steel shell. The results of the hypothetical accident testing reported elsewhere show the extent of crushing postulated to be far in excess of that actually occurring in drop tests, thus making this analysis ultra-conservative.

The original criticality safety analysis for the Model B fresh fuel shipping container was performed using the KENO-IV Monte Carlo code. Infinitely dilute cross sections were obtained from the 123 XSDRN master library cross section set. The NITAWL code was used to generate a 123 group KENO-IV working library that included resonance self shielding of the isotope uranium-238, the only resonance absorber present in the fresh fuel. The fuel lattice rod-to-rod self shielding for the U-238 resonance was accounted for with a Dancoff factor by NITAWL. The Dancoff factor was generated by the BWFC NULIF code, a neutron spectrum generator and spectrum weighted few group constant calculator.

For the revised SAR calculations fuel assembly specific enrichment limits were determined using the 27 group SCALE4.2 cross-section library with KENO-IV. Cross-sections were resonance treated using the SCALE4.2 CSAS25 module (BONAMI-S, NITAWL-II, and KENOVa; the KENOVa results were only used for number density input to KENO-IV). A KENO-IV bias was determined using the 27 group cross-section library and was conservatively shown to be 0.01159 $\pm$ 0.00347 (1.763σ). A description of the bias is contained in Section 6.3.

Three-dimensional heterogeneous geometry models were used for all KENO-IV criticality analyses. Individual components of the fuel assembly lattice were modeled along with the poison plates adjacent to each fuel assembly, surrounding moderator regions and container shell. Moderator was substituted for the fuel assembly grids, end fittings and structural steel components as a conservative geometry modeling simplification.

The accident condition package array was modeled as an infinite (symmetry boundary conditions) XY crushed container array of one active fuel column length (144 inches) with eight (8) inches of water on each end. For the most reactive moderation condition, 100% water moderator and water reflector, KENO-IV calculations were made for both an eight (8) and twelve (12) inch water reflector at each end of the active fuel stack. There was no significant difference in the K-effective due to the end reflector thickness.

The normal condition package array was modeled as an infinite (symmetry boundary conditions) XYZ container array with a 24 inch (a minimum) separation between fuel assemblies of different containers. For this analysis, the container walls were replaced by moderator.

K-effective was determined both as a function of moderator density and container array dimensions. Keeping the given crushed dimensions, K-effective was determined for moderator densities from 100% moderator density, down. The highest K-effective for these cases is 0.94994 at 100% moderator density for the Design 3 assembly at 4.98 wt% U-235 and 0.94348 for the Design 2 at 5.05 wt% U-235. All other assembly types were less limiting than Design 2. All original maximum K-effective values include 2 sigma + 0.02. The revised maximum K-effective values include $1.763\sigma + 0.01159$.

In the original analysis, a case was also run at nominal dimensions and 100% moderator density to represent an infinite xyz array of containers. K-effective was less than 0.927 and within 1 sigma of the reference case nominal K-effective.

The following conservatism adds a further factor of safety to the inherent safety of the system already demonstrated.

6.2.1 Credit has not been taken for the decrease in reactivity that results from other steel present; the "T" sections, the hold down bows, and strong-back structural members.

6.2.2 The nominal separative distances are smaller than can reasonably be predicted considering the actual drop test damage, and also recognizing that external structural members do in fact remain in place and contribute to separation.

6.3 LRC Critical Benchmark KENO-IV Results

Reference 1: Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel, BAW-1484-7, N.M. Baldwin, et al., July 1979.

The 21 critical LRC benchmark [1] calculations were evaluated using the 27 group SCALE 4.2 cross-section library. The KENO-IV calculations in Table 6.3.1 were performed using 625 neutrons per generation and 600 generations. The first 102 generations were skipped yielding a total of 311,250 neutron histories. Examination of Table 6.3.1 results indicates that the SCALE 27 group cross-section library with the CSASN (BONAMI/NITAWLII) cross-section treatment results in a maximum non-conservative bias of -0.01429 ± 0.00148 for core IX.

To test the adequacy of the neutron density per generation, core VI was rerun with a total of 850 generations and 2000 neutrons per generation. This case results in k-effective of 0.99781 ± 0.00053 with only the first 3 generations skipped. This case is shown in Table 6.3.2 along with other core VI results and indicates that larger neutron densities and generations are required to obtain meaningful results and statistics.

