SAFETY ANALYSIS REPORT
PIQUA-ELK RIVER SHIPPING CASK
(Packaging of Radioactive and Fissile Materials)

FINAL REPORT

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Abstract

Irradiated fuel from the NS Savannah can be loaded dry in the Piqua-Elk River cask equipped with a new poison insert and shipped as Fissile Class III material in compliance with AEC Manual Chapter 0529 (AECM 0529), "Safety Standards for the Packaging of Radioactive and Fissile Materials."

This report, as a supplement to the Knapp Mills report [1], presents the evaluation of the cask with the new insert for shipping the spent NS Savannah fuel.

Description of Shipment

Irradiated fuel assemblies discharged from the NS Savannah will be shipped from Todd Shipyards, Galveston, Texas, to the Savannah River Plant, Aiken, South Carolina.

All shipments will be made dry, as Fissile Class III material, in compliance with AECM 0529, in the 30-ton Piqua-Elk River Shipping Cask (figure 1) which has been equipped with a new poison insert (figures 2 and 3) for these shipments. This cask insert, which holds four 8-1/2-in.-square fuel assemblies (figure 4), consists of an outer steel cylindrical shell with four 9-3/4-in.-square fuel compartments with a steel-clad, Boral plate central divider serving as neutron absorber for criticality control.

Each fuel assembly consists of about 550 lb UO_2 plus some 200 lb of type 304 stainless steel contained in the fuel assembly structure. Four of these assemblies, weighing about 3000 lb, are held by the insert, which weighs about 1750 lb.

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Each fuel assembly (figure 4) consists of four subassemblies of 41 fuel rods each, a total of 164 rods, which are fastened top and bottom by the upper and lower support frames. These frames in turn are locked together with the central shaft support spine, which also provides a lifting bail for the assembly. All fuel rods are maintained at 0.66-inch-square-pitch spacing by a series of main and peripheral spacing ferrules.

The calculated minimum number of fuel pins needed to form a critical array when the assemblies are fully moderated by water is 386 (2.35 assemblies) of the most reactive type 2 fuel (4.596% ^{235}U by weight) or 432 (2.68 assemblies) of the less reactive type 1 fuel (4.194% ^{235}U by weight). These enrichments are before burnup. Four nonpoisoned fuel assemblies could be supercritical by nearly as much as 10%. In the absence of water moderation, an unlimited number of dry casks would be subcritical. The new insert for the Piqua cask is designed with the poison-cross configuration to ensure a substantial margin of subcriticality irrespective of the number of fuel assemblies in place in the insert and the amount of water present as neutron moderator.

The poison-cross neutron absorber is 1/4-in.-thick Boral (B_4C -Al matrix) plate, clad both sides with 0.120-in.-thick steel for protection against abrasion during shipping, and for added strength and stiffness. The analysis of the Boral plate has been certified by the vendor supplying this material. As a further check on its neutron absorbing properties, blackness tests were made on the plate by Savannah River Plant. A final check on the neutron absorbing properties of the poison insert is planned for the first underwater loading of the cask at Todd Shipyards, by means of neutron multiplication measurements and the determination of k_{eff} for the completely moderated and loaded cask. The cask and insert will then be drained of water and dried so that the shipment can be made dry. The multiplication measurement should be made with the most reactive NS Savannah fuel available; i.e., type 2 with minimum burnup of ^{235}U .

The temperature attained by the hottest fuel will be less than 1030°F during normal transport and less than 1070°F during a transport accident involving a 30-minute exposure of the cask to the regulatory 1475°F fire. At these temperatures, the pressures developed within the fuel pins as a result of thermal expansion of helium used for thermal bonding coupled with expansion

of fission gases released from the uranium oxide fuel are far below the pressure corresponding to the 1000-hour creep rupture strength of the type 304 stainless steel cladding of the fuel (Appendix 3).

The temperature attained by the steel-sheathed Boral cross that serves as nuclear poison and as a divider for the four fuel assemblies can attain a temperature in its central zone approaching that of the hottest fuel element (Appendix 4). This temperature is calculated conservatively to be less than 1070°F during the fire. At this temperature, there is no interaction between Boral and the steel sheathing, and no melting of the aluminum matrix of the Boral plate (mp of aluminum is 1220°F); hence no damage to the plate or loss of nuclear poison can occur.

The calculated temperature of the outer steel cross plates and stiffeners that support the fuel assemblies and the poison-cross divider (figure 2) is about 400°F during normal transport and about 650°F during the hypothetical fire (Appendix 4). At these temperatures, there is no significant loss of strength (carbon steel suffers a 10% loss of strength at a temperature of 750°F).

From experience gained on experimental drop tests of other casks and inserts, drop tests of unprotected Zircaloy-clad fuel elements, and calculations of stress in certain parts of the structure, it is concluded that the insert will not permit fuel assemblies to form a critical array if the cask were flooded after the drop and fire, and that the accident would not impair the integrity of the stainless steel cladding on NS Savannah fuel or result in escape of significant radioactivity to the cask or surroundings.

