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Draft for Review

**Safety Analysis Report for Packaging (SARP):
Models AL-M3 and AL-M6 Nuclear Packaging
(DOE C of C No. USA/5790/BLF and No. USA/
5791/BLF)**

Harold L. Coleman, Michael A. Whitney,
Monte A. Williams, Barbara M. Alexander,
and Alvin Shapiro

November 24, 1987

MOUND

Miamisburg, Ohio 45342

operated by

MONSANTO RESEARCH CORPORATION

a subsidiary of Monsanto Company

for the

U. S. DEPARTMENT OF ENERGY

Contract No. DE-AC04-76DP00053.

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Harold L. Coleman, Michael A. Whitney,
Monte A. Williams, Barbara M. Alexander,*
and Alvin Shapiro*

Reviewers: James F. Griffin, Ronald E. Wieneke,
Anatol Wojtowicz, and Richard K. Blauvelt

Issued: November 24, 1987

*Mound consultant.

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1. GENERAL INFORMATION

1.1. Introduction

This revised Safety Analysis Report for Packaging (SARP) satisfies the requirement of the U. S. Department of Energy (DOE) for an updated formal safety analysis of the two insulated drum shipping containers identified as USA/5790/BLF (DOE-AL) and USA/5791/BLF (DOE-AL). The report makes available to all potential users the technical information and limits pertinent to the construction and use of the shipping containers. This SARP includes discussions of structural integrity, thermal resistance, radiation shielding and radiological safety, nuclear criticality safety, and quality control.

Complete physical and technical descriptions of the packages are presented. Each package consists of a cylindrical steel inner container centered within an insulating steel drum assembly. The contents may be any radioactive materials that satisfy the requirements established in this SARP. A shipment of plutonium-238 in the form of a solid oxide is evaluated in this SARP as an example.

Design and development considerations, the tests and evaluations required to prove the ability of the containers to withstand normal transportation conditions, and the sequence of four hypothetical accident conditions (free drop, puncture, thermal, and water immersion) are discussed. Tables, graphs, dimensional sketches, photographs, technical references, loading and shipping procedures, Mound Facility experience in using the containers, and copies of the DOE Certificates of Compliance are included. Internal reviews of the original and revised SARPs have been performed in compliance with the requirement of DOE 1540.2.

The packagings are intended primarily for shipment of Type B solids in normal or special form; however, solid Type A quantities and lesser amounts would be authorized.

An all-new nuclear criticality safety analysis was performed for $^{238}\text{PuO}_2$, α -phase plutonium-239, uranium-233 metal, and uranium-235 metal. The quantities of these materials comprising Class I, Class II, and Class III shipments are summarized in this report.

Some of these quantities have changed significantly as a result of the new analysis. Users should be sure to consult Chapter 6 of this report for a full analysis. Where Class II limits were increased, the old Class II limits were retained. Where Class II limits decreased, the new limits have been adopted. Class III quantities are no longer authorized.

The package transport index, as a function of isotope mass per package, is plotted in Figure 1-1 for uranium-235, in Figure 1-2 for α -phase plutonium-239, and in Figure 1-3 for uranium-233. These values are significantly different from those of previous SARPs. The Class II mass limits for plutonium-238 are academic because of the high heat density of this material. Because of the heat dissipation characteristics of these containers, no more than 100 g of plutonium-238 can be shipped at any one time, an amount well below the calculated Class II mass limit for these packages.

1.2. Package Description

1.2.1. Packaging

1.2.1.1. General. In this section, sufficient information is given regarding design intent and design detail to accurately identify the shipping containers and to provide the basis for the evaluation of the packages. The

FIGURE 1-1 - Package transport index as a function of uranium-235 mass per package.

FIGURE 1-2 - Package transport index as a function of α -plutonium-239 mass per package.

FIGURE 1-3 - Package transport index as a function of uranium-233 mass per package.

two shipping containers (Figures 1-4 and 1-5) are basically the same design but differ in size. In both sizes, a steel inner container (the containment vessel) is centered within an insulated drum assembly. The packages are generally referred to in this report by the nominal size of the steel drum outer container or by an abbreviated form of the package identification. The 12-gal or type 5790 package is identified as USA/5790/BLF (DOE-AL) and the 55-gal size or type 5791 package is identified as USA/5791/BLF (DOE-AL). The maximum gross weights and overall dimensions of the two package types are given in Table 1-1. Drawings detailing the safety features considered in this analysis are included in the Appendix, section 1.3.

TABLE 1-1
MAXIMUM GROSS WEIGHTS AND OVERALL DIMENSIONS

Package Type	Nominal Size (gal)	Maximum Gross Weight (lb)	Overall Diameter (in.)	Overall Height (in.)
5790	12	90	14.6	21.1
5791	55	300	23.9	34.8

The containers are fabricated in accordance with the following MRC Dwg. Nos.:

- AYD 750138 - 12-gal shipping container, Model AL-M3 (or type 5790)
- AYD 750148 - 55-gal shipping container, Model AL-M6 (or type 5791)
- SPA 740977 - Acceptance and Reuse Inspections.

The design of the inner container of the type 5790 package has been changed significantly from the original issue of the drawings. Thus, users must take special care to consult the latest issue of the drawings. The design changes to the type 5790 package relate mainly to the inner container cover and seal.

Shipping container materials are not normally used as neutron absorbers or moderators. No shielding is normally required.

1.2.1.2. Design Intent. The two packages were designed to be used in much the same manner as Department of Transportation (DOT) specification 6M packages, except as specified in this report. The packages are intended primarily for shipment of solids. The quantities and types of materials are limited by the results of the tests and evaluations presented in this report. The packages are relatively inexpensive, easy to use and maintain (without special tools), and can be reused frequently. The packages are designed to provide the required containment during normal transportation and hypothetical accident conditions.

The packages are insulated to provide protection for the inner containers during hypothetical accident fire conditions. The type of insulation and its thickness are selected to preclude overheating when containing the maximum permissible decay heat load.

FIGURE 1-4 - Twelve gallon drum shipping container
Model AL-M3 USA/5790/BLF (DOE-AL).

FIGURE 1-5 - Fifty-five gallon drum shipping container
Model AL-M6 USA/5791/BLF (DOE-AL).

No special devices are required for tie-down or lifting. The packages may be strapped in place as required during transportation. The packages are designed to be lifted using drum handling equipment or a forklift; when using a forklift, the tines are placed under the bottom of the drum.

1.2.1.3. Steel Drums. The outer container for each package is an open head steel drum, as shown in Figures 1-6 and 1-7. The drums are vented to permit the escape of any vapors during the hypothetical accident fire, and the vents are sealed with waterproof tape during normal use. A metal identification plate is welded to the outside of the drum body, and security seals are fastened to the closure during shipment. The closure ring for both sizes has been modified by the addition of a metal "skirt" to keep the drum lid in place during the 30-ft drop test. The specifications for the outer steel drums are listed in Table 1-2.

TABLE 1-2
SPECIFICATIONS FOR OUTER STEEL DRUMS

<u>Package Type</u>	<u>Nominal Size (gal)</u>	<u>Drum Specification</u>	<u>Gauge Body/Heads</u>	<u>Inside Diameter (in.)</u>	<u>No. of 1/4-in. Diam. Vent Holes</u>
5790	12	MS-27684-8	20/20	13.8	4
5791	55	DOT-17C	16/16	22.5	12

1.2.1.4. Insulation. "Firedike" (Johns-Manville) insulating material was originally used in both size drums to protect the inner containers from impact and heat. Firedike has a density of approximately 24 lb/ft³ and consists of mineral wool fibers that offer exceptional fire protection. The flame resistance rating is Class A -- Incombustible -- and the flame spread rating is Class I (0-25). Firedike, which discolors significantly at 400°F, consists of 5/8-in. thick discs and rings glued together with B. F. Goodrich CA-108 nonflammable latex adhesive to eliminate air gaps. To provide a bearing surface for the inner container during shipment and protect the Firedike from wear, a Transite disc was glued to the bottom of the cavity that contains the inner container. The insulated cavity cover has a web strap attached to facilitate handling. The combined thickness of the insulation, at the top and bottom and the minimum thickness of the insulation, at the side, are given for each drum in Table 1-3.

TABLE 1-3
THICKNESS OF INSULATION IN EACH DRUM

<u>Package Type</u>	<u>Drum Nominal Size (gal)</u>	<u>Insulation Thickness Top and Bottom (in.)</u>	<u>Minimum Insulation Thickness at Sides (in.)</u>
5790	12	2	4
5791	55	7	6

FIGURE 1-6 - Outer container, package 5790; 12 gal steel drum, specification MS-27684-18.

FIGURE 1-7 - Outer container, package 5791, 55 gal steel drum, specification DOT-17C.

In 1977, however, Firedike and Transite insulating materials became unavailable. "Auratone Firecode" (U.S. Gypsum) was approved as a substitute for the Firedike. This material has thermal conductivity, density, fire resistance, and mineral fiber content comparable to Firedike (see section 3.4.2.1.).

At the same time, "Flexboard" (Johns-Manville) was approved as a substitute for Transite. This material provides a suitable bearing surface for the inner container in both packages.

1.2.1.5. Inner Container. The cylindrical steel inner containers, which are nested within the insulation, are the containment vessels for the radioactive materials. The inner containers are similar in design to DOT specification 2R containers, except as specified in this report. Two closure styles are used. In the 12-gal package (Figure 1-8), the inner container closure is a threaded cover that provides an O-ring seal. The inner closure for the 55-gal package (Figure 1-9) is a flanged assembly with an O-ring seal. Silicone and "Viton" O-rings are acceptable; no valves are used. Various packaging materials, such as aluminum pellets, nickel spheres, steel wool, and copper turnings, are used to position the materials being shipped within the inner containers. The inner containers must pass a helium leak test with no detectable leak greater than 1×10^{-5} std cm³/s when filled with 10 psig helium. The dimensions of the inner containers are given in Table 1-4.

TABLE 1-4
INNER CONTAINER DIMENSIONS

Package Type	Body o.d. (in.)	Body Wall (in.)	Flange o.d. (in.)	Overall Outside Height (in.)	Inside Height (in.)	Bottom Plate Thickness (in.)	Top Plate Thickness (in.)	Minimum Flange Thickness (in.)
5790	5.0	0.44	-	14.5	11.1	0.50	0.88	-
5791	7.0	0.25	9.88	18.25	16.0	0.25	0.48	0.44

1.2.2. Operational Features

The 5790 and 5791 packages are simple enclosures, involving no valves, connections, piping, or openings. The inner containment vessels are closed with O-ring seals.

1.2.3. Contents of Packaging

Materials that have been shipped in these packages include fissile and other radioactive materials that are dry solids in normal form not readily dispersible, and with no removable contamination. The radioactive material

FIGURE 1-8 - The inner container for package 5790 uses a threaded closure with an O-ring seal.

FIGURE 1-9 - The inner container for the 5791 package uses a flanged closure with an O-ring seal.

FIGURE 1-10 - Triply encapsulated heat standard.

consists of the same plutonium materials as previously noted. Other materials of normal form are uranium and thorium anhydrous oxide powders and small amounts (<20 mg) of thorium nitrate powder. The plutonium powder is placed in either center post cans or in nested cans with an O-ring seal. The pellet or cermet form is welded into stainless steel cans that are subjected to a helium leak rate test. The uranium and thorium powders are placed in glass vials which are in turn placed in O-ring sealed metal cans. These containers may additionally be packaged within hermetically sealed food pack cans (Figure 1-11) that provide secondary containment (see section 4.4).

Stabilized incinerator ash that contains a high level of plutonium-238 (up to 20 g) is also shipped to a recovery site. Prior to placing the ash in the type 5791 inner container, it is placed in two metal containers and a plastic bag as follows: the ash is placed in a metal can with a friction fit lid; this can is then placed in a plastic bag -- before it is put in a second metal can that has a rolled and crimped lid. Two of these assemblies are placed in one type 5791 inner container. Individual cans are assayed to ensure that the type 5791 shipping container limitation of 42 W(th) is not exceeded.

The thermal output of cans holding plutonium-238 material was determined by calorimetry to ensure that the thermal content placed in the shipping containers will not exceed the maximum allowable heat load for the containers as stated in section 3.4. The cans, after insertion into the inner container, are normally surrounded by packaging materials such as described in section 3.4.

1.2.3.1. Specific Criteria. Section B (page 3) of the ALO SARP Guide suggests certain criteria that should be considered in describing the contents. Information concerning the applicable criteria are given in this section.

1.2.3.1.1. Quantity of Isotopes. The quantity of isotopes shipped in each container is limited by the physical, chemical, and radiation characteristics of the isotope to be placed in the inner container.

The maximum quantities of uranium-235, plutonium-239, and uranium-233 that may be shipped in each package are listed in Table 1-5. The maximum quantities are based on the newly calculated Class II limits for each isotope or on the old package limits, whichever is smaller. Class III quantities are no longer authorized.

1.2.3.1.2. Maximum Amounts of Radioactivity. The amount of plutonium-238 permitted in each container is limited by the thermal output of the material. The maximum amount of plutonium-238 in each container is as follows:

Container Size (gal)	Plutonium-238 (Ci)
12	910
55	1,270

TABLE 1-5

MASS LIMITS^a FOR EACH PACKAGE

<u>Isotope</u>	<u>5790</u>	<u>5791</u>
Uranium-235	11.1	16.8
Plutonium-239	4.1	5.0
Uranium-233	4.9	6.7

^a In kg.

The plutonium-238 material normally consists of approximately 80 at. % plutonium-238, 16 at. % plutonium-239, 3 at. % plutonium-240, and considerably lesser quantities of plutonium-236, plutonium-241, and plutonium-242. The material also contains no more than 1% of other actinide impurities.

FIGURE 1-11 - Food pack can for secondary containment of plutonium.

1.2.3.1.3. Chemical and Physical Form. The radioactive material shipped was present as a nitrate or an oxide and existed either as loose powder or in pressed form, contained inside a sealed capsule or food pack can.

1.2.3.1.4. Material Density. The density of the radioactive material varied. The thorium and uranium isotopes were loosely packed powders, but the plutonium isotopes were pressed forms that had densities up to 9.5 g/cm^3 .

1.2.3.1.5. Moderating Ratios. Not applicable.

1.2.3.1.6. Configuration as Required for Nuclear Safety Evaluations. See section on criticality, Chapter 6.

1.2.3.1.7. Maximum Amount of Decay Heat. The maximum amount of decay heat for each container is as follows:

Container Size (gal)	Maximum Decay Heat (W)
12	30
55	42

1.2.3.1.8. Maximum Pressure Buildup in the Inner Container. All radioactive materials that may be shipped in the container do not decompose at the temperatures generated during the hypothetical accident fire condition, even when the maximum allowable wattage is present in the container. The plutonium-238 material does generate helium as part of its decay mode. Fortunately, the rate of generation is rather modest and amounts to 0.1 ml/mo/W. In addition, the helium is held tightly in the crystal lattice of the material and does not begin to escape until the material is heated to 2200°F -- which is well above the temperature that the inner container is expected to see under hypothetical accident fire conditions. The only pressure increase in the inner container, which will occur if the package is exposed to fire, will be from the heating of air trapped in the container at the time of sealing. This pressure increase will be nominal and should not exceed 20 psig. Therefore, there will be no appreciable pressure increase in the container.

1.2.3.1.9. Leak Tests. All material is tested for leakage before being packed into the inner container. The material is checked for radioactive contamination or tested for helium leaks to determine that it is free of leakage.

1.2.3.1.10. Loading Restrictions and Limitations. Loading procedures are given in Chapter 7.

1.3. Appendix

1.3.1. Detailed Drawings

AYD 750138	(5790)
AYC 750142	(Name Plate)
AYD 750140	(5790 inner)
AYC 750141	(5790 inner cover)
AYD 750148	(5791)
AYD 750150	(5791 inner)
AYD 750151	(5791 inner cover)
SPA 760488	(welding and inspection)

2. STRUCTURAL EVALUATION

2.1. Structural Design

2.1.1. Discussion

The outer container of the 5790 package is a 12-gal drum, specification MS-27684-8, fabricated of 20 gauge carbon steel. The outer container of the 5791 package is a 55-gal drum, specification DOT-17C, fabricated of 16 gauge carbon steel. The drums are vented to permit the escape of any vapors generated during the hypothetical accident fire; during normal use, the vents are sealed with waterproof tape. The drum covers are secured by a modified drum locking ring. A metal skirt is welded to the locking ring, which in turn is bolted to the drum body. This arrangement prevents separation of the drum cover if the package is dropped. Security seals are fastened to the closures during shipment.

The cavity between the inner and outer containers is filled with rigid insulating material to hold the inner containers in place, and to protect them from impact and heat.

The inner container for the 5790 package is a 4.12-in. inner diameter cylinder of 0.44-in. carbon steel. The closure is a threaded cover that provides an O-ring seal. The inner container for the 5791 package is a 6.5-in. inner diameter cylinder of 0.25 in. carbon steel. The closure is a flanged assembly with an O-ring seal. Silicone and Viton O-rings are acceptable. No valves are used.

2.1.2. Design Criteria

The internal pressure capability of both packagings at various temperatures was thoroughly evaluated. The weakest component was found to be the welded bottom plate of the inner steel container. The pressure capabilities are sufficiently high, however, to contain the pressure that would result from heating the air that was trapped during packaging operations (1 atm) to the temperatures reached during normal shipment and hypothetical accident conditions. Maximum allowable internal pressures for the two packages are summarized in Table 2-1. Details of the internal pressure calculations are given in the Appendix, section 2.10.

TABLE 2-1
MAXIMUM ALLOWABLE INTERNAL PRESSURE-INNER CONTAINERS

Nominal Package Size (gal)	Inside Radius (in.)	Internal Pressure Capability at -20 to 650°F (psig)	Maximum Bending Stress (psi)
12	2.06	220.4	364
55	3.25	37.0	363

When it is regarded as a simple beam supported at its ends along any major axis, the shipping container must be capable of withstanding a static load, normal to and uniformly distributed along its length, equal to five times the fully loaded container weight, without generating stresses in any material of the container in excess of the yield strength of that material. A sample calculation for the maximum bending stress is given in the Appendix, section 2.10. Results of these calculations are summarized in Table 2-1. Since the material yield stress of 27,000 psi is 74 times as great as the calculated maximum bending stress, the 55-gal drum container satisfies the load resistance requirement.

2.2. Weights and Centers of Gravity

The weights of the packages and individual subassemblies are tabulated in Table 2-2. The centers of gravity, shown in the same table, were calculated both on the basis of an empty inner container, and a container loaded with the maximum contents weight. For most shipments, the contents weight will be insignificant compared to the weight of the container. Centers of gravity for the 5790 and 5791 packages are shown in Figures 2-1 and 2-2.

TABLE 2-2
WEIGHTS OF PACKAGES AND COMPONENTS

<u>Assembly</u>	<u>5790 Package</u>		<u>5791 Package</u>	
	<u>Weight (lb)</u>	<u>Center of Gravity (in.)^(b)</u>	<u>Weight (lb)</u>	<u>Center of Gravity (in.)^(b)</u>
Inner Container	28	6.9	44.5	13.4
Outer Container	51.5	10.5	215	17.4
Total (empty)	79.5	10.4	259.5	18.1
Contents (maximum) ^(a)	8	-	44.5	-
Total (fully loaded)	87.5	10.4	304	18.7

^(a) Inner Container filled with lead shot.

^(b) Measured from outside bottom.

2.3. Mechanical Properties of Materials

Mechanical properties of the carbon steel used in the inner and outer containers are summarized in Table 2-3. Properties of the O-ring seals used in the two inner containers are shown in Table 2-4.

FIGURE 2-1 - Centers of gravity for 5790 package
 (■ inner container, ▲ outer container, ● overall container).

FIGURE 2-2 - Centers of gravity for 5791 package
 (■ inner container, ▲ outer container, ● overall container).

TABLE 2-3
MECHANICAL PROPERTIES OF CARBON STEEL

<u>Temp.</u>	<u>Yield Strength (10³ psi)</u>	<u>Charpy Impact Strength (ft·lbf)</u>	<u>Modulus of Elasticity (10⁶ psi)</u>	<u>Maximum Allowable Stress (10³ psi)</u>	<u>Density (lb/in.³)</u>	<u>Coefficient of Linear Thermal Expansion (μin./in.°F)</u>	
						-100°C to 20°C	20°C to 800°C
Normal	27.0(a)	34(b)	29.5(c)	15.0(d)	0.284(e)	5.73(f)	7.78(f)
-40°F	55(g)	33(b)	30(h)	-	-	-	-

(a) Reference 1, p. 208.

(b) Reference 4, p. 19.

(c) Reference 3, p. 217.

(d) Reference 1, p. 211, -20 to 650°F.

(e) Reference 3, p. 205.

(f) Reference 3, p. 207.

(g) Reference 4, p. 17.

(h) Reference 4, p. 22.

TABLE 2-4
O-RING PROPERTIES

<u>Material</u>	<u>Normal Temperature</u>		<u>High Temperature</u>		<u>Low Temperature</u>
	<u>Hardness, Shore A (pts)</u>	<u>Tensile Strength (psi)</u>	<u>Hardness, Shore A (pts)</u>	<u>Tensile Strength (psi)</u>	<u>Hardness, Shore A (pts)</u>
Silicone Rubber	75(a)	1195(a)	77(b)	1022(b)	76(c)
Viton Rubber	75(d)	1997(d)	76(e)	2046(e)	97(c)

(a) Reference 5.

(b) Reference 5, heat-aged 70 hr @ 437°F.

(c) Reference 6, p. A3-10.

(d) Reference 7.

(e) Reference 7, heat-aged 70 hr @ 528°F.

2.4. General Standards for All Packages

2.4.1. Minimum Package Size

10 CFR 71.43, part (a) states that the smallest overall dimension of a package must not be less than 10 cm (4 in.). For the 5790 and 5791 packages, the smallest overall dimension is the outer diameter, which is 14.6 in. for the 5790 and 23.9 in. for the 5791. Therefore, this condition is satisfied.

2.4.2. Tamperproof Feature

Seals are secured to the drum closures during shipment to deter unauthorized opening of the package. The unbroken seal also serves as evidence that the package has not been tampered with during shipment.

2.4.3. Positive Closure

Positive closures, bolts or screwed threads, are used on both the inner and outer containers to prevent accidental opening.

2.4.4. Chemical and Galvanic Reactions

The packaging materials and the package contents will not cause any significant reactions even at hypothetical accident conditions. Design materials were carefully selected on the basis of test data and past experience with container packaging, unpackaging, storage, and shipping.

2.5. Lifting and Tie-down Standards for All Packages

2.5.1. Lifting Devices

It is required that lifting devices, which are an integral part of the package, be capable of lifting three times the weight of the package and any attachments without generating stress in any package material in excess of that material's yield strength.

No lifting devices, as such, are provided on either shipping container. The drum covers and inner containers are generally removed manually and do not require any special devices. The assembled containers are commonly lifted with a forklift by positioning the forks beneath the bottom of the drum as shown in Figure 2-3. The following evaluations show that supporting three times the weight of the packages will not generate stresses in excess of the yield strength of the mild steel drums.

A sample calculation is provided for the 55-gal package. The gross weight of the 55-gal container is taken to be 400 lb and three times this weight is 1200 lb. This load is reacted at points as shown in Figure 2-3. Each fork is typically 4 in. wide and the drum wall is 0.0598 in. thick. The maximum stress in the drum is found by dividing the required load by the total cross-sectional area as follows:

$$S = \frac{1200 \text{ lb}}{(4 \text{ places})(4 \text{ in.} \times 0.0598 \text{ in.})} = 1254 \text{ psi} \quad ,$$

conservatively assuming the contact region to be 4 in. per each contact location.

FIGURE 2-3 - Lifting stresses in drum.

The lifting stress in the 12-gal package was determined in a similar manner, and the results are presented in Table 2-5.

TABLE 2-5
RESULTS OF FORKLIFT STRESS CALCULATIONS

Nominal Size Package (gal)	Drum Wall Thickness (in.)	Assumed Gross Weight (lb)	Maximum Compressive Stress (psi)
12	0.0359	150	781
55	0.0598	400	1254

The calculated stresses are far less than the 27,000 psi yield strength of the mild steel drums.

2.5.2. Tie-down Devices

DOE Manual Chapter 0529 specifies that the tie-down devices that are a structural part of the package must be capable of withstanding simultaneously 10 g longitudinal, 5 g lateral, and 2 g vertical loads without exceeding the yield strength of the material.

No tie-down devices are used on the packages discussed in this report. The packages are normally retained in place within the transport vehicle by other packages, or are strapped in place.

2.6. Normal Conditions of Transport

2.6.1. Heat

2.6.1.1. Summary of Pressures and Temperatures. Pressures and temperatures, calculated for conditions of normal transport in section 3.4, are summarized in Table 2-6.

TABLE 2-6
LIMITING TEMPERATURES AND PRESSURES^(a)
NORMAL CONDITIONS OF TRANSPORT

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	Maximum Temperatures in Shade (°F)			Maximum Temperatures in Sun (°F)			Maximum Pressure in Sun ^(c) (psig)
		Inside Inner Container ^(b)	Inner Container ^(b) Surface ^(b)	Drum Exterior	Inside Inner Container ^(b)	Inner Container ^(b) Surface ^(b)	Drum Exterior	
12	30	514	300	112	550	336	148	20.7
55	42	403	300	106	439	336	142	16.8

(a) Calculated at 100°F ambient, loaded with maximum contents heat load at -40°F.

(b) Add 30°F for Auratone Firecode insulation.

(c) Add 1 psig for Auratone Firecode insulation.

2.6.1.2. Differential Thermal Expansion. As stated in section 3.4.5., temperature differences within the packages under normal conditions of transport are small, and the stress from differential thermal expansion is negligible.

2.6.1.3. Stress Calculations. Differential thermal expansion under normal conditions of transport is small. Therefore, the resulting stresses are insignificant. In addition, the inner containers were heated during the steady state tests to temperatures higher than those that will be reached in direct sun as reported in section 3.4.2. No degradation of package performance was detected in these tests.

2.6.1.4. Comparison With Allowable Stresses. Stresses under normal conditions of transport will be negligible compared with the design criteria of section 2.1.2.

2.6.2. Cold

An ambient temperature of -40°F in still air and shade will not decrease the effectiveness of the packaging. This would reduce the temperature profile within the packages and would possibly be beneficial.

Even if the package is empty, and the temperature of the inner container falls as low as -40°F , brittle fracture will not be a problem. It is well known that will not be a problem because problems with brittle fracture increase as the size and thickness of the specimen increases [8]. It has been observed in large ships with heavy plating, although it has not affected small

ships with thin plating. In the small containers dealt with in this report, brittle fracture is not anticipated.

Reference 9 specifies fracture toughness requirements and criteria for protecting against brittle fracture. According to Category II requirements, a large margin of safety can be ensured without special testing of steel that is less than 0.19 in. thick [9]. Therefore, the outer containers of both packages should be free from the effects of brittle fracture.

The small size of the inner containers should also prevent problems of brittle fracture, even though the wall thicknesses are greater than 0.19 in. For added safety, full-scale containers were subjected to the 30-ft drop test. The inner containers of these packages were chilled to at least -40 F by dry ice. The containers used had been in service for some time, and would be expected to contain some flaws. Nevertheless, the inner containers of both packages were completely undamaged after the 30-ft drop (see section 2.7.1.3).

2.6.3. Reduced External Pressure

A reduced atmospheric pressure of 3.5 psia is well within the capability of the inner containers. This pressure is equivalent to an increased internal pressure of 11.2 psi above the maximum normal operating pressures calculated in section 3.4.4. The resulting internal pressures are summarized in Table 2-7. These pressures are well below the design criteria as determined in section 2.1.2.

TABLE 2-7
EQUIVALENT PRESSURES RESULTING FROM REDUCED EXTERNAL PRESSURE
OF 3.5 psia^(a)

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	Maximum Internal Pressure in Direct Sun (psig) ^(b)	Equivalent Pressure Rise at 3.5 psia Ambient (psi)	Equivalent Internal Pressure (psig) ^(b)
12	30	20.7	11.2	31.9
55	42	16.8	11.2	28.0

(a) Inner containers at ambient temperature of 100°F, loaded at -40°F.

(b) Add 1 psig for Auratone Firecode insulation.

2.6.4. Increased External Pressure

The containment vessel must be capable of withstanding an external pressure of 25 psi without any loss of contents. Conservatively, it is assumed that no loss of contents will result if the allowable pressure of the inner container body material is not exceeded and if local buckling does not occur even though these conditions would not necessarily cause the inner container to be breached. The inner container of the 55-gal package is constructed of mild steel with a maximum allowable stress of 15,000 psi, as specified per ASME Pressure Vessel Code. Sample calculations are presented for the 55-gal package, and the results are tabulated for both packages.

First, the allowable pressures for the inner container cover and the circular bottom end plate are considered. The actual boundary conditions for the top and bottom plate lie somewhere between fixed and simply supported. The top cover plate is bolted to a flanged body and is assumed to have simply

supported edges. The bottom plate is welded to the container body, and the edge of the plate is considered to be fixed. The allowable pressure loads for uniformly loaded circular plates are given by:

$$P_{\text{allow}} = \frac{4}{5} S_{\text{max}} \left(\frac{t}{R} \right)^2 \text{ for simply supported (top cover plate)}$$

$$P_{\text{allow}} = 2 S_{\text{max}} \left(\frac{t}{a} \right)^2 \text{ for fixed edge (bottom plate)}$$

where: P_{allow} = the allowable external working pressure (psi),
 S_{max} = maximum allowable stress, $S_{\text{max}} = 15,000$ psi,
 t = thickness of plate, $t = 0.44$ in. for top cover plate,
 and $t = 0.25$ in. for the bottom plate,
 R = outer radius of top cover plate, $R = 3.50$ in.,
 a = inner radius of bottom plate, $a = 3.25$ in.

The maximum allowable pressure for the top cover plate is:

$$P_{\text{allow}} = \frac{4}{5} \times 15,000 \times \left(\frac{0.44}{3.5} \right)^2$$

$$P_{\text{allow}} = 190 \text{ psi,}$$

and the maximum allowable pressure for the bottom plate is:

$$P_{\text{allow}} = 2 \times 15,000 \times \left(\frac{0.25}{3.25} \right)^2$$

$$P_{\text{allow}} = 177.5 \text{ psi.}$$

The third consideration is the buckling strength of the inner container. The allowable external pressure for the vessel is computed using the procedures specified in the ASME Pressure Vessel Code [1], which provides

conservative critical pressures. The Code states that the allowable external pressure is given by the expression:

$$P_{\text{allow}} = 4B/3(D_o/t),$$

where: D_o = outside diameter of the vessel, $D_o = 7.0$ in.,
 t = thickness of vessel, $t = 0.25$ in.,
 L = length of vessel, $L = 15.81$ in.,
 B = constant depending on the ratios L/D and D_o/t ,
 and for the 55-gal package inner container equals
 10,300.

For the 55-gal package the allowable pressure load is:

$$P_{\text{allow}} = \frac{4 \times 10,300}{3(7.0/0.25)}$$

$$P_{\text{allow}} = 490.5 \text{ psi.}$$

The required data and the results of the external pressure capability calculations for both sizes are summarized in Table 2-8.

TABLE 2-8
 SUMMARY OF EXTERNAL PRESSURE CAPABILITY DATA

<u>Required Data</u>	<u>Drum Size</u>	
	<u>12-gal</u>	<u>55-gal</u>
R, top cover, in.	2.50	3.50
t, top cover, in.	0.88	0.44
R, bottom plate, in.	2.06	3.313
t, bottom plate, in.	0.50	0.25
R, body, in.	2.50	3.50
t, body, in.	0.44	0.25
D, body, in.	5.00	7.00

(Cont.)

TABLE 2-8 - Continued

<u>Required Data</u>	<u>Drum Size</u>	
	<u>12-gal</u>	<u>55-gal</u>
L, body, in.	11.81	15.81
L/D, body	2.36	2.26
D/t, body	11.4	28
B, psi	12,500	10,300
Material Allowable Stress	15,000	15,000
<u>Calculated Maximum Allowable Pressures</u>		
Maximum Pressure for Top Cover Plate	2,190	190
Maximum Pressure for Bottom Plate	1,767	177.5
Maximum Pressure in Body	1,467	490.5
<u>Calculated External Pressure Capabilities (psi)</u>		
Minimum Required	25	25
ASME Allowable	1,467	177.5

2.6.5. Vibration

Vibration normally incident to the transport will not reduce the effectiveness of the packaging. The capability of these packages to withstand normal vibration is well established as a result of routine use and special tests that were performed for similar packages. Since 1968, over 461 shipments have been successfully completed with no evidence of damage from vibration. Also, the AL-M1 shipping container, which is similar in design, was subjected to laboratory vibration tests that did not cause any damage [9A].

2.6.6. Water Spray

A water spray sufficiently heavy to keep the entire exposed surface of the package, except the bottom, continuously wet during a period of 30 min will

not damage any of the packages in any way or have any effect other than slight cooling, on the contents. The packages are actually exempt from this test requirement since the external surfaces are of all metal construction and the vent holes are sealed with waterproof tape.

2.6.7. Free Drop

A free drop through a distance of 4 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected, would not reduce the effectiveness of the packaging.

This test was performed by Dow Chemical Co. for a 10-gal DOT-6C container [10] and for a 30-gal DOT-17H container [11], and by the University of California for a 55-gal container of 16-gauge wall thickness [12] with only minor damage to the drums.

In addition, when the 5790 and 5791 packages were subjected to the 30-ft drop test described in section 2.7.1., the subsequent water immersion test for fissile material (section 2.7.4.) resulted in no water in-leakage to the inner containers. These drop-tested containers were also pressurized to 20 psig with helium and reassembled using the original O-rings (section 2.7.6.). Helium leak testing showed no leaks greater than 1×10^{-4} std cm³/s for either package.

It is concluded that there would be no loss of radioactive contents greater than 10^{-6} A₂/hr, no significant increase in external radiation

levels, and no substantial reduction in the effectiveness of the packaging under these hypothetical accident test conditions; therefore, the packages will also meet these requirements under the less destructive normal conditions of transport.

2.6.8. Corner Drop

This test requires a free drop onto each corner of the package in succession, or, in the case of a cylindrical package, onto each quarter of each rim, from a height of 1 ft onto a flat, essentially unyielding, horizontal surface. This test applies only to packages that are constructed primarily of wood or fiberboard, and do not exceed 110-lb gross weight, and to all Fissile Class II packagings.

This test is applicable even though the packages are of metallic construction and the 55-gal size weighs in excess of 110 lb since the shipping containers can be used for Fissile Class II shipments.

This test was performed for a 10-gal DOT-6C container [10] and for a 30-gal DOT-17H container [11] by Dow Chemical Co. with only minor damage to the drums. The one foot corner drop is less severe than the four foot drop that was discussed in the previous section and would not damage the packages in any way so as to reduce the volume or effectiveness. As discussed above, these packages satisfied the requirements for normal conditions of transport even after being subjected to the sequence of four hypothetical accident tests. It is thus concluded that they would meet these requirements after the one foot corner drop.

2.6.9. Compression

This test requires a compressive load equal to either five times the weight of the package or 2 psi multiplied by the maximum horizontal cross section of the package, whichever is greater. The load must be applied during a period of 24 hr, uniformly against the top and bottom of the package, and in the position in which the package would normally be transported.

Previous testing by others indicates that the 55-gal size satisfies the compression requirements. Dow Chemical Co. [13] tested a package using two 55-gal drums welded together with 5,250 lb to qualify it for a 1,050-lb maximum gross weight, which far exceeds the 300-lb maximum gross weight of the 55-gal AL-M6 package.

The aforementioned testing has been supplemented with evaluations of both sizes and a sample calculation is provided for the 55-gal size*. The evaluation is based on a load of 2,000 lb, which is five times the assumed gross weight of 400 lb. The alternate criteria yield a value of only 804 lb. The wall thickness (t) of the 55 gal drum is 0.0598 in., and the outside diameter of the drum is 22.56 in. The longitudinal compressive stress (S) is calculated by dividing the load by the cross-sectional area of the drum wall as follows:

* The 55-gal drum was shown in Figure 1-5.

$$S = 2,000/\pi Dt$$

where, D is the diameter and t is the wall thickness,

The result is:

$$S = 2,000/\pi(22.62) (0.0598)$$

$$S = 471 \text{ psi.}$$

This stress value is only 2% of the yield stress which is 27,000 psi for mild steel. The results for both sizes are shown in Table 2-9. The bending stresses that occur in each of the drum rolling hoops are also determined.

TABLE 2-9
SUMMARY OF DRUM DIMENSIONS AND RESULTS OF COMPRESSION CALCULATION

Nominal Size Drum (gal)	Drum Diam. (in.)	Drum Wall Thick. (in.)	Rolling Hoop Depth (in.)	Compressive Load Requirement (lb)	Resulting Longitudinal Compressive Stress (psi)	Resulting Rolling Hoop Bending Stress (psi)	Ultimate Critical Buckling Stress Capability (psi)
12	13.85	0.0359	0.375	750	480	30,100	94,100
55	22.62	0.0598	0.625	2000	471	29,500	96,000

For a conservative approximation, the drum is assumed to act as a column rather than a shell; that is, the vertical load is supported axially by the drum's peripheral thickness. The bending moment and maximum bending stress are then found. The longitudinal stress resultant (S) for the 55-gal drum is given by:

$$S = \frac{2,000}{\pi D} = \frac{2,000}{\pi(22.62)} = 28.1 \text{ lb/in.}$$

The bending moment (M) at the hoop is then,

$$\begin{aligned} M &= S(e) = 28.1 (0.625 \text{ in.}) \\ &= 17.6 \text{ in.-lb/in.,} \end{aligned}$$

where e is the depth of the rolling hoop. The maximum bending stress is:

$$S_{\max} = \frac{6M}{t^2} = \frac{6(17.6)}{(0.0598)^2} = 29,530 \text{ psi.}$$

The above stress exceeds the 27,000 psi yield strength indicating that such large compressive loadings will yield the material. Such yielding, however, will be localized at the outer surface of the rolling hoop. The cross section of the drum is still capable of resisting a collapse because of this loading, and the drums are considered to adequately satisfy this requirement. The results of the calculations for both sizes are shown in Table 2-9.

The ultimate capability of each container is verified by considering the critical buckling stress of the cylindrical shell when subjected to uniform axial compression.

A sample calculation is provided for the 55-gal size. The critical buckling stress (S_{cr}) is given by the following equation (Reference 5, Table XVI, Case 7):

$$S_{cr} = \frac{Eh}{\sqrt{R^3(1-\mu^2)}},$$

where: E = modulus of elasticity, $E = 30 \times 10^6$ psi,
 h = drum wall thickness, $h = 0.0598$ in.,
 R = radius of drum, $R = 11.31$ in., and
 μ = poisson's ratio, $\mu = 0.3$.

Thus, the critical buckling stress is:

$$S_{cr} = \frac{30 \times 10^6}{11.31} \frac{(0.0598)}{\sqrt{3(1-0.09)}} = 96,000 \text{ psi.}$$

This value for the ultimate capability would never be reached in the drum. The critical buckling stresses for both sizes are shown in Table 2-9.

In summary, all the resulting longitudinal compressive stresses in the drum body are far below the 27,000 psi material yield stress. On the other hand, the resulting rolling hoop bending stresses are sufficiently high for the 12-gal and 55-gal sizes to cause localized yielding, but will not cause the drums to collapse. The ultimate critical buckling stress capabilities far exceed any stresses that could be developed from the compression tests. Thus, the drums are considered to satisfy the compression test requirement.

2.6.10. Penetration

It is necessary to evaluate the impact of the hemispherical end of a vertical steel cylinder, 1-1/4 in. in diameter and weighing 13 lb, dropped from a height of 40 in. onto the exposed surface of the package which is expected to be most vulnerable to puncture.

This test causes minor damage to drum surfaces but does not penetrate them. This fact was demonstrated by a test performed at Mound Facility [14]. This test was actually more stringent than required, as a weight of 13.2 lb was dropped from a height of 66 in. onto both the lid and the curved surface of a 55 gal drum with a wall thickness of 0.0478 in. This wall thickness is thinner than the 0.0598-in. wall thickness of the specified DOT 17C or 6C drum.

Figure 2-4 shows the rod suspended 66 in. above the drum lid. In Figure 2-5, the rod is shown impacting the drum lid. The rod is suspended 66 in.

FIGURE 2-4 - Rod suspended above drum lid for penetration test. (Photo No. 841009)

FIGURE 2-5 - Steel rod impacts drum lid during penetration test. (Photo No. 841007)

above the curved surface of the drum in Figure 2-6. In Figure 2-7, the rod is approaching the drum surface. In all tests, deformation of the drum surface was less than 0.375 in. and penetration did not occur. Consequently, the 5790 packages fulfill the requirements of this test.

A 10-gal DOT-6C and a 30-gal DOT-17H package were also tested by Dow Chemical Co. [10,11]. In every case, the test caused only minor damage to the drum and no penetration. The adequacy of the 55-gal drum was further substantiated by an engineering evaluation that compares the stress imposed by the dropped rod striking the drum to the ultimate strength of the drum material [15].

In this analysis, the impact velocity of the rod was compared to the ballistic limit velocity. The ballistic limit velocity (V_1) is defined as the lowest striking velocity that will result in perforation of the target with zero residual velocity. It can be estimated using the following equation (Reference 16, pp. 6-87):

$$V_1 = \frac{A_o}{W_s} A_p^m (t \sec \theta)^n,$$

where A_o , m , and n are empirical constants and:

W_s = striking weight, lb.

A_p = fragment area presented to plate, in.²

t = plate thickness, in.

θ = angle of obliquity

FIGURE 2-6 - Rod suspended above curved surface of drum for penetration test. (Photo No. 841011)

FIGURE 2-7 - Rod approaches curved surface of drum in penetration test. (Photo No. 841010)

The length of the dropped rod is 38.14 in. Therefore the length to diameter ratio $L/D = 30.5$. Then from Table 6.11, page 6-88 of Reference 16, for penetration of mild steel plates by compact steel fragments, $A_0 = 1261$, $m = 0.427$, and $n = 0.647$. This gives the ballistic limit velocity:

$$V_1 = \frac{1261}{13.2 \text{ lb}} [\pi \times (0.625 \text{ in.})^2]^{0.427} \times [(0.0598 \text{ in.})(\sec 0^\circ)]^{0.647} \\ = 61.22 \text{ ft/s}$$

The final velocity (V_f) of a steel rod dropped from a height of 40 in. is calculated by the equation:

$$V_f^2 = V_i^2 + 2 a h,$$

where:

- V_i = initial velocity, ft/s.
- a = acceleration of gravity, ft/s^2 .
- h = height of drop, in.