Table 6.3.3 shows the calculated bias for the eight most limiting core configurations identified from Table 6.3.1 using 2000 neutrons/generation and 847 active generations. The maximum calculated bias with uncertainty was -0.01335 $\pm$0.00197 for core XVI and represents a core with a water gap of 1.288 inches with borated aluminum isolation sheets in the water gap region. With the exception of core I and IX the other cases contained B_4C pins or borated aluminum isolation sheets. There is no apparent trend of the bias with separation distance or intervening materials. Therefore, the same 27 group bias and uncertainty is used for all problem types represented by these critical configurations.

6.3.1 Statistically Calculated Maximum Bias

The previous calculations defined the maximum 27 group bias plus uncertainty from using the worst single core configuration. A more precise understanding of the bias is to view it in a statistical sense. It is possible that any single measured or calculated core configuration could have included larger errors than those that would actually occur if the experiment were repeated. To state the case another way, is it appropriate to penalize all future criticality results because one of twenty-one core configurations appears to indicate a larger bias which could be the result of random measurement error? This type of problem is addressed in statistical analysis by considering the determination of the expected sample mean and is a valid approach to use when groups of calculations are done at different conditions (as is the case for the different core configurations). The sample mean approach would view the core critical experiments as separate entities. If each core configuration experiment (and KENOIV analysis) were repeated a very large number of times, all core configurations would converge on the true sample mean. Furthermore, the true sample mean would be the same for each of the experiments. The true or expected sample mean is defined as:

$$E(x) = \sum_{i=1}^{i=M} w_i x_i / \sum_{i=1}^{i=M} w_i$$

where w_i and x_i are the weighing factors and the core bias values, respectively. E(x) is the expected sample mean. The weighing factors are defined as:

$$w_i = n_i / \sigma_i^2$$

where n_i and σ_i are the number of KENOIV generations (sample size) and the combined measured and KENOIV calculated standard deviation, respectively.

The expected sample mean of the bias was conservatively computed to be -0.01159 using only the **worst eight core configurations,** while for comparison the average bias of the eight worst core configurations was computed to be -0.01189. Both values are very close. The standard deviation for the expected sample mean method is the maximum standard deviation computed for any individual core. In this case the 1σ value is $\pm$0.00197 from core XVI. The one-sided upper tolerance factor at the 95/95 confidence level is assumed to be the same as for the KENOIV results or 1.763. For the average bias method the standard deviation is computed directly from the worst eight core configurations to be $\pm$0.0009093 with a one-sided upper tolerance factor at the 95/95 confidence level of 3.188. To summarize; the expected sample mean method results in a bias of -0.01159 $\pm$0.00347 (1.763σ). The average bias method results in a bias of -0.01189 $\pm$0.00290 (3.188σ). For this analysis the expected sample mean bias was used since that statistical method is most appropriate.

Table 6.3.1. KENO-IV LRC Critical Results With CSASN
27 Group Library
(Neutrons per Generation = 625;
Number of Active Generations = 498)

Spacing Between Arrays (in.)	Core Number	KENOIV On IBM 6000 w/CSASN/27Gp (1σ Unc)	Measured (1σ Unc)	Calculated Minus Measured (1σ Unc)
None	I	0.98903 (0.00127)	1.0002 (0.0005)	-0.01117 (0.00136)
	II	1.00489 (0.00104)	1.0001 (0.0005)	+0.00479 (0.00115)
0.644	III	1.00438 (0.00099)	1.0000 (0.0006)	+0.00438 (0.00116)
	IV	0.98764 (0.00120)	0.9999 (0.0006)	-0.01226 (0.00134)
	XI	1.00013 (0.00108)	1.0000 (0.0006)	+0.00013 (0.00124)
	XIII	0.99377 (0.00120)	1.0000 (0.0010)	-0.00623 (0.00156)
	XIV	0.99323 (0.00115)	1.0001 (0.0010)	-0.00687 (0.00152)
	XV	0.99266 (0.00106)	0.9998 (0.0016)	-0.00712 (0.00192)
	XVII	0.99619 (0.00113)	1.0000 (0.0010)	-0.00381 (0.00151)
	XIX	1.00027 (0.00099)	1.0002 (0.0010)	+0.00007 (0.00141)
1.288	V	0.98603 (0.00117)	1.0000 (0.0007)	-0.01397 (0.00136)
	VI	0.99602 (0.00109)	1.0097 (0.0012)	-0.01368 (0.00162)
	XII	0.99439 (0.00116)	1.0000 (0.0007)	-0.00561 (0.00135)
	XVI	0.98777 (0.00121)	1.0001 (0.0019)	-0.01233 (0.00225)
	XVIII	0.99390 (0.00112)	1.0002 (0.0011)	-0.00630 (0.00157)
	XX	0.99767 (0.00113)	1.0003 (0.0011)	-0.00263 (0.00157)
1.932	VII	0.98589 (0.00116)	0.9998 (0.0009)	-0.01391 (0.00147)
	VIII	1.01234 (0.00123)	1.0083 (0.0012)	+0.00404 (0.00172)
	X	0.99469 (0.00119)	1.0001 (0.0009)	-0.00541 (0.00149)
	XXI	0.98649 (0.00117)	0.9997 (0.0015)	-0.01321 (0.00190)
2.576	IX	0.98871 (0.00118)	1.0030 (0.0009)	-0.01429 (0.00148)