In event of a transport accident involving fire, the average temperature of air in the cask can be expected to increase from <480°F to <770°F with possible increase in cask pressure to about 5 psig (Appendix 3). This pressure is below the design pressure of 100 psig and below the rupture disk relief setting of 75 psig.

The calculated temperature of the outer metal surface of the cask under the conditions of 130°F ambient air, solar heating, and an assumed 8000-Btu/hr

heat release from fuel is 200°F, corresponding to a 70°F differential between metal and air. Under expected conditions of 100°F maximum air temperature, solar heating, and a 5000-Btu/hr actual heat release, the outside cask metal temperature would reach a maximum of about 160°F.

Calculations indicate that the cask radiation level will be <55 mR/hr at the surface and <10 mR/hr at 6 feet. These values are within DOT regulatory limits for the exclusive use vehicle employed.

Comparison with Regulations

I. DEFINITIONS AND EXEMPTIONS

No comment is required.

II. PACKAGE STANDARDS

A. General Standards for All Packaging

1. No reaction between package and contents.
2. Positive closure on packaging. [1]
3. Lifting devices, *ibid*.
4. Tiedown devices, *ibid*.

B. Structural Standards for Large Quantity Packaging [1]

C. Criticality Standards for Fissile Material Packages

1. A package with complete moderation and reflection is subcritical. (Appendix 1)
2. Not applicable.
3. Not applicable.

D. Evaluation of a Single Package

1. (a) The package withstands conditions likely to occur in normal transport. (Mechanical: Appendix 2; Thermal: Appendix 4)

- (b) The package withstands conditions likely to occur in the hypothetical accident. (Mechanical: Appendix 3; Thermal: Appendix 4)

- 2. Evaluation of package and vehicle together.
Not applicable.
- 3. Approval of different accident conditions.
Not applicable.

E. Standards for Normal Conditions of Transport for a Single Package

Discussed with Item II,D,1,(a). No activity release.

F. Standards for Hypothetical Accident Conditions for a Single Package

Discussed with Item II,D,1,(b). No activity release.

G. Evaluation of an Array of Packages of Fissile Materials

Appendix 1.

H. Specific Standards for a Fissile Class I Package

Not applicable.

I. Specific Standards for a Fissile Class II Package

Not applicable.

J. Specific Standards for a Fissile Class III Package

The single cask will have exclusive use of the vehicle.

The criticality safety of four NS Savannah fuel assemblies in the cask was evaluated (Appendix 1). The enrichments of the two types of fuel assemblies (4.59 and 4.194%) are less than the 5.0% limit (ANSI N16.1 - 1969) for unmoderated systems; hence an unlimited number of dry casks would be subcritical.

Moderation by water cannot be excluded in an accident and will occur when the cask is unloaded. The calculations indicate that

the central cross of 1/4-in. Boral sheet will keep the cask subcritical by an adequate margin of safety when the cask and insert are flooded.

III. OPERATING PROCEDURES

A. Establishment and Maintenance of Procedures

A check list of items for preparation of procedures for the shipment of NS Savannah fuel has been supplied to the shipper for use in preparing Operating Procedures.

B. Assumptions as to Unknown Properties

Not applicable.

C. Preliminary Determination

1. The new insert constructed by Savannah River Plant for use in the Piqua-Elk River cask has been constructed in accordance with Du Pont and Savannah River Plant quality control procedures and standards governing both acquisition of materials and construction methods.
2. A final mechanical inspection prior to release of the insert indicated that there were no cracks, pinholes, or other defects which could significantly reduce its effectiveness for the intended use, and that the insert was fabricated in accordance with the approved design.
3. A criticality blackness test for neutron absorption prior to release of the insert indicated that the Boral poison plate member was fabricated in accordance with the approved design and performs its proper neutron absorbing (blackness) function for the intended use.
4. As a final check on the criticality-control measures adopted for these shipments, neutron multiplication measurements by Todd Shipyards are planned for the first underwater loading of the insert and cask in which the most reactive (highest enrichment) NS Savannah fuel available will be used.

5. Obtaining a good seal with the previous cask closure has presented some operating difficulties. Possible modifications to provide a simple and reliable seal for higher pressure and perhaps also for higher temperature are under investigation. Meanwhile, for currently proposed shipments, the cask has been equipped with a flat elastomeric gasket, 0.125-in.-thick, 60-durometer neoprene. With this gasket, the lid bolts, nuts, and underside of nuts should be lubricated with "Molykote" or other suitable lubricant and torqued to 80-90 ft-lb.

Before the cask is released from the Savannah River Plant for off-site shipments it will be leak-tested with 5-psig air.

D. Routine Determination

1. Pressure Test

To avoid during shipment the release of activity that may be present on internal cask surfaces, or on the surfaces of fuel assemblies, the cask should be sealed after loading and tested for leakage. The recommended test involves pressurizing the loaded and drained cask with air to about 5 psig, immersing in water, and observing for bubbles. Some bubbles of temporarily "trapped" air may be observed and are not of concern, but continued escape of bubbles indicative of leakage would be cause for corrective action and repeat of the leak test. An alternative method for searching for leaks is the conventional soap-bubble test with the cask pressurized but out of water.