Then:

$$V_f = \sqrt{2 \times 32.2 \frac{\text{ft}}{\text{s}^2} \times \frac{40}{12} \text{ ft}} \\ = 14.65 \text{ ft/s}$$

The actual final velocity is smaller than the ballistic limit velocity by a factor of 4.2. Therefore, penetration of the drum will not occur.

2.7. Hypothetical Accident Conditions

10 CFR 71 requires satisfactory performance of packaging when the shipping container is subjected to a series of four tests simulating accident

conditions. Escape of radioactive materials must be below defined limits, and the package must remain subcritical. The free drop, puncture, thermal, and water immersion tests must be performed in the listed sequence.

Full-scale 12-gal and 55-gal shipping containers were subjected to the complete series of four hypothetical accident tests. All test inner containers were packaged using lead shot to simulate maximum contents weight. Chromel/Alumel thermocouples were used for temperature measurements inside the inner containers. In addition, temperature sensitive paints were applied to the inside and outside walls of the inner containers. The paints and labels, which were color coded, were formulated to melt at 200°F (93°C), 300°F (149°C), 400°F (204°C), and 500°F (260°C). Each metal drum had a Chromel/Alumel thermocouple attached to its outer surface and one thermocouple placed near the top surface of the drum to measure flame temperature.

On final assembly of the packages, the bolt used to fasten the closure ring in place was aligned with the seam of the drum so this could be identified as the weakest part of the drum in a free-drop test.

2.7.1. Free Drop

This test requires a free drop through a distance of 30-ft onto a flat, essentially unyielding, horizontal surface. The package is positioned to strike the surface in a position for which maximum damage is expected.

The adequacy of the 5790 and 5791 packages under the conditions of the 30-ft drop test is demonstrated by a combination of analysis, prototype testing, and comparison to similar packages. These packages are typical of many cylindrical packages that have been tested and approved for shipment of Type B and fissile radioactive material, as well as spent fuel. At Mound Facility, the 5790 and 5791 packages were drop tested only in the corner orientation, but a large volume of data from tests performed elsewhere indicates that this is the most damaging orientation.

Casks for spent fuel, in particular, have been extensively tested to determine the drop orientation that produced the most damage. At Oak Ridge, 1.3-ton and 6-ton steel-shelled, lead-filled cylindrical casks were dropped in several orientations [17]. The investigators concluded that corner cask drops

onto welded zones produced the most damaging impact results. This is because the effects are localized and the maximum stress occurs at points that may be weaker than the surrounding metal plate; weld cracks appeared for drops of 15 ft or greater.

A study performed by the Franklin Institute Research Laboratories also concentrated on identifying more vulnerable impact attitudes for shipping casks [18]. One-tenth scale models of 15-ton steel-shelled, lead (or steel) filled cylindrical casks were tested at impact velocities and energies of approximately 30 miles per hour and 1000 ft-lb, respectively.

The results of this study agreed well with the testing of full-scale models at Oak Ridge. Once again, impacts on edges and corners were found to cause more extensive deformation than impacts on end or side walls. An impact on the edge of the closure region proved to be particularly damaging. However, even though impact energies in tests with partial-cask models reached values much higher than those specified in government regulations, in no case was there an obvious cask failure.

More comparable to the 5790 and 5791 packages are several drum-type packages tested for shipment of Type B quantities. An example is the DOT 17H, 30-gal package tested at the Rocky Flats Plant [19]. This package passed the 30-ft drop test in upright, flat, and corner orientations.

At Lawrence Livermore National Laboratory, a drum 28-3/4 in. o.d. by 28-1/4 in. high was tested in both 14-gauge and 16-gauge wall thicknesses [12]. Both drums were dropped on their rims for the 30-ft drop test. The

locking rings and rims of both drums were noticeably dented after the drop test. In addition, the ring on the 16-gauge package popped off because of insufficient tightening. Packages 5790 and 5791 use a modified closure in which a skirt is welded to the locking ring, to avoid this problem.

Another package tested at the Rocky Flats Plant is a 10-gal DOT 6C steel drum [10]. In this drop test, the package was dropped onto the bolt of the locking ring. The drop caused a 3-1/4 in. by 2-1/2 in. dent on the rim, but the locking ring held. In addition, this drum was dropped on its side, causing a decrease of 0.2 in. in its effective radius.

In another report, reserchers at Rocky Flats concluded that drum-type packages are most likely to break open if the impact is on the rim of the cover at the locking ring joint [11]. Four 55-gallon DOT 17C drums with 16-gauge wall thicknesses were tested in this fashion, and all remained intact. Additionally, a 30-gal DOT 17H drum was dropped on its side, reducing its effective diameter by less than 1/4 in.

Lawrence Livermore National Laboratory also tested a steel drum, 25 in. o.d. by 55 in. high, of 16-gauge wall thickness [20]. Five packages were dropped on the edge of the covers, one on the side, and one on both the edge and the side. The edge drop caused a 3 in. dent, but none of the drops impaired the leak-tightness of the containment vessel.

At the Savannah River Plant, drums of 30, 35, and 55 gal capacity, with 18-gauge wall thickness, were drop-tested on the bottom and the edge [21]. Researchers concluded that a drop on the edge of the closure region was most

damaging. When dropped in this orientation, a 3-3/4-in. deformation was produced; but the package remained intact.

At the Oak Ridge National Laboratory, three packages based on a 55-gal DOT 17H container of 18-gauge wall thickness were tested [22,23,24]. In the free drop test, the edge striking the surface was deformed, the ring clamp was dented, but the package remained intact. Package volume was reduced by about 1%.

Other packages tested at Oak Ridge include a 20-gal DOT 37A container of 20-gauge wall thickness [25], two 30-gal DOT 17H drums [26,27], and two packages fabricated from two 55-gallon DOT 17H drums of 16-gauge wall thickness, welded together bottom to top. In every case, when these packages were drop-tested, damage was minimal and the packages remained intact.

2.7.1.1. End Drop. The end drop test was not performed for these packages, as the corner drop test was determined to be the most damaging. In addition, a similar 30-gal DOT 17H package passed this test when performed at the Rocky Flats Plant [19]. Thirty and 55-gal packages passed this test at the Savannah River Plant [21] and at the Oak Ridge National Laboratory [23,27].

2.7.1.2. Side Drop. The side drop test was not performed for these packages as the corner drop test was determined to be the most damaging. In addition, a similar 30-gal DOT 17H and a similar 10-gal DOT 6C package passed

this test when performed at the Rocky Flats Plant [10,19]. Also, a 25 in. o.d. by 55 in. high package of 16-gauge wall thickness passed this test at the Lawrence Livermore National Laboratory [20].

2.7.1.3. Corner Drop. A specially designed, 50-ft high drop tower equipped with a 2-ton hoist was used to drop the packages from a height of 30 ft onto a steel-covered concrete drop pad. The packages were oriented upside down at a 45° angle so that the bolt on the bolt ring would strike the pad first as shown for the 55-gal package in Figure 2-8.

Figure 2-9 shows the container at the required 30-ft height just prior to manual actuation of the quick release hook. Figure 2-10, taken just prior to impact, shows that the container was dropped in precisely the initial orientation since no twisting motion was imparted to it by the quick release hook.

In a separate test, the effect of low temperatures on the package's inner containers was studied in a 30-ft drop test. The inner containers were chilled to at least -40°F by dry ice. Temperatures measured between -40 and -58°F both before and after the drop test. (Note that for hypothetical accident tests, a minimum temperature of -20°F is the lowest required by regulations.)

2.7.1.4. Oblique Drops. Oblique drop tests were not performed for these packages, as the corner drop test was determined to be the most damaging.

FIGURE 2-8 - Prior to the 30 ft drop, the 55 gal package is attached to a hoist. (Photo No. 750517)

FIGURE 2-9 - The 55 gal container at the requisite 30 ft height
(Photo No. 750514)

FIGURE 2-10 - The 55 gal drum prior to impact. (Photo No. 750518)

2.7.1.5. Summary of Results. NRC Regulatory Guide 7.9 requires, at a minimum, that drop orientations be considered in which the center of gravity is directly over the point of impact. In the corner drop tests performed at Mound, the shipping containers were dropped at an angle of 45°. This angle did not exactly position the center of gravity over the point of impact. The angle, α , at which the center of gravity is directly over the point of impact has been calculated, and is larger than 45° in both cases.

The effect of having this low angle of impact is to reduce the force of impact to:

$$F = \cos (\alpha - 45^\circ) F_{\max},$$

where F is the actual force and $F_{\max}$ would be the force if the center of gravity were directly over the point of impact. As shown in Table 2-10, the force is reduced by a very small amount. Table 2-10 also shows the equivalent weight of a package dropped with its center of gravity over the point of impact, that would fall with the same force.

TABLE 2-10
EQUIVALENT WEIGHT OF PACKAGES IF
DROPPED WITH CENTER OF GRAVITY OVER POINT
OF IMPACT

Nominal Package Size (gal)	Tested Weight (lb)	Angle to Position Center of Gravity Over Point of Impact(α)	Fraction of Maximum Impact Force at 45°	Equivalent Weight If Center of Gravity Over Point of Impact(lb)
12	87.5	55.7°	0.98	85.8
55	304	53.4°	0.99	301.0

The containers passed the drop tests to the extent that the drum covers remained intact and the inner containers did not suffer any observable damage. Modified closures were used since both sizes previously passed only marginally using standard drum closures.

In the 30-ft drop tests using inner containers at -40°F, the inner containers were completely undamaged. There was no evidence of any brittle failure in either inner container. The inner containers are shown after this 30-ft drop test in Figures 2-11 and 2-12. The 5790 container held a sealed tinplate food pack can during this test. (See Figure 2-11.) This food pack can was also completely unscathed after the 30-ft drop. We conclude that brittle fracture will not be a problem with these packages.

2.7.2. Puncture

This test requires a free drop through a distance of 40 in. striking (in such a position that maximum damage is expected) the top end of a vertical, cylindrical, mild steel bar mounted on an essentially unyielding horizontal surface. The bar must be 6-in. in diameter, with the top horizontal and the edge rounded to a radius of not more than 1/4 in.; and the bar must be long enough to cause maximum damage to the package, but not less than 8-in. long. The long axis of the bar must be perpendicular to the unyielding horizontal surface.

This test was conducted in a manner similar to the Free-Drop Test. Figure 2-13 shows the 55-gal package suspended 40 in. above the top of the 6-in. diameter cylinder. Figure 2-14 shows the container impacting on the

FIGURE 2-11 - Package 5790 inner container and food pack can following 30 ft drop at -40°F.

FIGURE 2-12 - Package 5791 inner container following 30 ft drop at -40°F.

FIGURE 2-13 - The 55 gal drum in place for puncture test.
(Photo No. 750512)

FIGURE 2-14 - Puncture test impact.

cylinder. The puncture test caused only a minor dent in the drum of the original type 5791 shipping container that was tested.

2.7.3. Thermal

2.7.3.1. Summary of Pressures and Temperatures. Limiting temperatures and pressures during the thermal accident test, as determined in section 3.5., are summarized in Table 2-11.

TABLE 2-11
MAXIMUM TEMPERATURES AND PRESSURES
HYPOTHETICAL ACCIDENT THERMAL TEST^(a)

<u>Nominal Package Size (gal)</u>	<u>Maximum Temperatures (°F)</u>			<u>Maximum Internal Pressure^(c) (psig)</u>
	<u>Drum Exterior</u>	<u>Outside Surface Inner Container^(b)</u>	<u>Inside Inner Container^(b)</u>	
12	1612	433	640	23.8
55	1606	442	536	20.2

(a) Ambient temperature 100°F, loaded with maximum contents heat load at -40°F.

(b) Add 30°F for Auratone Firecode insulation.

(c) Add 1 psig for Auratone Firecode insulation.

2.7.3.2. Differential Thermal Expansion. Based on the maximum pressures in Table 2-11, deformations and stresses were calculated on an IBM personal computer for both packages. The software program used is called "Stresspac" [28]. These calculations are summarized in Tables 2-12 and 2-13. The axial deformation caused by thermal expansion for each package is listed in Table 2-14. The program input/output is reproduced in section 2.10.3. The calculations were verified by application of the equations of section 2.10.1. to give the same results.

TABLE 2-12
STRESSES AND DEFORMATIONS
HYPOTHETICAL ACCIDENT THERMAL TEST - 23.8 PSIG

	<u>Package 5790</u>	
	<u>Inner Surface of Cylinder</u>	<u>Outer Surface of Cylinder</u>
Radial Stress	23.8 psi	0
Hoop Stress	125.2 psi	101.4 psi
Axial Stress	50.7 psi	50.7 psi
Radial Displacement	9.23×10^{-6} in.	8.60×10^{-6} in.
Factor of Safety	116.2	170.7

TABLE 2-13
STRESSES AND DEFORMATIONS
HYPOTHETICAL ACCIDENT THERMAL TEST - 20.2 PSIG

	<u>Package 5791</u>	
	<u>Inner Surface of Cylinder</u>	<u>Outer Surface of Cylinder</u>
Radial Stress	20.2 psi	7.63×10^{-6} psi
Hoop Stress	273.1 psi	252.9 psi
Axial Stress	126.4 psi	126.4 psi
Radial Displacement	3.07×10^{-5} in.	3.00×10^{-5} in.
Factor of Safety	59.1	68.5

TABLE 2-14
AXIAL DEFORMATION CAUSED BY THERMAL EXPANSION
HYPOTHETICAL ACCIDENT THERMAL TEST

<u>Package</u>	<u>Axial Deformation (in.)</u>
5790	0.044
5791	0.055

The following equation was used for the calculation of axial deformation (Reference 29, pp. 4-7):

$$\Delta l = l_1 \alpha \Delta t$$

where:

Δl = axial deformation
 l_1 = original length
 α = coefficient of linear thermal expansion
 Δt = temperature change.

Axial deformation is very small in every case.

2.7.3.3. Stress Calculations. The stresses caused by pressure are summarized in Tables 2-12 and 2-13. Temperature differences and the resulting secondary thermal stresses within the inner containers are not excessive under the requirements of 10 CFR 71.

2.7.3.4. Comparison with Allowable Stresses. Factors of safety calculated for each package are listed in Tables 2-12 and 2-13. As can be deduced from these safety factors, all stresses are well within the maximum allowable stresses for the materials of construction. Package performance is further confirmed by successful completion of the hypothetical accident tests performed on these containers.

2.7.4. Immersion - Fissile Material

This test is necessary for fissile material packages only. The test requires immersion in water to the extent that all portions of the package to be tested are under at least 3 ft of water for a period of not less than 8 hr.

A permanently installed, 10-ft diameter by 9-ft deep tank, equipped with a 2-ton hoist, was used. Prior to this test, the inner containers were removed from the insulated drum assemblies. The tank was filled to a depth of 51 in. to ensure immersion of all parts of the containers under at least 36 in. of water. Both inner containers were immersed overnight for 20 hr, since it was not convenient to withdraw them after the required 8-hr minimum period.

The 12-gal and 55-gal sizes passed the immersion tests with no evidence of water leakage into the inner containers. The inner containers were removed from the insulated drum assemblies prior to this test. In addition, after the immersion test, both inner containers were reassembled, using the same O-rings, and found to be helium leak tight at an instrument sensitivity of 10^{-4} std cm^3/s when containing 20 psig of helium pressure.

2.7.5. Immersion - All Packages

10 CFR 71.73(c)(5) requires that an undamaged specimen of each shipping container be subjected to water pressure equivalent to immersion under 15 m (50 ft) of water for a minimum of 8 hr. For the purposes of this regulation, a gauge pressure of 21 psig is considered to meet these conditions.

These tests were performed for the 5790 and 5791 packages by Bowser-Morner, Inc., Product and Materials Testing Laboratory, Dayton, Ohio. For these tests, the inner containers were removed from the package and immersed in a pressure vessel filled with water at ambient temperature [30]. The pressure regulator was adjusted to obtain a pressure of 21 psig, and this pressure was monitored for a period of at least 8 hr. At this time, the pressure was released and the shipping container was removed from the pressure vessel. After air-drying, the inner containers were disassembled and inspected. No trace of water was found on the inside of the containers, showing that they successfully met the requirements of this regulation.

2.7.6. Summary of Damage

The results of the hypothetical accident tests are summarized in Table 2-15. The drop and puncture tests produced only minor dents in the outer drums, leaving the inner containers undamaged. Both containers also survived the thermal test. The inner containers did not bulge or deform in any way, although the Firedike drum insulation was charred. The silicone O-ring seals were not damaged in any way. Therefore, the containers remained leak tight.

TABLE 2-15
RESULTS OF HYPOTHETICAL ACCIDENT TESTS

<u>Drum Size (gal)</u>	<u>Total Package Weight (lb)</u>	<u>30-ft Drop Results</u>	<u>40-in. Puncture Results</u>	<u>Maximum Temperature of Inner Container After 30 min Fire (°F)</u>	<u>Immersion Test Results</u>
12	87	Modified Cover Com- pletely in- tact (3-in. dent)	Minor Dent	237	No Leakage
55	304	Modified Cover Completely Intact (4-in. dent)	Minor Dent	244	No Leakage

Leak tightness was confirmed by the results of the 3-ft water immersion test. Both containers passed this test with no evidence of water leakage into the inner containers. In addition, after the immersion test, both inner containers were reassembled, using the same O-rings, and found to be helium leak tight at an instrument sensitivity of 10^{-4} std cm³/s when containing 20 psig of helium pressure. Therefore the integrity of the containment was not compromised in any way.

2.8. Special Form

Not applicable.

2.9. Fuel Rods

Not applicable.

2.10. Appendix

2.10.1. Internal Pressure Capability from Section 2.1.2.

2.10.1.1. General. The internal pressure capabilities of the inner containers were thoroughly evaluated. The evaluation clearly demonstrates that the vessels are capable of safe pressure containment during normal transportation and hypothetical accident conditions.

Sample calculations for the internal pressure capability of the type 5791 inner container are presented, and the results for both packages at

temperatures up to 650°F are then tabulated. A separate calculation is required for each of the four basic components, which are the cylindrical body, the welded bottom, the bolted cover, and the flange. The maximum pressure capability is then based on the values for the welded bottom since these are the lowest values for the four components.

2.10.1.2. Body Calculations. The inner container for the type 5791 package is a cylinder with a welded bottom plate and a bolted cover sealed with a 1/8 in. diameter cross section silicone O-ring. It is fabricated of mild steel, and the welds are examined after fabrication (see section 8). The maximum allowable working pressure of the seamless tubing body is calculated according to the following:

$$P = \frac{SEt}{R + 0.6t} \text{ (Ref. 1, p.19),}$$

where: P = maximum allowable working pressure, psi,
S = maximum allowable stress, S = 15,000 psi at -20 to 650°F,
E = welded joint efficiency, E = 1 for seamless tubing,
R = inside radius of inner container, R = 3.25 in.,
t = wall thickness, t = 0.25 in.

Substitution of the appropriate values into the above equation yields:

$$P = \frac{(15,000) (1) (0.25)}{[(3.25) + (0.6) (0.25)]}$$

$$P = 1103 \text{ psi.}$$

The required parameters and the results of the seamless body calculations for both packages are shown in Table 2-16.

TABLE 2-16
INNER CONTAINER BODY PRESSURE CAPABILITY CALCULATIONS

<u>Package Type</u>	<u>Drum Size (gal)</u>	<u>Inside Radius (in.)</u>	<u>Body Wall Thickness (in.)</u>	<u>Internal Pressure Capability (psig) @ -20 to 650°F</u>
5790	12	2.06	0.44	2840
5791	55	3.25	0.25	1103

2.10.1.3. Welded Bottom Plate Calculations. The inner container bottom for the type 5791 package is welded to the body. Its pressure capability is calculated as follows:

$$P = \frac{SE}{C} \left(\frac{t}{2R} \right)^2 \quad (\text{Ref. 1, p. 30}),$$

where: P = maximum allowable working pressure, psi,

S = maximum allowable stress, S = 15,000 psi,

C = factor accounting for method of attachment, C = 0.3,

d = inside diameter of inner container, d = 6.50 in.,

t = thickness of bottom plate, t = 0.25 in.

E = welded joint efficiency, E = 0.5.

The resulting calculation is as follows:

$$P = \left(\frac{15,000 \times 0.5}{0.3} \right) \left(\frac{0.25}{6.50} \right)^2$$

$$P = 37 \text{ psi.}$$

The required parameters and the results of the welded bottom plate calculations for both sizes are shown in Table 2-17.

TABLE 2-17
INNER CONTAINER WELDED BOTTOM
PLATE PRESSURE CAPABILITY CALCULATIONS

<u>Package Type</u>	<u>Drum Size (gal)</u>	<u>Inside Radius (in.)</u>	<u>Body Wall Thickness (in.)</u>	<u>Internal Pressure Capability (psig) @ -20 to 650°F</u>
5790	12	2.06	0.50	220.4
5791	55	3.25	0.25	37.0

2.10.1.4. Bolted Cover Calculations. The type 5791 inner container has a bolted cover and a silicone O-ring closure. The type 5790 inner container has, instead, a screwed-on cover with an O-ring seal. Since the cover is much thicker than the bottom plate, it is not the limiting factor and no additional calculations are required for the type 5790. A sample calculation is presented for the type 5791.

The calculation for the 55-gal inner container cover is similar to that for the bottom, except the edge moment must be accounted for because it is attached by bolts. The applicable equations are as follows:

$$P = \left[\frac{SE}{C} \left(\frac{t}{d} \right)^2 - \frac{1.9 W h}{SE (d)^3} \right] \text{ (Ref. 1, p. 30),}$$

$$W = H + H_p, \text{ (Ref. 1, p. 322), and}$$

$$H = \frac{\pi}{4} d_o^2 P,$$

where: P, S, C, t, and d are as defined above for the inner container bottom, and

E = welded joint efficiency, E = 0.55,

W = total bolt load, lb,

H = hydrostatic end force exerted by maximum allowable
working pressure, lb,

H_p = force to ensure tight joint, H_p = 0 for the silicone
O-ring since it is considered self-sealing, and

h_G = gasket moment arm, h_G = 0.6345 in.,

d_o = outside diameter of inner container, d_o = 7.0 in.

Combined, rearranged, and simplified, the pressure equations yield the
following:

$$P = SE \left(\frac{t}{d} \right)^2 \left/ \left(C + \frac{1.9\pi (d_o)^2 h_G}{4d^3} \right) \right.$$

$$P = \frac{(15,000 \times 0.55) \left(\frac{0.44}{6.5} \right)^2}{0.3 + \frac{1.9\pi}{4(6.5)^3} (7.0)^2 (0.6345)}$$

$$P = 80.6 \text{ psi}$$

The required parameters and the results of the calculations for the bolted
cover are shown in Table 2-18.

TABLE 2-18
INNER CONTAINER COVER PRESSURE
CAPABILITY CALCULATIONS

Package Type	Drum Size (gal)	Inside Radius (in.)	Cover Thickness (in.)	Gasket Moment Arm (in.)	Internal Pressure Capability (psig) @ -20 to 650°F
5790	12	2.06	0.57	-	2203.0
5791	55	3.25	0.4375	0.6345	80.6

2.10.1.5. Flange. The flange design was analyzed in detail, including O-ring gasket compression as related to bolt loading, bolt load (tension and shear) from internal pressure, longitudinal hub stress (S_H) and radial (S_R) and tangential (S_T) flange stress. Allowable stresses, from section UA-52 of Reference 1, are:

$$S_H < 1.5 S$$

$$S_R < S$$

$$S_T < S$$

where:

S = max. allowable material stress = 15,000 psi.

Under actual conditions, from section UA-50, using an O-ring seal:

$$M_o = 0.785 B^2 P \left(\frac{C-B - g_1}{2} \right) + 0.785 P (G^2 - B^2) \left[\frac{2 C-B-G}{4} \right]$$

where: M_o = total moment acting on flange in in.-lb.

B = inside diameter of flange, in.

P = internal pressure, psi

C = bolt-circle diameter, in.

g_1 = thickness of hub at back of flange, in.

G = diameter of O-ring seal, in.

For an actual pressure of 20.2 psi under accident conditions,

$$M_o = 0.785 (7)^2 (20.2) \left[\frac{9.125-7-.5}{2} \right]$$

$$+ .785 (20.2) \left[(7.86)^2 - (7)^2 \right] \left[\frac{2(9.125) - 7 - 7.86}{4} \right]$$

$$= 803.05 \text{ in.-lb.}$$

From section UA-47,

$$e = \frac{F}{B \sqrt{g_o}}$$

where:

e = factor for integral type flanges

F = factor for integral type flanges from Figure UA-51.2.

B = inside diameter of flange, in.

g_o = thickness of hub at small end, in.

$$e = \frac{.892}{(7) (.25)} = .674$$

Then from section UA-51,

$$S_H = \frac{f M_o}{L g_1^2 B}$$

$$S_R = \frac{(1.33 t e + 1) M_o}{L t^2 B}$$

$$S_T = \frac{Y M_o}{t^2 B} - Z S_R$$

where:

f = hub stress correction factor from Figure UA 51.6.

L = factor = $\frac{t e + 1}{T} + \frac{t^3}{d}$

d = factor for integral type flanges = $\frac{U}{V} h_o g_o^2$

t = flange thickness, in.

U,Y,Z = factors involving K (from Figure UA-51.1)

K = ratio of flange outside diameter to inside diameter

V = factor for integral type flanges (from Figure UA-51.3)

$h_o = \sqrt{B g_o}$, in.

all other variables as previously defined.

For accident conditions:

$$S_H = \frac{2.7(803.05)}{0.869(0.5)^2 (7)} = 1425.77 \text{ psi}$$

$$S_R = \frac{[1.33(0.5)(0.674) + 1] (803.05)}{0.869(0.5)^2 (7)}$$

$$= 764.74 \text{ psi}$$

$$S_T = \frac{(6) (803.05)}{(0.5)^2 (7)} - 3.1 (764.74)$$

$$= 382.62 \text{ psi.}$$

Because S_H is the largest fraction of allowable stress, this is the limiting stress. However, even this stress is less than 4% of the maximum allowable.

From section UA-49, the bolt load (W) can be calculated for an O-ring seal as follows:

$$W = 0.785 G^2 P$$

$$= 0.785 (7.86)^2 (20.2)$$

$$= 979.64$$

For eight bolts, the load on each bolt, W_B , is:

$$W_B = \frac{979.64}{8} = 122.45 \text{ lb/bolt}$$

All flange stress criteria were met, and actual bolt loads were found to be approximately 25% of the maximum allowable.

At the maximum allowable internal pressure of 37.0 psig for the 5791 welded bottom plate, the longitudinal hub stress is calculated as follows:

$$\begin{aligned} M_o &= 0.785(7)^2 (37.0) \left[\frac{9.125 (7) (0.5)}{2} \right] \\ &+ 0.785 (37.0) \left[(7.86)^2 - (7)^2 \right] \left[\frac{2(9.125) - 7 - 7.86}{4} \right] \\ &= 1470.93 \text{ in.-lb} \\ S_H &= \frac{2.7 (1470.93)}{0.869 (.5)^2 7} = 2611.55 \end{aligned}$$

Even at this high pressure, the stress is less than 12% of the maximum allowable.

The corresponding load on each bolt will be:

$$W_B = \frac{0.785(7.86)^2 (37)}{8} = 224.30,$$

which is less than half the maximum allowable.

The results of these calculations are summarized in Table 2-19. Because the 5790 inner container has a threaded cover, there are no flange calculations for this package.

Table 2-19
FLANGE STRESS CAPABILITY CALCULATIONS PACKAGE 5791

<u>Stress</u>	<u>Maximum Allowable (psi)</u>	<u>Actual for Accident Conditions (psi)</u>
Longitudinal (S_H)	22,500	1426
Radial (S_R)	15,000	765
Tangential (S_T)	15,000	383
Bolt Load (W_B)	448 lb/bolt	122 lb/bolt

2.10.1.6. Results and Conclusions. Comparison of the resulting internal pressure capability calculations for the four basic components indicates that the welded bottom plate is the weakest component for both inner containers. Thus, the ASME Code maximum allowable working pressures are those shown in Table 2-17. The pressure capabilities are sufficiently high to contain the pressure that would result from heating the air environment trapped during packaging operations at one atmosphere to the temperatures reached during normal shipment and during hypothetical accident conditions.

2.10.2. Load Resistance from Section 2.1.2

When it is regarded as a simple beam supported at its ends along any major axis, the shipping container must be capable of withstanding a static load, normal to and uniformly distributed along its length and equal to five times the fully loaded container weight, without generating stresses in any material of the container in excess of the yield strength of that material.

A sample calculation is presented for the 55-gal package and the results are then tabulated for both packages. The 55-gal size uses a DOT specification 17C drum, which is fabricated of 16 gauge mild steel. The drum with the required load is illustrated schematically in Figure 2-15. The drum

Gross Weight of Container = W, $W = 400 \text{ lb}$, $5 \times 400 = 2000 \text{ lb}$

$w = \text{Weight per unit length, lb/in.} = W/L$

$$wL = W = 2000 \text{ lb}$$

FIGURE 2-15 - Drum load resistance for 55 gal package.

material is mild steel with a minimum yield stress of 27,000 psi, as specified per ASME Pressure Vessel Code [1]. For the purpose of this evaluation, the gross weight is taken to be 400 lb, although the maximum gross weight is 300 lb. The gross weight used in this evaluation for the 12-gal calculation is also conservatively high.

Stresses in the drum resulting from the uniform load are determined, as recommended by Shappert [17], from the following equation:

$$S = MC/I = M/Z,$$

where: S = stress (psi)

M = maximum bending moment, $M = 5 WL/8$ (in.-lb)

$Z = I/C$ = section modulus of drum

I = mass moment of inertia

$$= \frac{\pi}{4} (R_o^4 - R_i^4) = \pi R_o^2 t (\text{in}^3).$$

For a large diameter, thin-walled cylinder,

C = maximum distance from neutral axis (in.), $C = R_o$

W = weight of drum, $W = 400$ lb

L = length of drum, $L = 34.81$ in.,

R_o = outside radius of drum, $R_o = 11.31$ in.,

R_i = inside radius of drum, $R_i = 11.25$ in., and

t = thickness of drum, $t = 0.0598$ in.

The computed maximum bending moment is:

$$M = 5(400)(34.81)/8 = 8,700 \text{ in.-lb}$$

The computed section modulus is:

$$\begin{aligned} Z &= \pi R_o^2 t = \pi (11.31)^2 (0.0598) \\ &= 24.0 \text{ in.}^3 \end{aligned}$$

The maximum bending stress is then:

$$S = 8,700/24.0 = 363 \text{ psi.}$$

Since the material yield stress of 27,000 psi is 74 times as great as the calculated maximum bending stress, the 55-gal drum container satisfies the load resistance requirement. The required data and the results of the load resistance calculations for both sizes are summarized in Table 2-20.

TABLE 2-20
SUMMARY OF LOAD RESISTANCE DATA AND RESULTS

<u>Required Data</u>	<u>Drum Size</u>	
	<u>12 gal</u>	<u>55 gal</u>
W, lb	150	400
L, in.	21.1	34.81
R, in.	6.94	11.31
t, in.	0.0359	0.0598
<u>Results</u>		
S, psi	364	363
<u>S_{yield}</u>	74	74
S actual		

2.10.3. Program Input/Output from Section 2.7.3.2.

Internal pressure 23.8 psi

External pressure 0

Inner radius 2.0625 in.
Outer radius 2.5 in.
Closed ends (Y/N) Y
Elastic modulus 2.95E7 psi
Poisson's ratio 0.287
Allowable stress 15000 psi

Stresses and Radial Displacements (5790)

At inner surface of cylinder

Radial stress = -23.8 psi
Hoop stress = 125.2411 psi
Axial stress = 50.72055 psi
Radial displacement = 9.233826E-06 in.
Factor of safety = 116.213

At outer surface of cylinder

Radial stress = 0
Hoop stress = 101.4411 psi
Axial stress = 50.72055 psi
Radial displacement = 8.596702E-06 in.
Factor of safety = 170.7445

Internal pressure 20.2 psi
External pressure 0

Inner radius 3.25 in.
Outer radius 3.5 in.
Closed ends (Y/N) Y
Elastic modulus 2.95E7 psi
Poisson's ratio 0.287
Allowable stress 15000 psi

Stresses and Radial Displacements (5791)

At inner surface of cylinder

Radial stress = -20.20001 psi
Hoop stress = 273.0741 psi
Axial stress = 126.4371 psi
Radial displacement = 3.072313E-05 in.
Factor of safety = 59.05911

At outer surface of cylinder

Radial stress = -7.629395E-06 psi
Hoop stress = 252.8741 psi
Axial stress = 126.4371 psi
Radial displacement = 3.000201E-05 in.
Factor of safety = 68.4946

References

1. J. M. Guy, et al., ASME Boiler and Pressure Vessel Code, Section VIII, Rules for Construction of Pressure Vessels, Division 1, The American Society of Mechanical Engineers, New York, N. Y., 1977.
2. R. J. Roark and W. C. Young, Formulas for Stress and Strain, McGraw-Hill, New York, N. Y., 5th Ed., 1975.
3. American Society for Metals, ASM Metals Reference Book, American Society for Metals, Metals Park, Ohio, 1983.
4. R. Barron, Cryogenic Systems, McGraw-Hill, New York, N.Y., 1966.
5. Parker Seal Group, "Technical Rept. KK1745", Lexington, Ky., 17 May 1985.
6. Parker Seal Group, Parker O-Ring Handbook, Lexington, Ky., 1982.
7. Parker Seal Group, "Technical Reprt. KB4465", Berea, Ky., January 1985.
8. M. E. Shank, Control of Steel Construction to Avoid Brittle Failure, Welding Research Council, New York (1957).
9. W. R. Holman and R. T. Langland, Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick, NUREG/CR-1815, Lawrence Livermore Laboratory (August 1981).

- 9A. R. A. Watkins, R. E. Bertram, R. K. Blauvelt, D. A. Edling, T. M. Flanagan, J. F. Griffin, and T. B. Rhinehammer, Safety Analysis Report for Packaging (SARP): USA/9507/BLF (ERDA-AL), Model A1-M1, MLM-2447, Monsanto Research Corporation, Miamisburg, Ohio (September 1977), 120 pp.
10. F. E. Adcock and W. F. Wackler, RDF Container - Model 1518 for Fissile Class II and Class III Shipments, RFP-1042, Dow Chemical Co., Rocky Flats (April 8, 1968).
11. F. E. Adcock, J. D. McCarthy, and W. F. Wackler, Rocky Flats Model 2030-1 Container (AEC-AL USA/5332/BLF) Safety Analysis Report for Packaging (SARP), RFP-1867 Rev. 1, Dow Chemical Co., Rocky Flats (February 27, 1974).
12. J. R. Gaskill and R. D. Taylor, An Improved Shipping Container for Fissile Material, UCRL-14903, University of California (June 3, 1968).
13. R. L. Delnay, "Description and Evaluation of Four $\text{Pu}(\text{NO}_3)_4$ Shipping Containers," Proceedings, International Symposium for Packaging and Transportation of Radioactive Materials, Albuquerque, New Mexico, January 1965. Published in 5C-RR-65-98, Technical Information Department, Sandia Corporation, Albuquerque, New Mexico, June 1965, p. 430.
14. Memo from C. M. Wires to J. N. Crawford (September 13, 1984).
15. Memo from C. M. Wires to J. N. Crawford (September 13, 1984).

16. U.S. Department of Energy, A Manual for the Prediction of Blast and Fragment Loadings on Structures, DOE/TIC-11268 (November 1980).
17. L. B. Shappert, "A Guide to the Design of Shipping Casks for the Transportation of Radioactive Material," ORNL-TM-681, Oak Ridge National Laboratory, April 1965.
18. H. G. Clarke, Jr. and S. P. Carfagno, "Structural Integrity of Shipping Containers for Radioactive Materials, Part VIII, Experimental Study of Crushing Impact," NYO-2539-7, Franklin Institute Research Laboratories, October 1968.
19. F. E. Adcock, J. T. Byrne, and R. L. Delnay, "Plutonium Oxide Shipping Packages," RFP-501, Dow Chemical Co., April 1965.
20. R. R. Sandberg, "Safety Analysis Report on Model UC-609 Shipping Package," UCRL-52424, Lawrence Livermore Laboratory, August 1977.
21. G. G. Chalfant, "Safety Analysis Report Packages USA/9965/B(U)F (DOE-SR), USA/9966/B(U)F (DOE-SR), USA/9967/B(U)F (DOE-SR), USA/9968/B(U)F (DOE-SR) (Packaging of Fissile and Other Radioactive Materials)," DPSPU-83-124-1, E. I. DuPont de Nemours and Co., Inc.
22. H. R. Dyer, W. T. Mee, "Safety Analysis Report for Packaging (Oak Ridge Y-12 Plant Model DT-2 Package for Enriched Uranium)," Y/DD-152, Oak Ridge Y-12 Plant, June 1974.

23. H. R. Dyer, "Safety Analysis Report for Packaging (Oak Ridge Y-12 Plant Model DT-5 Package for Enriched Uranium)," Y/DD-153, Oak Ridge Y-12 Plant, July 1974.
24. B. B. Klima, L. B. Shappert, R. D. Seagren, W. D. Box, "Safety Analysis Report for Packaging (SARP) of the Oak Ridge National Laboratory Foamglas Shipping Container (Revision 1)," ORNL-5407/R1, Oak Ridge National Laboratory, January 1979.
25. H. R. Dyer, "Safety Analysis Report for Packaging (Oak Ridge Y-12 Plant Model DT-7 Package for Enriched Uranium)," Y/DD-160, Oak Ridge Y-12 Plant, December 1974.
26. C. M. Hopper, "Safety Analysis Report for Packaging (Oak Ridge Y-12 Plant Model DT-14 Package for Enriched Uranium)," Y/DD-244, Oak Ridge Y-12 Plant, January 1978.
27. R. G. Taylor, "Safety Analysis Report for Packaging (Oak Ridge Y-12 Plant Model DT-14A Package for Enriched Uranium)," Y/DD-326, Oak Ridge Y-12 Plant, January 1984.
28. "Stresspac," Engineering Software Co., 1405 Porto Bello, Arlington, Texas, (1983).
29. Baumeister and Marks, Standard Handbook for Mechanical Engineers, 7th ed., McGraw-Hill.

30. Bowser-Morner, Inc., "Report of Immersion Test on Four Shipping Containers," Job Report No. 75285 (June 9, 1987).

3. THERMAL EVALUATION

3.1. Discussion

The 5790 and 5791 packages are designed to ensure safe containment of their contents under both normal conditions of transport and hypothetical accident conditions, as specified in 10 CFR Part 71. The packages are insulated to provide protection for the inner containers at hypothetical accident fire conditions. The type of insulation and the thickness are selected to preclude overheating when containing the maximum permissible decay heat load.

The internal pressure capability of each container is sufficient to contain the pressure buildup resulting from heating the atmospheric air, which is trapped during packaging operations, to the maximum temperatures at accident conditions.

The minimum decay heat load assumed in this evaluation is 0 for both the 5790 and 5791 packages. The maximum heat load is 30 W for the 5790 package and 42 W for the 5791 package.

3.2. Summary of Thermal Properties of Materials

The thermal conductivities, specific heats, and densities of all materials used in the thermal evaluation are summarized in Table 3-1.

TABLE 3-1
THERMAL PROPERTIES OF MATERIALS

<u>Material</u>	<u>Thermal Conductivity(BTU/hr ft⁰F)</u>	<u>Specific Heat (BTU/lb⁰F)</u>	<u>Density (lb/ft³)</u>
Carbon Steel	37 ^(a)	0.10 ^(b)	52.9 ^(b)
Firedike Insulation	0.033 ^(c)	Not Avail. ^(d)	19.2 ^(c)
Auratone Firecode Insulation	0.028 ^(c)	Not Avail. ^(d)	25.6 ^(c)

^(a)Reference 3, p. 210.

^(b)Reference 3, p. 205.

^(c)Data supplied by manufacturer.

^(d)Predicted range 0.20-0.25

3.3. Technical Specifications of Components

The 5790 and 5791 packages are simple containers, without valves, gauges, or moving parts. Properties of the insulation materials used in these packages are summarized in Table 3-1.

3.4. Thermal Evaluation for Normal Conditions of Transport

3.4.1. Thermal Model

3.4.1.1. Analytical Model. The procedure for calculating the temperatures resulting from direct sunlight consists of determining the heat load from the sun. This procedure is used to calculate the resulting external drum surface temperature that is required to dissipate the solar heat load to 100°F ambient air. The temperature increases resulting from the solar heat load throughout each package are less than, or equal to, the corresponding increase at the drum surface. Therefore the assumption that these temperature increases are equal throughout each package provides conservative estimates of the inner container temperature.

Shappert's [1] approach establishes the average solar heat load over a 24-hr period as 42 W/ft^2 of projected surface area. For a 24 hr period, this exposure is equivalent to $933 \frac{\text{gcal}}{\text{cm}^2}$, higher than the highest value suggested in the table in 10 CFR 71.71(c)(1). Therefore it is conservative.

The maximum possible projected surface area is estimated based on viewing the container, when in an upright position, at an angle perpendicular to a diagonal drawn through the drum. The resulting temperature increase at the drum surface is determined by linear extrapolation of the experimental, steady state, temperature profile data.

Solar heat load for the 55-gallon package [5].

$$\begin{aligned} A &= [D^2 + H^2]^{1/2} \times D \\ &= [(1.89 \text{ ft})^2 + (2.90 \text{ ft})^2]^{1/2} \times 1.89 \text{ ft} \\ &= 6.54 \text{ ft}^2, \end{aligned}$$

$$\begin{aligned} Q_S &= A \times Q_A \\ &= (6.54 \text{ ft}^2) (42 \text{ w/ft}^2) \\ &= 275 \text{ W}, \end{aligned}$$

where:

A = maximum projected surface area,

D = drum outer diameter,

H = drum height,

Q_A = solar heat load per unit area, and

Q_S = total solar heat load.