Table 6.3.2. KENO-IV LRC Core VI Results Using Variable Generations and Densities With CSASN 27 Group Library

Gen. Skipped	Act Gen/ Hist.	KENOIV On IBM 6000 w/CSASN/27Gp (1σ Unc)	Measured (1σ Unc)	Calculated Minus Measured (1σ Unc)
3	212/625	0.99322 (0.00189)	1.0097 (0.0012)	-0.01648
102	498/625	0.99602 (0.00109)	1.0097 (0.0012)	-0.01368
3	297/1000	0.99625 (0.00126)	1.0097 (0.0012)	-0.01345
102	198/1000	0.99806 (0.00149)	1.0097 (0.0012)	-0.01164
3	847/2000	0.99781 (0.00053)	1.0097 (0.0012)	-0.01189
102	748/2000	0.99736 (0.00056)	1.0097 (0.0012)	-0.01234

Table 6.3.3. KENO-IV LRC Critical Results Using CSASN 27 Group Library For Worst Eight Core Configurations (Neutrons per Generation = 2000; Number of Active Generations = 847)

Spacing Between Arrays (in.)	Core Number	KENO-IV On IBM 6000 w/CSASN/27Gp (1σ Unc)	Measured (1σ Unc)	Calculated Minus Measured (1σ Unc)
None	I	0.98964 (0.00053)	1.0002 (0.0005)	-0.01056 (0.00073)
0.644	IV	0.98892 (0.00052)	0.9999 (0.0006)	-0.01098 (0.00079)
1.288	V	0.98797 (0.00052)	1.0000 (0.0007)	-0.01203 (0.00087)
	VI	0.99715 (0.00049)	1.0097 (0.0012)	-0.01255 (0.00130)
	XVI	0.98675 (0.00051)	1.0001 (0.0019)	-0.01335 (0.00197)
1.932	VII	0.98689 (0.00050)	0.9998 (0.0009)	-0.01291 (0.00103)
	XXI	0.98896 (0.00050)	0.9997 (0.0015)	-0.01074 (0.00158)
2.576	IX	0.99100 (0.00051)	1.0030 (0.0009)	-0.01200 (0.00103)

7.0 Operating and Maintenance Procedures

7.1 Use and maintenance of the Model B fresh fuel shipping container is controlled by formal written procedures approved by appropriate plant management. These procedures specifically describe the sequence of operations for packaging, shipping, labeling, unloading, storing and maintaining the Model B shipping container to insure it meets the requirements set forth in its Certificate of Compliance.

Part of the requirements of these procedures dictate that a formal inspection is conducted prior to the use of each container. It consists of the following:

- The pressure pads and bow clamps (fuel assembly clamping mechanism) are properly aligned.
- The rubber pads are not damaged in any way.
- The neutron poison plates are present and acceptable.
- There are no broken welds, worn or stripped bolts, nuts, or "T" bolt slots, or excessive wear on support arm holes.
- The "O"-ring tank seat is acceptable and the rubber shock mounts have not deteriorated.
- There is no permanent set deflection in a shock mount.

- All threads, ball and sockets, ball lock pins, and friction parts are oiled when needed and all excessive oil is removed. All pillow block bearings are greased as necessary.

- Pressure pad (assembly clamps) adjustments meet their respective specifications.

- The out riggers (container supports) have no missing ball lock pins or damaged parts.

- The humidity indicators, when used, display blue in color.

- All shock indicators are properly installed, calibrated and correctly set.

- All sway indicator rods are properly set.

- If applicable, verify that the container is initially pressurized to two (2) psi, in accordance with the applicable procedure.

- The proper radiation surveys have been conducted in accordance with DOT regulations.

- Each container incorporates a tamperseal on the cover.

- All nuclear safety rules are strictly adhered to include enrichment verification. Health Safety shall be responsible to check to ensure that all these requirements are being met.

If any of the items are not in conformance, the proper corrective action is taken prior to the release of the container.