3.4.1.2. Test Model

3.4.1.2.1. Purpose. The steady state temperature profiles of each package and its contents were determined experimentally to ensure compliance with DOT, DOE, and NRC regulatory requirements, and conformance with any product specification, and to establish the appropriate temperatures for evaluation of the contents. Also, the steady state data obtained at the heat loads tested experimentally were used to determine the maximum heat load capability of each package.

3.4.1.2.2. Test Equipment and Procedures. The tests were performed using the equipment illustrated schematically in Figure 3-1. A 3/4-in. diameter x 7-3/4-in. long electric heater was installed inside the inner container to simulate a radioactive heat source. Separate tests were made for each size with the heater packaged in steel wool (see Figure 3-2) or simply suspended within the inner container air environment at atmospheric pressure, to determine the heater temperatures when these packaging methods were used.

FIGURE 3-1 - Schematic of thermal test equipment.

FIGURE 3-2 - Heater and steel wool packaging materials for
55 gal package. (Photo No. 750510)

Also, a series of tests was performed using the 12-gal size for evaluation of copper turnings, nickel spheres, lead shot, and aluminum pellets for packaging materials. The electrical wires and thermocouples were fed through small holes in the drum head and inner container cover. A wattmeter was used to determine the heat loads.

A digital thermometer was used to obtain temperature data at several locations. The thermocouples used to measure surface temperatures were welded in place. The eight type-K thermocouples (No. 2-9) were connected to a selector switch, and a single lead from the selector switch was connected to the digital thermometer. Temperatures were then obtained by simply dialing the thermocouple number and reading the temperature directly on the digital thermometer. The three type-T thermocouples (No. 10-12) were used similarly. The type-J thermocouple, which was fastened to the heater (No. 1), was connected to a temperature controller so that the entire system would safely shut down if the heater temperature exceeded a preset temperature.

The heating procedure required approximately one week for each set of data. The heat load was set at the desired value, and the temperature profile throughout the package was allowed to reach steady state. A complete set of temperature values was then obtained, and the procedure was repeated until a heat load of 60 to 70 W was reached. The temperature increases during the equilibration periods were studied to verify that steady state conditions were reached. The ambient temperature was determined at each heat load by averaging several readings taken at 0.5 to 1 ft away from the sides and top of the drum.

3.4.2. Maximum Temperatures

3.4.2.1. Test Results. The experimentally determined steady state temperatures are presented in Table 3-2 for the 12-gal size and in Table 3-3 for the 55-gal size; both packages are insulated with Firedike. All temperatures listed in these tables have been adjusted to an ambient temperature of 100°F (38°C) to represent normal conditions of transport on a hot day for comparison with DOT and DOE/NRC regulations. The maximum temperatures found for each drum exterior, inner container, and heater are plotted in Figure 3-3 for the 12-gal size and Figure 3-4 for the 55-gal size. The heater temperature data at steady state for the various packaging materials tested in the 12-gal package are presented in Table 3-4 and shown graphically in Figure 3-5.

TABLE 3-2
STEADY STATE TEMPERATURE TEST RESULTS FOR 12-GAL PACKAGE
AT 100°F AMBIENT TEMPERATURE

Thermocouple No. and Location	Steady State Temperature (°F)					
	<u>With Steel Wool</u>			<u>With Air</u>		
Watts =	<u>20</u>	<u>37</u>	<u>63</u>	<u>20</u>	<u>31</u>	<u>60</u>
<u>Heater</u>						
TC-1 Heater Surface	532	744	952	439	541	728
<u>Inside Inner Container</u>						
TC-2 Near Bottom	277	387	515	236	272	372
TC-3 Near Side	410	586	771	253	312	439
TC-4 Near Top	300	429	551	342	411	541
<u>Outside Inner Container</u>						
TC-5 Bottom Center	235	325	437	229	283	410
TC-6 Side Mid-Ht.	241	335	450	235	290	417
TC-7 Top Side Cover	234	321	427	234	286	404
TC-8 Top Center	230	312	413	239	290	406
<u>Firedike</u>						
TC-9 Inner Surface	216	302	413	201	246	362
<u>Drum Exterior</u>						
TC-10 Bottom Center	108	115	122	108	112	119
TC-11 Side Mid-Ht.	108	113	120	105	110	117
TC-12 Top Center	108	111	118	107	109	116

TABLE 3-3
STEADY STATE TEMPERATURE TEST RESULTS FOR 55-GAL PACKAGE
AT 100°F AMBIENT TEMPERATURE

Thermocouple No. and Location	Steady State Temperature (°F)				
	<u>With Steel Wool</u>			<u>With Air</u>	
Watts =	<u>30</u>	<u>43.5</u>	<u>64.5</u>	<u>30</u>	<u>66.5</u>
<u>Heater</u>					
TC-1 Heater Surface	651	809	993	497	715
<u>Inside Inner Container</u>					
TC-2 Near Bottom	241	298	382	253	402
TC-3 Near Side	330	412	525	253	400
TC-4 Near Top	277	338	422	269	405
<u>Outside Inner Container</u>					
TC-5 Bottom Center	237	297	374	238	384
TC-6 Side Mid-Ht.	246	305	390	241	382
TC-7 Top Side Cover	226	275	349	236	362
TC-8 Top Center	224	273	346	237	362
<u>Firedike</u>					
TC-9 Inner Surface	225	278	358	222	354
<u>Drum Exterior</u>					
TC-10 Bottom Center	104	104	104	101	104
TC-11 Side Mid-Ht.	105	106	109	104	109
TC-12 Top Center	104	105	107	104	108

FIGURE 3-3 - Steady state temperature variation with heat load for 12 gal package.

FIGURE 3-4 - Steady state temperature variation with heat load for 55 gal package.

TABLE 3-4
 HEATER SURFACE TEMPERATURES AT STEADY STATE
 IN VARIOUS PACKAGING MATERIALS FOR 12-GAL PACKAGE

<u>Packaging Material</u>	<u>Heat Load (W)</u>	<u>Temperature (°F)</u>
Steel Wool	20	532
	37	744
	63	952
Air	20	439
	31	541
	60	728
Copper Turnings	20	362
	40	527
	62	680
Nickel Spheres	20	322
	39	466
	58	584
Lead Shot	21	333
	41	474
	62	559
Aluminum Pellets	20	317
	40	449
	61	566

FIGURE 3-5 - Heater surface temperatures at steady state in various packaging materials for 12 gal package.

The two graphs (Figures 3-3 and 3-4) clearly show that a higher heat load may be packaged in the 55-gal size without exceeding inner container temperature capabilities. The exterior drum surface temperature was lower for the 55-gal size than for the 12-gal size because of the larger exterior surface area. Also, Figures 3-3 and 3-4 show that the resulting heater temperatures in steel wool and air are similar for both sizes. Figure 3-5 illustrates that the aluminum pellets performed the best and the steel wool performed the poorest of the packaging materials tested. The results are discussed in detail in section 3.4.2.2.

Another significant result of the tests was that there was no evidence that the heating and cooling caused any misfit, galling, or other damage to the packages except for discoloration of the Firedike at approximately 400°F.

The aforementioned steady state thermal test was repeated for a 5791 package insulated with Auratone Firecode [1]. The electric heater for this test was 5 in. long, and a voltage controller was used to adjust the electrical input to the heater. A multitest meter was used to measure the voltage and current for each test, and six thermocouples recorded temperatures on a chart recorder. A schematic of the wiring is illustrated in Figure 3-6.

The six thermocouples were positioned as shown in Figure 3-7. Table 3-5 describes their functions. The thermocouples were wired, taped, or inserted in place depending on their locations. The wires were routed through the small vent holes in the 55-gal drum and were then fed to a multipoint chart recorder that ran continuously through each test.

FIGURE 3-6 - Schematic of equipment and test setup for the 5791 package. Voltage was measured across points 1 and 2 with S1 closed. Current was measured across points 2 and 3 with S1 open.

FIGURE 3-7 - Thermocouple locations for 55 gal package.

TABLE 3-5
THERMOCOUPLE LOCATIONS AND FUNCTIONS

<u>Thermocouple^a</u>	<u>Location</u>	<u>Function</u>
TC-1	Top of exterior inner container over O-ring	Top temperature and O-ring temperature
TC-2	Middle of inner container, middle height	Container temperature
TC-3	Bottom of inner container exterior	Container temperature
TC-4	Under Auratone Firecode inside skin same height as TC-2	Maximum Auratone Firecode temperature
TC-5	Heater surface	Check for heater malfunction
TC-6	1 ft from drum midheight	Ambient temperature

The 10-W test was conducted first. The voltage controller was adjusted so that the applied heat load was approximately 10 W. The system was operated until the chart recorder verified that steady state had been reached. The package was then dismantled, and the Auratone Firecode was inspected for damage (charring). Since no damage was present, tests were conducted at 20 W and then 30 W. Still there was no signs of char or other damage. A plot of the data collected to this point showed points scattered about the Firedike curve (see Figure 3-8). Speculation that moisture (droplets of moisture collected on the relatively cool inner surfaces of the metal drum) may have affected these results prompted an attempt to dry the packing and retest at the 20-W and 30-W levels. The 20-W test ran significantly hotter this time; however, the 30-W test was not different within experimental error (see Figure 3-9).

FIGURE 3-8 - Plot of first three Auratone Firecode test points. Moisture may have affected these test results. Curve was drawn through upper points for most conservative result.

FIGURE 3-9 - Comparison of Auratone Firecode and Firedike steady state temperatures. Curve was drawn through upper points for most conservative result.

Next the 42-W test was conducted. During this test, which was conducted for 10 days, the Auratone Firecode reached a steady state temperature of 350°F on its interior surface. Visual inspection of the dismantled package showed that the paint on the surface of the Auratone Firecode had charred. There was no charring on any of the unpainted surfaces, including the cored area in which the inner container resides (see Figure 3-10).

The experimentally determined steady state temperatures for the 5791 container with Auratone Firecode packing are presented in Table 3-6. All temperatures listed have been normalized to an ambient temperature of 100°F. This adjustment is made by adding the difference between 100°F and the ambient temperature to each steady state temperature reading. Figure 3-9 is a graph showing the results. In Figure 3-9, a conservative approach was used when two tests were conducted for one wattage. In these cases, the test yielding the higher temperature was used to draw the curve; however, the other point is also shown.

FIGURE 3-10 - Cored area of Auratone Firecode after the
42 W test. (Photo No. 811160)

TABLE 3-6
STEADY STATE TEMPERATURES ATTAINED BY THE 5791 PACKAGE AT 100°F
AMBIENT TEMPERATURE WITH AURATONE FIRECODE^a

<u>Thermocouple Location</u>	<u>Temperature, °F</u>					
	<u>10.9 W</u>	<u>20.2 W</u>	<u>19.7 W</u>	<u>30.7 W</u>	<u>29.3 W</u>	<u>42.9 W</u>
<u>Outside Inner</u>						
<u>Container</u>						
Top Over O-Ring	178	188	213	253	257	321
Side Midheight	174	181	206	246	251	316
Bottom Center	169	176	197	240	242	305
<u>Insulation Packing</u>						
Inner Surface	169	174	197	237	242	302
Heater	291	357	395	447	437	561

^aHeater in air atmosphere during test.

Finally, to make certain these results are accurate, all instruments used in the tests were calibrated by the in-house standards laboratory. The voltages read are accurate within 0.1 V; the current is accurate to 0.01 A. These errors would result in a deviation of only 0.5 W at the 42-W level. The chart recorder and thermocouples were calibrated simultaneously and have a combined deviation of 2.5°C. The accuracy to which the chart records could be read was approximately 5°C (9°F).

It was concluded the Auratone Firecode is an acceptable replacement for Firedike. Auratone Firecode can withstand a 42-W heat load for an extended time with absolutely no degradation. For temperature estimation, the temperatures of inner containers, specifically of packages insulated with Auratone Firecode, may be estimated to be 30°F higher than the corresponding temperatures for packages insulated with Firedike [2].

3.4.2.2. Maximum Heat Load Capability. The maximum heat load capability is determined for each package based on the maximum capability of the inner containers and the Firedike -- since these are the limiting factors for these packages. The drum exterior surface temperatures at the maximum heat loads are determined. Also in this section, guidance is provided regarding packaging materials based on the heater surface temperature data.

When the package is in 100°F ambient air, the maximum permissible temperature for the inner-container exterior surface during normal transportation has been selected as 300°F (149°C). The 300°F temperature was selected to ensure the Firedike insulation will not be discolored. (It will discolor significantly above 400°F.) This temperature is sufficiently low so that the inner containers will not exceed design capabilities even at the hypothetical accident fire temperature of 500°F (260°C). The pressure capability of the inner containers is based on 500°F. The life of silicone or Viton O-rings is indefinitely long at 300°F. At 500°F the life is 20 hr for silicone and 4 hr for Viton. Based on the 300°F maximum temperature for the inner container surface during normal transportation, and the resulting temperature increases at hypothetical accident conditions (see section 3.5), the resulting maximum heat load capabilities and external drum temperatures at 100°F ambient temperature, as shown in Figures 3-3 and 3-4, are given in Table 3-7. The maximum capabilities of the 12- and 55-gal sizes are based on the 300°F limitation, and the estimated maximum temperatures in a fire are approximately 60°F below the specified maximum of 500°F.

TABLE 3-7
 DRUM AND INNER CONTAINER SURFACE TEMPERATURES AT 100°F AMBIENT
 IN SHADE AND IN DIRECT SUNLIGHT

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	<u>In Shade</u>		<u>Intermed. Calculations</u>			<u>In Sun</u>	
		Drum (°F)	Inner Container ^a (°F)	Proj. Area (ft ²)	Solar Load (W)	Drum (°F)	Inner Container ^a (°F)	
12	30	112	300	2.44	102	148	336	
55	42	106	300	6.54	275	142	336	

^aAdd 30°F for Auratone Firecode insulation.

The maximum temperatures for the exterior drum surface in shade shown in Table 3-7 are less than the permitted maximum accessible external surface temperature of 122°F (50°C); thus, this is not a limiting factor for determination of the maximum heat load capability. The maximum heat loads listed are considered sufficiently high to provide for anticipated requirements and sufficiently low to provide a reasonable margin of safety.

The heater surface temperature data shown in Figure 3-5 provide a comparison between the packaging materials tested and a basis for establishing guidelines that may be used to ensure that package contents will not reach excessive temperatures. The aluminum pellets, lead shot, and nickel spheres performed well, as indicated by the lower heater temperatures achieved when these materials were used. For most uses, the aluminum pellets are ideal because of their heat transfer capability, light weight, and low price. Use of lead shot is somewhat limited because of its low melting temperature of 621°F (327°C) and weight. The highest heater temperature resulted when steel wool was used and, as a result, it may be necessary to use other packing materials when high wattage amounts of some products are shipped. In addition to the results shown in Figure 3-5, a configuration was tested in

which a metal can containing copper turnings was placed inside the innercontainer. The results were similar to the "air" curve but cannot be compared directly since a smaller heater was used for this test.

The results of the tests using steel wool and air were approximately the same for both packages. Thus, the results for all of the packaging materials tested in the 12-gal size provide reliable guidance for these materials when used in the 55-gal size. However, the actual temperature of an item being shipped will not necessarily be the same as the experimental heater temperatures. Even though the products typically shipped from Mound Facility are similar in geometry to the 7-3/4 in. long x 3/4 in. diameter electrical heater used for the testing, the actual temperature of a product being shipped depends on the particular characteristics of that product; for example, a smaller surface area will result in a higher surface temperature.

Most of the products typically shipped from Mound Facility can be shipped safely at a surface temperature of up to 750°F (399°C). It is not intended that the 750°F value be interpreted either as a limiting temperature for the contents, or that all materials can be shipped safely at 750°F. For example, a product with the same heat transfer characteristics as the heater and with a maximum surface temperature capability of 750°F can be packaged in steel wool in the 55-gal package if the heat load is limited to approximately 38 W. Also, it could be packaged in any of the other materials tested when containing up to the maximum permissible heat loads of 30 W for the 12-gal size and 42 W for the 55-gal size. The foregoing example illustrates how to use the data shown in Figure 3-4. In order to ensure compliance with requirements, it is generally necessary to consider the

specific characteristics of the material being shipped and its temperature capability.

For the 55-gal size, the solar heat load is calculated to be 275 W. Since the drum surface temperature increased 8.6°F above ambient when 66.5 W was dissipated, the surface temperature increase is estimated to be 36°F from the solar heat load. The temperatures for both sizes at 100°F ambient in shade and in direct sunlight are summarized in Table 3-7.

3.4.3. Minimum Temperatures

The minimum temperature of each component in this analysis is assumed to be the ambient temperature. According to 10 CFR 71, the minimum ambient temperature to be considered for normal conditions of transport is -40°F. Therefore, this will be the minimum temperature used in this analysis.

3.4.4. Maximum Internal Pressures

The maximum internal pressures for the inner containers can be simply estimated from the ideal gas law:

$$\frac{P_2}{P_1} = \frac{T_2}{T_1}$$

The inner container is assumed to be loaded at a pressure of 1 atm absolute (14.7 psia), at the minimum design temperature of -40°F. If the container holds the maximum permissible heat load, and is assumed to be subjected to direct sunlight at an ambient temperature of 100°F, the maximum internal pressure for normal conditions of transport will be reached. The maximum pressures for the two container sizes are summarized in Table 3-8.

TABLE 3-8
MAXIMUM INTERNAL PRESSURE - INNER CONTAINERS
NORMAL CONDITIONS OF TRANSPORT

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	Loading Temp. (°F)	Max Inner Container Temp. at 100°F in Direct Sun (°F) (a)	Maximum Internal Pressure (psig) (b)
12	30	-40	550	20.7
55	42	-40	439	16.8

(a) Add 30°F for Auratone Firecode insulation.

(b) Add 1 psig for Auratone Firecode.

At the calculated temperatures and pressures, no phase change or decomposition of the contents or packaging materials will occur, and no gas generation will take place.

3.4.5. Maximum Thermal Stresses

The limiting combination of thermal gradient and isothermal stresses is achieved in direct sunlight at 100°F ambient temperature. The resulting temperature distribution is as given in section 3.4.2 and Table 3-7.

Temperature differences within the packages under normal conditions of transport are small and corresponding stresses are insignificant. Differential thermal expansion is negligible.

3.4.6. Evaluation of Package Performance for Normal Conditions of Transport

Heat input from the sun will not cause the inner container temperatures to exceed design capabilities. In fact, the inner containers are designed to

withstand hypothetical accident fire conditions, as discussed elsewhere in this report. The effectiveness of the steel drums and insulation is not expected to be reduced as a result of the sunlight; and the insulation temperatures remain well below 400°F, which is estimated to be the temperature at which Firedike insulation will begin to char slightly.

Comparison of the resulting internal pressure capability calculations for the four basic components indicates that the welded bottom plate is the weakest component for both inner containers. Thus, the ASME Code maximum allowable working pressures are those shown in Table 3-9. The pressure capabilities are sufficiently high to contain the 20 psig pressure that would result from heating the air environment trapped during packaging operations at one atmosphere and also contain the temperatures normally encountered during shipment and during hypothetical accident conditions.

TABLE 3-9
MAXIMUM ALLOWABLE INTERNAL PRESSURE-INNER CONTAINERS

<u>Nominal Package Size (gal)</u>	<u>Inside Radius (in.)</u>	<u>Internal Pressure Capability at -20 to 650°F (psig)</u>	<u>Maximum Bending Stress (psi)</u>
12	2.06	220.4	364
55	3.25	37.0	363

Maximum temperatures that could be reached in direct sun at 100°F are summarized in Table 3-10.

TABLE 3-10
MAXIMUM TEMPERATURES
NORMAL CONDITIONS OF TRANSPORT^(a)

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	Maximum Temperatures (°F)		
		Inside Inner Container ^(b)	Inner Container Surface ^(b)	Drum Exterior
12	30	550	336	148
55	42	439	336	142

(a) Direct sun at 100°F ambient containing maximum contents heat load.

(b) Add 30°F for Auratone Firecode insulation.

3.5. Hypothetical Accident Thermal Evaluation

3.5.1. Thermal Model

3.5.1.1. Analytical Model. Not applicable.

3.5.1.2. Test Model. Packages were prepared for testing as detailed in section 2.7. This test requires exposure to a thermal test in which the heat input to the package is not less than the heat input that would result from exposure of the whole package to a radiation environment of 1475°F (800°C) for 30 min with an emissivity coefficient of 0.9, assuming the surfaces of the package have an absorption coefficient of 0.8. The package may not be cooled artificially until 3 hr after the test period, unless it can be shown that the temperature on the inside of the package has begun to fall in less than 3 hr.

The fire-test facility was designed to meet DOT/DOE hypothetical accident conditions. To simulate actual conditions, the test facility provides an open, aviation-gasoline-fueled fire. A water spray system was designed into the facility to eliminate the huge volumes of smoke normally associated with

open gasoline fires. Wind effects are reduced by 8-ft high firebrick walls on three sides and on the fourth side a 4-ft wall to permit viewing and ease of handling the shipping containers. The base of the fire pit is poured concrete measuring 10 x 10 x 0.5 ft deep [4]. Fuel and seven water-spray nozzles are located in the fire pit base, which is flooded with water 5-in. deep to avoid excessively heating the pit. A 5-HP fan supplies approximately 8000 ft³/min of air through the air manifold outlets located in two opposing 8-ft sides just above the fire pit. The 100-octane aviation gasoline is continuously gravity-fed to the distribution system from a 5,000-gal, buried tank located approximately 100 ft from the fire pit. The gasoline floats to the surface of the water and burns. The nozzle spray is directed horizontally, providing complete coverage of the burning aviation fuel surface. The water spray reduces the smoke plume far below maximum allowable requirements. The test facility was used to test the 12-gal and 55-gal packages. The 55-gal package is shown after the test in Figure 3-11; it was mounted on a stand 2-ft above the water surface and was centered within the burning area approximately 3 ft from the sides.

The flame temperatures obtained throughout the tests are plotted as a function of time in Figure 3-12. The temperatures were measured at the side of the package. Chromel/alumel thermocouples and a multipoint recorder were used to monitor the tests.

The type 5791 alternate design inner container was not subjected to the thermal test. The maximum surface temperature reached (244°F) on the original type 5791 inner container during the thermal test was not high enough to cause damage to the gauge on the alternate design. Therefore, the thermal test was not administered to the 5791 alternate design.

FIGURE 3-11 - The 55 gal container after fire test.
(Photo No. 750511)

FIGURE 3-12 - Temperatures as a function of time in fire test.

3.5.2. Package Conditions and Environment

During the free drop test, the 12-gal package sustained a 3-in. dent to the upper edge; the 55-gal package sustained a 4-in. dent to the upper edge. The covers of both containers were completely intact, and remained securely fastened to the drum body. The puncture test produced a minor dent in each package tested. Such damage probably had little, if any, effect on package thermal performance.

3.5.3. Package Temperatures

Temperatures as a function of time in the thermal test are shown in Figure 3-12.

3.5.4. Maximum Internal Pressures

All radioactive materials which may be shipped in the container do not decompose at temperatures generated during the hypothetical accident fire condition, even when the maximum allowable wattage is present in the container. The plutonium-238 material does generate helium as part of its decay mode. Fortunately, the rate of generation is rather modest amounting to 0.1 ml/mo/W. In addition, the helium is held tightly in the crystal lattice of the material and does not begin to escape until the material is heated to 2200°F which is well above the temperature that the inner container is expected to see under hypothetical accident fire conditions. The only pressure increase in the inner container, which will occur if the package is exposed to fire, will be from the heating of air trapped in the container at the time of sealing.

The maximum internal pressures inside the package inner container during the thermal accident test are calculated using the ideal gas law as in section 3.4.4. As before, the inner container is assumed to be loaded at an initial pressure of one atmosphere absolute (14.7 psia) at the minimum design temperature of -40°F. If the containers hold the maximum permissible heat load at an ambient temperature of 100°F, the maximum temperatures and pressures summarized in Table 3-11 result.

TABLE 3-11
MAXIMUM INTERNAL PRESSURE DURING
HYPOTHETICAL ACCIDENT THERMAL TEST^a

Nominal Package Size (gal)	Maximum Contents Heat Load (W)	Maximum Internal Inner Container Temperature ^b (°F)	Maximum Temperature Rise from Thermal Accident Test (°F)	Maximum Internal Temperature ^b (°F)	Maximum Internal Pressure ^c (psig)
12	30	514	126	640	23.8
55	42	403	133	536	20.2

^aCalculated at an ambient temperature of 100°; inner container assumed loaded at -40°F.

^bAdd 30°F for Auratone Firecode insulation.

^cAdd 1 psig for Auratone Firecode insulation.

3.5.5. Maximum Thermal Stresses

The most severe thermal stress conditions occur near the end of the thermal test, when inner container temperatures are highest.

3.5.6. Evaluation of Package Performance for Hypothetical Accident Thermal Condition

Both sizes passed the fire test. The outside surface (side at mid-height) of the 55-gal inner container reached a maximum temperature of

194°F and the inside surface reached 183°F. The silicone O-ring seals used to seal the inner container did not appear damaged in any way. The Firedike insulation was charred as a result of the fire test. The average temperature of the package prior to the fire test was estimated to be 50°F. Thus, a 50°F correction must be added to the resulting temperatures to adjust to the required initial package temperature of 100°F. Based on this, the inner container surface temperature reached $194^{\circ}\text{F} + 50^{\circ}\text{F} = 244^{\circ}\text{F}$ when corrected to 100°F ambient. The results for the 12-gal size were similar except that the inner container reached 205°F and the initial package temperature was 68°F. Thus, a 32°F correction is added to give a resulting maximum temperature of 237°F at 100°F ambient.

The maximum temperatures reached during the thermal accident test for the two packages are summarized in Tables 3-12 and 3-13. The maximum temperatures inside the inner containers will result in the maximum pressures listed in Table 3-11. These pressures are below the allowable pressures of 220.4 and 37 psig for the 12- and 55-gal packages, as determined in section 2.1.2., and therefore, will not adversely affect package performance.

The maximum surface temperatures of the inner containers are less than 500°F. At 500°F, the seal life of the silicone and Viton O-rings are 20 and 4 hr, respectively, as shown in section 4.1.3. Therefore, no breach of containment will result from thermal accident conditions.

TABLE 3-12
MAXIMUM TEMPERATURES REACHED DURING
HYPOTHETICAL ACCIDENT THERMAL TEST^a

Package 5790

<u>Location</u>	<u>Maximum Normal Temperature (°F)</u>	<u>Maximum Temperature Rise from Thermal Accident Test (°F)</u>	<u>Maximum Accident Temperature (°F)</u>
Inside Inner Container	514	126	640
Outside Surface Inner Container	296	137	433
Drum Exterior	112	1500	1612

^a Ambient temperature of 100°F, containing maximum contents heat load.

TABLE 3-13
MAXIMUM TEMPERATURES REACHED DURING
HYPOTHETICAL ACCIDENT THERMAL TEST^a

Package 5791

<u>Location</u>	<u>Maximum Normal Temperature (°F)</u>	<u>Maximum Temperature Rise from Thermal Accident Test (°F)</u>	<u>Maximum Accident Temperature (°F)</u>
Inside Inner Container	403	133	536
Outside Surface Inner Container	298	144	442
Drum Exterior	106	1500	1606

^a Ambient temperature of 100°F, containing maximum contents heat load.

References

1. J. N. Rigano, R. A. Watkins, and H. B. Kreider, "Temperature Tests of the USA/5791/BLF (DOE-AL) Shipping Container with Auratone Firecode Insulation", 30 October 1981, MLM-2868.
2. J. F. Griffin, memo to H. B. Kreider, 1 April 1982.

3. American Society for Metals, ASM Metals Reference Book, American Society for Metals, Metals Park, Ohio, 1983.
4. H. Williams and J. F. Griffin, "Smokeless Gasoline Fire Test," presented at the Fourth International Symposium on the Packaging and Transportation of Radioactive Materials, Miami Beach, Florida, September 1974, MLM-2159(OP).
5. L. B. Shappert, Cask Designers Guide, ORNL-NSIC-68, Union Carbide Corp., Oak Ridge, Tennessee, February 1970, p. 143.

4. CONTAINMENT

4.1. Containment Boundary

The inner containment vessels of the 5790 and 5791 packages provide the containment boundary. These vessels have been proven capable of retaining their contents under conditions normally encountered during transport and under hypothetical accident conditions.

4.1.1. Containment Vessel

Inner container dimensions are summarized in Table 4-1. Maximum allowable temperatures, pressures, and decay heat of contents are summarized in Table 4-2.

TABLE 4-1
INNER CONTAINER DIMENSIONS

Package Type	Body o.d. (in.)	Body Wall (in.)	Flange o.d. (in.)	Overall Outside Height (in.)	Inside Height (in.)	Bottom Plate Thickness (in.)	Top Plate Thickness (in.)	Minimum Flange Thickness (in.)
5790	5.0	0.44	-	14.5	11.1	0.50	0.88	-
5791	7.0	0.25	9.88	18.25	16.0	0.25	0.48	0.44

TABLE 4-2
DESIGN SPECIFICATIONS FOR CONTAINMENT VESSELS

Nominal Container Size (gal)	Maximum Allowable Outside Temperatures (°F)		Maximum Allowable Pressure (psig)		Maximum Decay Heat (W)
	Normal Transport	Accident Conditions	<u>Internal</u> <u>External</u>		
12	400	500	220.4	1467	30
55	400	500	37.0	177.5	42

4.1.2. Containment Penetrations

There are no containment penetrations for packages 5790 and 5791.

4.1.3. Seals and Welds

Requirements for welds on the carbon steel inner containers for the 5790 and 5791 packages are detailed in drawing SPA 760488, found in Appendix 1.3.

The O-ring seals used in the inner containers are silicone or Viton rubber. The properties of these seals are summarized in Table 4-3.

TABLE 4-3
O-RING MINIMUM SPECIFICATIONS

<u>Material</u>	<u>Hardness</u>	<u>Tensile</u>	<u>Elongation</u>	<u>Seal Life (hr)</u>	
	<u>Shore A(pts)</u>	<u>Strength(psi)</u>	<u>(%)</u>	<u>(@ 300°F)</u>	<u>(@ 500°F)</u>
Silicone	70(a)	500(a)	100(a)	>1000(c)	20(c)
Viton	75(b)	1400(b)	125(b)	>1000(c)	4(c)

(a) ASTM D2000/SAE J200 5GE 705 B37, E016, E036, F19, G11 Spec.

(b) MIL-R-83248A, Class 1, Type 1 Spec.

(c) Parker O-ring Handbook, Parker Seal Group, ORD 5700, Lexington, KY, 1982, pp. A3-36.

4.1.4. Closure

The inner container of the 5790 package is closed by a threaded cover which provides an O-ring seal. The inner closure for the 5791 package is a flanged assembly with an O-ring seal.

4.2. Requirements for Normal Conditions of Transport

4.2.1. Containment of Radioactive Material

Section 51 of 10 CFR 71 requires loss of radioactive material to be less than 10^{-6} A_2 per hour under normal conditions of transport. The A_2 quantities for isotopes typically shipped in these packages are listed in Table 4-4.

TABLE 4-4
CONTAINMENT CRITERIA - NORMAL CONDITIONS OF TRANSPORT

<u>Isotope</u>	<u>A_2 quantity^(a) (Ci)</u>	<u>Maximum loss(Ci/hr)</u>
Pu 238	0.003	3×10^{-9}
Pu 239	0.002	2×10^{-9}
U 233	0.1	1×10^{-7}
U 235	0.2	2×10^{-7}

(a) 10 CFR 71, p. 719, Table A-1.

4.2.2. Pressurization of Containment Vessel

All radioactive materials that may be shipped in the container do not decompose at temperatures generated during the hypothetical accident fire condition, even when the maximum allowable wattage is present in the container. The plutonium-238 material does generate helium as part of its decay mode. Fortunately, the rate of generation is rather modest amounting to 0.1 ml/mo/W. In addition, the helium is held tightly in the crystal lattice of the material and does not begin to escape until the material is heated to 2200°F -- which is well above the temperature that the inner container is expected to see under hypothetical accident fire conditions. The only

pressure increase in the inner container that will occur if the package is exposed to fire will be from the heating of air trapped in the container at the time of sealing. This pressure increase will be nominal and should not exceed 20 psig. Therefore, there will be no appreciable pressure increase in the container.

4.2.3. Containment Criterion

All inner containers must pass a helium leak test with no detectable leak greater than 1×10^{-5} std cm³/s when filled with 10 psig helium.

Fulfillment of this requirement demonstrates that the containment criteria of section 4.2.1 are satisfied.

4.3. Containment Requirements for Hypothetical Accident Conditions

10 CFR 71.51(a)(2) specifies that, under hypothetical accident conditions, loss of radioactive material must be less than the A₂ amount for each isotope. In addition, the external radiation dose rate must be less than one rem per hour at one meter from the external surface of the package. A₂ amounts for each isotope shipped in these packages are shown in Table 4-4.

The 5790 and 5791 packages meet these requirements, as shown by the successful completion of the series of hypothetical accident tests for these containers. The inner containers were undamaged following the free-drop, puncture and thermal tests, and they passed the water-immersion test without leakage. In addition, after the immersion test, both inner containers were reassembled, using the same O-rings, and were found to be helium leak tight at

an instrument sensitivity of 10^{-4} std cm³/s when containing 20 psig of helium pressure. Therefore, these containers meet the criteria of 10 CFR 71.51.

4.3.1. Fission Gas Products

Not applicable.

4.3.2. Containment of Radioactive Material

Section 51 of 10 CFR 71 requires loss of radioactive material to be less than A₂ per week under hypothetical accident conditions. The A₂ quantities for isotopes typically shipped in these packages are listed in Table 4-4.

4.3.3. Containment Criterion

Both packages 5790 and 5791 have been demonstrated to satisfy the requirements of 10 CFR 71.51 for containment of radioactive material under hypothetical accident conditions. This is demonstrated by these packages passing the water immersion test at the end of the sequence of hypothetical accident tests with no leakage. In addition, after the immersion test, both inner containers were reassembled, using the same O-rings, and found to be helium leak tight at an instrument sensitivity of 10^{-4} std cm³/s when containing 20 psig of helium pressure.

4.4. Special Requirements

10 CFR 71.63 requires that plutonium in excess of 20 Ci be packaged within a secondary containment vessel (inside the package inner container). If the entire package is subjected to normal conditions of transport, this separate inner container must restrict loss of plutonium to less than 10^{-6} A₂ per hour. Under hypothetical accident conditions, the secondary containment must limit plutonium loss to less than A₂ in one week. These two quantities for plutonium-238 and plutonium-239 are listed in Table 4-4.

At Mound Facility, the following vessels are currently accepted as secondary containment.

4.4.1. Food Pack Can

Sealed food pack cans with rolled and crimped lids are used in several types of packages to provide secondary containment [1,2,3]. The lids fold around a flange on the can body to produce a double seam (see Figure 4-1). To form a hermetic seal, the lids are then crimped, using a sealing machine that forces butyl rubber into the space between the metal folds.

In Reference 1, it is stated that a tinplate food pack can can be heated to 350°F and pressurized to 15 psig and still remain helium leak-tight to less than 10^{-5} atm cm³/s. At 75°F, the helium leak rate was found to be less than 10^{-8} atm cm³/s. These leak rates are low enough to prevent releases of radioactive materials in excess of allowable limits for both normal and accident conditions [1].

FIGURE 4-1 - Food pack can used as secondary containment.

In Reference 3, the leak testing of food pack cans was described. After sealing, the cans were vacuum bubble tested and gas pressure tested. Cans that passed the vacuum bubble test at a pressure difference of one atmosphere showed helium leak rates of less than 10^{-7} cm³/s. Therefore, they can be considered leak-tight. Cans that leaked could be sealed with silicone rubber or Glyptal enamel if the leak rate was less than 2 cm³/s and visual inspection of the mechanical seal showed no gross defects. Cans coated with silicone rubber remained leak-tight up to 35 psig internal pressure.

Impact and thermal tests were conducted on loaded 30-gal DOT 6M containers containing food pack cans inside the inner containers [3]. For the test, the cans were loaded with sand and lead shot to provide contents weights between 1.1 and 2.3 kg each. The package was dropped on one end from a height of 30 ft. After impact, the inner container was heated to 300°F for one hour, and the metal cans were then tested for leak-tightness. When empty slip lid cans and lid spacers were used to keep the food pack cans in place, no impact damage to the cans was found. The cans were still leak-tight, even after heating to 300°F.

The investigators concluded that pressures in excess of 15 psig will cause plastic deformation of the can lids, which could cause leakage [3]. To maintain internal pressures lower than 15 psig at high temperature, water must be strictly excluded from the can contents.

Stainless steel cans are used in the PAT-1 package developed by Sandia Albuquerque [2]. The cans are crimp-sealed, like the tinplate cans, then

double-sealed by applying an epoxy sealant to the roll-crimped joint. Sealed cans must pass a vacuum leak test with a leak rate of less than 10^{-3} cm^3/s .

Two complete PAT-1 packages, containing filled food pack cans, were subjected to the hypothetical accident conditions of 10 CFR 71. The can in one package was filled with depleted UO_2 for a total weight of 1.75 kg; the can in the second package contained depleted UO_2 and lead shot for a total weight of 3.15 kg. The total package weight was approximately 500 lb.

In both cases, the food pack cans remained intact, suffering several minor dents. Uranium detection measurements detected no release of uranium greater than 10^{-8} g, although the temperature of the can reached 170-180°F [3]. We therefore conclude that food pack cans provide adequate secondary containment for plutonium contents in the 5790 and 5791 packages.

4.4.2. Encapsulated Heat Standards

Plutonium-238 heat standards, encapsulated in two or three separate welded capsules, are often shipped in the 5790 and 5791 packages (see Figure 4-2). The heat standards are right circular cylinders of various sizes and thermal outputs. Dimensions and capsule materials for three of the heat standards are summarized in Tables 4-5, 4-6, and 4-7.

All capsules are gas tungsten arc (GTA) welded, and all welds are examined both visually and by dye penetrant test to ensure that there are no visible cracks or surface pores greater than 0.13 mm diameter in the weld. The welds

FIGURE 4-2 - Triply encapsulated heat standard used as secondary containment.

TABLE 4-5
3.0-4.0 W HEAT STANDARD CAPSULE DIMENSIONS

<u>Capsule</u>	<u>Material</u>	<u>Length (in.)</u>	<u>Diameter (in.)</u>	<u>Wall Thickness (in.)</u>	<u>Cap Thickness (in.)</u>
Liner	T-111	0.538	0.607	0.020	0.020
Strength Member	T-111	0.628	0.697	0.040	0.060
Clad	Hastelloy C	0.744	0.753	0.020	0.060

TABLE 4-6
0.5 W HEAT STANDARD CAPSULE DIMENSIONS

<u>Capsule</u>	<u>Material</u>	<u>Length (in.)</u>	<u>Diameter (in.)</u>	<u>Wall Thickness (in.)</u>	<u>Cap Thickness (in.)</u>
Liner	Ta-10W	0.780	0.250	0.020	0.020
Strength Member	Haynes 25	0.935	0.378	0.058	0.060

TABLE 4-7
1.5 W HEAT STANDARD CAPSULE DIMENSIONS

<u>Capsule</u>	<u>Material</u>	<u>Length (in.)</u>	<u>Diameter (in.)</u>	<u>Wall Thickness (in.)</u>	<u>Cap Thickness (in.)</u>
Liner	Ta-10W	2.530	0.250	0.020	0.020
Strength Member	Haynes 25	2.685	0.378	0.058	0.060

are also examined by radiography and ultrasonic inspection to ensure acceptability.

Each capsule is also helium leak checked. For the triply encapsulated heat standard, the liner must have a leak rate less than $5 \times 10^{-10} \text{ cm}^3/\text{s}$. The strength member and clad must leak less than $1.0 \times 10^{-11} \text{ cm}^3/\text{s}$ helium. The operating temperature of this heat standard is 750°F. The product acceptance specification for the heat standard is summarized in MRC Dwg. No. PS 314643 (see Appendix, section 4.5).

For the doubly encapsulated heat standard, the welded assembly must have a leak rate less than or equal to 1×10^{-6} std cm³/s as determined by a mass spectrometer-type helium leak detector. The product acceptance specification for this heat source is summarized in an internal Mound document, "Specifications for Fabricating Calorimeter Thermal Standards," (see Appendix, section 4.5). Both the liner and strength member are designed for a 20 yr life. They are required to withstand a 1900°F fire for 30 min at the end of their life span, and also to withstand a drop from a height of 3.0 m.

Triply encapsulated heat sources fueled with UO₂ or ThO₂ fuel simulant were subjected to the following DOT accident tests [4]:

- Free drop. A free drop through a distance of 30 ft on to a flat, essentially unyielding horizontal surface, striking the surface in such a position as to suffer maximum damage.
- Percussion. Impact of the flat circular end of a one-inch diameter steel rod weighing three pounds, dropped through a distance of 40 in. The capsule or material shall be placed on a sheet of lead, of hardness number 3.5 to 4.5 on the Vickers scale, and not more than one-inch thick, supported by a smooth, essentially unyielding surface.
- Heating. Heating in air to a temperature of 1475°F and remaining at that temperature for a period of 10 min.
- Immersion. Immersion for 24 hr in water at room temperature. The water shall be at pH 6 - pH 8, with a maximum conductivity of 10 $\mu\Omega$.

Following this series of tests, radiographic and helium leak check inspection of these capsules showed no detectable damage to the liner or strength member, and helium leak rates less than $5.3 \times 10^{-9} \text{ cm}^3/\text{s}$. It was concluded that the tested capsules complied with the requirements of the designated DOT tests.

In a separate study, 12 triply encapsulated heat sources were subjected to two impact tests [5]. Ten assemblies were tested at $\sim 80 \text{ m/s}$, and two were tested at $\sim 105 \text{ m/s}$. The impact angles were different in each of the tests. The heat sources were heated to 450°C prior to each impact. None of the heat source strength members was breached in these tests. Therefore, no fuel would have been released as a result of these tests.

It is apparent that the superior strength and leak-tightness of these standards provides reliable secondary containment for the plutonium contents.

4.5. Appendix

4.5.1. Specifications for Fabricating Doubly Encapsulated Calorimeter Thermal Standards

DESIGN AGENCY CONTROL NO.		DWG CLASSIFICATION LEVEL UNCLASSIFIED				REVISIONS		CHKR.	ENGR.
		ISSUE	DESCRIPTION	PREPARED BY	DATE				
PS314643-000		A	F. MORRIS MD/R. T. JANKOWSKI 2531 W. AMOS/DTER 761647SC		7/6/77				
			CER 7718875C		8-4-77				
		B	F. MORRIS MD/R. T. JANKOWSKI 2531 W. AMOS/AC0770220MD-C		8/29/77				
		C	F. MORRIS MD/R. T. JANKOWSKI 2531 W. AMOS/AC0770257MD-C		9-28-77				
		D	F. MORRIS MD/R. T. JANKOWSKI 2531 W. AMOS/AC0770353MD-C		1-25-78				
		E	F. MORRIS MD/R. T. JANKOWSKI 2531 W. AMOS/AC0780127MD-C		5-4-78				
		F	D. MORRIS MD/C. A. MCKEVER 2561 P. TEANEY/AC0841116MD-C.		07-11-84				

SHEET											
INDEX											
SHEET	ISSUE	1	2	3	4	5	6	7			
ISSUE		F	F	G	C	A	F	A			

AGENCY APPROVALS			TITLE	
ORG	DATE	INITIALS	PRODUCT ACCEPTANCE SPECIFICATION, MC2893	

DWG CLASSIFICATION LEVEL		SIZE	CODE IDENT NO.	DWG NUMBER
UNCLASSIFIED			14215	PS314643
				SHEET 1 OF 7

TIE/SC/MD

SC 5300-AC (5-70)

DECKS - 50, 52, 53, 01, 60, 40, 98, CM

1. GENERAL.

1.1 Scope. This specification defines the acceptance criteria and quality control requirements for the fueled heat source.

2. DOCUMENTS.

2.1 Required. The current issues of the following documents form a part of this specification.

2.1.1 Sandia Laboratories, Albuquerque, Drawing:

314643 MC2893

2.1.2 DOE document:

DOE 5480.1, Chg. 3 Packaging and Shipping Requirements

2.1.3 MRC - Mound Facility documents:

MD-90077 MC2893 Heat Source Reference Manual

MD-21394 MC2893 Material Control Manual

SPA740588 Material Specification For Yttrium

2.2 Precedence of Documents. In event of conflict between this specification and any other documents, this specification shall prevail.

3. REQUIREMENTS.

MRC-Mound Facility shall conform to the following requirements.

3.1 Component Identification. The serial number of each component shall be used to control the part through all pre-assembly operations and material control.

ISSUE	A	B	C	D	E	F	DWG CLASSIFICATION LEVEL	SIZE	CODE IDENT NO.	DWG NUMBER
							UNCLASSIFIED	A	14213	PS314643
										SHEET 2 OF

MRC-ML-4074

SC 510-44 5.7

4. H E A T S O U R C E A S S E M B L Y.

- 4.1 Fuel. The fuel shall be plutonium-238 dioxide meeting the requirements of MRC Specification SPA740209.
- 4.2 Getter. The oxygen getter shall be in the form of yttrium chips meeting the impurity requirements of MRC Specification SPA740588. The oxygen getter capsule loading shall be 0.544 ± 0.005 grams.
- 4.3 Thermal Output. The capsule fuel loading shall be 4.0 ± 0.05 watts.
- 4.4 Contamination. The outer surface contamination of the completed fueled heat source shall be less than 20 disintegrations per minute.
- 4.5 Total Neutron Emission Rate. The total neutron emission rate shall be determined and reported on each fueled heat source and shall not be greater than 4.60×10^4 n¹/sec.
- 4.6 Gamma and Neutron Spectra. The gamma and neutron spectra shall be determined and reported on one fueled heat source from the first production batch of fuel.
- 4.7 Gamma and Neutron Dose Rates. The gamma and neutron dose rates shall be determined and reported on one fueled heat source for each shipment of sources. The gamma dose rate should be less than 5 mr/hour at 10 centimeters and shall be measured at a minimum on one heat source from each batch of fuel.
- 4.8 Welding Gases. The liner and strength member welds shall be performed with nominally 25% argon, 75% helium with no more than 0.01% impurities. The clad weld shall be performed with argon with no more than 0.01% impurities.
- 4.9 Example Weld. One example weld will be made prior to each production run to assure that the welding machine is operating properly and the correct fixture setup has been made to meet the weld requirements. A production run is defined as a continuous work period on a particular component assembly (liner, strength member, or clad) with the same operators and equipment.

DATE	A	B	D	CLASSIFICATION LEVEL	100	DISSEMINATION	DWG NUMBER
				UNCLASSIFIED	A	4-16	PS314643
							REVISION 3

4.10 Fuel Loading and Welding Atmosphere. The helium atmosphere for fuel and getter loading and welding operations shall have a monitored oxygen value not exceeding 25 ppm and a monitored water value not exceeding 60 ppm. The nitrogen reading, which is taken prior to the example weld and used for the entire production run, shall not exceed 160 ppm. In the event of a gas analyzer breakdown where the atmospheric purity cannot be determined by instrumentation, a 105°, 2T bend test will be performed on a 0.51 mm thick specimen which has been welded with torch gas. The atmosphere is suitable for welding if there are no cracks in the convex surface of the bent specimen. Cracks occurring on the corners of the specimen during testing are not to be considered. If it becomes necessary to pass additional parts or supplies into the welding box after a production run has started, the box shall be purged to obtain a new nitrogen reading and the nitrogen, oxygen, and water analyzers shall be monitored to assure the atmospheric requirements above are met. Another example weld is then made and evaluated prior to resuming production.

4.11 Reencapsulation of Subassemblies.

4.11.1 Liner Assembly. The strength member may be removed from the liner assembly if the strength member assembly is unacceptable. Reencapsulation of the liner assembly is permitted if the involved liner assembly exhibits no visual anomalies after strength member removal.

4.11.2 Strength Member Assembly. The clad may be removed from the strength member if the clad assembly is unacceptable. Reencapsulation of the clad assembly is permitted if the involved strength member assembly exhibits no visual anomalies after clad removal.

5. WELD ACCEPTANCE - EXAMPLE WELDS.

5.1 Visual Inspection. All example welds will be given a visual inspection to determine if each weld meets the requirements of 5.1.1 and 5.1.2. The liner example weld will be examined at 1X, while the strength member and clad example welds will be examined at 10 to 30X. The liner, strength member, and clad example welds shall be removed from the body approximately 6 mm from the edge of the weld to facilitate the inspection of the weld underbead. If the welds meet the visual inspection requirements (5.1.1 and 5.1.2) and the weld joints are free from lack of penetration, production welding may begin.

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DWG CLASSIFICATION LEVEL

5.1.1 Cracks. There shall be no visible cracks in the weld or in the base metal adjacent to the weld.

5.1.2 Surface Pores. There shall be no detectable surface pores in the weld exceeding 0.13 mm diameter.

6. WELD ACCEPTANCE - PRODUCTION WELDS.

6.1 Visual Inspection. All production welds will be given a visual inspection at 1X to assure the requirements of 6.1.1 and 6.1.2 are met.

6.1.1 Cracks. There shall be no visible cracks in the weld or in the base metal adjacent to the weld.

6.1.2 Surface Pores. There shall be no detectable surface pores in the weld exceeding 0.13 mm diameter.

6.2 Dye Penetrant Inspection. All production welds will be given a dye penetrant inspection to determine if each weld meets the requirements of 6.1.1 and 6.1.2.

ISSUE	A					DWG CLASSIFICATION LEVEL	SIZE	CODE IDENT NO.	DWG NUMBER
						UNCLASSIFIED	A	14213	PS314643
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6.3 Radiography. Four tangential radiographs of the entire capsule, one at every 45°, shall be made to assure the strength member and clad welds have 100% joint penetration with no lack of fusion. The welds shall have no evidence of a square corner to assure the occurrence of burn-through. The minimum thickness of the weld shall be equal to or greater than the minimum body wall thickness.

6.4 Ultrasonic Inspection. The strength member weld shall be free from ultrasonic responses which exceed by 50% the signal from a .051 mm simulated lack of penetration weld defect. Strength member welds having ultrasonic responses exceeding these parameters will be subjected to tangential radiographic evaluation of the area exhibiting anomolous signals. These welds will be acceptable only if there is radiographic evidence of a concave fillet and minimum wall thickness.

6.5 Leak Test. The liner, strength members, and clad assemblies shall be helium leak checked. The liner shall have a leak rate equal to or less than 5×10^{-10} cc/sec helium, while the strength member and clad shall each have leak rates equal to or less than 1.0×10^{-11} cc/sec helium.

7. A D D I T I O N A L T E S T I N G .

MRC-Mound Facility shall perform any additional operations or tests which are considered necessary to insure the quality of the completed product. SLA shall be informed of any additional work. It will be documented and available for review at Mound Facility.

8. H I S T O R I C A L S A M P L E S .

Representative historical samples of each lot of yttrium, Hastelloy C, and T-111 shall be retained for a minimum of 26 1/2 years after the particular lot has been completely used for production units. The quantity of yttrium shall be 5 grams, and the quantity of Hastelloy C and T-111 shall each be 7742 square millimeters.

9. P A C K A G I N G A N D S H I P P I N G .

The fueled heat sources shall be packaged and shipped according to DOE 5480.1, Chg. 3 and applicable DOT regulations.

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10. R E C O R D S.

Appropriate quality documentation and records shall be retained to confirm that all specifications have been met. These records shall be kept in enough detail to allow the following of a logical course of action in solving any MC2893 production or MC2730 problems which may arise.

10.1 Records shall be retained for a period of 26 1/2 years after unit fabrication has been completed. Retained records shall consist of:

10.1.1 Substantiation by supplier's data and/or MRC verification that the materials used in the manufacture of the MC2893 heat source meet the applicable specification(s).

10.1.2 Documentation showing traceability of all hardware components, fuel, and getter used to fabricate each MC2893.

10.1.3 Analytical and fuel preparation data on the fuel with which the assembly was loaded.

10.1.4 All manufacturing and fabrication data.

10.1.5 Complete inspection and test results obtained from component and assembly processing.

10.1.6 Unusual occurrences during the manufacture of each unit.

ISSUE	A				DWG CLASSIFICATION LEVEL		SIZE	CODE IDENT NO.	DWG NUMBER
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SPECIFICATIONS FOR FABRICATIONS DOUBLY ENCAPSULATED CALORIMETER THERMAL
STANDARDS

1. GENERAL

- 1.1 Scope - This specification defines the design criteria, acceptance criteria and quality control requirements for primary and secondary encapsulation of plutonium-238 for calorimeter thermal standards. Re-encapsulation of existing heat sources for use as calorimeter standards lies within the scope of this document provided the previously-used containment vessels and fuels meet specifications as defined in paragraph 2.1 below, or are compatible with same.

2. DOCUMENTS

- 2.1 Applicable Documents: The following documents of exact issue shown or of latest issue if none is shown, shall form a part of this specification to the extent specified herein.

2.1.1 Material Specifications - Primary Encapsulation

- a. ASTM-B364 Specification for Tantalum-10% Tungsten alloy plate sheet, bar and seamless tubing.
- b. SPA-750034 Mound Material Specification for T-111 alloy plate, sheet, bar and seamless tubing.

2.1.2 Material Specifications - Secondary Encapsulation

- a. AMS-5537 Material specification for cobalt-chromium-tungsten-nickel alloy, welded and seamless tubing, sheet and bar.
- b. SPA-750029 Mound material specification for Hastelloy-C sheet and plate.
- c. SPA-790139 Mound material specification, platinum-30% rhodium alloy, plate, sheet, bar and seamless tubing.

2.1.3 Fuel Specification

- a. SPA-780744 Mound plutonium-238 dioxide specification.
- b. 26Y-318000 LANL fuel specifications for MC2893A and MC-3599 heat sources.

2.1.4 Standards and Procedures

- a. ASTM-E190 Guided bend test for ductility of welds.
- b. ASTM-E165 Liquid penetrant inspection.

2.2 Precedence of Documents: In the event of conflict between this specification and any other document, this specification shall prevail.

3. DESIGN REQUIREMENTS

3.1 Primary Encapsulation: The primary encapsulation container shall be fabricated from any of the materials specified in paragraph 2.1.1 above, and must meet or exceed:

- a. Design life = 20 years.
- b. Capsule to withstand a 1900°F fire for 30 minutes at end of life.
- c. Capsule to withstand a drop from a height of 3.0 meters.

3.2 Secondary Encapsulation: The secondary encapsulation container shall be fabricated from any of the materials specified in paragraph 2.1.2 above and must meet or exceed:

- a. Design life = 20 years.
- b. Capsule to withstand a 1900°F fire for 30 minutes at end of life.
- c. Capsule to withstand a drop from a height of 3.0 meters.

3.3 Design Information: The supplier of calorimeter thermal standards shall maintain design information in such a form as to assure compliance to known design intent or acquire MRC-Mound's approval of encapsulation designs.

4. ENCAPSULATION REQUIREMENTS

4.1 Fuel: The fuel shall be plutonium-238 dioxide as specified in paragraph 2.1.3 a or b above.

4.2 Thermal Output: The heat source fuel loading shall be specified via MRC-Mound's procurement contract.

4.3 Contamination: The outer surface of the primary container shall be <2000 disintegrations per minute. The outer surface of the secondary container shall be <10 disintegrations per minute.

4.4 Total Neutron Emission Rate: The total neutron emission rate shall be determined on at least one completed calorimeter thermal standard of a unique design.

4.5 Gamma and Neutron Dose Rates: The gamma and neutron dose rates shall be determined on at least one completed calorimeter thermal standard of an unique design.

4.6 Fuel Loading and Welding Atmosphere: The helium atmosphere for fueling, and welding operations shall have a monitored oxygen value <25 ppm, a monitored water value <60 ppm, and a monitored nitrogen value <160 ppm. In the event the atmosphere purity cannot be maintained, a longitudinal face bend test will be performed per ASTM-E190 on a representative sample of the base material welded without torch gas (per paragraph 4.7). The atmosphere is usable for welding if there are no cracks in the convex surface of the bent specimen (cracks occurring on corners of specimen during testing are not to be considered).

4.7 Welding Gases: The primary and secondary containment welds shall be performed using a GTAW torch gas with a purity of 99.996% minimum (gas shall be argon, helium or a mixture of the two).

5. WELD ACCEPTANCE

5.1 Example Weld: An example weld will be made immediately prior to and following each production run to assure that the welding machine and fixturing is operating properly to meet weld design requirements. (A production run is defined as a continuous work period with the same operators, equipment and weld-joint design). The example welds shall be removed from parent material approximately 6mm from edge of weld to facilitate inspection of weld underbeads. Example weld hardware will be visually and dye penetrant inspected (per "ASTM-E165 using an ultra-sensitive fluorescent penetrant such as Tracor Tech P-149, or MRC-Mound approved equivalent) at 10 to 30X to determine:

5.1.1 Cracks: There shall be no visible cracks in the weld or base metal adjacent to the weld, and

5.1.2 Surface Pores: There shall be no detectable surface pores in the weld exceeding 0.005" diameter. The total surface porosity (internal and external to weld penetration) in one inch of weld length shall not exceed 5% of an area equal to (1*t) square inches, where t equals thickness of the thinnest base material.

5.2 Production Welds: All production welds will be visually and dye penetrant inspected at 1X or greater to assure external surfaces comply to paragraphs 5.1.1 and 5.1.2 above. Additionally, each production weld will be subjected to:

5.2.1 Radiography: Four tangential radiographs (one at every 45°) shall be made of the capsule welds. The following criteria shall be used to evaluate the radiographs.

5.2.1.1 Cracks: There shall be no cracks in the weld or in the adjacent base metal.

5.2.1.2 Incomplete Penetration: There shall be no evidence of a square corner remaining on the underbead side of the weld joint. The minimum thickness of the weld shall be equal to or greater than the thickness of the thinnest member at the weld joint.

5.2.1.3 Lack of Fusion: There shall be no indication of lack of fusion in the weld.

5.2.1.4 Tungsten Inclusions: There shall be no indications of tungsten inclusions in the weld.

5.2.1.5 Porosity: The following are the acceptance criteria for internal weld porosity. All pores must be generally circular in shape with no sharp corners or tails. There shall be no pore with a diameter greater than 40% of the thickness, t , of the thinnest member. The total projected area of all the pores in one inch of weld length shall not exceed 5% of the vertical cross-sectional area of " t " square inches (1-inch length times the thickness, t , of the thinnest member at the weld joint times 5%).

5.2.2 Leak Test: The welded assembly shall have a leak rate less than or equal to 1×10^{-6} std cc/sec as determined by a mass spectrometer type helium leak detector.

6. CAPSULE MARKING

Each final assembly will be clearly and permanently marked with the identity as prescribed in the procurement contract. Laser engraving shall be used. Other methods may be used only by prior agreement with MRC-Mound.

7. QUALITY ASSURANCE PROVISIONS

7.1 Quality Control Program: The supplier shall establish and maintain a quality control program satisfactory to MRC-Mound that will provide product quality consistent with this specification and known design intent. MRC-Mound shall be accorded the privilege of auditing procurement, fabrication and inspection processes to assure that supplier's quality control program is functioning properly.

7.2 Procurement: Control over procurement sources shall be maintained by the supplier of calorimeter thermal standards to ensure that raw materials, ancillary materials and/or purchased services conform to this specification. Purchase orders and/or subcontracts shall be controlled to ensure incorporation of pertinent technical and quality requirements including MRC-Mound authorized changes. Records of supplier's inspections performed on purchased materials shall be maintained.

7.3 Task Information: The supplier shall provide, document and maintain a task definition which will completely describe all activities pertinent to the manufacture, inspection/testing, cleaning and welding of calorimeter thermal standards hardware components and assemblies.

- 7.4 Component Identification: The serial number of each component shall be used to control the parts and assembly to assure traceability throughout the supplier's manufacturing pipeline from time of procurement of raw materials to final disposition as a calorimeter thermal standard.
- 7.5 Rework to Specification: Units which have failed a test or requirement may be reworked per paragraph 6.3 (task definition) provided the final product meets all requirements of this specification. Reworking efforts performed outside the conventional manufacturing techniques must be documented and submitted to MRC-Mound at the time of product acceptance.
- 7.6 Record Retention: The supplier of calorimeter thermal standards shall maintain retrievable records containing all processing information for a period of life of the source from date of shipment of final products.
- 7.7 Responsibility for Inspection: The supplier of calorimeter thermal standards shall be responsible for performance of tests to determine production conformance prior to submission of material to MRC-Mound. MRC-Mound reserves the right to perform any test deemed necessary to assure compliance with requirements of this specification. When there is conflict between supplier's result and MRC-Mound's, MRC-Mound shall have the authority for final disposition of acceptance or rejection.
- 7.8 Data Package: The supplier of calorimeter thermal standards must provide substantiating evidence of product quality at the time the materials are submitted to MRC-Mound for acceptance. The data package shall consist of:
- a. A listing of all non-conformances to product drawings and this specification.
 - b. A listing of all non-conformances to product drawings and this specification.
 - c. The total plutonium-238 inventory of heat sources.
 - d. The specific neutron emission rate for each heat source.
 - e. The Health Physics neutron and gamma dose rate measurements at time of shipment.
 - f. The surface contamination wipe counts.
 - g. Helium leak check reports.
 - h. Dye penetrant and visual inspection reports.
 - i. Radiography reports.
- 7.9 Notification of Tests: If the procurement contract states that tests are to be monitored by a MRC-Mound representative at the source, the calorimeter thermal standards supplier shall notify MRC-Mound at least three working days prior to scheduled start of tests.

8. PACKAGING AND SHIPPING

8.1 Requirements: The fueled heat sources shall be packaged and shipped according to DOE-0529 and applicable DOT regulations.

References

1. DOT-6M Safety Analysis Report - Draft Report, USDOE, March 1986.
2. J. A. Andersen, T. A. Duffey, S. A. Dupree, and R. H. Nilson, PARC (Plutonium Accident Resistant Container) Program Research, Design, and Development, NUREG/CR-0030, Sandia Laboratory (July 1978).
3. J. M. Taylor, Gas Leak Characteristics of Inner Packaging Components Used in the DOT-Spec 6M Container, PNL-5591, Pacific Northwest Laboratory (September 1985).
4. W. R. Amos, Milliwatt Generator Heat Source Progress Report, July 16, 1973 - August 15, 1973, Mound Laboratory.
5. P. E. Teaney, W. B. Cartmill, and R. L. Wise, Testing and Evaluation of Doubly Impacted Simulant Fueled Milliwatt Generator Heat Sources, MLM-MU-82-64-0001, Mound Facility (April 9, 1982).

5. SHIELDING EVALUATION

5.1. Discussion and Results

The USA/5790/BLF(DOE-AL) and USA/5791/BLF(DOE-AL) packages meet the radiation dose-rate standards prescribed in the NRC 10 CFR regulation. Tables 5-1 and 5-2 summarize the gamma dose rates for the side and end geometries, respectively, for both the 5790 and 5791. Table 5-3 gives the total dose rate (gamma and neutron) for plutonium-238 (oxide and nitrate forms) for both the 5790 and 5791. Table 5-4 summarizes the gamma spectra for the appropriate materials and quantities whereas Tables 5-5, 5-6, 5-7, 5-8, and 5-9 illustrate the detailed gamma spectra. Table 5-10 illustrates the detailed neutron source spectrum for plutonium-238. Tables 5-11, 5-12, 5-13, and 5-14 illustrate the detailed dose rates (mrem/hr) per energy range, quantity of material, and density of material for both the oxide and nitrate forms. Table 5-15 provides the atom densities (atoms/barn-cm) per material and density.

The materials that will be transported in the 5790 and 5791 packages will not require shielding for the dose rates to be within allowable limits. Dose calculations were made one inch (2.54 cm) inside of the outside drum, to simulate the worst possible accident condition, and three feet (91.44 cm) from the container to simulate a normal condition. As the data in Tables 5-1, 5-2, and 5-3 show, the doses are well within allowable limits for both conditions.

Dose rates have been calculated for the former package curie limits. In some cases, these limits have been reduced. Although the packages meet dose requirements even for the former quantities, contents are restricted to the following amounts: for the 5790, authorized contents are up to 20.5 kg of uranium-235, up to 4.1 kg of plutonium-239, up to 4.9 kg of uranium-233, or up to 52 g of plutonium-238. The 5791 is limited to 16.8 kg of uranium-235, 5.0 kg of plutonium-239, 6.7 kg of uranium-233, or 73 g of plutonium-238.

5.2. Source Specification

The source strengths used [uranium-233 6.7 kg 5790/5791, uranium-235 20.5 kg 5790/5791, plutonium-239 5.0 kg 5790/5791, plutonium-238 910 Ci (52 g)-5790, and 1270 Ci (73 g)-5791] are greater than or equal to the maximum authorized quantities. As shown in the attendant tables, both the oxide and nitrate forms were analyzed.

A maximum density of 9.5 g/cm^3 is allowed, but the analysis was not performed at that density because of self absorption/attenuation. To produce a conservative approach and to reduce the effect of self attenuation, 2.375 g/cm^3 was chosen as a benchmark density (refer to Table 5-1 for actual densities). The exception to this approach is uranium-235 for the 5790. The minimum density necessary to fit the entire maximum authorized quantity of 20.5 kg into the 5790 package is 8.54 g/cm^3 , and an analysis was performed at this density. An examination was also performed with 11.4 kg of uranium-235 in the 5790 at a density of 4.75 g/cm^3 to see if there was an impact upon dose (refer to Table 5-1).

5.2.1. Gamma Source

Gamma source strengths and spectra used in this evaluation were derived by examining the decay of the radioactive material from the Table of Isotopes [2,3].

Uranium-235 (half-life 7.1×10^8 yr) decays to thorium-231 (half-life 25.5 hr) and then to protactinium-231 (half-life 3.2×10^4 yr). Table 5-9 reflects this decay by the conservative addition of a 0.290 MeV gamma from protactinium-231. Spontaneous fission gammas [4] from uranium-235 were acknowledged by the addition of a 0.750 MeV gamma.

Plutonium-239 (half-life 2.44×10^4 yr) decays to uranium-235, and it was assumed that plutonium-239 constitutes 90% of the radioactive material. In addition, it was conservatively assumed that the other 10% was plutonium-241 (half-life 13.2 yr) and that it had been 13.2 yr since the creation of plutonium-241. This approach leads to plutonium-239 constituting 90% of the radioactive material with plutonium-241 and americium-241 constituting 5% each. The decay scheme for the plutonium-239 family is shown in Table 5-6.

Spontaneous fission gammas from plutonium-239 were considered but, because of that element's long spontaneous fission half-life (5.5×10^{15} yr) and low gamma production (fission gammas 2.1×10^{-12} Ci/g and fission product gammas 1.4×10^{-12} Ci/g compared to the 0.12 and 0.38 MeV gammas which contribute 6×10^{-7} Ci/g each) [5], their contribution to the source team was deemed insignificant.

Plutonium-238 was analyzed at two quantities: 910 Ci (52 g) for the 5790 and 1270 Ci (73 g) for the 5791. The plutonium-238 family consisted of 80 at.% plutonium-238, 16 at.% plutonium-239, 3 at.% plutonium-240, and considerably lesser quantities of plutonium-236, plutonium-241, and plutonium-242. For this analysis, only plutonium-238, plutonium-239, and plutonium-240 were considered; and it was assumed that it had been 15 yr since the creation of these elements. This age was used to calculate the uranium-234 daughter, which is presented in Tables 5-5 and 5-7. Because uranium-234 (half-life 2.47×10^5 yr), plutonium-239 (half-life 2.44×10^4 yr), and plutonium-240 (half-life 6.6×10^3 yr) have such long half-lives, their daughter contributions were not examined. Fission and fission product gammas are also presented in Tables 5-5 and 5-7.

The gamma decay scheme for 6.7 kg of uranium-233 is shown in Table 5-8. Uranium-233 decays (half-life 1.62×10^5 yr) to thorium-229. Because of its (uranium-233) long half-life, the thorium-229 contribution will be limited and is not considered in this study.

5.2.2. Neutron Source

The neutron source for both uranium-235 and plutonium-239 are from spontaneous fission. The spontaneous fission half-lives for uranium-235 and plutonium-239 are 9×10^{17} yr and 5.5×10^{15} yr, respectively [3]. Calculating the spontaneous fission activity (from the half-lives, decay constant, and number of atoms at time = 0) gives six fission events per second for uranium-235 and 50 fission events per second for plutonium-239. This produces an approximate three neutron per fission event [2,4] that yields 18 neutrons per second for uranium-235 and 150 neutrons per second for plutonium-239. These neutrons contribute very little to the total dose rate and were disregarded.

Uranium-233 and its daughter thorium-229 are not neutron emitters. Table 5-10 gives the neutron source terms for plutonium-238, PuO_2 (both

plutonium-238 and plutonium-240), and plutonium-240. The average neutron energy is approximately 2 MeV [5]; the (α , n) average energy is $\approx$ 2.25 MeV, the fission spectrum average energy $\approx$ 1.00 MeV, and the spontaneous fission average energy $\approx$ 1.00 MeV. Table 5-16 shows the calculations for the neutron dose rates. For these calculations, the geometry of the source was taken to be a point source. Table 5-3 illustrates the total (gamma and neutron) dose rate for the plutonium-238 family.

5.3. Model Specification

5.3.1. Description of Geometric Configuration

Both the 5790 and 5791 packages were modeled on the MICROSHIELD two-dimensional computer code. Point-kernal multigroup integration, with a minimum of 2400 differential source elements, was the calculational method. Dose rates were calculated for both the side and end of the cylinder. As illustrated in Figures 5-1 and 5-2, there were two dose points. The first point (one inch inside the outer container) was chosen to simulate the worst accident condition, with the second dose point being three feet from the outer container.

Table 5-17 gives the source dimensions for the materials and containers. Because the radioactive material is in a powder form, the source geometry could be of varying dimensions and shape. The philosophy was to choose a shape that increased surface area and reduced self attenuation with respect to the dose point thereby yielding the highest dose. Note that the two dose points (side and end) "view" different configurations. The side dose point viewed a cylinder with as small as reasonable radius and as long as possible height. Whereas the end dose point "sees" a dish with a relatively large radius and small height. This approach could not be adhered to when the quantity of material filled or nearly filled the available volume, generally the 5790. For both the end and side view, the source was placed in the center of the container. Center being defined as the appropriate middle of the container either the horizontal or vertical midline.

5.3.2. Shield Regional Densities

The inner container for both packages (the 5790 and the 5791) is steel. The main components were input into the program at the appropriate densities. (The remainder of the space not filled by the source or steel was designated as air. This approach gives no credit for the Firedike insulation as a gamma attenuator.)

For both the 5790 and 5791, the top plates of the inner containers were thicker than the bottom plates. To produce a conservative dose rate, the bottom plate thickness (5790-0.5 in., 5791-0.48 in.) was used for the appropriate end dose point.

5.4. Shielding Evaluation

Radiation dose calculations were made using MICROSHIELD [1], a two-dimensional, point-kernal multigroup integration, microcomputer program. MICROSHIELD is a PC version of ISOSHLI II (Engel, et al., 1966). Point-kernal integration involves: first, dividing the source into small differential sources (for these calculations never less than 2,400

FIGURE 5-1 - Shielding dimensions for the 5790 package.

FIGURE 5-2 - Shielding dimensions for the 5791 package.

differential sources, this number of differential sources produces a less than 2% variations between the number of differential sources and a larger number of differential sources), and second, the dose from each volume element is calculated as a point source. The procedure considers the amount of shield material between each volume element and the dose point and takes buildup and attenuation into account. The total dose is the sum of the doses from all the volume elements. Dose calculations were conducted for both the side and end of a cylindrical model of the shipping container.

5.5. Benchmark

The ISOSHL D radiation shielding computer code is being used throughout the industry. Applications include spent fuel handling, storage and shipping, waste management, emergency planning, and operational health physics.

The accuracy of MICROSHIELD, which was used for this report, is essentially that for the public domain code ISOSHL D II, which it emulates. MICROSHIELD is a personal computer version of ISOSHL D programmed in BASIC.

In his paper, "ISOSHL D: Evolution, Experience, and Benchmarking," [8] Jess Greenberg, INPO, Atlanta, Georgia, compares MICROSHIELD to other programs with respect to ANSI/ANS 6.6.1-1979 "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Plants."

Benchmarking for this SARP was accomplished by duplicating ANSI/ANS 6.6.1 Problem II.2 and comparing the results to those presented by J. Greenberg.

The problem consists of a 12 ft diameter tank, 35 ft high, containing water with a uniformly distributed source emitting 0.8 MeV photons at a rate of $4.31 \text{ E9 photons/s}$.

Figure 5-3 shows the geometry and Table 5-18 gives the results. The reference codes referred to are COHORT II, QADMOD, and DOT. As can be seen, the MICROSHIELD used for 5790 and 5791 compares very well with other published and accepted calculations.

References

1. MICROSHIELD - A program for analyzing gamma radiation shielding - Version 2.1., Grove Engineering Inc., Washington Grove, Maryland.
2. S. Lederer et al., Table of Isotopes, 7th Ed., John Wiley and Sons, 1978..
3. S. Lederer et al., Table of Isotopes, 6th Ed., John Wiley and Sons, 1967.
4. W. Cember, Introduction to Health Physics, 2nd Ed., Pergammon, 1976.
5. Plutonium Handbook - A Guide to the Technology, Volumes I and II, O. J. Wick (ed.), American Nuclear Society, 1980.
6. T. Rockwell II, Reactor Shielding Design Manual, 1st Ed., D. Van Nostrand Co., Inc., 1956.

7. ANSI/ANS 6.6.1 - 1979 "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Plants," ANS, La Grange, IL.
8. J. Greenborg "ISOSHL: Evaluation, Experience, and Benchmarking," presented at the Conference on Theory and Practices in Radiation Protection and Shielding, 22-24 April 1987.

FIGURE 5-3 - Benchmark geometry.

TABLE 5-1

GAMMA DOSE RATES^a FOR THE SIDE OF THE 5790 AND 5791 PACKAGES

²³⁵ U (4.75 g/cm ³ - 11.4 kg) UO ₂ UO ₂ (NO ₃) ₂		²³³ U (2.37 g/cm ³ - 6.7 kg) UO ₂ UO ₂ (NO ₃) ₂		²³⁹ Pu (2.375 g/cm ³ - 5.0 kg) PuO ₂ PuO ₂ (NO ₃) ₂		²³⁸ Pu (2.9 g/cm ³ - 52 g) (2.4 g/cm ³ - 52 g) PuO ₂ PuO ₂ (NO ₃) ₂	
00.71 ^b 00.90 ^b (00.71) (00.90)		1.59 2.05		5790 <u>Inside Outer Container</u> 60.3 63.8		34.31 33.05	
00.00 ^b 00.00 ^b (00.00) (00.00)		00.04 00.05		5790 <u>3 ft from Container</u> 1.6 1.5		00.59 00.57	
(2.37 g/cm ³ - 20.5 kg) (2.37 g/cm ³ - 20.5 kg) 1.2 1.5		1.8 1.9		5791 <u>Inside Outer Container</u> 39.1 40.6		(2.37 g/cm ³ - 73 g) (2.37 g/cm ³ - 73 g) 25.36 24.53	
0.1 0.1		0.1 0.1		5791 <u>3 ft from Container</u> 1.7 1.8		00.9 00.9	

^aIn mrem/hr.^b²³⁵U 8.54 g/cm³ - 20.5 kg.

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TABLE 5-2

GAMMA DOSE RATES^a FOR THE END OF THE 5790 AND 5791 PACKAGES

²³⁵ U (4.75 g/cm ³ - 11.4 kg) <u>UO₂</u> <u>UO₂(NO₃)₂</u>		²³³ U (2.37 g/cm ³ - 6.7 kg) <u>UO₂</u> <u>UO₂(NO₃)₂</u>		²³⁹ Pu (2.375 g/cm ³ - 5.0 kg) <u>PuO₂</u> <u>PuO₂(NO₃)₂</u>		²³⁸ Pu (2.9 g/cm ³ - 52 g) (2.4 g/cm ³ - 52 g) <u>PuO₂</u> <u>PuO₂(NO₃)₂</u>	
1.4 ^b (1.41) ^b		3.1 4.0		5790 <u>Inside Outer Container</u> 72.6 71.1		40.9 39.8	
0.0 ^b (0.00) ^b		0.0 0.0		5790 <u>3 ft from Container</u> 0.7 0.6		0.8 0.7	
(2.37 g/cm ³ - 20.5 kg) (2.37 g/cm ³ - 20.5 kg) 0.8 1.1		0.8 0.7		5791 <u>Inside Outer Container</u> 21.6 22.2		(2.4 g/cm ³ - 73 g) (2.4 g/cm ³ - 73 g) 24.9 28.1	
0.0 0.0		0.0 0.0		5791 <u>3 ft from Container</u> 1.2 1.2		1.2 0.6	

^aIn mrem/hr.^b²³⁵U 8.54 g/cm³ - 20.5 kg.

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TABLE 5-3

^{238}Pu TOTAL END DOSE RATE^a FOR THE 5790 AND 5791 PACKAGES^b

Package	Dose Point Location	PuO_2			$\text{PuO}_2(\text{NO}_3)_2$		
		Gamma	Neutron	Total	Gamma	Neutron	Total
5790	Inside Container	40.9	74.8	115.7	39.8	74.8	114.6
	3 ft from Container	0.8	1.0	1.8	0.7	1.0	1.7
5791	Inside Container	24.9	40.7	65.6	28.1	40.7	68.8
	3 ft from Container	1.2	1.3	2.5	0.6	1.3	1.9

^aIn mrem/hr.

^bThe highest dose rates (end of package for ^{238}Pu) were chosen for tabulation in this chart.

TABLE 5-4

GAMMA SPECTRA^a FOR THE APPROPRIATE RADIOISOTOPES AND QUANTITIES

Energy Group MeV	²³⁵ U		²³³ U	²³⁹ Pu	²³⁸ Pu	
	(20.5 kg)	(11.4 kg)	(6.7 kg)	(5.0 kg)	(52 g)	(73 g)
0.150	1.218×10^9	6.792×10^8	3.691×10^8	1.762×10^9	2.769×10^9	3.237×10^9
0.250	1.624×10^8	9.056×10^7	1.766×10^8	2.133×10^8	1.431×10^6	1.998×10^6
0.350				5.100×10^8	3.503×10^5	4.885×10^5
0.450				6.660×10^7	1.562×10^5	2.178×10^5
0.650	1.22×10^2	0.69×10^2		2.400×10^8	9.394×10^5	1.311×10^6
0.825					2.320×10^7	3.237×10^7

^aIn photon/s.

²³⁸Pu FAMILY DECAY GAMMAS FOR THE 5790 PACKAGE

²³⁸ Pu (52 g) ^a				²³⁴ U (1.339 x 10 ⁹ dis/s)		²³⁹ Pu (11 g)		
MeV	Ci/g	Ci	Photons/s	Fraction	Photons/s	Ci/g	Ci	Photons/s
0.100	1 x 10 ⁻³	5.2 x 10 ⁻²	1.924 x 10 ⁹					
0.117				4 x 10 ⁻⁷	5.33 x 10 ²			
0.120						6 x 10 ⁻⁷	6.312 x 10 ⁻⁶	2.34 x 10 ⁵
0.150	2 x 10 ⁻⁴	0.010	3.87 x 10 ⁸					
0.203	7 x 10 ⁻⁷	3.66 x 10 ⁻⁵	1.35 x 10 ⁶					
0.207						2 x 10 ⁻⁷	2.1 x 10 ⁻⁶	7.78 x 10 ⁴
0.340						3 x 10 ⁻⁷	3.16 x 10 ⁻⁶	1.17 x 10 ⁵
0.380						6 x 10 ⁻⁷	6.31 x 10 ⁻⁶	2.34 x 10 ⁵
0.420						4 x 10 ⁻⁷	4.21 x 10 ⁻⁶	1.56 x 10 ⁵
0.480				4 x 10 ⁻⁷	5.33 x 10 ²			
0.580				1.2 x 10 ⁻⁷	1.60 x 10 ²			
0.750 ^b	4.8 x 10 ⁻⁷	2.51 x 10 ⁻⁵	9.28 x 10 ⁵					
	1.53 x 10 ⁻⁷	1.53 x 10 ⁻⁷	1.12 x 10 ⁴					
			(²⁴⁰ Pu)					
0.760	9 x 10 ⁻⁶	4.70 x 10 ⁻⁴	1.74 x 10 ⁷					
0.875	3 x 10 ⁻⁶	1.57 x 10 ⁻⁴	5.8 x 10 ⁶					

^a910 Ci.

^bFission and fission product gammas.

TABLE 5-6

 ^{239}Pu FAMILY DECAY GAMMAS FOR THE 5790 AND 5791 PACKAGES

^{239}Pu (4500 g) ^a				^{241}Am (250 g)		^{241}Pu (250 g)	
MeV	Ci/g	Ci	Photons/s	Fraction	Photons/s	Fraction	Photons/s
0.120	5.6×10^{-6}	2.52×10^{-2}	9.32×10^7				
0.145						1.6×10^{-6}	1.66×10^9
0.207	2.0×10^{-7}	9×10^{-4}	3.33×10^7				
0.208				6×10^{-6}	1.8×10^8		
0.335				8×10^{-6}	2.4×10^8		
0.340	3.0×10^{-7}	1.35×10^{-3}	4.99×10^7				
0.380	6.0×10^{-7}	2.7×10^{-3}	9.99×10^7				
0.370				4×10^{-6}	1.2×10^8		
0.420	4.0×10^{-7}	1.8×10^{-3}	6.66×10^7				
0.663				5×10^{-6}	1.5×10^8		
0.722				3×10^{-6}	9.0×10^7		

^aContents of the 5790/5791 limited to 5.0 kg ^{239}Pu .

TABLE 5-7

 ^{238}Pu FAMILY DECAY GAMMAS FOR THE 5791 PACKAGE

^{238}Pu (73 g) ^a				^{234}U (1.86×10^9 dis/s)		^{239}Pu (15 g)		
MeV	Ci/g	Ci	Photons/s	Fraction	Photons/s	Ci/g	Ci	Photons/s
0.100	1.0×10^{-3}	7.29×10^{-2}	2.7×10^9	4×10^7	7.44×10^2			
0.117								
0.120						6×10^{-7}	8.8×10^{-6}	3.26×10^5
0.150	2.0×10^{-4}	1.58×10^{-2}	5.4×10^8					
0.203	7.0×10^{-4}	5.1×10^{-5}	1.9×10^6					
0.207						2×10^{-7}	2.9×10^{-6}	1.09×10^5
0.340						3×10^{-7}	4.4×10^{-6}	1.23×10^5
0.380						6×10^{-7}	8.8×10^{-6}	3.26×10^5
0.420						4×10^{-7}	5.87×10^{-6}	2.17×10^5
0.480				4×10^{-7}	7.44×10^2			
0.580				1.2×10^{-7}	2.31×10^2			
0.750 ^b	4.8×10^{-7}	3.5×10^{-5}	1.3×10^6					
	1.5×10^{-7}	4.2×10^{-7}	1.6×10^4					
			(^{240}Pu)					
0.760	9.0×10^{-6}	6.6×10^{-4}	2.4×10^7					
0.875	3×10^{-6}	2.2×10^{-4}	8.1×10^6					

^a1270 Ci.^bFission and fission product gammas.

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TABLE 5-8

^{233}U (6.7 kg) DECAY GAMMAS FOR THE 5790 AND 5791 PACKAGES

<u>MeV</u>	<u>Fraction</u>	<u>Photons/s</u>
0.100	3.8×10^{-5}	8.89×10^7
0.119	4.0×10^{-5}	9.36×10^7
0.121	2.0×10^{-5}	4.68×10^7
0.124	7.0×10^{-6}	1.64×10^7
0.140	1.0×10^{-6}	2.34×10^6
0.145	9.0×10^{-6}	2.11×10^7
0.146	4.0×10^{-5}	9.36×10^7
0.163	2.4×10^{-6}	5.62×10^6
0.208	2.4×10^{-5}	5.62×10^7
0.216	8.0×10^{-6}	1.87×10^7
0.217	3.5×10^{-5}	8.19×10^7
0.256	3.0×10^{-6}	7.02×10^6
0.259	2.1×10^{-6}	4.91×10^6
0.262	3.2×10^{-6}	7.49×10^6

TABLE 5-9

^{235}U DECAY GAMMAS FOR 20.5 kg (8.54 g/cm³) AND 11.4 kg (4.75 g/cm³)

<u>MeV</u>	<u>Fraction</u>	<u>20.5 kg</u>	<u>11.4 kg</u>
0.110	0.04	6.50×10^7	3.62×10^7
0.143	0.11	1.79×10^8	9.96×10^7
0.163	0.05	8.12×10^7	4.53×10^7
0.185	0.55	8.93×10^8	4.98×10^8
0.205	0.06	9.74×10^7	5.43×10^7
0.290	0.04(^{231}Pa)	6.50×10^7	3.62×10^7
0.750 ^a		1.22×10^2	0.69×10^2

^aSpontaneous fission.