Other key points covered in these procedures include:

- The proper equipment needed.

- The operator's qualifications.

- The acceptance criteria for routine inspections, maintenance requirements and maintenance records.

- Precautions that should be taken when handling the Model B shipping container.

- Step-by-step sequential instructions for loading and unloading the container.

Procedures have also been written to instruct the appropriate personnel at the reactor sites on the proper and safe use of the Model B shipping containers.

7.2 Records pertaining to Model B container shipments as required by 10 CFR 71 are retained for a minimum of 3 years.

7.3 BWFC NRC approved QA Manual covers the design, fabrication, testing, inspection, use and repair of radioactive materials shipping containers subject to the QA requirements of 10 CFR 71.

EXHIBIT A

MODEL B - FRESH FUEL SHIPPING CONTAINER DRAWING LIST

DRAWING NO.	DRAWING TITLE	PAGES
1215464 E	Shipping Container Strongback Assembly and Details	Sheets 1 - 4
1215465 E	Shipping Container Upper Weldment, Lower Weldment and Details	Sheets 1 - 2
1215466 E	Shipping Container Assembly and Components	Sheets 1 - 4

FIGURE WITHHELD UNDER 10 CFR 2.390

BILL OF MATERIALS

TOLERANCES
UNLESS OTHERWISE SPECIFIED
FRACTIONS: 0-6" ±1/32
OVER 6"-24" ±1/16
OVER 24" ±1/8
DECIMAL: 2-PLACE - .XX ±.02
3-PLACE - .XXX ±.005
ANGLE: ±1/2°
ANGULAR DIM. FOR CHAM. & C'SNKS ±5°
FILLET RADII .03 R MAX.
BREAK CORNERS .02 MAX. R OR CHAM.
ALL MACHINED SURFACES 125/AA FINISH

DR. BY DA BARGER
DATE 11-22-94
ME/92-32/5/01

SHIPPING CONTAINER
STRONGBACK ASSEMBLY
AND DETAILS

SHEET 1 OF 4

B & W FUEL COMPANY

AS SHOWN

DRAWING NO. 1215464 E
REV. 0

A

FIGURE WITHHELD UNDER 10 CFR 2.390

TOLERANCES
UNLESS OTHERWISE SPECIFIED
FRACTIONS: 0-6" ±1/32
OVER 6"-24" ±1/16
OVER 24" ±1/8
DECIMAL: 2-PLACE - .XX ±.02
3-PLACE - .XXX ±.005
ANGLE: ±1/2°
ANGULAR DIM. FOR CHAM. & C'SINKS ±5°
FILLET RADII .03 R MAX.
BREAK CORNERS .02 MAX. R OR CHAM.
ALL MACHINED SURFACES 125/AA Finish

DA BARGER
DATE 11-22-04
CONTRACT NO
ME/92-52/5/01

SHIPPING CONTAINER
STRONGBACK ASSEMBLY
AND DETAILS

SHEET 2 OF 4

B & W FUEL COMPANY
SCALE AS SHOWN
1215464 E 0

FIGURE WITHHELD UNDER 10 CFR 2.390

TOLERANCES
UNLESS OTHERWISE SPECIFIED
FRACTIONS: 0-6" ±1/32
OVER 6"-24" ±1/16
OVER 24" ±1/8
DECIMAL: 2-PLACE - .XX ±.02
3-PLACE - .XXX ±.005
ANGLE: ±1/2°
ANGULAR DIM. FOR CHAM. & C'SINKS ±5°
FILLET RADII .03 R MAX.
BREAK CORNERS .02 MAX. R OR CHAM.
ALL MACHINED SURFACES 125/AA FINISH

DRN BY DA BARGER
DATE 11-22-94
CONTRACT NO.
ME/92-52/5/01

SHIPPING CONTAINER STRONGBACK ASSEMBLY AND DETAILS

SHEET 3 OF 4

THIS DRAWING IS THE PROPERTY OF B & W FUEL COMPANY

SCALE AS SHOWN

DRAWING NO. 1215464 E

REV 0

FIGURE WITHHELD UNDER 10 CFR 2.390

TOLERANCES UNLESS OTHERWISE SPECIFIED FRACTIONS: 0-6" ±1/32 OVER 6"-24" ±1/16 OVER 24" ±1/8 DECIMAL: 2-PLACE - .XX ± .02 3-PLACE - .XXX ± .005 ANGLE: ±1/2° ANGULAR DIM. FOR CHAM. & C'SINKS ±5° FILLET RADII .03 R MAX. BREAK CORNERS .02 MAX. R OR CHAM ALL MACHINED SURFACES 125/AA Finish	DA BARGER 11-22-06 CONTRACT NO. ME/02-32/3/01	SHIPPING CONTAINER STRONGBACK ASSEMBLY AND DETAILS SHEET 4 OF 4	B & W FUEL COMPANY SCALE AS SHOWN DRAWING NO. 1215464 E REV 0