Table 5-10

NEUTRON SOURCES FOR ^{238}Pu FAMILY FOR THE 5791 AND 5790 PACKAGES

5791				
^{238}Pu (73 g) <u>(neutrons/s)</u>	PuO_2 (73 g ^{238}Pu) <u>(neutrons/s)</u>	^{240}Pu (2.76 g) <u>(neutrons/s)</u>	PuO_2 (2.76 g ^{240}Pu) <u>(neutrons/s)</u>	
8.4 x 10 ³	6.13 x 10 ⁵			
1.4 x 10 ⁴	1.022 x 10 ⁶			
1.02 x 10 ³		2.815 x 10 ³		
1.7 x 10 ²			4.692 x 10 ²	
5790				
^{238}Pu (52 g) <u>(neutrons/s)</u>	PuO_2 (52 g ^{238}Pu) <u>(neutrons/s)</u>	^{240}Pu (2.00 g) <u>(neutrons/s)</u>	PuO_2 (2.00 g ^{240}Pu) <u>(neutrons/s)</u>	
8.4 x 10 ³	4.37 x 10 ⁵			
1.4 x 10 ⁴	7.28 x 10 ⁵			
1.02 x 10 ³		2.04 x 10 ³		
1.7 x 10 ²			3.4 x 10 ²	

TABLE 5-11

GAMMA DOSE RATES PER ENERGY GROUP FOR ^{233}U AT 6.7 kg - 2.77 g/cm³ FOR BOTH PACKAGES

<u>Energy Group</u>		<u>5791 Dose Rate (mrem/hr)</u>			
		<u>UO₂</u>		<u>UO₂ (NO₃)₂</u>	
		<u>Side</u>	<u>End</u>	<u>Side</u>	<u>End</u>
<u>MeV</u>	<u>Photons/s</u>				
0.10-0.20	3.69×10^8	0.444	0.199	0.450	0.169
0.20-0.30	1.77×10^8	1.40	0.581	1.42	0.514
TOTAL	5.46×10^8	1.8	0.8	1.9	0.68

<u>Energy Group</u>		<u>5791 Dose Rate (mrem/hr)</u>			
		<u>UO₂</u>		<u>UO₂ (NO₃)₂</u>	
		<u>Side</u>	<u>End</u>	<u>Side</u>	<u>End</u>
<u>MeV</u>	<u>Photons/s</u>				
0.10-0.20	3.69×10^8	0.294	0.574	0.393	0.774
0.20-0.30	1.77×10^8	1.29	2.52	1.65	3.23
TOTAL	5.46×10^8	1.6	3.1	2.0	4.0

TABLE 5-12

GAMMA DOSE RATES PER ENERGY GROUP FOR ^{235}U AT 11.4 kg - 4.75 g/cm³ AND 20.5 kg - 8.54 g/cm³ FOR BOTH PACKAGES

5790 Dose Rate (mrem/hr)										
^{235}U (11.4 kg - 4.75 g/cm ³)						^{235}U (20.5 kg - 8.54 g/cm ³)				
<u>Energy Group</u> <u>MeV</u>	<u>UO₂</u>		<u>UO₂(NO₃)₂</u>			<u>UO₂</u>		<u>UO₂(NO₃)₂</u>		
	<u>Photons/s</u>	<u>Side</u>	<u>End</u>	<u>Side</u>	<u>End</u>	<u>Photons/s</u>	<u>Side</u>	<u>End</u>	<u>Side</u>	<u>End</u>
0.10-0.20	6.79 x 10 ⁸	0.324	0.625	0.418	0.814	1.22 x 10 ⁹	0.318	0.617	0.413	0.809
0.20-0.30	9.06 x 10 ⁷	0.388	0.779	0.486	0.979	1.62 x 10 ⁸	0.392	0.789	0.492	0.995
TOTAL	7.69 x 10 ⁸	0.7	1.4	0.9	1.8	1.38 x 10 ⁹	0.7	1.4	0.9	1.8

5791 Dose Rate (mrem/hr)						
^{235}U (20.5 kg - 2.37 g/cm ³)						
<u>Energy Group</u> <u>MeV</u>	<u>UO₂</u>		<u>UO₂(NO₃)₂</u>			
	<u>Photons/s</u>	<u>Side</u>	<u>End</u>	<u>Side</u>	<u>End</u>	
0.10-0.20	1.22 x 10 ⁹	0.621	0.457	0.792	0.575	
0.20-0.30	1.62 x 10 ⁸	0.600	0.393	0.738	0.475	
TOTAL	1.38 x 10 ⁹	1.2	0.8	1.5	1.0	

TABLE 5-13

GAMMA DOSE RATES PER ENERGY GROUP FOR ^{239}Pu AT 5 kg - 2.37 g/cm^3 FOR BOTH PACKAGES

5790 Dose Rate (mrem/hr)						5791 Dose Rate (mrem/hr)					
MeV	Photons/s	$^{239}\text{PuO}_2$		$^{239}\text{PuO}_2 (\text{NO}_3)_2$		Photons/s	$^{239}\text{PuO}_2$		$^{239}\text{PuO}_2 (\text{NO}_3)_2$		
		Side	End	Side	End		Side	End	Side	End	
0.10-0.20	1.76×10^7	1.71×10^{-3}	2.66×10^{-3}	2.06×10^{-3}	2.81×10^{-3}	1.76×10^9	1.70	0.941	2.50	1.09	
0.20-0.30	2.13×10^8	1.88	2.74	2.17	2.83	2.13×10^8	1.48	0.841	1.60	0.911	
0.30-0.40	5.10×10^8	15.6	20.8	16.9	20.5	5.10×10^8	10.6	5.82	11.1	6.06	
0.40-0.55	6.66×10^7	4.82	5.87	5.10	5.75	6.66×10^7	3.06	1.66	3.15	1.69	
0.55-0.75	2.40×10^8	38.0	43.2	39.5	40.9	2.40×10^8	22.2	12.3	22.8	12.4	
TOTAL	1.05×10^9	60.3*	72.6	63.8	71.1	1.05×10^9	39.1	21.6	40.6	22.2	

TABLE 5-14

GAMMA DOSE RATES PER ENERGY GROUP FOR ^{238}Pu AT 910 Ci (52 g - 2.37 g/cm^3 - 5790)
AND 1270 Ci (73 g - 2.37 g/cm^3 - 5791)

5790 Dose Rate (mrem/hr)					
MeV	Photons/s	$^{238}\text{PuO}_2$		$^{238}\text{PuO}_2(\text{NO}_3)_2$	
		Side	End	Side	End
0.10-0.20	2.77×10^9	21.6	28.7	20.9	27.8
0.20-0.30	1.43×10^6	8.00×10^{-3}	9.77×10^{-3}	7.52×10^{-3}	9.35×10^{-3}
0.30-0.40	3.50×10^5	5.07×10^{-3}	5.57×10^{-3}	4.75×10^{-3}	5.35×10^{-3}
0.40-0.55	1.56×10^5	3.89×10^{-3}	3.98×10^{-3}	3.67×10^{-3}	3.85×10^{-3}
0.55-0.75	9.39×10^5	0.376	0.370	0.357	0.359
0.75-0.90	2.32×10^7	12.2	11.8	11.7	11.5
TOTAL	2.79×10^9	34.3	40.9	33.1	39.8

5790 Dose Rate (mrem/hr)					
MeV	Photons/s	$^{238}\text{PuO}_2$		$^{238}\text{PuO}_2(\text{NO}_3)_2$	
		Side	End	Side	End
0.10-0.20	3.24×10^9	17.2	19.1	16.7	16.7
0.20-0.30	1.99×10^6	6.25×10^{-3}	5.97×10^{-3}	5.89×10^{-3}	6.16×10^{-3}
0.30-0.40	4.89×10^5	3.58×10^{-3}	2.97×10^{-3}	3.37×10^{-3}	3.93×10^{-3}
0.40-0.55	2.18×10^5	2.64×10^{-3}	2.00×10^{-3}	2.49×10^{-3}	3.23×10^{-3}
0.55-0.75	1.31×10^5	0.246	0.178	0.234	0.327
0.75-0.90	3.24×10^7	7.78	5.53	7.51	10.9
TOTAL	3.27×10^9	25.4	24.9	24.5	28.1

TABLE 5-15

ATOM DENSITY (atoms/barn-cm) PER SOURCE MATERIAL QUANTITY AND DENSITY

<u>^{235}U (5790)</u>		<u>^{233}U (5790/5791)</u>		<u>^{239}Pu (5790/5791)</u>	
(11.4 kg - 4.75 g/cm ³)	(20.5 kg - 8.54 g/cm ³)	(6.7 kg - 2.77 g/cm ³)	(5.0 kg - 2.37 g/cm ³)	<u>^{241}Pu</u>	<u>^{241}Am</u>
atoms/barn-cm	1.22×10^{-2}	2.19×10^{-2}	6.6×10^{-3}	5.98×10^{-3}	3.29×10^{-4}
				3.29×10^{-4}	3.29×10^{-4}

<u>^{238}Pu (5790)</u>				
	(52 g - 2.37 g/cm ³)	<u>^{239}Pu</u>	<u>^{240}Pu</u>	<u>^{234}U</u>
atoms/barn-cm	6.02×10^{-3}	1.22×10^{-3}	2.26×10^{-4}	6.81×10^{-4}

<u>^{238}Pu (5791)</u>				
	(73 g - 2.37 g/cm ³)	<u>^{239}Pu</u>	<u>^{240}Pu</u>	<u>^{234}U</u>
atoms/barn-cm	5.96×10^3	1.19×10^3	2.24×10^4	6.74×10^{-4}

<u>^{235}U (5791)</u>	
	(20.5 kg - 2.735 g/cm ³)
atoms/barn-cm	6.08×10^{-3}

TABLE 5-16

NEUTRON DOSE RATE CALCULATIONS FOR THE ^{238}Pu FAMILY5791

$$\text{(Inside Container)} D_N = \frac{S}{4\pi R^2 C_f} = \frac{1.6383 \times 10^6}{4\pi(19.64)^2(8.3)} = 40.7 \text{ mrem/hr}$$

$$\text{(3 ft from Container)} D_N = \frac{S}{4\pi R^2 C_f} = \frac{1.6383 \times 10^6}{4\pi(111.08)^2(8.3)} = 1.3 \text{ mrem/hr}$$

5790

$$\text{(Inside Container)} D_N = \frac{S}{4\pi R^2 C_f} = \frac{1.1674 \times 10^6}{4\pi(12.23)^2(8.3)} = 74.8 \text{ mrem/hr}$$

$$\text{(3 ft from Container)} D_N = \frac{S}{4\pi R^2 C_f} = \frac{1.167 \times 10^6}{4\pi(103.67)^2(8.3)} = 1.0 \text{ mrem/hr}$$

Where: $C_f = 8.3 \text{ N/S-cm}^2$ per 1 mrem/hr (Ref. 6) at 2 MeV.
 S = Neutron sources from Table 5-10.

TABLE 5-17

SOURCE DIMENSIONS (cm) PER DOSE POINT, MATERIAL, AND CONTAINER

	^{235}U						^{233}U			
	(11.4 kg)		(20.5 kg)				(6.7 kg)			
	5790		5790		5791		5790		5791	
	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>
Side	5.2	28	5.2	28	8.3	40	5.2	28	4.7	40
End	5.2	28	5.2	28	8.3	40	5.2	28	8	14

	^{239}Pu				^{238}Pu			
	(5.0 kg)				(52 g)		(73 g)	
	5790		5791		5790		5791	
	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>	<u>Radius</u>	<u>Hgt.</u>
Side	4.6	28	3.9	40	1.0	7.0	1.0	9.8
End	5.2	22	8.0	9.5	2.7	1.0	3.1	1.0

TABLE 5-18

BENCHMARK RESULTS

Dose Point (ft)	Isoshld II (rad/yr)	Greenborg Microshield (rad/yr)	5790/5791 Microshield (rad/yr)	Reference Code Range (rad/yr)
20	9.33E-1	9.32E-1	9.34E-1	6.0E-1 to 1.2E-0
50	2.11E-1	2.08E-1	2.09E-1	1.2E-1 to 2.2E-1
200	1.16E-2	1.16E-2	1.19E-2	7.5E-3 to 1.2E-2
500	1.16E-3	1.17E-3	1.22E-3	5.4E-3 to 1.2E-3

6. CRITICALITY EVALUATION

6.1. Discussion and Results

The USA/5790/BLF (DOE-AL) and USA/5791/BLF (DOE-AL) containers were analyzed for criticality using the KENO-IV [1] Monte Carlo computer code along with the 16 group Hansen Roach cross section set [2] supplied with the code. This computer code and cross-section set combination is recognized as a standard for criticality analysis. The code has been used extensively in the past, throughout the DOE and elsewhere, and has been successfully checked against many benchmark experiments. Benchmark evaluations conducted for this work indicated that the code is highly accurate in predicting criticality, since the calculated criticality was very close to experimental determinations.

The containers were analyzed for shipping plutonium-238 oxide, uranium-235 metal, plutonium-239 metal, and uranium-233 metal. Fissile Class I, II, and III limits were established in accordance with the criteria set forth in 10 CFR Part 71, and the NRC Regulatory Guide 7.9.

The containers are designed to prevent criticality in that the fissile material is placed in an inner container which, in turn, is fixed in the center of Firedike insulation. The inner container is sealed with an O-ring, and water submersion tests have shown that no in-leakage of water occurs in the inner container. In addition, after the fissile material is placed in the inner container and the container is sealed, the inner container is leak tested prior to each shipment. As an additional precaution associated with plutonium, and in accordance with 10 CFR 71.63, plutonium is placed in sealed capsules before being placed in the inner container. As a result of the water submersion tests and the leak testing of the inner container prior to each shipment, the containers are considered to meet the exemptions for in-leakage as stated in 10 CFR 71.55 part (c). Therefore, in-leakage of water into the inner container was not considered credible and was not analyzed.

The Firedike insulation is sufficiently thick to produce a relatively large spacing between fissile units. The geometry is such that interspersed water between undamaged packages results in a decrease in criticality, since the water absorption outweighs moderation. Any interspersed water for plutonium-238 results in a decrease in criticality under all conditions, undamaged or damaged, since this isotope can become critical only in a fast neutron spectrum. Consequently, no interspersed water was assumed for the evaluation of undamaged packages. The damaged condition assumed that all of the Firedike insulation and the outer drum were removed and the inner containers were in contact. Although drop tests have indicated that this is an almost incredible damaged configuration, it was the most reactive configuration and, for the sake of conservatism, was chosen as the assumed damaged geometry. With this damaged package geometry, interspersed water of 10% of full density water between the inner containers was most reactive with plutonium metal as the fissile material, whereas interspersed water of 20% of full density was most reactive for uranium metal in the damaged condition. Table 6-1 summarizes the parameters used for Fissile Class I, II and III analyses. Infinite arrays for Fissile Class I undamaged package analysis utilized a 2x2x2 geometry with specular reflection, since KENO does not allow specular reflection for a single package. The number of damaged packages

TABLE 6-1

CRITICALITY EVALUATION PARAMETERS

Isotope	Parameter	Fissile Class					
		I		II		III	
		Undam.	Dam.	Undam.	Dam.	Undam.	Dam.
All	No. of pkg.	Infinite	280	25	10	2	1
^{239}Pu	Interspersed Water Density	0	0.1	0	0.1	0	0.1
^{235}U		0	0.2	0	0.2	0	0.2
^{233}U		0	0.2	0	0.2	0	0.2
^{238}Pu Oxide		0	0	0	0	0	0
<u>Reflector</u>							
^{239}Pu	Water (cm)	N/A	30.0	30.0	30.0	30.0	30.0
^{235}U	Water (cm)	30.0	30.0	30.0	30.0	30.0	30.0
^{233}U	Water (cm)	30.0	30.0	30.0	30.0	30.0	30.0
^{238}Pu Oxide	Steel (in.)	N/A	6.0	6.0	6.0	6.0	6.0
All	Transport Index			10.0	10.0		
All	Package Size (cm ³)						
	5790	51,864	4,016	51,864	4,016	51,864	4,016
	5791	227,161	10,551	227,161	10,551	227,161	10,551

analyzed for Fissile Class I was taken as 280, in accordance with NRC practice, rather than the 250 given in the federal regulations. A near cubical system can be produced with an 8x7x5 array, that is, 280 containers, whereas no such combination can be found for 250 containers. This is a result of the height of the containers being about 1 1/2 times the diameter. As indicated earlier, no interspersed water was used for plutonium-238 oxide, since the presence of such water would decrease criticality. Thirty centimeters of full density water was used as a reflector for reflected arrays, except for plutonium-238, which used six inches of stainless steel as the reflector. In all applicable cases, Fissile Class II limits were dictated by the damaged configuration.

Parametric studies resulted in additional assumptions to ensure conservative results. The most reactive fissile material shape was found to be a solid sphere, so wherever possible, the analyses used a sphere centered in the shipping drum for the fissile material geometry. Triangular arrays, rather than square arrays, were found to be the most reactive, and were therefore used in the analysis. Except for plutonium-238 oxide, which is not limited by criticality, Fissile Class limits were established by determining a k_{eff} plus a two sigma value less than or equal to 0.95 with bias applied. The plutonium-238 oxide limits were established by seeking a k_{eff} of unity, and then reducing the fissile mass by 25%.

A summary of simplifications and assumptions used in the analysis which produced conservative results is as follows:

- Except for the higher masses of uranium-235 in the 5790 container, the fissile material was assumed spherical, which is the most reactive geometry. The higher mass values of uranium-235 could not be made spherical and still fit in the 5790 inner container. In those cases, the uranium-235 was assumed cylindrical.
- Pure plutonium-239 was taken as the plutonium metal at the full density of 19.6 g/cm³, and pure uranium-235 and uranium-233 metal were used at their full density of 18.9 g/cm³.
- The damaged condition, which limited Class II and Class III shipments, was assumed to have the outer container and Firedike stripped away, with the inner cylinders in contact. Although this hypothetical configuration is by far the most reactive, its occurrence, based on container test results, is almost incredible.
- Nonfissile materials were assumed not to be present inside the inner container. Therefore, the effect of the steel capsules, which hold the fissile material inside the inner container, and the effect of packaging materials within the inner container, were not considered. Also, the outer steel drum was neglected in the analysis. Since all of these materials would act as poisons, neglecting them would produce conservative results.
- All k_{eff} values were less than or equal to 0.95 with bias and two sigma applied.
- Triangular arrays for undamaged package analysis, rather than square arrays, were used since triangular arrays produced the highest criticality. As a result of the very conservative assumed damaged condition, square arrays were used for the damaged package analysis.

Details concerning the aforementioned findings are given in later sections.

6.2. Package Fuel Loading.

Fissile Class I limits, along with the conditions producing these limits (damaged or undamaged), are given in Table 6-2. The Fissile Class I analysis of plutonium-238 oxide allowed for kilogram quantities of this isotope to be shipped. Heat dissipation requirements, however, prohibit shipping more than

TABLE 6-2

FISSILE CLASS I LIMITS

<u>Package</u>	<u>Isotope</u>	<u>Mass (kg)</u>	<u>Condition</u>	<u>No. of Packages</u>
5790	^{238}Pu Oxide	1.87	Undam.	Infinite
	^{239}Pu Metal	0.63	Undam.	Infinite
	^{235}U Metal	1.0	Undam.	Infinite
	^{233}U Metal	0.59	Undam.	Infinite
5791	^{238}Pu Oxide	3.70	Dam.	280
	^{239}Pu Metal	1.69	Undam.	Infinite
	^{235}U Metal	3.15	Undam.	Infinite
	^{233}U Metal	1.70	Undam.	Infinite

a few hundred grams of $^{238}\text{PuO}_2$ in the drum-type containers considered in this analysis. $^{238}\text{PuO}_2$ has a very high heat density at about 0.5 W/g. As a result, shipments of plutonium-238 oxide are limited by heat dissipation and not criticality; consequently, no additional criticality analyses were required for this material.

Table 6-3 shows Fissile Class II limits, that is, packages with the maximum transport index of 10. In all cases, the damaged package arrays produced the highest multiplication and were therefore limiting. Curves of the transport index versus the mass per package for each isotope and both containers are given in Figures 6-1 through 6-3, and the values are tabulated in Table 6-4.

Fissile Class III limits are shown in Table 6-5 and are plotted in Figures 6-4 through 6-6. Again, the damaged configuration limits Fissile Class III shipments. Please note that Fissile Class III quantities are not authorized for shipment in these packages.

Tables 6-6 and 6-7 compare results for undamaged and damaged packages for the 5790 and 5791 containers, respectively. The tables indicate the limiting condition based on the array that produced the highest k_{eff} value.

TABLE 6-3

FISSILE CLASS II LIMITS

<u>Package</u>	<u>Isotope</u>	<u>Mass (kg)</u>	<u>Condition</u>	<u>No. of Packages</u>
5790	^{239}Pu Metal	4.1	Dam.	10
	^{235}U Metal	11.1	Dam.	10
	^{233}U Metal	4.9	Dam.	10
5791	^{239}Pu Metal	5.4	Dam.	10
	^{235}U Metal	16.8	Dam.	10
	^{233}U Metal	6.7	Dam.	10

6.3. Calculational Model

6.3.1. Model Geometry for Inner and Outer Containers

The geometry used for the undamaged 5790 and 5791 containers is shown in Figure 6-7. The fissile material was centered in the inner container and was surrounded by void in the inner container.

The cylindrical inner container for the 5790 drum was made of stainless steel, with an inside diameter of 10.5 cm and an inside height of 28.2 cm. The inner container had a sidewall thickness of 1.12 cm, a top plate thickness of 2.24 cm, and a bottom plate thickness of 1.27 cm. Firedike insulation surrounded the inner container with a diameter of 35.1 cm and a height of 53.6 cm. The outer metal drum was neglected in the analysis. Reflected arrays assumed a full density water reflector 30 cm thick.

The stainless steel inner container for the 5791 configuration had an inner diameter of 16.5 cm and an inside height of 40.6 cm. The sidewall thickness was 0.64 cm, with a top plate thickness of 1.12 cm and a bottom plate thickness of 0.64 cm. The Firedike surrounding the inner container had an inner diameter of 57.2 cm and a height of 88.4 cm. Again, the outer drum was neglected, and array reflectors were 30 cm of water.

The geometry for the damaged package studies included the inner container only, since it was assumed that accident conditions stripped away all of the Firedike insulation material.

Figure 6-1 - Transport index as a function of package mass for ^{235}U metal.

Figure 6-2 - Transport index as a function of package mass for ^{239}Pu metal.

Figure 6-3 - Transport index as a function of package mass for ^{233}U metal.

TABLE 6-4

TRANSPORT INDEX VERSUS MASS

Transport Index	Array	Condition	Mass (kg)/Container					
			5790			5791		
			^{235}U	^{239}Pu	^{233}U	^{235}U	^{239}Pu	^{233}U
0.5	10x5x4	Dam.	4.4	2.0	2.2	8.0	3.1	3.6
2.0	5x5x2	Dam.	6.6	2.8	3.2	11.0	4.1	4.8
5.0	5x2x2	Dam.	9.4	3.6	4.5	14.8	4.9	6.0
10.0	5x2x1	Dam.	11.1	4.1	4.9	16.8	5.4	6.7

TABLE 6-5

FISSILE CLASS III LIMITS

Package	Isotope	Mass (kg)	Condition	No. of Packages
5790	^{239}Pu Metal	6.0	Dam.	1
		4.5	Dam.	4
		4.1	Dam.	8
	^{235}U Metal	24.5	Dam.	1
		13.6	Dam.	4
		11.4	Dam.	8
	^{233}U Metal	8.0	Dam.	1
		5.8	Dam.	4
		4.8	Dam.	8
5791	^{239}Pu Metal	6.7	Dam.	1
		5.6	Dam.	4
		5.5	Dam.	8
	^{235}U Metal	25.8	Dam.	1
		19.2	Dam.	4
		17.0	Dam.	8
	^{233}U Metal	9.2	Dam.	1
		7.3	Dam.	4
		6.8	Dam.	8

Figure 6-4 - Number of packages as a function of package mass for ^{235}U metal, Fissile Class III limits.

Figure 6-5 - Number of packages as a function of package mass for ^{239}Pu metal, Fissile Class III limits.

Figure 6-6 - Number of packages as a function of package mass for ^{233}U metal, Fissile Class III limits.

TABLE 6-6

COMPARISON OF UNDAMAGED VERSUS DAMAGED CONTAINERS
FOR ESTABLISHING FISSILE CLASS LIMITS

5790 Container							
Fissile Class	Isotope	Mass (kg)	Array		k _{eff} Limit		Limit Cond.
			Undam.	Dam.	Undam.	Dam.	
I	239Pu	0.63	2x2x2 Specular Reflection	8x7x5 Reflected Opt Inter- dispersed Water	0.92138 (0.00482)	0.56137 (0.00466)	UD
	235U	1.0			0.92408 (0.00339)	0.50755 (0.00368)	UD
	233U	0.59			0.91761 (0.00431)	0.54895 (0.00357)	UD
II	239Pu	4.1	5x5x1 Reflected Opt Inter- dispersed Water	5x2x1 Reflected Opt Inter- dispersed Water	0.82105 (0.00566)	0.91623 (0.00589)	D
	235U	11.1			0.80814 (0.00414)	0.92722 (0.00438)	D
	233U	4.9			0.79814 (0.00464)	0.91477 (0.00537)	D
III	239Pu	6.0	2x1x1 Reflected	1 Reflected Opt Inter- dispersed Water	0.90353 (0.00609)	0.91933 (0.00512)	D
	235U	24.5			0.88678 (0.00492)	0.92648 (0.00483)	D
	233U	8.0			0.89137 (0.00536)	0.91150 (0.00548)	D

TABLE 6-7

COMPARISON OF UNDAMAGED VERSUS DAMAGED CONTAINERS
FOR ESTABLISHING FISSILE CLASS LIMITS

5791 Container							
Fissile Class	Isotope	Mass (kg)	Array		k _{eff} Limit		Limit Cond.
			Undam.	Dam.	Undam.	Dam.	
I	239Pu	1.69	2x2x2 Specular Reflection	8x7x5 Reflected Opt Inter- dispersed Water	0.92119 (0.00523)	0.72997 (0.00577)	UD
	235U	3.15			0.92897 (0.00422)	0.92692 (0.00434)	UD
	233U	1.70			0.91869 (0.00443)	0.71198 (0.00506)	UD
II	239Pu	5.4	5x5x1 Reflected Opt Inter- dispersed Water	5x2x1 Reflected Opt Inter- dispersed Water	0.86486 (0.00479)	0.92384 (0.00596)	D
	235U	16.8			0.79917 (0.00409)	0.92759 (0.00470)	D
	233U	6.7			0.84196 (0.00425)	0.91084 (0.00540)	D
III	239Pu	6.7	2x1x1 Reflected	1 Reflected Opt Inter- dispersed Water	0.90895 (0.00571)	0.91921 (0.00587)	D
	235U	25.8			0.87992 (0.00474)	0.91122 (0.00455)	D
	233U	9.2			0.90055 (0.00547)	0.90285 (0.00513)	D

Figure 6-7 - Model geometry for the 5790 and 5791 containers. (All dimensions are in cm.)

6.3.2. Fissile Shape and Array Spacing

Prior to evaluating the Fissile Class requirements, several preliminary studies were made. First, to determine the most reactive geometry for the fissile material within the inner container, a study was conducted for each isotope and container. Tables 6-8 and 6-9 show the results of that study. An 8x7x5 array of undamaged containers was used for the analysis. No inter-spersed water was assumed, but the array was fully reflected with 30 cm of water. The first configuration assumed a solid sphere of fissionable material in the center of the inner container, with the containers in a square array. The second configuration maintained the sphere in the center of the container, but changed the array to a triangular one. Instead of a square between container centers, a rhombus was assumed with each container placed between, and contiguous with, two containers in an adjacent row. An equivalent square array with a reduced center-to-center distance, such that the area of the equivalent square was equal to that of the rhombus, was then used for the KENO analysis. For example, the 5790 container has a diameter of 35.1 cm, making the center-to-center distance in a square array 35.1 cm. In a triangular configuration, the rhombus area is $(35.1)^2 \sin(60)$, which is the area of the equivalent square. The center-to-center distance for an equivalent square array is then the square root of the rhombus area, or 32.66 cm. The outside diameter of the drum was then reduced to 32.66 cm, and the analysis was conducted for a square array at this center-to-center distance. As might be surmised, the triangular mesh produced a higher k_{eff} as a result of the reduced spacing between units.

Additional geometries looked at are given in Table 6-4, and include cylinders of various height-to-diameter ratios, a couple of hollow spheres, and three stacked cylinders that filled the height of the inner container. The density of the stacked cylinders was reduced to maintain a constant mass of fissile material. The stacked cylinders were run to simulate a distribution of fissile material within the inner container. As indicated, by Tables 6-8 and 6-9, the highest k_{eff} was obtained for the solid sphere in a triangular array and was the geometry chosen for undamaged package analyses.

As a result of the extreme conservatism assumed for damage package arrays, that is, contiguous inner cylinders, square arrays were used for damaged package analyses. The results for the stacked cylinders indicate that a reduction in density of the fissile material within the inner container reduces criticality. All computer runs assumed the fissile material was spherical in shape, except for mass values of uranium-235 greater than 12 kg in the 5790 container. At mass values greater than 12 kg, uranium-235 spheres would not fit into the inner 5790 container; consequently, the fissile material was then assumed to be cylindrical. The spheres or cylinders were centered in the inner container.

6.3.3. Effect of Volume Reduction

To establish a geometry for damaged packages, a study was conducted to determine the effect of losing the insulation material surrounding the inner container in an accident. In addition to the full volume, the volume of the insulation was reduced by 50%, and then completely withdrawn so that the inner containers were in contact. The array was set at 8x7x5, with 30 cm. of water surrounding the array as a reflector. The results are given for each isotope in both containers in Table 6-10. It is clear that the complete removal of all insulation material between containers, with the inner containers in contact, results in the highest effective Multiplication Factor. The

TABLE 6-8

EFFECTS OF FISSILE SHAPE, 8x7x5 ARRAY,
FULLY REFLECTED, NO INTERSPERSED WATER

Isotope	Mass (kg)	5790 Container		
		Mesh	Fissile Shape	k_{eff}
^{235}U	6.0	Square Triangular	Sphere	0.70632(0.00441)
			Sphere	0.72138(0.00376)
			Cylinder (D=H)	0.72060(0.00420)
			Cylinder (D=H/2)	0.69354(0.00430)
			Cylinder (D=0.65H)	0.68810(0.00433)
			Hollow Sphere (i.d.=4 cm.)	0.68242(0.00419)
			Hollow Sphere (i.d.=6 cm.)	0.64654(0.00460)
			3 Stacked Cylinders	0.63441(0.00409)
^{239}Pu	4.8	Square Triangular	Sphere	0.95800(0.00598)
			Sphere	0.96436(0.00612)
			Cylinder (D=H)	0.96318(0.00537)
			Cylinder (D=H/2)	0.95307(0.00535)
			Cylinder (D=1.08H)	0.96153(0.00573)
			Hollow Sphere (i.d.=1.0 cm.)	0.93700(0.00527)
			Hollow Sphere (i.d.=2.0 cm.)	0.94885(0.00623)
			3 Stacked Cylinders	0.87857(0.00613)
^{233}U	4.0	Square Triangular	Sphere	0.83344(0.00549)
			Sphere	0.84159(0.00560)
			Cylinder (D=H)	0.83559(0.00530)
			Cylinder (D=H/2)	0.80678(0.00539)
			Cylinder (D=0.75H)	0.82451(0.00502)
			Hollow Sphere (i.d.=4.0 cm.)	*0.78349(0.00470)
			Hollow Sphere (i.d.=6.0 cm.)	0.72762(0.00438)
			3 Stacked Cylinders	0.74700(0.00502)

TABLE 6-9

EFFECTS OF FISSILE SHAPE, 8x7x5 ARRAY,
FULLY REFLECTED, NO INTERSPERSED WATER

Isotope	Mass (kg)	5791 Container		
		Mesh	Fissile Shape	k _{eff}
²³⁵ U	10.0	Square Triangular	Sphere	0.74745(0.00410)
			Sphere	0.76139(0.00447)
			Cylinder (D=H)	0.75078(0.00390)
			Cylinder (D=H/2)	0.70691(0.00384)
			Cylinder (D=0.91H)	0.74393(0.00441)
			Hollow Sphere (i.d.=6 cm.)	0.67625(0.00360)
			Hollow Sphere (i.d.=6 cm.)	0.63055(0.00481)
			3 Stacked Cylinders	0.64487(0.00413)
²³⁹ Pu	6.4	Square Triangular	Sphere	0.95662(0.00525)
			Sphere	0.96278(0.00565)
			Cylinder (D=H)	0.93620(0.00583)
			Cylinder (D=H/2)	0.91417(0.00501)
			Cylinder (D=1.08H)	0.94817(0.00590)
			Hollow Sphere (i.d.=1.0 cm.)	0.95387(.00597)
			Hollow Sphere (i.d.=2.0 cm.)	0.94059(0.00606)
			3 Stacked Cylinders	0.85832(0.00517)
²³³ U	4.0	Square Triangular	Sphere	0.91179(0.00470)
			Sphere	0.91286(0.00574)
			Cylinder (D=H)	0.89777(0.00510)
			Cylinder (D=H/2)	0.85864(0.00497)
			Cylinder (D=0.75H)	0.89333(0.00510)
			Hollow Sphere (i.d.=8.0 cm.)	0.72208(0.00477)
			Hollow Sphere (i.d.=10.0 cm.)	0.66172(0.00489)
			3 Stacked Cylinders	0.78944(0.00557)

TABLE 6-10

EFFECTS OF VOLUME REDUCTION FOR DAMAGED CONFIGURATION
8X7X5 ARRAY, FULLY REFLECTED

<u>Container</u>	<u>Isotope</u>	<u>Mass (kg)</u>	<u>Insulation Volume Reduction (%)</u>	<u>k_{eff}</u>
5790	²³⁵ U	6.0	0	0.72138(0.00376)
			50	0.78048(0.00417)
			100	1.10180(0.00452)
	²³⁹ Pu	4.8	0	0.96436(0.00612)
			50	1.00289(0.00643)
			100	1.32964(0.00710)
	²³³ U	4.0	0	0.84159(0.00560)
			50	0.89712(0.00520)
			100	1.21142(0.00553)
5791	²³⁵ U	10.0	0	0.76139(0.00447)
			50	0.80052(0.00452)
			100	1.05392(0.00473)
	²³⁹ Pu	6.4	0	0.96278(0.00565)
			50	0.97919(0.00646)
			100	1.21365(0.00594)
	²³³ U	7.0	0	0.96278(0.00565)
			50	0.97919(0.00646)
			100	1.21365(0.00594)

Multiplication Factor increases as the distance between fissile units decrease. The damaged array configuration assumed was then an array with the inner cylinders in contact. As a result of this extreme condition, a square array was utilized instead of a triangular array.

6.3.4. Effect of Interspersed Water

The effect of interspersed water for damaged packages is shown in Table 6-11. Water of variable density was placed between the contacting inner containers of an 8x7x5 reflected array, to evaluate the effect of interspersed water on the Effective Multiplication Factor. Water, at 20% of full density, produced the highest k_{eff} for the uranium isotopes, whereas water at 10% of full density was most reactive for plutonium. These results were consistent for both containers.

Interspersed water was also introduced between a 2x2x2 specularly reflected undamaged package array, which is required for Fissile Class I analysis. The results are shown in Table 6-12. It is clear that interspersed water between undamaged packages markedly reduces criticality. The geometry of undamaged packages is such that additions of water between the fissile units acts as an absorber rather than an effective moderator.

6.3.5. Package Regional Densities

The materials used in the KENO calculations are shown in Table 6-13. The fissile isotopes were chosen after self-shielding corrections were made for resonance absorption. This is accomplished with the Hansen Roach cross section set by calculating a potential scattering cross-section per fissile nuclide and selecting the cross section set that is closest to this value. The cross sections are given as a function of potential scattering cross section per fissile nucleus. The plutonium-238 oxide composition was well defined, as shown in Table 6-13. The metal compositions of plutonium and uranium were taken at their full density for the sake of conservatism. Unfortunately, the composition of Firedike is no longer available since it is no longer manufactured. The Firedike insulation was assumed to have the same composition as that of the Superex insulation of reference [3], at the known Firedike density of 0.385 g/cm³. Since this density is relatively small, the effect of a different composition of insulating material should not be significant.

6.3.6. Criticality Calculation

The simplified geometry options that are available with KENO proved adequate for this analysis. Undamaged array studies were conducted with a slightly reduced inter-unit spacing dictated by triangular arrays. The outer drum diameter was reduced to this spacing to prevent intersecting geometries, which would necessitate a much more complicated calculation. As stated earlier, the use of triangular arrays in this fashion produced a higher criticality than a square array with full dimensions and is, therefore, conservative. Since interspersed water between undamaged packages produced a reduction in criticality, no interspersed water was used for undamaged package analyses. Damaged configurations assumed a square array of contiguous inner cylinders, with 10% of full density water between the cylinders for plutonium metal and 20% for uranium metal.

Fissile Class I undamaged calculations were performed with a 2x2x2 array of packages specularly reflected to simulate an infinite number of packages.

TABLE 6-11

EFFECTS OF INTERSPERSED WATER
8X7X5 DAMAGED PACKAGE ARRAY

Container:	5790			5791		
Isotope:	<u>^{239}Pu</u>	<u>^{235}U</u>	<u>^{233}U</u>	<u>^{239}Pu</u>	<u>^{235}U</u>	<u>^{233}U</u>
Mass (kg):	<u>4.8</u>	<u>6.0</u>	<u>4.0</u>	<u>6.4</u>	<u>10.0</u>	<u>7.0</u>
Full Density Water (%)	k_{eff}			k_{eff}		
100	1.19584 (0.00708)	0.98241 (0.00431)	1.07943 (0.00497)	1.12483 (0.00632)	0.95487 (0.00452)	1.07859 (0.00511)
80	1.21268 (0.00658)	1.01699 (0.00477)	1.11085 (0.00613)	1.14853 (0.00682)	0.98457 (0.00459)	1.11009 (0.00579)
60	1.26378 (0.00658)	1.04762 (0.00447)	1.14919 (0.00613)	1.16017 (0.00682)	1.00983 (0.00465)	1.13498 (0.00517)
40	1.30046 (0.00652)	1.09801 (0.00479)	1.19642 (0.00592)	1.18472 (0.00614)	1.03187 (0.00483)	1.17804 (0.00625)
20	1.32864 (0.00653)	1.11907 (0.00504)	1.23128 (0.00476)	1.21582 (0.00686)	1.07857 (0.00538)	1.20952 (0.00591)
10	1.33260 (0.00756)			1.21979 (0.00723)		
0	1.32964 (0.00710)	1.10180 (0.00452)	1.21142 (0.00553)	1.21365 (0.00594)	1.05392 (0.00473)	1.19257 (0.00644)

TABLE 6-12

EFFECTS OF INTERSPERSED WATER
2X2X2 SPECULARLY REFLECTED UNDAMAGED ARRAY

Container:	5790			5791		
Isotope:	^{239}Pu	^{235}U	^{233}U	^{239}Pu	^{235}U	^{233}U
Mass (kg):	1.0	0.63	0.59	1.69	3.1	1.75
Full Density Water (%)	k_{eff}			k_{eff}		
100	0.45424 (0.00446)	0.33424 (0.00303)	0.40541 (0.00341)	0.59694 (0.00447)	0.46363 (0.00324)	0.54126 (0.00385)
80	0.45826 (0.00398)	0.33469 (0.00291)	0.40757 (0.00357)	0.60344 (0.00522)	0.45973 (0.00363)	0.55341 (0.00466)
60		0.34701 (0.00303)	0.41267 (0.00333)		0.46160 (0.00363)	0.55327 (0.00413)
50	0.47122 (0.00433)			0.61922 (0.00473)		
40		0.36501 (0.00294)	0.42865 (0.00381)		0.48037 (0.00334)	0.55539 (0.00385)
20	0.50738 (0.00411)	0.40652 (0.00301)	0.45942 (0.00453)	0.62822 (0.00491)	0.50435 (0.00360)	0.58501 (0.00413)
10	0.55621 (0.00495)			0.59694 (0.00447)		
0	0.92230 (0.00530)	0.92398 (0.00357)	0.91761 (0.00431)	0.92255 (0.00538)	0.91623 (0.00400)	0.91869 (0.00443)

TABLE 6-13.

MATERIALS USED IN ANALYSIS

<u>Mixture</u>	<u>Density</u> <u>(g/cm³)</u>	<u>Isotope</u> or <u>Element</u>	<u>KENO</u> <u>No.</u>	<u>Number</u> <u>Density</u> <u>(barn-cm)⁻¹</u>
²³⁸ Pu Oxide	9.96	²³⁸ Pu	94801	1.7754-2 ^a
		²³⁹ Pu	94904	3.5509-3
		²⁴⁰ Pu	94005	6.6579-4
		²⁴¹ Pu	94100	2.2193-4
		Oxygen	8100	4.4386-2
²³⁹ Pu Metal	19.6	²³⁹ Pu	94901	4.9383-2
²³⁵ U Metal	18.7	²³⁵ U	92501	4.8431-2
²³³ U Metal	18.9	²³³ U	92301	4.8848-2
Stainless Steel	7.9	Premixed	200	1.0
Superex ^b	0.385	Silicon	14100	2.9621-3
		Oxygen	8100	7.0726-3
		Aluminum	13100	3.4056-4
		Calcium	20100	2.3760-4
		Iron	26100	1.5840-4
		Titanium	22100	1.1880-5
		Magnesium	12100	3.1680-5
		Sodium	11100	1.0692-4
		Potassium	19100	1.08004-4
Interspersed Water	0.9982	Premixed	501	1.0
Reflector Water	0.9982	Premixed	502	1.0

^a1.7754-2 is to be read as 1.7754×10^{-2} .

^bSuperex composition by weight percent: SiO₂ 74.8%; Al₂O₃ 4.3%; CaO 6%; Fe₂O₃, 2%; TiO₂, 0.3%; MgO, 0.8%; Na₂O₂+K₂O, 2.7%.

Since KENO-IV will not calculate a single package specularly reflected, a 2x2x2 array was chosen.

Damaged package Fissile Class I calculations were conducted with an 8x7x5 array of 280 packages, which provided a near cubical array. Since the 280 packages are more than the 250 required by 10 CFR 71, the results are conservative. This array for damaged Fissile class I analysis is used by the NRC. The radius of the fissile sphere, or cylinder when applicable for higher uranium masses in the 5790 container, was varied until the k_{eff} value was less than or equal to the value required for a k_{eff} of 0.95 with bias and two sigma applied.

Fissile Class II analyses were made by selecting the appropriate damaged array to provide a given transport index, as shown in Table 6-4. The radius of the fissile material was varied until the desired k_{eff} was established. The undamaged array for the same transport index and mass was then calculated to affirm the limiting condition, as given in Table 6-6 and 6-7. In all cases the damaged condition was limiting, and the Fissile Class II limit was established for damaged packages with a transport index of 10.

Fissile Class III limits were established by selecting package numbers between the Fissile Class II limit at a transport index of 10, which was 10 packages, and a single package. Again, the fissile radius was varied until the desired k_{eff} was established.

6.3.7. Specifications on Effective Multiplication Factor

The required Multiplication Factor should be less than or equal to 0.95 after the benchmark bias is applied and after two standard deviations are added. From preliminary analyses, the standard deviation for almost all runs near critical was less than 0.006. Using this value as the standard deviation, a k_{eff} value was established for each isotope based on the bias and standard deviation obtained from the benchmark studies. This value of k_{eff} was used as the search value. The calculations are shown in Table 6-14. The k_{eff} values used from the KENO results are shown in Table 6-15, along with the search value, that is, that value of k_{eff} with bias applied plus two standard deviations that will result in a Multiplication Factor less than 0.95. In almost all cases, the k_{eff} value used was less than the search value, and the standard deviation was always less than 0.006, ensuring that the k_{eff} value used was less than 0.95 with a 95% confidence level. In those few cases where k_{eff} was greater than the search value, the corrected k_{eff} calculated to less than or equal to 0.95.

6.4. Validation of Calculational Method

The KENO-IV code used in the analysis was tested against measured criticality data to ascertain the accuracy of the calculations. The plutonium-239 calculations were tested against the measured criticality of the Jezebel plutonium metal spherical reactor of reference [4], the uranium-235 calculations were tested against the uranium metal sphere data of reference [5], and the uranium-233 calculations were tested against the uranium-233 solution results of reference [6]. These references were chosen on the basis of applicability, availability, and on geometries that can be easily simulated with KENO.

TABLE 6-14

APPLICATION OF BIAS TO MULTIPLICATION FACTOR

I. ^{235}U

Calculated k_{eff} : $k_{\text{eff}} + 0.006$

Benchmark Calculation: 0.99483 ± 0.00454

Bias from Benchmark: 0.00517

Bias Application: $k_{\text{eff}}(1.00517) + 2(k_{\text{eff}} \times 0.00454 + 0.006) \leq 0.95$

$$k_{\text{eff}} (1.00517 + 0.00908) + 2 \times 0.006 \leq 0.95$$

$$k_{\text{eff}} \leq 0.92482$$

II. ^{239}Pu

Benchmark Calculation: 0.99579 ± 0.00574

Bias from Benchmark: 0.00421

Bias Application: $k_{\text{eff}} (1.000421 + 0.01148) + 2 \times 0.006 \leq 0.95$

$$k_{\text{eff}} \leq 0.92351$$

III. ^{233}U

Benchmark Calculation: 1.00071 ± 0.00653

Bias from Benchmark: 0.00071

Bias Application: $k_{\text{eff}} (1.00071 + 0.01306) + 2 \times 0.006 \leq 0.95$

$$k_{\text{eff}} \leq 0.92526$$

TABLE 6-15

APPLICATION OF THE SEARCH VALUE TO THE MULTIPLICATION FACTOR

	5790			5791		
	^{235}U	^{239}Pu	^{233}U	^{235}U	^{239}Pu	^{233}U
Search Value	0.92482 (0.006)	0.92351 (0.006)	0.92526 (0.006)	0.92482 (0.006)	0.92351 (0.006)	0.92526 (0.006)
Fissile Class I	0.92408 (0.00339)	0.92138 (0.00482)	0.91761 (0.00431)	0.92897 (0.00422)	0.92119 (0.00523)	0.91869 (0.00443)
Fissile Class II						
TI ^a = 0.5	0.92651 (0.00483)	0.92420 (0.00584)	0.90671 (0.00547)	0.92837 (0.00541)	0.92457 (0.00580)	0.90671 (0.00547)
TI = 2.0	0.92462 (0.00494)	0.92625 (0.00572)	0.90804 (0.00525)	0.92283 (0.00507)	0.92232 (0.00573)	0.90440 (0.00534)
TI = 5.0	0.92267 (0.00473)	0.92105 (0.00524)	0.91613 (0.00463)	0.92501 (0.00487)	0.92539 (0.00545)	0.90278 (0.00538)
TI = 10.0	0.92722 (0.00483)	0.91623 (0.00589)	0.91477 (0.00537)	0.92759 (0.00470)	0.92384 (0.00596)	0.91084 (0.00540)
Fissile Class III						
1x1x1	0.92648 (0.00483)	0.91933 (0.00512)	0.91150 (0.00548)	0.91122 (0.00455)	0.91921 (0.00587)	0.90285 (0.00513)
2x2x1	0.92799 (0.00469)	0.91815 (0.00532)	0.91632 (0.00471)	0.92309 (0.00454)	0.91891 (0.00499)	0.90993 (0.00553)
4x2x1	0.92396 (0.00456)	0.92325 (0.00554)	0.91182 (0.00511)	0.92484 (0.00467)	0.92032 (0.00532)	0.91119 (0.00510)

^aTransport index.

The data associated with the criticality of plutonium-238 is very sparse and, as mentioned earlier, plutonium-238 is not limited by criticality, but by heat dissipation requirements. Nevertheless, an attempt was made to match the calculations against data given in reference [7].

The materials used in all of the verification analyses are given in Table 6-16. The results are shown in Table 6-17.

The geometry for the plutonium-239 calculation consisted of an unreflected sphere of plutonium-239 metal of radius 6.28 cm.

The geometry for the uranium-235 verification consisted of a sphere of enriched uranium metal of radius 5.948 cm, held in place with a stainless steel tie bolt 0.36 cm. in radius through the center of the sphere. The uranium metal sphere was centered in a cylindrical tank filled to a height of 30 cm with uranyl nitrate solution. The tank had an inner radius of 25.6 cm, with a radial thickness of 0.48 cm and top and bottom thicknesses of 0.64 cm. The cylindrical tank was placed in a glovebox, with the following inner dimensions as measured from the center of the sphere: $x = 82$ cm; $-x = 242$ cm; $y = 37$ cm; $-y = 45$ cm; $z = 82.5$ cm; $-z = 166.5$ cm. Three sides and the bottom of the glovebox were stainless steel, 1.3 cm in thickness. The fourth side was Plexiglas and the top was opened. The entire system was reflected with 100 cm of concrete, the inner walls of which were at the following distances from the center of the sphere: $x = 326$ cm; $-x = 680$ cm; $y = 55.3$ cm; $-y = 1083.3$ cm; $z = 844.5$ cm; and $-z = 198.3$ cm.

The uranium metal sphere was enriched to 93.16 wt % uranium-235, with the remainder taken as uranium-238. The density of the uranium was 18.012 g/cm³. The uranyl nitrate solution contained 13.12 g of uranium per liter, with the uranium containing 93.18 wt % uranium-235, and the remainder uranium-238. The solution density was 1.0163 g/cm³. The nitric acid normality was 0.0288.

The geometry for the uranium-233 validation consisted of a 3x3 array of polyethylene bottles containing uranium-233 solution, reflected by Lucite. The solution was contained in the bottles of inner radius 5.78 cm at a solution height of 29.57 cm. The height of the bottle interior was 44.58 cm, and the bottle thickness was 0.254 cm. The array was square with the bottles separated by a surface-to-surface distance of 10.11 cm. The side reflectors, 15.24 cm thick, were placed against the outer bottles of the array; and the top and bottom reflectors, 11.43 cm thick, were placed directly on the top and bottom of the bottles.

The solution was uranyl nitrate hexahydrate, $\text{UO}_2(\text{NO}_3) \cdot 6\text{H}_2\text{O}$, at a density of 330 g of uranium-233 per liter. This uranium solution was combined with nitric acid and water to make a 0.53 M nitric acid concentration. The uranium isotopic composition was: ^{233}U , 98.2%; ^{234}U , 0.8%; and ^{238}U , 1%.

The plutonium-238 calculations were tested against multiplication data taken with plutonium-238 isotopic heat sources. An unreflected 2x2 array inserted in water provided the maximum multiplication in the experiments conducted, and this was the configuration selected to test the calculations. The plutonium was in the form of $^{238}\text{PuO}_2$ inside a heat source container 5.38 cm in outside diameter, with an outer height of 13.703 cm. The thickness

TABLE 6-16

MATERIALS USED IN VALIDATION ANALYSIS

<u>Validated</u> <u>Isotope</u>	<u>Mixture</u>	<u>Isotope</u> <u>or</u> <u>Element</u>	<u>KENO</u> <u>No.</u>	<u>Atom</u> <u>Density</u> <u>-1</u> <u>(barn-cm)</u>
^{239}Pu Metal	^{239}Pu	^{239}Pu	94901	$3.8902 \cdot 2^a$
^{235}U Metal	Uranium	^{235}U	92501	$4.2999 \cdot 2$
		^{238}U	92825	$3.1172 \cdot 3$
		Premixed	200	1.0
	Uranyl	^{235}U	92500	$3.1320 \cdot 5$
	Nitrate	^{238}U	92800	$2.2706 \cdot 6$
		Oxygen	8100	$3.3539 \cdot 2$
		Nitrogen	7100	$8.4527 \cdot 5$
		Hydrogen	1102	$6.6432 \cdot 2$
	Concrete	Premixed	301	1.0
	Uranyl	^{233}U	92309	$8.3532 \cdot 4$
^{233}U	Nitrate	^{234}U	92400	$6.8051 \cdot 6$
		^{238}U	92852	$8.5063 \cdot 6$
		Oxygen	8100	$3.6767 \cdot 2$
		Nitrogen	7100	$1.9899 \cdot 3$
		Hydrogen	1101	$5.8570 \cdot 2$
	Polyethylene	Premixed	402	1.0
	Plexiglass ^b	Premixed	601	1.0
	Plutonium	^{238}Pu	94801	$1.3253 \cdot 2$
		^{239}Pu	94904	$2.6506 \cdot 3$
		^{240}Pu	94005	$4.9699 \cdot 4$
		^{241}Pu	94100	$1.6566 \cdot 4$
		Oxygen	8100	$3.2676 \cdot 2$
^{238}Pu Oxide	SS	Premixed	200	1.0
	H_2O	Premixed	502	1.0

^a $3.8902 \cdot 2$ is to be read as 3.8902×10^{-2} .

^bPlexiglass was substituted for the Lucite reflector.

TABLE 6-17

RESULTS OF VALIDATION ANALYSIS

<u>Isotope</u>	<u>System</u>	<u>KENO k_{eff}</u>	<u>Reference</u>
^{239}Pu	Jezebel, fast bare ^{239}Pu metal sphere 6.28 cm radius, critical	0.99579 (0.00574) ^a	4
^{235}U	Uranium metal sphere 5.548 cm in radius, in uranium solution, critical	0.99483 (0.00454)	5
^{233}U	^{233}U solution in polyethylene bottles in 3x3 array closely reflected with Lucite, critical	1.00071 (0.00653)	6
^{238}Pu	2x2 array of heat sources immersed in water multiplication = 1.19	Multiplication = 1.58	7

^a0.99574 is to be read as 0.99574 + or - 0.00574 (0.00574).

of the heat source was taken as 0.508 cm of stainless steel; the heat sources were separated by 0.318 cm.

The composition of the plutonium consisted of 80% ^{238}Pu , 16% ^{239}Pu , 3% ^{240}Pu , and 1% ^{241}Pu . Each heat source contained 994 g of ^{238}Pu . The density of the oxide was calculated to be 7.32 g/cm³.

As shown in Table 6-16, excellent agreement was achieved between the experimental and KENO results for plutonium-239, uranium-235 and uranium-233. Good agreement was not shown between the plutonium-238 measurements and KENO, although the KENO result is conservative in comparison to the measurement. This discrepancy could be from the small multiplication measured and the sensitivity of the experiment to small errors in spacing between the heat sources, which occurs for unreflected arrays as pointed out in reference [6]. Also, the plutonium-oxide geometry and the internals of the heat sources were not defined in the references. Since criticality is not limiting for

plutonium-238, however, and since the quantities that can be shipped are well below values associated with criticality, a more exact verification was not deemed necessary.

6.5. Sample KENO Input and Results

Representative KENO output for the benchmark and Fissile Class limit calculations are given in Appendix 6.6. This output includes the input parameters, materials and geometry, which is sufficient to specify and reproduce the calculation. The results include the multiplication values of the three to ten generations skipped and the corresponding frequency distribution. The results are given with the following hierarchy: benchmarks followed by 5790 and 5791 results; ^{235}U , followed by ^{239}Pu and ^{233}U ; Class I followed by Class II and then Class III. The title associated with each output is sufficiently descriptive to identify the calculation.

References

1. L.M. Petrie and N.F. Cross, KENO-IV. An Improved Monte Carlo Criticality Program, ORNL-4938, November 1975.
2. G.E. Hansen and W.H. Roach, Six and Sixteen Group Cross sections for Fast and Intermediate Critical Assemblies, Los Alamos Scientific Laboratory Report, LAMS-2543, December 1964.
3. R.A. Watkins et al., Safety Analysis Report for Packaging (SARP): USA/9507/BLF(ERDA-AL), Model AL-M1, MLM-2447, September 1977, 120 pp.
4. G.A. Jarvis, G.A. Linenberger, J.D. Orndorff, and H.C. Paxton, "Two Plutonium-Metal Critical Assemblies," Nuclear Science and Engineering, **8**, 525-531 (1960).
5. D.C. Hunt and R.E. Rothe, "Criticality Measurements on Uranium Metal Spheres Immersed in Uranium Solution," Nuclear Science and Engineering, **46**, 76-87 (1971).
6. R.C. Lloyd, E.D. Clayton, and J.H. Chalmers, "Criticality of Arrays of ^{233}U Solution," Nuclear Applications, **4**, 136-141 (March 1968).
7. R.A. Wolfe, "The Nuclear Criticality Safety Aspects of Plutonium-238," Nuclear Applications and Technology, **9**, 218-228 (August 1970).

6.6. Appendix

6.6.1. Selected KENO Output for Benchmarks and Fissile Class Limits

BENCHMARK FOR U-235 U-METAL SPHERE IN URANIUM SOLUTION

NUMBER OF GENERATIONS	103	START TYPE	0
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	10	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	10	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	28	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	10	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 30.0000 MINUTES			

BENCHMARK FOR U-235 U-METAL SPHERE IN URANIUM SOLUTION

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.29990E-02
1	-92825	3.11720E-03
2	200	1.00000E+00
3	-92500	3.13200E-05
3	-92800	2.27060E-06
3	8100	3.35370E-02
3	7100	8.45270E-05
3	1102	6.64320E-02
4	301	1.00000E+00
5	601	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL	7.96/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	301	ORDINARY CONCRETE	
NUCLIDE =	601	X(E) PLEXIGLAS	1.1826/CC H=.056884, C=.035552, O=.014221
NUCLIDE =	1102	HYDROGEN DE/E	HANSEN ROACH
NUCLIDE =	7100	NITROGEN	HANSEN ROACH
NUCLIDE =	8100	OXYGEN	HANSEN ROACH
NUCLIDE =	92500	U-235 YR	HANSEN ROACH
NUCLIDE =	92501	U-235-1R	SIG P = 20 HANSEN ROACH
NUCLIDE =	92800	U-238 Y	HANSEN ROACH
NUCLIDE =	92825	U-238	SIG P = 180 HANSEN ROACH JRK MOD

BENCHMARK FOR U-235 U-METAL SPHERE IN URANIUM SOLUTION
GEOMETRY DESCRIPTION

REGION

1	CYLINDER	2	RADIUS = 3.5500E-01	+Z = 5.9000E+00	-Z = -5.9000E+00			
2	SPHERE	1	RADIUS = 5.9480E+00					
3	CYLINDER	3	RADIUS = 2.5550E+01	+Z = 1.5000E+01	-Z = -1.5000E+01			
4	CYLINDER	2	RADIUS = 2.6030E+01	+Z = 1.5640E+01	-Z = -1.5640E+01			
5	CUBOID	0	+X = 8.2000E+01	-X = -2.4200E+02	+Y = 3.7000E+01	-Y = -4.5000E+01	+Z = 8.2500E+01	-Z = -1.6650E+02
6	CUBOID	2	+X = 8.3300E+01	-X = -2.4330E+02	+Y = 3.8300E+01	-Y = -4.5000E+01	+Z = 8.2500E+01	-Z = -1.6780E+02
7	CUBOID	5	+X = 8.3300E+01	-X = -2.4330E+02	+Y = 3.8300E+01	-Y = -4.6300E+01	+Z = 8.2500E+01	-Z = -1.6780E+02
8	CUBOID	0	+X = 3.2600E+02	-X = -6.8000E+02	+Y = 5.5300E+01	-Y = -1.0833E+03	+Z = 8.4450E+02	-Z = -1.9830E+02
REFLECTOR		4	+X = 1.0000E+02	-X = 1.0000E+02	+Y = 1.0000E+02	-Y = 1.0000E+02	+Z = 1.0000E+02	-Z = 1.0000E+02

WT. ID = 301

16 GROUP WEIGHTS FOR CONCRETE , HAVING 20 INTERVALS, EACH

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.99483	+ OR - 0.00454	0.99029 TO 0.99936	0.98575 TO 1.00390	0.98121 TO 1.00844	30000
4	0.99567	+ OR - 0.00450	0.99116 TO 1.00017	0.98666 TO 1.00468	0.98215 TO 1.00918	29700
5	0.99723	+ OR - 0.00427	0.99236 TO 1.00150	0.98869 TO 1.00577	0.98442 TO 1.01004	29400
6	0.99820	+ OR - 0.00420	0.99400 TO 1.00240	0.98981 TO 1.00660	0.98561 TO 1.01080	29100
7	0.99875	+ OR - 0.00421	0.99454 TO 1.00295	0.99033 TO 1.00716	0.98612 TO 1.01137	28800
8	0.99871	+ OR - 0.00425	0.99446 TO 1.00296	0.99021 TO 1.00721	0.98595 TO 1.01147	28500
9	0.99846	+ OR - 0.00429	0.99417 TO 1.00275	0.98988 TO 1.00704	0.98559 TO 1.01133	28200
10	0.99889	+ OR - 0.00432	0.99457 TO 1.00320	0.99026 TO 1.00752	0.98594 TO 1.01183	27900

BENCHMARK FOR U-235 U-METAL SPHERE IN URANIUM SOLUTION

FREQUENCY FOR GENERATIONS 4 TO 103

0.8216 TO 0.8447	*
0.8447 TO 0.8678	
0.8678 TO 0.8909	
0.8909 TO 0.9140	***
0.9140 TO 0.9371	***
0.9371 TO 0.9602	*****
0.9602 TO 0.9833	*****
0.9833 TO 1.0064	*****
1.0064 TO 1.0295	*****
1.0295 TO 1.0526	*****
1.0526 TO 1.0757	*****
1.0757 TO 1.0988	**

PU-239 BENCHMARK JEZEBEL SPHERE

NUMBER OF GENERATIONS	103	START TYPE	
	1		
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	1	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	1	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	1	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	2	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	1	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 30.0000 MINUTES			

PU-239 BENCHMARK JEZEBEL SPHERE

MIXTURE	NUCLIDE	DENSITY
1	-94901	3.89020E-02

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 94901 PU-239-1 SIG P = 20

HANSEN ROACH

PU-239 BENCHMARK JEZEBEL SPHERE

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 6.3140E+00

2 CUBOID 0 +X = 6.3140E+00 -X = -6.3140E+00 +Y = 6.3140E+00 -Y = -6.3140E+00 +Z = 6.3140E+00 -Z = -6.3140E+00

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.99579	+ OR - 0.00574	0.99004 TO 1.00153	0.98430 TO 1.00727	0.97856 TO 1.01301	30000
4	0.99574	+ OR - 0.00580	0.98994 TO 1.00154	0.98414 TO 1.00734	0.97834 TO 1.01314	29700
5	0.99613	+ OR - 0.00585	0.99028 TO 1.00197	0.98443 TO 1.00782	0.97859 TO 1.01366	29400
6	0.99624	+ OR - 0.00591	0.99033 TO 1.00214	0.98443 TO 1.00805	0.97852 TO 1.01395	29100
7	0.99590	+ OR - 0.00596	0.98995 TO 1.00186	0.98399 TO 1.00782	0.97803 TO 1.01378	28800
8	0.99648	+ OR - 0.00599	0.99049 TO 1.00247	0.98449 TO 1.00847	0.97850 TO 1.01446	28500
9	0.99622	+ OR - 0.00605	0.99017 TO 1.00228	0.98412 TO 1.00833	0.97807 TO 1.01438	28200
10	0.99656	+ OR - 0.00611	0.99045 TO 1.00266	0.98434 TO 1.00877	0.97823 TO 1.01488	27900

PU-239 BENCHMARK JEZEBEL SPHERE

FREQUENCY FOR GENERATIONS 4 TO 103

0.8688 TO 0.8919	***
0.8919 TO 0.9150	*****
0.9150 TO 0.9381	*****
0.9381 TO 0.9611	*****
0.9611 TO 0.9842	*****
0.9842 TO 1.0073	*****
1.0073 TO 1.0304	*****
1.0304 TO 1.0535	*****
1.0535 TO 1.0766	*****
1.0766 TO 1.0997	****
1.0997 TO 1.1228	**
1.1228 TO 1.1459	
1.1459 TO 1.1690	*

U-233 BENCHMARK 3X3X1 SOLUTION BOTTLES REFLECTED

NUMBER OF GENERATIONS	103	START TYPE	0
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	8	LIST FISSION AND ABS. BY REGION	NO
NUMBER OF MIXTURES	3	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	8	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	45	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	9	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	3	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	3	LIST FISSION PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	8	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

MAXIMUM TIME = 30.0000 MINUTES

U-233 BENCHMARK 3X3X1 SOLUTION BOTTLES REFLECTED

MIXTURE	NUCLIDE	DENSITY
1	-92309	8.35320E-04
1	-92400	6.80510E-06
1	-92852	8.50630E-06
1	8100	3.67670E-02
1	7100	1.98990E-03
1	1101	5.85700E-02
2	402	1.00000E+00
3	601	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	402	DE/E POLYETHYLENE	0.926/CC	H=.079433, C=.039716
NUCLIDE =	601	X(E) PLEXIGLAS	1.1826/CC	H=.056884, C=.035552, O=.014221
NUCLIDE =	1101	HYDROGEN X(E)		HANSEN ROACH
NUCLIDE =	7100	NITROGEN		HANSEN ROACH
NUCLIDE =	8100	OXYGEN		HANSEN ROACH
NUCLIDE =	92309	U-233-9	SIG P = 2000	HANSEN ROACH
NUCLIDE =	92400	U-234		MIHALCZO MOD OF H-R U-238
NUCLIDE =	92852	U-238	SIG P = 20000	HANSEN ROACH

U-233 BENCHMARK 3X3X1 SOLUTION BOTTLES REFLECTED

GEOMETRY DESCRIPTION

BOX TYPE 1

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

BOX TYPE 2

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 6.0330E+00	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 2.1273E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

BOX TYPE 3

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01				
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01				
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01				
4	CUBOID	0	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01	
5	CUBOID	3	+X = 1.1087E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -2.1273E+01	+Z = 3.3973E+01	-Z = -3.3973E+01	

BOX TYPE 4

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01				
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01				
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01				
4	CUBOID	0	+X = 1.1087E+01	-X = -6.0330E+00	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01	
5	CUBOID	3	+X = 1.1087E+01	-X = -2.1273E+01	+Y = 2.1273E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01	

BOX TYPE 5

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01				
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01				
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01				
4	CUBOID	0	+X = 1.1087E+01	-X = -6.0330E+00	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01	
5	CUBOID	3	+X = 1.1087E+01	-X = -2.1273E+01	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01	

BOX TYPE 6

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 1.1087E+01	-X = -6.0330E+00	+Y = 1.1087E+01	-Y = -6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 1.1087E+01	-X = -2.1273E+01	+Y = 1.1087E+01	-Y = -2.1273E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

BOX TYPE 7

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 6.0330E+00	-X = -1.1087E+01	+Y = 6.0330E+00	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 2.1273E+01	-X = -1.1087E+01	+Y = 2.1273E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

BOX TYPE 8

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 6.0330E+00	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 2.1273E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -1.1087E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

BOX TYPE 9

REGION

1	CYLINDER	1	RADIUS = 5.7790E+00	+Z = 7.2770E+00	-Z = -2.2289E+01			
2	CYLINDER	0	RADIUS = 5.7790E+00	+Z = 2.2289E+01	-Z = -2.2289E+01			
3	CYLINDER	2	RADIUS = 6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01			
4	CUBOID	0	+X = 6.0330E+00	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -6.0330E+00	+Z = 2.2543E+01	-Z = -2.2543E+01
5	CUBOID	3	+X = 2.1273E+01	-X = -1.1087E+01	+Y = 1.1087E+01	-Y = -2.1273E+01	+Z = 3.3973E+01	-Z = -3.3973E+01

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	1.00071	+ OR - 0.00653	0.99418 TO 1.00723	0.98765 TO 1.01376	0.98113 TO 1.02029	30000
4	1.00040	+ OR - 0.00659	0.99381 TO 1.00699	0.98723 TO 1.01357	0.98064 TO 1.02016	29700
5	1.00103	+ OR - 0.00662	0.99441 TO 1.00766	0.98779 TO 1.01428	0.98117 TO 1.02090	29400
6	1.00096	+ OR - 0.00669	0.99427 TO 1.00765	0.98758 TO 1.01434	0.98089 TO 1.02103	29100
7	0.99944	+ OR - 0.00658	0.99285 TO 1.00602	0.98627 TO 1.01260	0.97969 TO 1.01919	28800
8	1.00050	+ OR - 0.00657	0.99394 TO 1.00707	0.98737 TO 1.01363	0.98081 TO 1.02020	28500
9	1.00159	+ OR - 0.00654	0.99505 TO 1.00814	0.98850 TO 1.01468	0.98196 TO 1.02122	28200
10	1.00054	+ OR - 0.00653	0.99401 TO 1.00707	0.98748 TO 1.01360	0.98096 TO 1.02013	27900

U233 BENCHMARK 3X3X1 SOLUTION BOTTLES REFLECTED

	FREQUENCY FOR GENERATIONS	4 TO 103
0.8506 TO 0.8737	**	
0.8737 TO 0.8968	***	
0.8968 TO 0.9199	*****	
0.9199 TO 0.9430	*****	
0.9430 TO 0.9661	*****	
0.9661 TO 0.9892	*****	
0.9892 TO 1.0123	*****	
1.0123 TO 1.0353	*****	
1.0353 TO 1.0584	*****	
1.0584 TO 1.0815	*****	
1.0815 TO 1.1046	*****	
1.1046 TO 1.1277	****	
1.1277 TO 1.1508	*	

579025 FISSILE CLASS I UNDAMAGED 1.0kg 2X2X2 ARRAY SPECULAR REFLECTION

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	13	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 120.0000 MINUTES

579025 FISSILE CLASS I UNMAGED 1.0kg 2X2X2 ARRAY SPECULAR REFLECTION

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04
4	501	1.00000E+00
5	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE =	8100	OXYGEN HANSEN ROACH
NUCLIDE =	11100	SODIUM HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN (TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN (TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM HANSEN ROACH
NUCLIDE =	20100	CALCIUM GAM-2
NUCLIDE =	22100	TITANIUM GAM-2
NUCLIDE =	26100	IRON HANSEN ROACH
NUCLIDE =	92501	U-235-1R SIG P = 20 HANSEN ROACH

579025 FISSILE CLASS I UNMAGED 1.0kg 2X2X2 ARRAY SPECULAR REFLECTION

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 2.3289E+00						
2	CYLINDER	0	RADIUS = 5.2500E+00	+Z = 1.4100E+01	-Z = -1.4100E+01				
3	CYLINDER	2	RADIUS = 6.3500E+00	+Z = 1.5850E+01	-Z = -1.5850E+01				
4	CYLINDER	3	RADIUS = 1.6320E+01	+Z = 2.6800E+01	-Z = -2.6800E+01				
5	CUBOID	0	+X = 1.6320E+01	-X = -1.6320E+01	+Y = 1.6320E+01	-Y = -1.6320E+01	+Z = 2.6800E+01	-Z = -2.6800E+01	

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92408	+ OR - 0.00339	0.92069 TO 0.92747	0.91731 TO 0.93085	0.91392 TO 0.93424	30000
4	0.92419	+ OR - 0.00342	0.92078 TO 0.92761	0.91736 TO 0.93103	0.91394 TO 0.93445	29700
5	0.92402	+ OR - 0.00345	0.92057 TO 0.92747	0.91712 TO 0.93091	0.91367 TO 0.93436	29400
6	0.92352	+ OR - 0.00345	0.92007 TO 0.92697	0.91662 TO 0.93041	0.91317 TO 0.93386	29100
8	0.92363	+ OR - 0.00348	0.92014 TO 0.92711	0.91666 TO 0.93059	0.91318 TO 0.93408	28500
9	0.92396	+ OR - 0.00350	0.92046 TO 0.92746	0.91695 TO 0.93097	0.91345 TO 0.93447	28200
10	0.92390	+ OR - 0.00354	0.92036 TO 0.92744	0.91682 TO 0.93098	0.91328 TO 0.93453	27900

579025 FISSILE CLASS I UN4MAGED 1.0kg 2X2X2 ARRAY SPECULAR REFLECTION

FREQUENCY FOR GENERATIONS 4 TO 103

0.8202 TO 0.8433	*
0.8433 TO 0.8663	***
0.8663 TO 0.8894	*****
0.8894 TO 0.9125	*****
0.9125 TO 0.9356	*****
0.9356 TO 0.9587	*****
0.9587 TO 0.9818	*****
0.9818 TO 1.0049	*****

579025 FISSILE CLASS II DAMAGED 11.1 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(03)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579025 FISSILE CLASS II DAMAGED 11.1 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(03)

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 92501 U-235-1R SIG P = 20 HANSEN ROACH

579025 FISSILE CLASS II DAMAGED 11.1 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(03)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 5.1950E+00
2 CYLINDER 0 RADIUS = 5.2500E+00 +Z = 1.4100E+01 -Z = -1.4100E+01
3 CYLINDER 2 RADIUS = 6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
4 CUBOID 3 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 6.3500E+00 -Y = -6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92722	+ OR - 0.00438	0.92283 TO 0.93160	0.91845 TO 0.93599	0.91406 TO 0.94037	30000
4	0.92728	+ OR - 0.00443	0.92286 TO 0.93171	0.91843 TO 0.93614	0.91400 TO 0.94057	29700
5	0.92768	+ OR - 0.00446	0.92323 TO 0.93214	0.91877 TO 0.93659	0.91431 TO 0.94105	29400
6	0.92789	+ OR - 0.00450	0.92340 TO 0.93239	0.91890 TO 0.93689	0.91440 TO 0.94138	29100
7	0.92805	+ OR - 0.00454	0.92351 TO 0.93259	0.91897 TO 0.93713	0.91443 TO 0.94167	28800
8	0.92783	+ OR - 0.00458	0.92324 TO 0.93241	0.91866 TO 0.93699	0.91408 TO 0.94158	28500
9	0.92808	+ OR - 0.00463	0.92346 TO 0.93271	0.91883 TO 0.93733	0.91421 TO 0.94196	28200
10	0.92851	+ OR - 0.00466	0.92386 TO 0.93317	0.91920 TO 0.93782	0.91455 TO 0.94248	27900

579025 FISSILE CLASS II DAMAGED 11.1 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(03)

FREQUENCY FOR GENERATIONS 4 TO 103

0.8233 TO 0.8464	***
0.8464 TO 0.8695	*****
0.8695 TO 0.8926	*****
0.8926 TO 0.9157	*****
0.9157 TO 0.9388	*****
0.9388 TO 0.9619	*****
0.9619 TO 0.9850	*****
0.9850 TO 1.0080	*****
1.0080 TO 1.0311	***

579025 FISSILE CLASS III DAMAGED 24.5 kg CYL 1X1X1 MESH OPT H20(01)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579025 FISSILE CLASS III DAMAGED 24.5 kg CYL 1X1X1 MESH OPT H2O(01)

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE = 92501 U-235-1R SIG P = 20 HANSEN ROACH

579025 FISSILE CLASS III DAMAGED 24.5 kg CYL 1X1X1 MESH OPT H2O(01)

GEOMETRY DESCRIPTION

REGION

1 CYLINDER 1 RADIUS = 5.0200E+00 +Z = 8.1900E+00 -Z = -8.1900E+00
2 CYLINDER 0 RADIUS = 5.2500E+00 +Z = 1.4100E+01 -Z = -1.4100E+01
3 CYLINDER 2 RADIUS = 6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
4 CUBOID 3 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 6.3500E+00 -Y = -6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92648	+ OR - 0.00483	0.92164 TO 0.93131	0.91681 TO 0.93615	0.91198 TO 0.94098	30000
4	0.92560	+ OR - 0.00480	0.92080 TO 0.93040	0.91600 TO 0.93520	0.91119 TO 0.94001	29700
5	0.92626	+ OR - 0.00481	0.92145 TO 0.93106	0.91664 TO 0.93587	0.91184 TO 0.94067	29400
6	0.92490	+ OR - 0.00466	0.92024 TO 0.92955	0.91558 TO 0.93421	0.91092 TO 0.93887	29100
7	0.92421	+ OR - 0.00465	0.91955 TO 0.92886	0.91490 TO 0.93352	0.91025 TO 0.93817	28800
8	0.92465	+ OR - 0.00468	0.91996 TO 0.92933	0.91528 TO 0.93401	0.91060 TO 0.93870	28500
9	0.92529	+ OR - 0.00469	0.92061 TO 0.92998	0.91592 TO 0.93467	0.91123 TO 0.93936	28200
10	0.92459	+ OR - 0.00468	0.91991 TO 0.92928	0.91522 TO 0.93396	0.91054 TO 0.93865	27900

579025 FISSILE CLASS III DAMAGED 24.5 kg CYL 1X1X1 MESH OPT H20(01)

FREQUENCY FOR GENERATIONS 4 TO 103

0.8226 TO 0.8456	****
0.8456 TO 0.8687	*****
0.8687 TO 0.8918	*****
0.8918 TO 0.9149	*****
0.9149 TO 0.9380	*****
0.9380 TO 0.9611	*****
0.9611 TO 0.9842	*****
0.9842 TO 1.0073	*****
1.0073 TO 1.0304	*****

579049 INFINITE ARRAY FISSILE CLASS I UNDAMAGED NO WATER (R=1.98)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	11	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	3	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	11	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	11	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 120.0000 MINUTES

579049 INFINITE ARRAY FISSILE CLASS I UNMAGED NO WATER (R=1.98)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL	7.96/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	8100	OXYGEN	HANSEN ROACH
NUCLIDE =	11100	SODIUM	HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN	(TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM	HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN	(TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM	HANSEN ROACH
NUCLIDE =	20100	CALCIUM	GAM-2
NUCLIDE =	22100	TITANIUM	GAM-2
NUCLIDE =	26100	IRON	HANSEN ROACH
NUCLIDE =	94901	PU-239-1	SIG P = 20 HANSEN ROACH

579049 INFINITE ARRAY FISSILE CLASS I UNMAGED NO WATER (R=1.98)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 1.9800E+00					
2	YCYLINDER	0	RADIUS = 5.2500E+00	+Y = 1.4100E+01	-Y = -1.4100E+01			
3	YCYLINDER	2	RADIUS = 6.3500E+00	+Y = 1.6340E+01	-Y = -1.5370E+01			
4	YCYLINDER	3	RADIUS = 1.6330E+01	+Y = 2.6800E+01	-Y = -2.6800E+01			
5	CUBOID	0	+X = 1.6330E+01	-X = -1.6330E+01	+Y = 2.6800E+01	-Y = -2.6800E+01	+Z = 1.6330E+01	-Z = -1.6330E+01

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92138	+ OR - 0.00482	0.91656 TO 0.92621	0.91174 TO 0.93103	0.90691 TO 0.93585	30000
4	0.92171	+ OR - 0.00486	0.91685 TO 0.92657	0.91199 TO 0.93143	0.90713 TO 0.93630	29700
5	0.92134	+ OR - 0.00490	0.91644 TO 0.92623	0.91154 TO 0.93113	0.90665 TO 0.93603	29400
6	0.92151	+ OR - 0.00494	0.91657 TO 0.92645	0.91162 TO 0.93140	0.90668 TO 0.93634	29100
7	0.92163	+ OR - 0.00499	0.91663 TO 0.92662	0.91164 TO 0.93161	0.90664 TO 0.93661	28800
8	0.92095	+ OR - 0.00500	0.91595 TO 0.92595	0.91095 TO 0.93095	0.90595 TO 0.93595	28500
9	0.92052	+ OR - 0.00504	0.91549 TO 0.92556	0.91045 TO 0.93060	0.90542 TO 0.93563	28200
10	0.92123	+ OR - 0.00504	0.91620 TO 0.92627	0.91116 TO 0.93131	0.90612 TO 0.93635	27900 579049

INFINITE ARRAY FISSILE CLASS I UNDamaged NO WATER (R=1.98)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7713 TO 0.7944	**
0.7944 TO 0.8175	
0.8175 TO 0.8406	**
0.8406 TO 0.8636	*****
0.8636 TO 0.8867	*****
0.8867 TO 0.9098	*****
0.9098 TO 0.9329	*****
0.9329 TO 0.9560	*****
0.9560 TO 0.9791	*****
0.9791 TO 1.0022	*****
1.0022 TO 1.0253	*
1.0253 TO 1.0484	**

579049 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 WATER \$ REFL.(R=3.68)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	15	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579049 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 WATER \$ REFL.(R=3.68)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	502	1.00000E+00
4	501	1.00000E-01

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=.066742, O=.033371
94901 PU-239-1 SIG P = 20 HANSEN ROACH

579049 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 WATER \$ REFL.(R=3.68)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 3.6800E+00
2 YCYLINDER 0 RADIUS = 5.2500E+00 +Y = 1.4100E+01 -Y = -1.4100E+01
3 YCYLINDER 2 RADIUS = 6.3500E+00 +Y = 1.6340E+01 -Y = -1.5370E+01
4 CUBOID 4 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 1.6340E+01 -Y = -1.5370E+01 +Z = 6.3500E+00 -Z = -6.3500E+00
REFLECTOR 3 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91623	+ OR - 0.00589	0.91034 TO 0.92212	0.90445 TO 0.92801	0.89856 TO 0.93390	30000
4	0.91692	+ OR - 0.00591	0.91101 TO 0.92283	0.90510 TO 0.92874	0.89919 TO 0.93465	29700
5	0.91705	+ OR - 0.00597	0.91108 TO 0.92302	0.90512 TO 0.92899	0.89915 TO 0.93496	29400
6	0.91628	+ OR - 0.00598	0.91030 TO 0.92226	0.90432 TO 0.92824	0.89834 TO 0.93422	29100
7	0.91558	+ OR - 0.00600	0.90958 TO 0.92158	0.90358 TO 0.92758	0.89757 TO 0.93358	28800
8	0.91565	+ OR - 0.00606	0.90959 TO 0.92172	0.90353 TO 0.92778	0.89746 TO 0.93385	28500
9	0.91637	+ OR - 0.00609	0.91028 TO 0.92245	0.90419 TO 0.92854	0.89811 TO 0.93463	28200
10	0.91744	+ OR - 0.00606	0.91138 TO 0.92350	0.90533 TO 0.92955	0.89927 TO 0.93561	27900

579049 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 WATER \$ REFL.(R=3.68)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7430 TO 0.7661	*
0.7661 TO 0.7892	*
0.7892 TO 0.8123	*
0.8123 TO 0.8354	*****
0.8354 TO 0.8585	*****
0.8585 TO 0.8816	*****
0.8816 TO 0.9047	*****
0.9047 TO 0.9278	*****
0.9278 TO 0.9509	*****
0.9509 TO 0.9740	*****
0.9740 TO 0.9971	*****
0.9971 TO 1.0202	*****
1.0202 TO 1.0432	**
1.0432 TO 1.0663	*

579049 1X1X1 FISSILE CLASS III DAMAGED 0.1 WATER & REFL.(R=4.2)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	15	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579049 1X1X1 FISSION CLASS III DAMAGED 0.1 WATER & REFL.(R=4.2)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	502	1.00000E+00
4	501	1.00000E-01

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 94901 PU-239-1 SIG P = 20 HANSEN ROACH

579049 1X1X1 FISSION CLASS III DAMAGED 0.1 WATER & REFL.(R=4.2)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 4.2000E+00
2 YCYLINDER 0 RADIUS = 5.2500E+00 +Y = 1.4100E+01 -Y = -1.4100E+01
3 YCYLINDER 2 RADIUS = 6.3500E+00 +Y = 1.6340E+01 -Y = -1.5370E+01
4 CUBOID 4 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 1.6340E+01 -Y = -1.5370E+01 +Z = 6.3500E+00 -Z = -6.3500E+00
REFLECTOR 3 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91933	+ OR - 0.00512	0.91422 TO 0.92445	0.90910 TO 0.92957	0.90398 TO 0.93469	30000
4	0.91976	+ OR - 0.00515	0.91461 TO 0.92491	0.90946 TO 0.93006	0.90431 TO 0.93522	29700
5	0.91891	+ OR - 0.00513	0.91378 TO 0.92404	0.90865 TO 0.92918	0.90351 TO 0.93431	29400
6	0.91857	+ OR - 0.00517	0.91340 TO 0.92375	0.90822 TO 0.92892	0.90305 TO 0.93410	29100
7	0.91872	+ OR - 0.00523	0.91350 TO 0.92395	0.90827 TO 0.92918	0.90304 TO 0.93440	28800
8	0.91788	+ OR - 0.00521	0.91267 TO 0.92310	0.90746 TO 0.92831	0.90224 TO 0.93352	28500
9	0.91836	+ OR - 0.00525	0.91311 TO 0.92360	0.90786 TO 0.92885	0.90262 TO 0.93410	28200
10	0.91834	+ OR - 0.00530	0.91303 TO 0.92364	0.90773 TO 0.92894	0.90243 TO 0.93425	27900