4 3 2 1 A

FIGURE WITHHELD UNDER 10 CFR 2.390

BILL OF MATERIALS

TOLERANCES
UNLESS OTHERWISE SPECIFIED
FRACTIONS: 0-6" ±1/32
OVER 6"-24" ±1/16
OVER 24" ±1/8
DECIMAL: 2-PLACE - .XX ±.02
3-PLACE - .XXX ±.005
ANGLE: ±1/2°
ANGULAR DIM. FOR CHAM. & C'SINKS ±5°
FILLET RADII .03 R MAX.
BREAK CORNERS .02 MAX. R OR CHAM.
ALL MACHINED SURFACES 125/AA FINISH

DWN. BY GF MCDANIEL
DATE 11-22-04
ME/92-52/5/02

SHIPPING CONTAINER, UPPER WELDMENT, LOWER WELDMENT AND DETAILS

SHEET 1 OF 2

THIS DRAWING IS THE PROPERTY OF B & W FUEL COMPANY

SCALE NONE

DRAWING NO. 1215465 E 0

FIGURE WITHHELD UNDER 10 CFR 2.390

TOLERANCES UNLESS OTHERWISE SPECIFIED FRACTIONS: 0-6" ±1/32 OVER 6"-24" ±1/16 OVER 24" ±1/8 DECIMAL: 2-PLACE - .XX ±.02 3-PLACE - .XXX ±.005 ANGLE: ±1/2° ANGULAR DIM. FOR CHAM. & C'SINKS ±5° FILLET RADII .03 R MAX. BREAK CORNERS .02 MAX. R OR CHAM. ALL MACHINED SURFACES 125/AA Finish	DRN BY DF MCDANIEL CHD BY PASSED BY APPD BY DATE 11-22-94 CONTRACT NO ME/92-52/5/02	SHIPPING CONTAINER UPPER WELDMENT, LOWER WELDMENT AND DETAILS SHEET 2 OF 2	THIS DRAWING IS THE PROPERTY OF B & W FUEL COMPANY DO NOT SCALE / USE DIMENSIONS ONLY SCALE NONE DRAWING NO. 1215465 E REV. NO. 0

A

FIGURE WITHHELD UNDER 10 CFR 2.390

BILL OF MATERIALS

TOLERANCES UNLESS OTHERWISE SPECIFIED FRACTIONS 0-6" ±1/32 OVER 6"-24" ±1/16 OVER 24" ±1/8 DECIMAL: 2-PLACE - .XX ± .02 3-PLACE - .XXX ± .005 ANGLE ±1/2° ANGULAR DIM. FOR CHAM. & C'SINKS ±5° FILLET RADII .03 R MAX. BREAK CORNERS .02 MAX. R OR CHAM. ALL MACHINED SURFACES 125/AA Finish	DWN BY DA BARGER CHKD BY PASSED BY APPD BY DATE 11-22-94 CONTRACT NO ME/92-52/5/03	SHIPPING CONTAINER ASSEMBLY AND COMPONENTS SHEET 1 OF 4	THIS DRAWING IS THE PROPERTY OF B & W FUEL COMPANY DO NOT SCALE USE DIMENSIONS ONLY SCALE NTS DRAWING NO. 1215466 E REV NO 0

4 3 2 1

FIGURE WITHHELD UNDER 10 CFR 2.390

TOLERANCES UNLESS OTHERWISE SPECIFIED FRACTIONS: 0-6" ±1/32 OVER 6"-24" ±1/16 OVER 24" ±1/8 DECIMAL: 2-PLACE - .XX ±.02 3-PLACE - .XXX ±.005 ANGLE: ±1/2° ANGULAR DIM. FOR CHAM. & C'SINKS ±5° FILLET RADII .03 R MAX. BREAK CORNERS .02 MAX. R OR CHAM. ALL MACHINED SURFACES 125/AA FINISH	DR BARNES DATE 11-22-04 CONTRACT NO. ME/92-52/5/03	SHIPPING CONTAINER ASSEMBLY AND COMPONENTS SHEET 2 OF 4	B & W FUEL COMPANY SCALE NTS 1215466 E 0
4	3	2	1

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390