579049 1X1X1 FISSILE CLASS III DAMAGED 0.1 WATER & REFL.(R=4.2)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7923 TO 0.8154	**
0.8154 TO 0.8385	**
0.8385 TO 0.8616	*****
0.8616 TO 0.8847	*****
0.8847 TO 0.9078	*****
0.9078 TO 0.9309	*****
0.9309 TO 0.9540	*****
0.9540 TO 0.9771	*****
0.9771 TO 1.0002	*****
1.0002 TO 1.0233	*****
1.0233 TO 1.0464	*

579023 FISSILE CLASS I UNDATED 0.595 kg 2X2X2 ARRAY SPECULAR REFLECTION

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	13	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 150.0000 MINUTES

579023 FISSILE CLASS I UNDAAGED 0.595 kg 2X2X2 ARRAY SPECULAR REFLECTION

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04
4	501	1.00000E+00
5	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL 7.96/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE =	8100	OXYGEN HANSEN ROACH
NUCLIDE =	11100	SODIUM HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN (TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN (TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM HANSEN ROACH
NUCLIDE =	20100	CALCIUM GAM-2
NUCLIDE =	22100	TITANIUM GAM-2
NUCLIDE =	26100	IRON HANSEN ROACH
NUCLIDE =	92301	U-233-1 SIG P = 20 HANSEN ROACH

579023 FISSILE CLASS I UNDAAGED 0.595 kg 2X2X2 ARRAY SPECULAR REFLECTION

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 1.9588E+00					
2	CYLINDER	0	RADIUS = 5.2500E+00	+Z = 1.4100E+01	-Z = -1.4100E+01			
3	CYLINDER	2	RADIUS = 6.3500E+00	+Z = 1.5850E+01	-Z = -1.5850E+01			
4	CYLINDER	3	RADIUS = 1.6320E+01	+Z = 2.6800E+01	-Z = -2.6800E+01			
5	CUBOID	0	+X = 1.6320E+01	-X = -1.6320E+01	+Y = 1.6320E+01	-Y = -1.6320E+01	+Z = 2.6800E+01	-Z = -2.6800E+01

579023 FISSILE CLASS 1 UNDAMAGED 0.595 kg 2X2X2 ARRAY SPECULAR REFLECTION

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91761	+ OR - 0.00431	0.91336 TO 0.92192	0.90898 TO 0.92623	0.90467 TO 0.93054	30000
4	0.91720	+ OR - 0.00434	0.91286 TO 0.92154	0.90853 TO 0.92587	0.90419 TO 0.93021	29700
5	0.91768	+ OR - 0.00435	0.91333 TO 0.92203	0.90897 TO 0.92639	0.90462 TO 0.93074	29400
6	0.91812	+ OR - 0.00438	0.91374 TO 0.92250	0.90937 TO 0.92687	0.90499 TO 0.93125	29100
7	0.91725	+ OR - 0.00433	0.91292 TO 0.92159	0.90858 TO 0.92592	0.90425 TO 0.93026	28800
8	0.91637	+ OR - 0.00429	0.91209 TO 0.92066	0.90780 TO 0.92495	0.90351 TO 0.92924	28500
9	0.91616	+ OR - 0.00433	0.91183 TO 0.92049	0.90750 TO 0.92482	0.90317 TO 0.92915	28200
10	0.91623	+ OR - 0.00438	0.91185 TO 0.92061	0.90748 TO 0.92498	0.90310 TO 0.92936	27900

579023 FISSILE CLASS 1 UNDAMAGED 0.595 kg 2X2X2 ARRAY SPECULAR REFLECTION

FREQUENCY FOR GENERATIONS	4 TO 103
0.8137 TO 0.8368	*****
0.8368 TO 0.8599	****
0.8599 TO 0.8830	*****
0.8830 TO 0.9061	*****
0.9061 TO 0.9292	*****
0.9292 TO 0.9522	*****
0.9522 TO 0.9753	*****
0.9753 TO 0.9984	*****
0.9984 TO 1.0215	***

579023 FISSILE CLASS II DAMAGED 4.9 kg 5X2X1 TRI MESH(T1=10.0) OPT H2O(10)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579023 FISSILE CLASS II DAMAGED 4.9 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(10)

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=0.066742, O=0.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=0.066742, O=0.033371
92301 U-233-1 SIG P = 20 HANSEN ROACH

579023 FISSILE CLASS II DAMAGED 4.9 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(10)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 3.9556E+00
2 CYLINDER 0 RADIUS = 5.2500E+00 +Z = 1.4100E+01 -Z = -1.4100E+01
3 CYLINDER 2 RADIUS = 6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
4 CUBOID 3 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 6.3500E+00 -Y = -6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91477	+ OR - 0.00537	0.90940 TO 0.92014	0.90403 TO 0.92551	0.89866 TO 0.93088	30000
4	0.91482	+ OR - 0.00542	0.90940 TO 0.92024	0.90397 TO 0.92567	0.89855 TO 0.93109	29700
5	0.91534	+ OR - 0.00545	0.90989 TO 0.92080	0.90443 TO 0.92625	0.89898 TO 0.93170	29400
6	0.91559	+ OR - 0.00550	0.91009 TO 0.92110	0.90458 TO 0.92660	0.89908 TO 0.93210	29100
7	0.91580	+ OR - 0.00556	0.91024 TO 0.92136	0.90468 TO 0.92692	0.89913 TO 0.93247	28800
8	0.91588	+ OR - 0.00562	0.91026 TO 0.92150	0.90465 TO 0.92711	0.89903 TO 0.93273	28500
9	0.91583	+ OR - 0.00568	0.91016 TO 0.92151	0.90448 TO 0.92719	0.89880 TO 0.93286	28200
10	0.91550	+ OR - 0.00573	0.90977 TO 0.92123	0.90405 TO 0.92696	0.89832 TO 0.93268	27900

579023 FISSILE CLASS II DAMAGED 4.9 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(10)

	FREQUENCY FOR GENERATIONS	4 TO 103
0.7647 TO 0.7878	*	
0.7878 TO 0.8108	**	
0.8108 TO 0.8339	**	
0.8339 TO 0.8570	*****	
0.8570 TO 0.8801	*****	
0.8801 TO 0.9032	*****	
0.9032 TO 0.9263	*****	
0.9263 TO 0.9494	*****	
0.9494 TO 0.9725	*****	
0.9725 TO 0.9956	*****	
0.9956 TO 1.0187	*****	
1.0187 TO 1.0418	**	
1.0418 TO 1.0649		
1.0649 TO 1.0880	*	

579023 FISSILE CLASS III DAMAGED 8.03 KG 1X1X1 TOUCHING OPT H20(10)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

MAXIMUM TIME = 120.0000 MINUTES

579023 FISSILE CLASS III DAMAGED 8.03 KG 1X1X1 TOUCHING OPT H20(10)

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.96/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.99826/CC H=0.66742, O=0.33371
NUCLIDE = 502 DE/E WATER 0.99826/CC H=0.66742, O=0.33371
NUCLIDE = 92301 U-233-1 SIG P = 20 HANSEN ROAD

579023 FISSILE CLASS III DAMAGED 8.03 KG 1X1X1 TOUCHING OPT H20(10)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 4.6636E+00
2 CYLINDER 0 RADIUS = 5.2500E+00 +Z = 1.4100E+01 -Z = -1.4100E+01
3 CYLINDER 2 RADIUS = 6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
4 CUBOID 3 +X = 6.3500E+00 -X = -6.3500E+00 +Y = 6.3500E+00 -Y = -6.3500E+00 +Z = 1.5850E+01 -Z = -1.5850E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3

579023 FISSILE CLASS III DAMAGED 8.03 KG 1X1X1 TOUCHING OPT H20(10)

LIFETIME = 1.60258E-04 + OR - 1.62161E-05
-06 + OR - 2.18310E-07

GENERATION TIME = 4.47787E

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91150	+ OR - 0.00548	0.90602 TO 0.91698	0.90054 TO 0.92245	0.89506 TO 0.92793	30000
4	0.91182	+ OR - 0.00552	0.90629 TO 0.91734	0.90077 TO 0.92287	0.89524 TO 0.92839	29700
5	0.91225	+ OR - 0.00556	0.90668 TO 0.91781	0.90112 TO 0.92337	0.89555 TO 0.92894	29400
6	0.91237	+ OR - 0.00562	0.90675 TO 0.91799	0.90113 TO 0.92361	0.89551 TO 0.92923	29100
7	0.91202	+ OR - 0.00567	0.90635 TO 0.91769	0.90068 TO 0.92335	0.89501 TO 0.92902	28800
8	0.91212	+ OR - 0.00573	0.90639 TO 0.91785	0.90067 TO 0.92357	0.89494 TO 0.92930	28500
9	0.91123	+ OR - 0.00572	0.90551 TO 0.91694	0.89979 TO 0.92266	0.89407 TO 0.92838	28200
10	0.91132	+ OR - 0.00578	0.90554 TO 0.91710	0.89977 TO 0.92288	0.89399 TO 0.92866	27900

579023 FISSILE CLASS III DAMAGED 8.03 KG 1X1X1 TOUCHING OPT H20(10)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7383 TO 0.7614	*
0.7614 TO 0.7845	**
0.7845 TO 0.8076	**
0.8076 TO 0.8307	***
0.8307 TO 0.8538	***
0.8538 TO 0.8769	*****
0.8769 TO 0.9000	*****
0.9000 TO 0.9230	*****
0.9230 TO 0.9461	*****
0.9461 TO 0.9692	*****
0.9692 TO 0.9923	*****
0.9923 TO 1.0154	****
1.0154 TO 1.0385	
1.0385 TO 1.0616	**

579125 FISSILE CLASS I UNDAAGED 3.15 kg WITH SPECULAR REFL. NO H2O

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	13	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 100.0000 MINUTES

579125 FISSILE CLASS I UNMAGED 3.15 kg WITH SPECULAR REFL. NO H2O

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04
4	501	1.00000E+00
5	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER 0.99826/CC H=0.066742, O=0.033371
NUCLIDE =	502	DE/E WATER 0.99826/CC H=0.066742, O=0.033371
NUCLIDE =	8100	OXYGEN HANSEN ROACH
NUCLIDE =	11100	SODIUM HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN (TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN (TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM HANSEN ROACH
NUCLIDE =	20100	CALCIUM GAM-2
NUCLIDE =	22100	TITANIUM GAM-2
NUCLIDE =	26100	IRON HANSEN ROACH
NUCLIDE =	92501	U-235-1R SIG P = 20 HANSEN ROACH

579125 FISSILE CLASS I UNMAGED 3.15 kg WITH SPECULAR REFL. NO H2O

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 3.4139E+00					
2	CYLINDER	0	RADIUS = 8.2500E+00	+Z = 2.0300E+01	-Z = -2.0300E+01			
3	CYLINDER	2	RADIUS = 8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01			
4	CYLINDER	3	RADIUS = 2.6615E+01	+Z = 4.4200E+01	-Z = -4.4200E+01			
5	CUBOID	0	+X = 2.6615E+01	-X = -2.6615E+01	+Y = 2.6615E+01	-Y = -2.6615E+01	+Z = 4.4200E+01	-Z = -4.4200E+01

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92897	+ OR - 0.00422	0.92475 TO 0.93319	0.92052 TO 0.93741	0.91630 TO 0.94163	30000
4	0.92844	+ OR - 0.00423	0.92420 TO 0.93267	0.91997 TO 0.93690	0.91574 TO 0.94113	29700
5	0.92839	+ OR - 0.00427	0.92412 TO 0.93266	0.91984 TO 0.93694	0.91557 TO 0.94121	29400
6	0.92834	+ OR - 0.00432	0.92403 TO 0.93266	0.91971 TO 0.93698	0.91539 TO 0.94130	29100
7	0.92910	+ OR - 0.00429	0.92481 TO 0.93340	0.92052 TO 0.93769	0.91622 TO 0.94199	28800
8	0.92973	+ OR - 0.00429	0.92544 TO 0.93403	0.92114 TO 0.93832	0.91685 TO 0.94261	28500
9	0.92970	+ OR - 0.00434	0.92536 TO 0.93404	0.92102 TO 0.93838	0.91669 TO 0.94272	28200
10	0.92896	+ OR - 0.00432	0.92463 TO 0.93328	0.92031 TO 0.93760	0.91599 TO 0.94192	27900 579125

FISSILE CLASS I UNDAAGED 3.15 kg WITH SPECULAR REFL. NO H2O

FREQUENCY FOR GENERATIONS 4 TO 103

0.8250 TO 0.8481	***
0.8481 TO 0.8712	****
0.8712 TO 0.8943	*****
0.8943 TO 0.9174	*****
0.9174 TO 0.9405	*****
0.9405 TO 0.9636	*****
0.9636 TO 0.9867	*****
0.9867 TO 1.0098	*****
1.0098 TO 1.0329	**
1.0329 TO 1.0560	*

579125 FISSILE CLASS II DAMAGED 16.80 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(01)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX	
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579125 FISSILE CLASS II DAMAGED 16.80 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(01)

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 92501 U-235-1R SIG P = 20 HANSEN ROACH

579125 FISSILE CLASS II DAMAGED 16.80 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(01)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 5.9600E+00
2 CYLINDER 0 RADIUS = 8.2500E+00 +Z = 2.0300E+01 -Z = -2.0300E+01
3 CYLINDER 2 RADIUS = 8.9000E+00 +Z = 2.1200E+01 -Z = -2.1200E+01
4 CUBOID 3 +X = 8.9000E+00 -X = -8.9000E+00 +Y = 8.9000E+00 -Y = -8.9000E+00 +Z = 2.1200E+01 -Z = -2.1200E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92759	+ OR - 0.00472	0.92287 TO 0.93231	0.91815 TO 0.93703	0.91342 TO 0.94175	30000
4	0.92757	+ OR - 0.00477	0.92280 TO 0.93233	0.91803 TO 0.93710	0.91326 TO 0.94187	29700
5	0.92919	+ OR - 0.00453	0.92466 TO 0.93372	0.92013 TO 0.93825	0.91560 TO 0.94278	29400
6	0.92873	+ OR - 0.00455	0.92418 TO 0.93329	0.91963 TO 0.93784	0.91507 TO 0.94239	29100
7	0.92838	+ OR - 0.00459	0.92379 TO 0.93297	0.91921 TO 0.93755	0.91462 TO 0.94214	28800
8	0.92801	+ OR - 0.00462	0.92339 TO 0.93263	0.91877 TO 0.93725	0.91415 TO 0.94187	28500
9	0.92793	+ OR - 0.00467	0.92326 TO 0.93260	0.91859 TO 0.93727	0.91392 TO 0.94194	28200
10	0.92832	+ OR - 0.00470	0.92362 TO 0.93303	0.91892 TO 0.93773	0.91422 TO 0.94243	27900

579125 FISSILE CLASS II DAMAGED 16.80 Kg SX2X1 TRI MESH(TI=10.0) OPT H20(01)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7544 TO 0.7775	*
0.7775 TO 0.8006	
0.8006 TO 0.8237	
0.8237 TO 0.8468	*
0.8468 TO 0.8699	*****
0.8699 TO 0.8929	*****
0.8929 TO 0.9160	*****
0.9160 TO 0.9391	*****
0.9391 TO 0.9622	*****
0.9622 TO 0.9853	*****
0.9853 TO 1.0084	*****
1.0084 TO 1.0315	**
1.0315 TO 1.0546	*
1.0546 TO 1.0777	*

579125 FISSILE CLASS III DAMAGED 25.80 kg 1X1X1 TRI MESH REFL OPT H2O(01)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579125 FISSILE CLASS III DAMAGED 25.80 kg 1X1X1 TRI MESH REFL OPT H2O(01)

MIXTURE	NUCLIDE	DENSITY
1	-92501	4.84310E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL	7.9G/CC	CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER	0.99826/CC	H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER	0.99826/CC	H=.066742, O=.033371
NUCLIDE =	92501	U-235-1R	SIG P = 20	HANSEN ROACH

579125 FISSILE CLASS III DAMAGED 25.80 kg 1X1X1 TRI MESH REFL OPT H2O(01)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 6.8816E+00						
2	CYLINDER	0	RADIUS = 8.2500E+00	+Z = 2.0300E+01	-Z = -2.0300E+01				
3	CYLINDER	2	RADIUS = 8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01				
4	CUBOID	3	+X = 8.9000E+00	-X = -8.9000E+00	+Y = 8.9000E+00	-Y = -8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01	
REFLECTOR		4	+X = 3.0000E+01	-X = 3.0000E+01	+Y = 3.0000E+01	-Y = 3.0000E+01	+Z = 3.0000E+01	-Z = 3.0000E+01	

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91122	+ OR - 0.00455	0.90667 TO 0.91577	0.90212 TO 0.92032	0.89756 TO 0.92487	30000
4	0.91037	+ OR - 0.00452	0.90586 TO 0.91489	0.90134 TO 0.91941	0.89682 TO 0.92392	29700
5	0.91049	+ OR - 0.00456	0.90593 TO 0.91506	0.90137 TO 0.91962	0.89681 TO 0.92418	29400
6	0.91015	+ OR - 0.00460	0.90555 TO 0.91474	0.90096 TO 0.91934	0.89636 TO 0.92394	29100
7	0.90989	+ OR - 0.00464	0.90525 TO 0.91452	0.90061 TO 0.91916	0.89598 TO 0.92380	28800
8	0.91044	+ OR - 0.00465	0.90579 TO 0.91509	0.90113 TO 0.91974	0.89648 TO 0.92440	28500
9	0.90994	+ OR - 0.00468	0.90527 TO 0.91462	0.90059 TO 0.91929	0.89592 TO 0.92397	28200
10	0.90990	+ OR - 0.00473	0.90517 TO 0.91463	0.90045 TO 0.91935	0.89572 TO 0.92408	27900

579125 FISSILE CLASS III DAMAGED 25.80 kg 1X1X1 TRI MESH REFL OPT H2O(01)

FREQUENCY FOR GENERATIONS 4 TO 103

0.8073 TO 0.8304	*****
0.8304 TO 0.8535	*****
0.8535 TO 0.8766	*****
0.8766 TO 0.8997	*****
0.8997 TO 0.9228	*****
0.9228 TO 0.9459	*****
0.9459 TO 0.9690	*****
0.9690 TO 0.9920	*****
0.9920 TO 1.0151	***
1.0151 TO 1.0382	*

579149 INFINITE ARRAY FISSILE CLASS I UNDAMAGED NO WATER (R=2.74)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	14	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	11	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	3	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	11	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	11	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00+00

MAXIMUM TIME = 120.0000 MINUTES

579149 INFINITE ARRAY FISSILE CLASS 1 UNDEMANAGED NO WATER (R=2.74)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL	7.96/CC CONTAINS C, CR, MN, FE, NI, SI
NUCLIDE =	8100	OXYGEN	HANSEN ROACH
NUCLIDE =	11100	SODIUM	HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN	(TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM	HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN	(TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM	HANSEN ROACH
NUCLIDE =	20100	CALCIUM	GAM-2
NUCLIDE =	22100	TITANIUM	GAM-2
NUCLIDE =	26100	IRON	HANSEN ROACH
NUCLIDE =	94901	PU-239-1	SIG P = 20

579149 INFINITE ARRAY FISSILE CLASS 1 UNDEMANAGED NO WATER (R=2.74)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 2.7400E+00		
2	YCYLINDER	0	RADIUS = 8.2500E+00	+Y = 2.0300E+01	-Y = -2.0300E+01
3	YCYLINDER	2	RADIUS = 8.9000E+00	+Y = 2.1420E+01	-Y = -2.0940E+01
4	YCYLINDER	3	RADIUS = 2.6615E+01	+Y = 4.4200E+01	-Y = -4.4200E+01
5	CUBOID	0	+X = 2.6615E+01	-X = -2.6615E+01	+Y = 4.4200E+01 -Y = -4.4200E+01 +Z = 2.6615E+01 -Z = -2.6615E+01

579149 INFINITE ARRAY FISSILE CLASS I UNDAHAGED NO WATER (R=2.74)

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92119	+ OR - 0.00523	0.91596 TO 0.92642	0.91073 TO 0.93165	0.90550 TO 0.93688	30000
4	0.92062	+ OR - 0.00525	0.91537 TO 0.92588	0.91012 TO 0.93113	0.90486 TO 0.93638	29700
5	0.92006	+ OR - 0.00528	0.91478 TO 0.92533	0.90951 TO 0.93061	0.90423 TO 0.93589	29400
6	0.92056	+ OR - 0.00531	0.91525 TO 0.92586	0.90994 TO 0.93117	0.90464 TO 0.93648	29100
7	0.91992	+ OR - 0.00532	0.91460 TO 0.92524	0.90927 TO 0.93057	0.90395 TO 0.93589	28800
8	0.91975	+ OR - 0.00538	0.91437 TO 0.92513	0.90899 TO 0.93050	0.90362 TO 0.93588	28500
9	0.91936	+ OR - 0.00542	0.91394 TO 0.92479	0.90852 TO 0.93021	0.90310 TO 0.93563	28200
10	0.91847	+ OR - 0.00540	0.91307 TO 0.92387	0.90766 TO 0.92928	0.90226 TO 0.93468	27900

579149 INFINITE ARRAY FISSILE CLASS I UNDAHAGED NO WATER (R=2.74)

FREQUENCY FOR GENERATIONS 4 TO 103	
0.7942 TO 0.8173	**
0.8173 TO 0.8404	***
0.8404 TO 0.8635	*****
0.8635 TO 0.8866	*****
0.8866 TO 0.9096	*****
0.9096 TO 0.9327	*****
0.9327 TO 0.9558	*****
0.9558 TO 0.9789	*****
0.9789 TO 1.0020	*****
1.0020 TO 1.0251	***
1.0251 TO 1.0482	*
1.0482 TO 1.0713	
1.0713 TO 1.0944	
1.0944 TO 1.1175	
1.1175 TO 1.1406	*

579149 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 & WATER REFL.(R=4.05)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	15	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579149 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 & WATER REFL.(R=4.05)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	502	1.00000E+00
4	501	1.00000E-01

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.9982G/CC H=.066742, O=.033371
94901 PU-239-1 SIG P = 20 HANSEN ROACH

579149 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 & WATER REFL.(R=4.05)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 4.0500E+00
2 YCYLINDER 0 RADIUS = 8.2500E+00 +Y = 2.0300E+01 -Y = -2.0300E+01
3 YCYLINDER 2 RADIUS = 8.9000E+00 +Y = 2.1420E+01 -Y = -2.0940E+01
4 CUBOID 4 +X = 8.9000E+00 -X = -8.9000E+00 +Y = 2.1420E+01 -Y = -2.0940E+01 +Z = 8.9000E+00 -Z = -8.9000E+00
REFLECTOR 3 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.92384	+ OR - 0.00596	0.91788 TO 0.92981	0.91192 TO 0.93577	0.90596 TO 0.94173	30000
4	0.92371	+ OR - 0.00602	0.91769 TO 0.92973	0.91167 TO 0.93575	0.90565 TO 0.94177	29700
5	0.92422	+ OR - 0.00606	0.91816 TO 0.93028	0.91209 TO 0.93634	0.90603 TO 0.94240	29400
6	0.92467	+ OR - 0.00611	0.91856 TO 0.93078	0.91246 TO 0.93688	0.90635 TO 0.94299	29100
7	0.92490	+ OR - 0.00617	0.91873 TO 0.93106	0.91256 TO 0.93723	0.90640 TO 0.94340	28800
8	0.92566	+ OR - 0.00618	0.91948 TO 0.93185	0.91330 TO 0.93803	0.90711 TO 0.94421	28500
9	0.92571	+ OR - 0.00625	0.91946 TO 0.93196	0.91321 TO 0.93820	0.90696 TO 0.94445	28200
10	0.92569	+ OR - 0.00632	0.91937 TO 0.93201	0.91306 TO 0.93832	0.90674 TO 0.94464	27900

579149 5X2X1(TI=10.0) FISSILE CLASS II DAMAGED 0.1 & WATER REFL.(R=4.05)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7506 TO 0.7737	*
0.7737 TO 0.7968	**
0.7968 TO 0.8199	*
0.8199 TO 0.8430	**
0.8430 TO 0.8661	*****
0.8661 TO 0.8892	*****
0.8892 TO 0.9123	*****
0.9123 TO 0.9354	*****
0.9354 TO 0.9585	*****
0.9585 TO 0.9816	*****
0.9816 TO 1.0047	*****
1.0047 TO 1.0278	**
1.0278 TO 1.0509	*
1.0509 TO 1.0740	*
1.0740 TO 1.0971	**

579149 1X1X1 FISSILE CLASS III DAMAGED 0.1 WATER & REFL.(R=4.35)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	15	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579149 1X1X1 FISSILE CLASS III DAMAGED 0.1 WATER \$ REFL.(R=4.35)

MIXTURE	NUCLIDE	DENSITY
1	-94901	4.93830E-02
2	200	1.00000E+00
3	502	1.00000E+00
4	501	1.00000E-01

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL	7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER	0.99826/CC H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER	0.99826/CC H=.066742, O=.033371
NUCLIDE =	94901	PU-239-1	SIG P = 20 HANSEN ROACH

579149 1X1X1 FISSILE CLASS III DAMAGED 0.1 WATER \$ REFL.(R=4.35)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 4.3500E+00						
2	YCYLINDER	0	RADIUS = 8.2500E+00	+Y = 2.0300E+01	-Y = -2.0300E+01				
3	YCYLINDER	2	RADIUS = 8.9000E+00	+Y = 2.1420E+01	-Y = -2.0940E+01				
4	CUBOID	4	+X = 8.9000E+00	-X = -8.9000E+00	+Y = 2.1420E+01	-Y = -2.0940E+01	+Z = 8.9000E+00	-Z = -8.9000E+00	
REFLECTOR		3	+X = 3.0000E+01	-X = 3.0000E+01	+Y = 3.0000E+01	-Y = 3.0000E+01	+Z = 3.0000E+01	-Z = 3.0000E+01	

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91921	+ OR - 0.00587	0.91334 TO 0.92507	0.90747 TO 0.93094	0.90161 TO 0.93681	30000
4	0.91847	+ OR - 0.00588	0.91259 TO 0.92435	0.90671 TO 0.93023	0.90083 TO 0.93611	29700
5	0.91899	+ OR - 0.00592	0.91307 TO 0.92490	0.90716 TO 0.93082	0.90124 TO 0.93674	29400
6	0.91935	+ OR - 0.00597	0.91339 TO 0.92532	0.90742 TO 0.93128	0.90145 TO 0.93725	29100
7	0.92052	+ OR - 0.00591	0.91460 TO 0.92643	0.90869 TO 0.93234	0.90278 TO 0.93825	28800
8	0.92047	+ OR - 0.00598	0.91449 TO 0.92644	0.90852 TO 0.93242	0.90254 TO 0.93839	28500
9	0.92025	+ OR - 0.00603	0.91421 TO 0.92628	0.90818 TO 0.93232	0.90214 TO 0.93835	28200
10	0.92074	+ OR - 0.00608	0.91466 TO 0.92682	0.90858 TO 0.93290	0.90250 TO 0.93898	27900

579149 IXIX1 FISSILE CLASS III DAMAGED 0.1 WATER & REFL.(R=4.35)

FREQUENCY FOR GENERATIONS 4 TO 103

0.7229 TO 0.7460	*
0.7460 TO 0.7691	
0.7691 TO 0.7922	
0.7922 TO 0.8153	*****
0.8153 TO 0.8384	***
0.8384 TO 0.8615	*****
0.8615 TO 0.8846	*****
0.8846 TO 0.9077	*****
0.9077 TO 0.9308	*****
0.9308 TO 0.9538	*****
0.9538 TO 0.9769	*****
0.9769 TO 1.0000	*****
1.0000 TO 1.0231	*****
1.0231 TO 1.0462	**

579123 FISSILE CLASS I UNFAMAGED 1.75 kg WITH SPECULAR REFL. (H2O 0.0)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	5	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	2	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	2	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	13	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	1	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

THIS PROBLEM WILL BE RUN WITH SPECULARLY REFLECTING BOUNDARY CONDITION

THE ALBEDOS ARE +X = 1.00000E+00 -X = 1.00000E+00 +Y = 1.00000E+00 -Y = 1.00000E+00 +Z = 1.00000E+00 -Z = 1.00000E+00

MAXIMUM TIME = 150.0000 MINUTES

579123 FISSILE CLASS I UNDAMAGED 1.75 kg WITH SPECULAR REFL. (H2O 0.0)

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04
4	501	1.00000E+00
5	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL 7.96/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE =	8100	OXYGEN HANSEN ROACH
NUCLIDE =	11100	SODIUM HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN (TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN (TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM HANSEN ROACH
NUCLIDE =	20100	CALCIUM GAM-2
NUCLIDE =	22100	TITANIUM GAM-2
NUCLIDE =	26100	IRON HANSEN ROACH
NUCLIDE =	92301	U-233-1 SIG P = 20 HANSEN ROACH

579123 FISSILE CLASS I UNDAMAGED 1.75 kg WITH SPECULAR REFL. (H2O 0.0)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 2.8064E+00						
2	CYLINDER	0	RADIUS = 8.2500E+00	+Z = 2.0300E+01	-Z = -2.0300E+01				
3	CYLINDER	2	RADIUS = 8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01				
4	CYLINDER	3	RADIUS = 2.6615E+01	+Z = 4.4200E+01	-Z = -4.4200E+01				
5	CUBOID	0	+X = 2.6615E+01	-X = -2.6615E+01	+Y = 2.6615E+01	-Y = -2.6615E+01	+Z = 4.4200E+01	-Z = -4.4200E+01	

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT		95 PER CENT		99 PER CENT		NUMBER OF HISTORIES
			CONFIDENCE INTERVAL	CONFIDENCE INTERVAL	CONFIDENCE INTERVAL	CONFIDENCE INTERVAL			
3	0.91869	+ OR - 0.00443	0.91426 TO 0.92312	0.90983 TO 0.92756	0.90540 TO 0.93199	30000			
4	0.91949	+ OR - 0.00440	0.91508 TO 0.92389	0.91068 TO 0.92830	0.90627 TO 0.93270	29700			
5	0.91843	+ OR - 0.00432	0.91411 TO 0.92275	0.90979 TO 0.92707	0.90547 TO 0.93139	29400			
6	0.91882	+ OR - 0.00435	0.91448 TO 0.92317	0.91013 TO 0.92752	0.90578 TO 0.93187	29100			
7	0.91907	+ OR - 0.00439	0.91468 TO 0.92345	0.91029 TO 0.92784	0.90591 TO 0.93222	28800			
8	0.91864	+ OR - 0.00441	0.91423 TO 0.92306	0.90982 TO 0.92747	0.90541 TO 0.93188	28500			
9	0.91883	+ OR - 0.00446	0.91437 TO 0.92328	0.90991 TO 0.92774	0.90546 TO 0.93219	28200			
10	0.91874	+ OR - 0.00450	0.91424 TO 0.92324	0.90973 TO 0.92775	0.90523 TO 0.93225	27900			

579123 FISSILE CLASS I UNDAHAGED 1.75 kg WITH SPECULAR REFL. (H2O 0.0)

FREQUENCY FOR GENERATIONS 4 TO 103

0.8148 TO 0.8379 **
0.8379 TO 0.8610 *****
0.8610 TO 0.8841 *****
0.8841 TO 0.9071 *****
0.9071 TO 0.9302 *****
0.9302 TO 0.9533 *****
0.9533 TO 0.9764 *****
0.9764 TO 0.9995 ***
0.9995 TO 1.0226 ***
1.0226 TO 1.0457 **
1.0457 TO 1.0688 *

579123 FISSILE CLASS II DAMAGED 6.70 kg 5X2X1 TRI MESH(TI=10.0) OPT H20(01)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	4	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	4	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	4	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	5	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	2	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	4	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO

MAXIMUM TIME = 120.0000 MINUTES

579123 FISSILE CLASS II DAMAGED 6.70 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(01)

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	501	2.00000E-01
4	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE = 200 TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE = 501 X(E) WATER 0.99826/CC H=.066742, O=.033371
NUCLIDE = 502 DE/E WATER 0.99826/CC H=.066742, O=.033371
92301 U-233-1 SIG P = 20 HANSEN ROACH

579123 FISSILE CLASS II DAMAGED 6.70 kg 5X2X1 TRI MESH(TI=10.0) OPT H2O(01)

GEOMETRY DESCRIPTION

REGION

1 SPHERE 1 RADIUS = 4.3904E+00
2 CYLINDER 0 RADIUS = 8.2500E+00 +Z = 2.0300E+01 -Z = -2.0300E+01
3 CYLINDER 2 RADIUS = 8.9000E+00 +Z = 2.1200E+01 -Z = -2.1200E+01
4 CUBOID 3 +X = 8.9000E+00 -X = -8.9000E+00 +Y = 8.9000E+00 -Y = -8.9000E+00 +Z = 2.1200E+01 -Z = -2.1200E+01
REFLECTOR 4 +X = 3.0000E+01 -X = 3.0000E+01 +Y = 3.0000E+01 -Y = 3.0000E+01 +Z = 3.0000E+01 -Z = 3.0000E+01

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.91084 + OR - 0.00540		0.90543 TO 0.91624	0.90003 TO 0.92164	0.89463 TO 0.92704	30000
4	0.91028 + OR - 0.00543		0.90485 TO 0.91570	0.89942 TO 0.92113	0.89399 TO 0.92656	29700
5	0.90970 + OR - 0.00545		0.90424 TO 0.91515	0.89879 TO 0.92060	0.89334 TO 0.92606	29400
6	0.90954 + OR - 0.00551		0.90404 TO 0.91505	0.89853 TO 0.92056	0.89302 TO 0.92607	29100
7	0.91008 + OR - 0.00554		0.90454 TO 0.91562	0.89901 TO 0.92116	0.89347 TO 0.92670	28800
8	0.91005 + OR - 0.00560		0.90445 TO 0.91564	0.89885 TO 0.92124	0.89326 TO 0.92684	28500
9	0.91012 + OR - 0.00566		0.90446 TO 0.91577	0.89881 TO 0.92143	0.89315 TO 0.92709	28200
10	0.91060 + OR - 0.00570		0.90491 TO 0.91630	0.89921 TO 0.92200	0.89351 TO 0.92769	27900

579123 FISSION CLASS II DAMAGED 6.70 kg SX2X1 TRI MESH(1=10.0) OPT H20(01)

	FREQUENCY FOR GENERATIONS	4 TO	103
0.7376 TO 0.7607	*		
0.7607 TO 0.7838	*		
0.7838 TO 0.8069	*		
0.8069 TO 0.8300	***		
0.8300 TO 0.8531	****		
0.8531 TO 0.8762	*****		
0.8762 TO 0.8993	*****		
0.8993 TO 0.9224	*****		
0.9224 TO 0.9455	*****		
0.9455 TO 0.9686	*****		
0.9686 TO 0.9917	*****		
0.9917 TO 1.0148	***		
1.0148 TO 1.0379	*		
1.0379 TO 1.0609	*		

579123 FISSILE CLASS III DAMAGED 9.20 kg 1X1X1 TOUCHING OPT H20(01)

NUMBER OF GENERATIONS	103	START TYPE	3
NUMBER PER GENERATION	300	GENERATIONS BETWEEN CHECKPOINTS	0
NUMBER OF GENERATIONS TO BE SKIPPED	3	LIST INPUT X-SECTIONS READ FROM TAPE	NO
NUMBER OF ENERGY GROUPS	16	LIST 1-D MIXTURE X SECTIONS	NO
MAX. NUMBER OF ENERGY TRANSFERS	6	LIST 2-D MIXTURE X-SECTIONS	NO
NUMBER OF INPUT NUCLIDES	13	LIST FISS. AND ABS. BY REGION	NO
NUMBER OF MIXTURES	5	USE X-SECTIONS FROM PREVIOUS CASE	NO
NUMBER OF MIXING TABLE ENTRIES	13	USE GEOMETRY FROM PREVIOUS CASE	NO
NUMBER OF GEOMETRY CARDS	14	USE VELOCITIES FROM PREVIOUS CASE	NO
NUMBER OF BOX TYPES	1	COMPUTE MATRIX K-EFFECTIVE BY UNIT	NO
NUMBER OF UNITS IN X DIRECTION	1	COMPUTE MATRIX K-EFFECTIVE BY BOX TYPE	NO
NUMBER OF UNITS IN Y DIRECTION	1	LIST FISS PROB MATRIX BY UNIT	NO
NUMBER OF UNITS IN Z DIRECTION	1	ADJOINT CALCULATION	NO
NUMBER OF NUCLIDES READ FROM TAPE	13	USE EXPONENTIAL TRANSFORM	NO
ALBEDO TYPE	0	CALCULATE FLUX	NO
SEARCH TYPE	0	CALCULATE FISSION DENSITIES	NO
MAXIMUM TIME = 120.0000 MINUTES			

579123 FISSILE CLASS III DAMAGED 9.20 kg 1X1X1 TOUCHING OPT H2O(01)

MIXTURE	NUCLIDE	DENSITY
1	-92301	4.88560E-02
2	200	1.00000E+00
3	14100	2.96210E-03
3	8100	7.07260E-03
3	13100	3.40560E-04
3	20100	2.37600E-04
3	26100	1.58400E-04
3	22100	1.18800E-05
3	12100	3.16800E-05
3	11100	1.06920E-04
3	19100	1.08040E-04
4	501	2.00000E-01
5	502	1.00000E+00

CROSS SECTIONS READ FROM TAPE

NUCLIDE =	200	TYPE 304 STAINLESS STEEL 7.9G/CC CONTAINS C,CR,MN,FE,NI,SI
NUCLIDE =	501	X(E) WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE =	502	DE/E WATER 0.9982G/CC H=.066742, O=.033371
NUCLIDE =	8100	OXYGEN HANSEN ROACH
NUCLIDE =	11100	SODIUM HANSEN ROACH
NUCLIDE =	12100	MAGNESIUM XSDRN (TOTAL ADJUSTED JRJ 6/76)
NUCLIDE =	13100	ALUMINUM HANSEN ROACH
NUCLIDE =	14100	SILICON XSDRN (TOTAL ADJUSTED JRJ 8/76)
NUCLIDE =	19100	POTASSIUM HANSEN ROACH
NUCLIDE =	20100	CALCIUM GAM-2
NUCLIDE =	22100	TITANIUM GAM-2
NUCLIDE =	26100	IRON HANSEN ROACH
NUCLIDE =	92301	U-233-1 SIG P = 20 HANSEN ROACH

579123 FISSILE CLASS III DAMAGED 9.20 kg 1X1X1 TOUCHING OPT H2O(01)

GEOMETRY DESCRIPTION

REGION

1	SPHERE	1	RADIUS = 4.8799E+00						
2	CYLINDER	0	RADIUS = 8.2500E+00	+Z = 2.0300E+01	-Z = -2.0300E+01				
3	CYLINDER	2	RADIUS = 8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01				
4	CUBOID	4	+X = 8.9000E+00	-X = -8.9000E+00	+Y = 8.9000E+00	-Y = -8.9000E+00	+Z = 2.1200E+01	-Z = -2.1200E+01	
REFLECTOR		5	+X = 3.0000E+01	-X = 3.0000E+01	+Y = 3.0000E+01	-Y = 3.0000E+01	+Z = 3.0000E+01	-Z = 3.0000E+01	

WT. ID = 500

16 GROUP WEIGHTS FOR WATER , HAVING 10 INTERVALS, EACH 3.0 CM THICK WERE USED.

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	0.90285	+ OR - 0.00513	0.89772 TO 0.90798	0.89259 TO 0.91311	0.88746 TO 0.91824	30000
4	0.90297	+ OR - 0.00518	0.89779 TO 0.90815	0.89261 TO 0.91333	0.88742 TO 0.91851	29700
5	0.90213	+ OR - 0.00517	0.89696 TO 0.90729	0.89180 TO 0.91246	0.88663 TO 0.91762	29400
6	0.90139	+ OR - 0.00516	0.89622 TO 0.90655	0.89106 TO 0.91172	0.88589 TO 0.91688	29100
7	0.90166	+ OR - 0.00521	0.89644 TO 0.90687	0.89123 TO 0.91208	0.88602 TO 0.91729	28800
8	0.90182	+ OR - 0.00526	0.89655 TO 0.90708	0.89129 TO 0.91235	0.88602 TO 0.91761	28500
9	0.90244	+ OR - 0.0091301	0.88659 TO 0.91829	0.89716 TO 0.90773	0.89188 TO 0.91301	29000
10	0.90305	+ OR - 0.00531	0.89774 TO 0.90835	0.89244 TO 0.91366	0.88713 TO 0.91896	27900
FREQUENCY FOR GENERATIONS 4 TO 103						

0.7527 TO 0.7758 *
 0.7758 TO 0.7989 *
 0.7989 TO 0.8220 ***
 0.8220 TO 0.8451 *****
 0.8451 TO 0.8682 *****
 0.8682 TO 0.8913 *****
 0.8913 TO 0.9144 *****
 0.9144 TO 0.9375 *****
 0.9375 TO 0.9606 *****
 0.9606 TO 0.9837 *****
 0.9837 TO 1.0068 *****
 1.0068 TO 1.0299 *
 1.0299 TO 1.0530 *

7. OPERATING PROCEDURES

Practices and procedures for safely handling and properly packaging radioactive materials at MRC-Mound are covered in three basic program documents, which are described below (other DOE sites utilizing Type B nuclear packagings are expected to have equivalent arrangements).

- (1) Nuclear Packaging Manual, MD-10279. This manual has been developed to serve as a consistent and definitive reference for packaging radioactive products and materials for shipment, thereby providing Mound packaging personnel with instructive guidelines for packaging radioactive materials for offsite shipment, in full compliance with all applicable Department of Energy (DOE) orders, the federal regulations of both the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT), and Mound or other agency or customer requirements.
- (2) Quality Plan for Type B Reusable Nuclear Packaging, MD-10236. This quality plan is intended to provide MRC-Mound management, the design agencies, and the Department of Energy with assurance that the reusable radioactive materials packages employed for offsite shipment meet the established specifications and requirements. All aspects of acquisition, inspection, handling, maintenance, utilization, and control for Type B nuclear shipping packages used for offsite shipment of radioactive materials, except U. S. DOT Spec. 6M (refer to Technical Manual MD-10233), are covered in this quality plan, which is structured to comply with U. S. DOE Order 5700.6, Quality Assurance Requirements (equivalent to the U. S. NRC quality program requirements of 10 CFR Part 71).
- (3) Nuclear Radiation Safety, MD-10019. This manual describes guidelines and refers to documents that describe work methods in environments where radioactive materials are present or contained, establishes responsibilities and guidelines for Mound employees in radiation areas, and ensures radiation safety in accordance with established regulatory standards within Mound's "as low as reasonably achievable" (ALARA) policy.

7.1. Procedures for Loading Packages

Operating procedures and instructions for loading or unloading nuclear packagings at MRC-Mound are contained in approved "operation sheets," which are collected into technical manuals in order to maintain publication control. Such operation sheets are formatted as instructional checklists to be completed by the operator(s) as the packages are being loaded, assembled, and shipped. Completed operation sheets, along with any related forms or documentation, are retained on file to verify compliance with applicable procedures and requirements.

The following MRC-Mound forms are included as examples of the forms to be completed during packaging activities (in the absence of printed forms, these could be copied and used as needed).

- (1) MD-70152, OP. 1 (Figure 7-1). This is a checklist for the operators' reuse inspection for fit, function, and condition during packing operations and is also a generalized assembly process sheet when either loaded or empty containers are being prepared for shipment.
- (2) MD-70152, OP. 15 (Figure 7-2) and Form MRC-ML-1245 (Figure 7-3). For all radioactive shipments, these forms are used by MRC-Mound to record surface contamination levels of the outer shipping package and to provide documentation to show compliance with all shipping regulations. Equivalent forms will be completed by other users.

Prior to each use, Type B packaging used at MRC-Mound is inspected and accepted by Quality Control and released for use by Quality Engineering (see section 8.2).

For loading the 5790 or 5791 package, the contents are first wipe tested for removable contamination before release by Health Physics. The contents are placed within the inner container and any void space is filled with metal shot, metal wool, or other suitable inert material. The inner container lid is fastened to the body, and this assembly is installed within the outer drum assembly on the fire retardant filler support. The fire retardant plug is inserted, the drum lid is put into position and the closure ring is put on and fastened tightly with the ring bolt. The outer package is checked by Health Physics for both fixed and removable contamination, and a radiation survey is conducted to ensure that external radiation from the package is within U. S. Department of Transportation regulatory limits.

As each operation is completed, the Form MD-70152, OP. 1, is so marked; in addition, Form MRC-ML-1245 is completed giving contamination and radiation levels and Form MD-70152, OP. 15, is also completed for verification of the condition of the package. The security seal is installed, its number recorded on Form MRC-ML-1245, radioactive labels are completed and attached, and the package is then ready for shipment.

7.2. Procedure for Unloading Packages

Unloading a package is basically the reverse of loading. Upon receipt, the package is surveyed for contamination before removal of the cover, and each component is checked as it is removed from the package. After the contents are removed from the inner container and checked for contamination, the packaging is placed in "hold" storage pending the reuse inspection and Quality Engineering disposition.

7.3. Preparation of Empty Package for Transport

To be prepared for shipment, empty packaging must first be surveyed and released by Health Physics. It must then be reassembled completely, with all component parts in proper order. The radioactive material label (namely, the yellow-III labels) should be removed, "empty" labels applied in compliance with DOT requirements, and a security seal installed as necessary.

7.4. Appendix

Not applicable.

MONSANTO RESEARCH CORP.
MOUND LABORATORY

SHIPPING CONTAINER

OPERATION SHEET

PROGRAM REUSABLE RADIOACTIVE SHIPPING CONTAINERS		SHEET 1 of 1	MANUAL NUMBER MD-70152	OPERATION 1
AUTHORIZATION <i>N. H. Brunel</i>	CLASSIFICATION Unclassified	EFFECTIVITY 10-12-77	ECN(S) INCORPORATED M5-C	
OPERATION TITLE Use, Inspection, and Loading (for use with DOT-6M, 9506, 5790, 5791)				

USE INSPECTION

New _____ Reuse _____ Specification _____ Serial No. _____

FINDING

1. Remove labels.
2. Closure ring and locking device in working condition.
3. Check all containers for contamination and decontaminate as necessary.
4. Thread appears in good condition (remove foreign matter).
5. Centering material in good condition.
6. Overall condition (satisfactory):

Date _____ Signature _____

LOADING PROCEDURE (GENERAL)

COMPLETED

1. Radioactive material form (solid, liquid or gas).
2. Lute inner container threads/gasket.
3. Torque inner container closure (if required).
4. Install insulation.
5. Place compression pad (as required) and sketch under drum lid.
6. Close drum.
7. Install seal.
8. Apply labels (Radioactive).

Date _____ Signature _____

DISTRIBUTION:

White - Container File
Yellow - Nuclear QC
Goldenrod - Shipper File
Pink - Originator

MRC-ML-128
T

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FIGURE 7-1 - MD-70152, Op. 1. Operators' use, inspection, and loading checklist.

MONSANTO RESEARCH CORPORATION
MOUND FACILITY

SHIPPING CONTAINER
OPERATION SHEET

PROGRAM REUSABLE NUCLEAR PACKAGING SHIPPING CONTAINER		SHEET 1 of 1	MANUAL NUMBER MD-70152	OPERATION 15
TECH. RESP. <i>[Signature]</i>	DATE 4/10/87	QUALITY ENGINEER <i>[Signature]</i>	DATE 4/10/87	EFFECTIVITY 12-1-86
OPERATION TITLE DOT Quality Requirements Check Sheet		ICNIR INCORPORATED 860407MD-C		

DOT QUALITY REQUIREMENTS CHECK SHEET

Unclassified

SHIPPING REQUEST NUMBER _____ Date _____

SHIPPING CONTAINER S/N(S): _____

CHECK OFF

1. Verify that the proper packaging is being used.
 Identity of packaging used _____
2. Verify that packaging is in unimpaired physical condition, except for superficial marks. _____
3. Verify that each closure device of the packaging, including any required gasket, is properly installed, secured, and free of defects. _____
4. For fissile material only, verify that each moderator and neutron absorber, if required, is present and in proper position. _____
5. Verify that each special instruction for filling, closing, and preparation of the packaging for shipment has been followed. _____
- Operation Sheet Form number _____
6. Verify that each closure, valve, or other opening of the containment system through which the radioactive content might escape is properly closed and sealed. _____
7. Verify that each packaging containing liquid in excess of an A2 quantity and intended for air shipment has been tested to show that it will not leak under an ambient atmospheric pressure of not more than 0.25 atmosphere, absolute, (0.25 kilograms per square centimeter or 3.6 psia). The test must be conducted on the entire containment system, or on any receptacle or vessel within the containment system, to determine compliance with this requirement.
 SARF number or other test document ID _____
8. Verify that the internal pressure of the containment system will not exceed the design pressure during transportation. _____
- Explain how verified: _____
9. Verify that external radiation and contamination levels are within the allowable limits specified in parts 173.441 and 173.443 of 49 CFR. _____
- Form 1245 ML number _____

SIGNATURE _____ DATE SIGNED _____

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MRC-ML-128 (7-83)

FIGURE 7-2 - MD-70152, Op. 15. DOT quality requirements checklist.

SHIPPING RADIOACTIVE AND FISSILE MATERIAL

MONSANTO RESEARCH CORPORATION
MRC-MOUND
MIAMISBURG, OHIO 45342

*Revised
fig. 73
Reduce and
mount
~ 75%*

PO NUMBER	MATL
	☐ U

Material Req'd.	NO. OF PKGS.	"M.L." NUMBER

SHIP TO:

I Modes of Transportation	Courier Agency (if any)	Escort Agency (if any)
---------------------------	-------------------------	------------------------

SECURITY SEAL NUMBERS:	1.	2.	3.	4.	5.	6.	7.	8.
------------------------	----	----	----	----	----	----	----	----

PACKAGE IDENTITY: MODEL & SERIAL NO.	CONTENTS: ISOTOPE CHEMICAL/ PHYSICAL FORM	CURIE AMOUNT (Ci)	RADIATION (mRem/hr.)				SURFACE CONTAMINATION dis/min/100 cm ²						TRANSPORT INDEX (TI)			
			SURFACE		1 METER		PRIMARY		OUTER				MAX. 10/PKG. 50/SHIPMENT			
							REMOVABLE		REMOVABLE		FIXED		Radiation		FISSILE	
			β - γ	η	β - γ	η	α	β - γ	α	β - γ	α	β - γ	$\gamma + \eta @ 1m$	Index	Class	
1.																
2.																
3.																
4.																
5.																
6.																
7.																

RADIATION SURVEY INSTRUMENTS USED α _____ β _____ γ _____ η _____
(Record Identity Numbers)

III RADIOACTIVE LABELS REQUIRED	None Inner Only	W-1	Y-11	Y-111	White Empty	Others	IV Warehouse Shipping Document No.
---------------------------------	-----------------	-----	------	-------	-------------	--------	------------------------------------

Comments:	H.P. Surveyor
-----------	---------------

Approved for Shipment:	Signature	Date
Requestor		
Nuclear Criticality		
Health Physics		
Special Material Handling		
Other		

**PERSONNEL INVOLVED IN
LOADING OF CONTAINERS**

INSTRUCTIONS:

1. Requestor complete Part I, except M.L. Number; and Part II columns 2 and 3.
2. Health Physics complete Part II columns 4 and 5, Radiation portion of column 6, and Part III; and insert M.L. No.
3. Nuclear Matl. Control complete Part II, column 1. Enter Security Seal numbers.
4. Nuclear Criticality complete Criticality column 6 of Part II.
5. Special Matl. Handling complete Part IV and make distribution. For Empty Containers, no M.L. No. is to be issued.

- DISTRIBUTION:
1. Special Matl. Handling
 2. Nuclear Matl. Control
 3. Nuclear Criticality
 4. Health Physics
 5. Quality Engineering

MRC-ML-1245 (8-87)
17-1435

THIS SECTION MUST BE FILLED IN SECURITY CLASSIFICATION	
☐	CLASSIFIED - Must be signed by Authorized Classifier
☐	UNCLASSIFIED - Must be signed by Supervisor or Designated Alternate
Classification Category	
Authorized Signature	Date

FIGURE 7-3 - MRC-ML-1245. Radioactive and fissile material shipment form.

8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1. Acceptance Tests Performed Prior to First Use

Prior to the first use of the 5790 and 5791 packages, extensive test procedures, both destructive and nondestructive, were performed on prototype units to verify compliance with design criteria. Such tests and evaluations are described, and the results reported, in preceeding chapters of this document.

8.1.1. Visual Inspection

Visual inspections were performed on each part, component, subassembly, and assembly of each prototype test unit to ensure correct fit, finish, and function, as described in preceeding chapters and in the Appendix, section 8.3.

8.1.2. Structural and Pressure Tests

The structural and pressure tests are described and discussed in Chapter 2.

8.1.3. Leak Tests

Leak tests are described and discussed in the preceeding chapters and in the Appendix, section 8.3.

8.1.4. Component Tests

For those components to be tested, the related tests are described and discussed in the preceeding chapters and in the Appendix, section 8.3. Furthermore, all aspects of acquisition, inspection, handling, storage, maintenance, utilization, and control for type B nuclear packagings, or packaging components used for offsite shipment of radioactive materials (except U.S. DOT Spec. 6M) are covered under the Mound "Quality Plan for Type B Reusable Nuclear Packaging," MD-10236. This plan complies with U.S. DOE Order 5700.6b, "Quality Assurance Requirements" (equivalent to U.S. NRC quality program requirements of 10 CFR Part 71).

8.1.5. Shielding Integrity Tests

Since there is no shielding installed in either the 5790 or 5791 package, no tests for shielding integrity are performed.

8.1.6. Thermal Acceptance Tests

Thermal performance is established in the prototype testing and qualification described and reported in Chapter 3 of this report. Any damage, deviation, or inadequacy of materials, parts, or components related to the thermal characteristics of these packagings would be determined by the visual inspections; therefore, thermal acceptance testing on individual units is not performed.

8.2. Maintenance Program

Fabrication acceptance inspections are to be performed before first use of each new or remanufactured 5790 or 5791 packaging unit. The nondestructive inspections, measurements, and determinations specified in the Appendix, section 8.3., shall be completed. These include applicable visual and dimensional inspections, structural, pressure, and leak tests and examinations for fit and function of all parts, components, subassemblies, valves, gauges, gaskets, and other items that could impair the effectiveness of the packaging unit.

Reuse inspection, acceptance, and QC release for use are to be performed on each packaging unit prior to any reuse. Each 5790 or 5791 package shall be disassembled, inspected, and accepted by a qualified quality control inspector (that is, someone from the Mound Receiving Inspection group) according to the applicable inspection operation sheet (see Figures 8-1 and 8-2). Any indication of damage, wear, degradation, or the defective, inadequate or deviant condition of any item, part, component, subassembly or assembly during this reuse inspection, shall cause the packaging unit to be placed in a "hold" status pending resolution of the observed condition. The cognizant quality engineer convenes a material review board if needed, for resolution of the nonconforming conditions. The inspection/acceptance report prepared and issued by the QC inspector is reviewed by the cognizant quality engineer, who issues a release for use of the packaging units identified as acceptable. This release is transmitted to, and retained on file by the Nuclear Packaging group for utilization as needed.

Empty 5790 or 5791 packages shall be marked with "empty" labels and stored in controlled and protected areas, to minimize environmental effects, pilferage, or other incidental or accidental damage. During storage, packagings should be placed under the control of the Nuclear Materials Control group.

Routine basic maintenance on the 5790 and 5791 packaging units such as cleaning and painting, replacement of gaskets, nuts, bolts, and so forth, is the responsibility of the functional group performing the nuclear packaging operations. Such maintenance activities shall be performed in accordance with the applicable quality program requirements of MD-10236. A reuse inspection/acceptance is required prior to quality engineering release for use of repaired or corrected packaging units.

8.2.1. Structural and Pressure Tests

In addition to the reuse inspection required prior to each use (including the mass spectrometer helium leak test which is part of the loading process), the containment vessels shall be examined at least once every four years. Damage, defects, or abnormalities in the weldments are determined using a dye penetrant process equivalent to that used for the initial acceptance inspection, as described in the Appendix, section 8.3.

Upon completion of this testing with acceptable findings, the cognizant quality engineer shall issue an acceptance report and release the vessels for use to the Nuclear Packaging group. The quality engineer shall also ensure that the vessels are permanently and legibly marked with the acceptance date

RECEIVING INSPECTION/PRODUCT ACCEPTANCE
OPERATION SHEET

COMPONENT	DRAWING NO.	ISSUE	P R NO.	QC NUMBER
5790 SHIPPING CONTAINER (USED)	AYD750138	E		
	AYD750140	D		
	AYC750141	C		
	AYC750142	C		

REUSABLE
NUCLEAR SHIPPING
CONTAINERS

MANUAL NO.	OP.	ISSUE	SHEET
MD30072	5790A	1	1 of 2

APPROVAL/EFFECTIVITY/SIGNATURES			
PRODUCT ENGINEER	DATE	QC ENGINEER	DATE
<i>Max D. Williams</i>	5/5/84	<i>[Signature]</i>	5/5/84

VENDOR:

(USE BLACK INK)

ATTRIBUTES		SAMPLE	ACCEPT	NEEDS MATERIAL REVIEW	ACCEPTANCE			REMARKS
					EQUIPMENT	DRAWING NO.	RECORD ISSUE	
	HEALTH PHYSICS SURVEY							
1	Exterior wipe test before opening package	100%			Appropriate Alpha, Beta or Gamma Radiation detection instruments			
2	Interior direct survey or wipe test as components are opened	100%						
3	Wipe test survey inside inner container AYC750140	100%						
4	Exterior finish	100%			Visual			No severe rusting, scrapes, scratches, gouge which compromise metal thickness
5	Dents and deformities	100%			Visual			Do not interfere with the function/contents.
6	Gasket	100%			Visual			Good condition, flexible make good seal
7	Identification plate	100%			Visual			Note serial number
8	Weld, seam	100%			Visual			No split
9	Closure ring and locking device	100%			Functional Fit			Good working condition

MRC-ML-8117

FIGURE 8-1 - MD-30072, Op. 5790A, sheets 1 and 2.

RECEIVING INSPECTION/PRODUCT ACCEPTANCE
OPERATION SHEET

COMPONENT	DRAWING NO.	ISSUE	P R NO.	QC NUMBER
5790 SHIPPING CONTAINER (USED)	AYD750138	E		
	AYD750140	D		
	AYC750141	C		
	AYC750142	C		

REUSABLE
NUCLEAR SHIPPING
CONTAINERS

MANUAL NO.	OP.	ISSUE	SHEET
MD30072	5790A	1	2 of 2

APPROVAL/EFFECTIVITY/SIGNATURES			
PRODUCT ENGINEER	DATE	QC ENGINEER	DATE
<i>Monde Williams</i>	5/5/86	<i>Abdel</i>	5/5/86

VENDOR:

(USE BLACK INK)

ATTRIBUTES		SAMPLE	ACCEPT	NEEDS MATERIAL REVIEW	ACCEPTANCE			REMARKS
					EQUIPMENT	DRAWING NO.	RECORD ISSUE	
10	Insulation	100%			Visual			No cracks or breakage
11	Cylinder serial number	100%			Visual			Cover and cylinder same serial number
12	Foreign materials	100%			Visual			No foreign material inside inner container
13	"O" Ring	100%			Visual			No cracks and in good condition
14	Sealing surface	100%			Visual			Clean and smooth
15	Overall condition of container assembly	100%			Visual			Good condition

MRC-ML-8117

FIGURE 8-1 - (continued).

RECEIVING INSPECTION/PRODUCT ACCEPTANCE
OPERATION SHEET

COMPONENT	DRAWING NO.	ISSUE	P R NO.	QC NUMBER
5791 SHIPPING CONTAINER (USED)	AYD750148	E		

REUSABLE
NUCLEAR SHIPPING
CONTAINERS

MANUAL NO.	OP.	ISSUE	SHEET
MD30072	5791	1	1 of 2

APPROVAL/EFFECTIVITY/SIGNATURES			
PRODUCT ENGINEER	DATE	QC ENGINEER	DATE
<i>Max D. Williams</i>	5/5/86	<i>John Doe</i>	5/5/86

VENDOR:

(USE BLACK INK)

ATTRIBUTES		SAMPLE	ACCEPT	NEEDS MATERIAL REVIEW	ACCEPTANCE			REMARKS
					EQUIPMENT	DRAWING NO.	RECORD ISSUE	
	HEALTH PHYSICS SURVEY				Appropriate alpha, beta or gamma radiation detection instruments			
1	Exterior wipe test before opening package	100%						
2	Interior direct survey or wipe test as components are opened	100%						
3	Wipe test survey inside inner container AYD750150	100%						
4	Exterior finish	100%			Visual			No servere rusting, scrapes, scratches, gouges, which compromise metal thickness.
5	Dents and deformities	100%			Visual			Do not interfere with the function/contents
6	Gasket	100%			Visual			Good condition, flexible make good seal
7	Identification Plate	100%			Visual			Note Serial Numbers
8	Weld, seam	100%			Visual			No split
9	Closure ring and locking device	100%			Functional Fit			Good working condition
10	Insulation	100%			Visual			No cracks or breakage
11	Foreign materials	100%			Visual			No foreign material inside inner container

MRC-ML-8117

FIGURE 8-2 - MD-30072, Op. 5791, sheets 1 and 2.

REUSABLE NUCLEAR SHIPPING CONTAINERS

MANUAL NO.	OP.	ISSUE	SHEET
MD30072	5791	1	2 of 2

VENDOR:

(USE BLACK INK)

MRC-ML-8117

of test, the facility where the test was performed, and the name of the responsible quality engineer.

8.2.2. Leak Tests

Mass spectrometer helium leak tests for the containment vessels of the 5790 and 5791 packages are part of the loading processes and are described in the respective loading operation sheets included in Chapter 7.

8.2.3. Subsystem Maintenance

There are no subsystems in the 5790 or 5791 packages.

8.2.4. Valves, Rupture Disks, and Gaskets on Containment Vessels

Gaskets on containment vessels are inspected visually and functionally for each usage and are replaced as needed. There are no rupture discs or valves in the 5790 or 5791 packages.

8.2.5. Shielding

There is no shielding installed in the 5790 or 5791 packages, other than materials of construction.

8.2.6. Thermal

Any damage, deviation, or inadequacy of materials, parts, or components related to the thermal characteristics of these packaging units would be determined by visual examination during reuse inspections.

8.2.7. Miscellaneous

There are no additional tests specified for the 5790 or 5791 packages.

8.3. Appendix

8.3.1. Detailed Drawings

SPA 740977, Issue E, 7 shts.
SPA 790424, Issue A, 4 shts.
SPA 790425, Issue A, 4 shts.
SPA 790426, Issue A, 4 shts.
SPA 790427, Issue A, 4 shts.

SP740977		W O NO	
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ISSUE	DATE	REVISION	BY	CHK'D	APPRO
A	4-3-75	ORIGINAL ISSUE	CP	WVD	GAB
B	4-15-77	CHANGED PER LD-2-14876-M47 & M51	DM	WVD	GAB
C	5-13-77	CHANGED PER LD-2-14876-M52-C	DM	WVD	GAB
D	6-28-77	CHANGED PER LD-2-14876-M53-C	DM	WVD	GAB
E	9-10-79	CHANGED PER LD-2-14876-M67	PG	WVD	GAB

SHEET											
ISSUE											
SHEET	1	2	3	4	5	6	7				
ISSUE	E	B	B	B	B	B	B				

<div style="text-align: center;"> DWG CLASSIFICATION LEVEL & CAT UNCL. BY <i>D. Osmer</i> TITLE _____ DATE <i>1-2-80</i> </div>	MONSANTO RESEARCH CORPORATION MOUND FACILITY MIAMISBURG, OHIO	
	NUCLEAR SHIPPING CONTAINER 5790/91 ACCEPTANCE AND REUSE INSPECTION	
	DRAWN PG DATE <i>9/10/79</i> CHECKED <i>WVD</i> DATE <i>1-2-80</i> SPONSOR <i>WVR</i> DATE <i>1-4-80</i> APPROVED <i>DVB</i> DATE <i>1-2-80</i>	APPROVALS <div style="border: 1px solid black; padding: 2px; text-align: center;"> <i>WVR</i> </div> <div style="border: 1px solid black; padding: 2px; text-align: center;"> 2 </div> <div style="border: 1px solid black; padding: 2px; text-align: center;"> 3 </div> <div style="border: 1px solid black; padding: 2px; text-align: center;"> 4 </div>
	JOB NO _____	

SP	CODE IDENT NO	
DWG TYPE	14065	

DWG NO SPA740977 SHT 1 OF 7	
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1.1. This is an inspection criteria document for the following radioactive materials shipping containers:

<u>Type</u>	<u>Model</u>	<u>Certificate of Compliance</u>	<u>Drum Size</u>
5790	AL-M3	USA/5790/BLF(DOE-AL)	12 gallon
5791	AL-M6	USA/5791/BLF(DOE-AL)	55 gallon

2. INTRODUCTION.

2.1. The Type 5790 and 5791 shipping containers are designed for offsite shipping of radioactive materials, including Pu 238 oxide. They meet the safety requirements of the Department of Energy and the unique requirements of the various materials which are shipped in them.

2.2. The two sizes of containers are illustrated in Figures 1 and 2. All have similar basic designs consisting of an outer steel drum and inner container which is packed with insulation to provide for mechanical and thermal shock during accident conditions

[illegible]

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DAC-73-SACB **MRC-ML-5317 (12.73)**

SPA740977

SHIPPING CONTAINER
MODEL 103 A1-M3
USA/5790/BLF (DOE-A1)
FIGURE 1

ISSUE

B

CODE IDENT NO
14065

DWG. NO. SPA740977 SHT. 3

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DAC-72-MASCB MRC-ML-5317

SPA 740977

SHIPPING CONTAINER
MODEL 131 A1-M6
USA-5791/BLF (DOE/AL)
FIGURE 2

ISSUE

B

CODE IDENT NO
14065

DWG. NO. SPA740977 SHT. 4

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MRC-ML-5317
DAC-72-MASCB

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3. A P P L I C A B L E D O C U M E N T S.

Monsanto Research Corporation Drawings:

AYD750138 - 103/AL-M3 Shipping Container Assembly

AYD750140 - 103/AL-M3 Inner Container Body

AYD750141 - 103/AL-M3 Inner Container Cover

AYD750148 - 131/AL-M6 Shipping Container Assembly

AYD750150 - 131/AL-M6 Inner Container Body

AYD750151 - 131/AL-M6 Inner Container Cover

AYD790558 - 131/AL-M6 Inner Container Cover with Pressure Gauge

Other Documents

Department of Energy Manual, Chapter 0529

Safety Analysis Report for Packing (SARP): USA/5790/BLF
(DOE-AL) and USA/5791/BLF (DOE-AL) MLM-2242 (Rev. 1),
Monsanto Research Corp., Miamisburg, Ohio, December 21, 1979.

4. GENERAL PROVISIONS.

4.1. Responsibilities. The manufacturer is responsible for assuring that all specific requirements are met and shall utilize accepted quality control measures during manufacture. Monsanto Research Corporation reserves the right to audit the manufacturer's facilities and procedures as required. MRC Engineering is responsible for establishing inspection criteria for receiving acceptance and reuse, and for providing this document.

MRC Nuclear Operations Quality Control is responsible for auditing the Quality Control Program and review of applicable drawings and specifications to assure that the Quality Control Program is carried out as defined. MRC Nuclear Material

[illegible]

DAC-73-SACB

DAC-73-SACB **MRC-ML-5317 (12-73)**

LL604740977		
<p>5.4. <u>Inspection of Final Assembly.</u> The final assembly shall be examined visually and functionally with regard to meeting the requirements of the Final Assembly. Defects which shall be cause for rejection are listed in MRC Drawing No. SPA 790426 for the type 5790 and SPA 790427 for the type 5791 Initial Acceptance Inspection Criteria and Data Sheets. The MRC inspector is required to fill out the data sheets specified.</p> <p>5.5. <u>Inspection Plan.</u> Each shipping container shall be inspected for all attributes listed in MRC Drawing No. SPA 790426 for the type 5790 and SPA 790427 for the type 5791.</p> <p>6. <u>U S E I N S P E C T I O N S.</u></p> <p>6.1. <u>Criteria.</u> Inspections prior to use of the container are required and are specified by MRC Shipping Container Operation Sheet, Manual Number MD-70152, Operation 1. That document also specifies loading procedures.</p> <p>6.2. <u>Documentation.</u> The Operation Sheet, MD-70152 Operation 1, referred to in Section 6.1 above specifies the MRC form which the inspector is required to fill out.</p> <p>6.3. <u>Inspection Plan.</u> Each shipping container, new or used, is to be inspected prior to each use of that container.</p>		
ISSUE	B	
CODE IDENT NO	14065	DWG NO SPA740977 SHT 7

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DAC-73-SACB MRC-ML-5317 (12 73)

				W O N O			
ISSUE	DATE	R E V I S I O N			BY	CHK'D	APPRO
A	9/10/79	ORIGINAL ISSUE			PG	WVD	GAB
SHEET							
ISSUE							
SHEET	1	2	3	4			
ISSUE	A	A	A	A			
		DWG CLASSIFICATION			MOUSANTO RESEARCH CORPORATION		
		LEVEL & CAT UNCL			MOUND FACILITY		
		BY <i>Ed Green</i>			MIAMSBURG, OHIO		
		TITLE			NUCLEAR SHIPPING CONTAINER		
		DATE 1-2-80			FABRICATION & INSPECTION		
					CERTIFICATION FOR TYPE 5790		
					CONTAINERS		
		DRAWN PG			DATE 9/10/79		APPROVALS
		CHECKED WVD/			DATE 1-2-80		1 <i>JHR</i>
		SPONSOR			DATE		2
		APPROVED <i>WVD</i>			DATE 1/2/80		3
		JOB NO					4
SC1	CODE IDENT NO				DWG NO SPA790424 SHT 1 OF 4		
DWG TYPE	14065						

DAC-78-SA

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<p>1. <u>S C O P E</u></p> <p>1.1 This document specifies the fabrication and inspection certification forms for Type 5790 shipping containers as required by Section 5.3 of MRC Drawing No. SPA740977, the document which defines the general inspection program for these containers.</p> <p>2. <u>F O R M S R E Q U I R E D</u></p> <p>2.1 The fabricator of the inner container must complete Form MRC-ML-6796 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 3 of this specification.</p> <p>2.2 The fabricator of an outer container must complete Form MRC-ML-6797 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 4 of this specification.</p>		
ISSUE	A	
CODE IDENT NO 14065		DWG. NO. 790424 SHT. 2

DAC-72-MASCB MRC-ML-5317

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Reference MRC Dwg. No. AYD750138

OUTER CONTAINER SERIAL NUMBER _____

DATE _____

1. Drum is MS-27684-8.	_____	_____
2. Insulation and other materials and components as specified.	_____	_____
3. Insulation glued as specified.	_____	_____
4. Dimension as specified.	_____	_____
5. Container can be assembled properly.	_____	_____
6. Container tagged and dated to indicate inspection results satisfactory.	_____	_____
7. Fabricator overall review and approval of this certification.	_____	_____

DISTRIBUTION: White - Nuclear QC
Yellow - Shipping Container Engineer
Pink - File

MRC-ML-6797

[illegible]

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DAC-72-MASCB MRC-ML-5317

1.1 This document specifies the fabrication and inspection certification forms for Type 5791 shipping containers as required by Section 5.3 of MRC Drawing No. SPA740977, the document which defines the general inspection program for these containers.

- 2.1 The fabricator of the inner container must complete Form MRC-ML-6798 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 3 of this specification.
- 2.2 The fabricator of an outer container must complete Form MRC-ML-6799 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 4 of this specification.

[illegible]

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Reference MRC Dwg. Nos. AYD750150 (Body) and AYD750151 (Cover)

INNER CONTAINER SERIAL NUMBER _____

DATE _____

1. Components and materials as specified.	_____	_____
2. Welded as specified.	_____	_____
3. Visual weld examination acceptable.	_____	_____
4. Dye penetrant examination acceptable.	_____	_____
5. Dimensions as specified.	_____	_____
6. O-ring surface finish as specified.	_____	_____
7. Container can be assembled properly.	_____	_____
8. Helium leak test shows no leak greater than 1×10^{-6} std. cc/sec at 15 psig.	_____	_____
9. Container tagged and dated to indicate inspection results satisfactory.	_____	_____
10. Fabricator overall review and approval of this container.	_____	_____

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Pink - File

MRC-ML-6798

[illegible]

DWG. NO. SPA790425 SHT. 3

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0AC-72-MA5CB MRC-ML-5317

Reference MRC Dwg. No. AYD150148

OUTER CONTAINER SERIAL NUMBER _____

DATE _____

1. Drum is 55-gal., DOT-17C.	_____	_____
2. Insulation and other materials and components as specified.	_____	_____
3. Insulation glued as specified.	_____	_____
4. Dimensions as specified.	_____	_____
5. Container can be assembled properly.	_____	_____
6. Container tagged and dated to indicate inspection results satisfactory.	_____	_____
7. Fabricator overall review and approval of this certification.	_____	_____

DISTRIBUTION: White - Nuclear QC
Yellow - Shipping Container Engineer
Pink - File

MRC-ML-6799

[illegible]

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DAC-72-MASCB MRC-ML-5317

PAC-18-SA

8-23

1.1 This document specifies the initial acceptance criteria and data sheets for Type 5790 shipping containers as required by Section 5.4 of MRC Drawing No. SPA740977, the document which defines the general inspection program for these containers.

- 2.1 The fabricator of the inner container must complete Form MRC-ML-6800 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 3 of this specification.
- 2.2 The fabricator of an outer container must complete Form MRC-ML-6801 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 4 of this specification.

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1. _____ Inner container serial number.
2. _____ Manufacturer of inner container.
3. _____ Manufacturer has satisfactorily completed and submitted a "Fabrication and Inspection Certification" form MRC-ML-6796.
4. _____ Metal plate securely attached with the words "RADIOACTIVE MATERIAL" stamped or etched into the plate. (Major)
5. _____ Welds appear acceptable. (Major)
6. _____ Threads appear to be in good condition. (Major)
7. _____ Workmanship acceptable with no sharp edges or burrs. (Critical)
8. _____ Silicon or viton O-ring has no visible damage. (Major)
9. _____ O-ring sealing surfaces appear to be in good condition. (Major)

10. Dimensions:
 - A. _____ Container outside diameter is $5 \pm 1/8$ in. (Major)
 - B. _____ Container overall height with cover on and including pins on cover is $14\ 9/16$ in. maximum and $14\frac{1}{2}$ in. minimum. (Major)
 - C. _____ Container body wall thickness is $3/8$ in. minimum. (major)
 - D. _____ Cover thickness is $3/4$ in. minimum. (Major)
 - E. _____ Container body inside depth is $11\frac{1}{2} \pm \frac{1}{2}$ in. (Minor)
 - F. _____ Container bottom plate is $\frac{1}{2} \pm 1/16$ in. (Major)
11. _____ Cover screws into body such that a minimum of five complete threads are engaged. (Major)
12. _____ Inner container completely disassembled and assembled satisfactorily. (Major)
13. _____ Helium leak test performed with no detectable leak greater than 1×10^{-5} std. cc/sec when pressurized to 10 psig with helium. (Major)
14. _____ Inner container tagged and dated to indicate satisfactory inspection results. (Minor)

COMMENTS: _____

Signature _____ Date _____

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MRC-ML-6800

ISSUE

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DWG. NO. SPA790426 SHT. 3

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DAC-72-MASCB MRC-ML-5317

INITIAL ACCEPTANCE INSPECTION CRITERIA AND DATA SHEET

TYPE 5790 SHIPPING CONTAINER INSULATING DRUM ASSEMBLY

(Reference MRC Dwg. AYD750138)

1. _____ Drum assembly serial number.
2. _____ Manufacturer of drum assembly.
3. _____ Manufacturer has certified by letter that all items are as specified or MRC Engineering approved equal. (Major)
4. _____ Workmanship acceptable with no sharp edges or burrs. (Critical)
5. _____ Insulation cap assembly can be lifted into and out of the drum easily. (Major)
6. _____ The drum assembly can be assembled and disassembled easily. (Major)
7. _____ Drum cover fits with no interference with bolts on insulation cap. (Critical)
8. _____ Drum permanent head marked MS-27684-18. (Major)
9. _____ Drum cover marked MS-27684-23. (Major)
10. _____ Name plate (MRC Drawing No. AYC750142) welded to drum side reads USA/5790/BLF(ERDA-AL), auth.gr.wt. 90 lb, Model AL-M3, and shows serial number. (Major)
11. _____ Monsanto logo securely attached to side of drum. (Minor)
12. _____ Drum has four vent holes sealed with tape. (Major)
13. _____ Drum closure ring has skirt, lugs and tab or other provision for attachment of security seal. (Major)
14. _____ Web strap for lifting insulation cap is securely attached. (Major)

15. Measured Dimensions:
 - A. _____ Drum overall diameter with locking ring off is $14\frac{1}{2} \pm \frac{1}{4}$ in. (Major)
 - B. _____ Drum overall height with cover off is $20\frac{7}{8} \pm \frac{1}{4}$ in. (Major)
 - C. _____ Maximum gap between insulation and side of drum is less than $\frac{1}{4}$ in. (Major)
 - D. _____ Maximum gap between top of insulation and top lip of drum with cover and cerawool pad removed is less than $1\frac{1}{2}$ in. (Major)
 - E. _____ Diameter of cavities in insulation cap and body both are $5\frac{5}{8} \pm \frac{1}{4}$ in. (Major)
 - F. _____ Depth of cavity in insulation cap.
 - G. _____ Depth of cavity in insulation body.
 - H. _____ Overall height (exclusive of lifting strap) of insulation cap.
 - I. _____ Overall height of insulation body.
16. Calculated Dimensions:
 - J. _____ Overall cavity depth (as calculated from dimensions F + G) is $14\frac{5}{8}$ in. minimum. (Major)
 - K. _____ Thickness of insulation body below cavity (as calculated from dimensions I-G) is 1.75 in. minimum. (Major)
 - L. _____ Thickness of insulation cap above cavity (as calculated from dimensions H-F) is 1.75 in. minimum. (Major)

17. _____ Drum tagged and dated to indicate inspection results satisfactory. (Major)

COMMENTS: _____

Signature _____

Date _____

MRC ML 6801

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ISSUE

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1.1 This document specifies the initial acceptance criteria and data sheets for Type 5791 shipping containers as required by Section 5.3 of MRC Drawing No. SPA740977, the document which defines the general inspection program for these containers.

- 2.1 The fabricator of the inner container must complete Form MRC-ML-6802 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 3 of this specification.
- 2.2 The fabricator of an outer container must complete Form MRC-ML-6803 and submit it to MRC Nuclear Operations Quality Control. This form shall be as shown on Sheet 4 of this specification.

DAC-72-MASCB MRC-ML-5317

8-28

1. _____ Inner container serial number.
2. _____ Manufacturer of inner container.
3. _____ Manufacturer has satisfactorily completed and submitted a "Fabrication and Inspection Certification" form MRC-ML-6798.
4. _____ Metal plate securely attached with the words "RADIOACTIVE MATERIAL" stamped or etched into the plate. (Major)
5. _____ Welds appear acceptable. (Major)
6. _____ Workmanship acceptable with no sharp edges or burrs. (Critical)
7. _____ O-ring sealing surfaces appear to be in good condition. (Major)
8. _____ Silicon or viton o-ring has no visible damage. (Major)
9. _____ Lifting handle for lid securely attached. (Major)
10. _____ Gauge and gauge protection are securely fastened to lid (alternate design only). (Major)

11. Dimensions:
 - A. _____ Container tube outside diameter is $7 \pm 1/16$ in. (Major)
 - B. _____ Container exclusive of handle but including installed closure bolts is $17 \frac{1}{8} \pm 1/8$ in. (Critical)
 - C. _____ Inner container handle height is 1½ in. maximum. (Critical)
 - D. _____ Hex head closure bolts are 3/8 in. dia. X 1½ in. long. (Major)
 - E. _____ Height of gauge with protection is 1½ in. maximum (alternate design only). (Critical)
12. _____ Lifting handle is securely fastened. (Major)
13. _____ Inner container completely disassembled and assembled satisfactorily. (Major)
14. _____ Helium leak test performed with no detectable leak greater than 1×10^{-5} std. cc/sec when filled to 5 ± 1 psig helium. (Major)
15. _____ Inner container tagged and dated to indicate inspection results satisfactory. (Minor)

COMMENTS: _____

Signature _____ Date _____

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MRC-ML-6802

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DAC-72-MASCB MRC-ML-5317

INITIAL ACCEPTANCE INSPECTION CRITERIA AND DATA SHEET
TYPE 5791 SHIPPING CONTAINER INSULATING DRUM ASSEMBLY
(Reference MRC Dwg. AYD150148)

1. _____ Drum assembly serial number.
2. _____ Manufacturer of drum assembly.
3. _____ Vendor has certified by letter that all items are as specified or MRC Engineering approved equal. (Major)
4. _____ Workmanship acceptable with no sharp edges or burrs. (Critical)
5. _____ Insulation cap assembly can be lifted into and out of the drum easily. (Major)
6. _____ The drum assembly can be assembled and disassembled easily. (Major)
7. _____ Drum cover fits with no interference with bolts on insulation cap. (Critical)
8. _____ Drum permanent head marked DOT 17C, 16-55-year manufacture, name or symbol of manufacture. (Major)
9. _____ Name plate (MRC Dwg. No. AYD750142) welded to drum side reads USA/5791/BLF(ERDA-AL). Auth. Gr. Wt. 300 lb., Model AL-M8, and shows serial number. (Major)
10. _____ Monsanto logo securely attached to side of drum. (Minor)
11. _____ Drum has twelve vent holes sealed with tape. (Major)
12. _____ Drum closure ring has skirt, lugs and tab or other provision for attachment of security seal. (Major)
13. _____ Web strap for lifting insulation cap is securely attached. (Major)

14. Dimensions:
- A. _____ Drum inside diameter is $22\frac{1}{2} \pm \frac{1}{8}$ in. (Major)
 - B. _____ Drum inside height with cover off is $33\frac{3}{8}$ in. minimum. (Major)
 - C. _____ Maximum gap between insulation and side of drum is less than $\frac{1}{8}$ in. (Major)
 - D. _____ Gap between installed insulation assembly (exclusive of lifting strap) to top of open drum is $1\frac{1}{8}$ in. max. (Major)
 - E. _____ Inside diameter of lower body cavity in insulation assembly is $7\frac{1}{8}$ in. maximum, $7\frac{1}{8}$ in. minimum. (Major)
 - F. _____ Inside diameter of upper body cavity in insulation assembly is $10\frac{1}{8}$ in. maximum, 10 in. minimum. (Major)
 - G. _____ Height of cavity in insulation cap assembly is $1\frac{1}{8} \pm \frac{1}{8}$ in. (Major)
 - H. _____ Inside diameter of cavity in insulation cap assembly is $6\frac{1}{2} \pm \frac{1}{8}$ in. (Major)
 - I. _____ Depth of lower body cavity in insulation assembly.
 - J. _____ Depth of upper body cavity in insulation assembly is $2\frac{1}{8}$ in. minimum. (Critical)
 - K. _____ Insulation cap height (exclusive of lifting strap).
 - L. _____ Overall height of insulation body.

15. Calculated Dimensions:
- M. _____ Overall depth of stepped, dual diameter body cavity in insulation assembly (as calculated from dimensions I + J) is 17 3/8 in. minimum. (Critical)
- N. _____ Thickness of insulation below body cavity (as calculated from dimensions L-M) is 7 in. minimum. (Major)
- O. _____ Thickness of insulation cap above cavity (as calculated from dimensions K-G) is 7 in. minimum. (Major)
16. _____ Drum tagged and dated to indicate inspection results satisfactory. (Major)

COMMENTS:

Signature _____ Date _____

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MFC-ML-8803

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