2. Dry Shipment (at Thermal Equilibrium)

It is desired to make the shipment dry at approximate thermal equilibrium.

After the cask is leak-tested and the seals are found to be tight, suitable hose connections should be made and the cask

should be purged with a downward flow of air at the rate of 1 to 2 cfm for 12 hours or more, to remove residual water and let the contents approach thermal equilibrium. This purge should be carried out at atmospheric pressure. When the purge is complete, the purge inlet and outlet connections should be sealed, and the cask shipped at atmospheric pressure.

E. Records

Not applicable to present analysis.

APPENDIX 1

Criticality Safety Evaluation for NS Savannah Core Shipment

FOREWORD

The initial criticality evaluation of the new insert for use in the Piqua-Elk River cask when shipping NS Savannah fuel was based on the use of an insert with an unsymmetrical arrangement of fuel compartments, two of which were 9×9 in. internal dimensions and two of which were $9\text{-}7/8 \times 9$ in. The final design adopted makes use of a symmetrical arrangement where each of the four fuel assembly compartments is $9\text{-}3/4$ in. square, internal dimensions.

Using these changed dimensions, the results of the initial criticality analysis were revised. The original analysis (starting with Introduction and Summary) then follows because it serves as point of departure for the revision and is needed to complete this record.

The dimensions of the compartments in the cask insert have been changed. All compartments will now be the same size and will be $9\text{-}3/4$ in. square, internally. The average thickness of the space separating the outside of the compartment walls from the inner wall of the cylindrical shell enclosing the insert will be 2.96 in. This space will contain some steel bracing, which occupies only a small volume fraction and which was ignored in the calculations, and will be filled with water whenever the cask is filled. By using Hansen-Roach cross sections, together with the ORNL cross sections for lead, KENO monte carlo calculations for type 2 fuel in a flooded, cuboidal cask gave a k_{eff} of 0.8728 ± 0.0038 with the fuel in the near corners, adjacent to the Boral. With the fuel in the far corners, k_{eff} was calculated to be 0.8335 ± 0.0046 . These values may be compared with previous values of 0.8591 ± 0.0059 for the flooded cask and 0.8619 ± 0.0038 for the water-reflected insert. Homogenization of the fuel in all these calculations, as pointed out previously, gives it too low a reactivity. The critical value of k_{eff} should be about 0.9510 as indicated by table 4. The margin of subcriticality is thus about $0.9510 - 0.8728 \approx 0.08$, which is more than sufficient to compensate for uncertainties in the calculation.

INTRODUCTION AND SUMMARY

The criticality safety of shipping four NS Savannah fuel assemblies in the Piqua-Elk River cask was evaluated. The enrichments of the two types of fuel assemblies (4.59 and 4.194%) are less than the 5.0% limit [2] for unmoderated systems; hence an unlimited number of dry casks will be subcritical. Moderation by water cannot be excluded in an accident, however, and will occur at SRP when the cask is unloaded. Calculations indicate that a central cross of 1/4-in. Boral sheet will keep the cask subcritical by an adequate margin of safety when it is flooded.

It is recommended that the subcriticality of the cask be confirmed by neutron multiplication measurements when it is filled with water for the first time. An attempt should be made to have the most reactive fuel (type 2 with minimum burnup) present when these measurements are made so that it will not be necessary to repeat them.

DISCUSSION

The fuel rods are described in table 1. An assembly contains 164 rods, a 13×13 array with the center and four corner rods missing. The missing rods were ignored in the evaluation. The evaluation was made for fresh fuel. The actual fuel is depleted some 20% and contains fission products. Plutonium built in only partially compensates for the loss in reactivity. [3] Since the enrichment is below 5.0%, the fuel rods cannot be made critical in the absence of moderation.¹ Water, however, cannot be excluded under accident conditions and will be introduced prior to unloading the fuel at SRP. MGBS-TGAN [4] calculations were made for a water-reflected cylinder of rods at the lattice pitch in the assemblies. For enrichments between 1.5 and 5.0% correlations [5] of MGBS-TGAN with experiment indicate that k_{eff} calculated for a critical configuration at this pitch should be 0.9872. (Experiments at 4.02% enrichment with a prototype of the NS Savannah fuel indicate a value of about 0.9912 [5] for k_{eff} ; experiments at 5.742%, with similar fuel, indicate a value of 0.9967. [4]) The critical number of rods (i.e., the number at 0.9872) was calculated to be 432 and 386 for types 1 and 2, respectively. Thus the four assemblies (656 rods) proposed for a

¹ This limit applies to metal. [2] Hansen-Roach cross sections [6] give a k of 0.926 for metal with this enrichment. For UO_2 or UO_3 , the enrichment at this same k is calculated to be about 5.65%. For clad type 1, k is 0.679; for clad type 2, k is 0.718.

shipment will be critical in water unless sufficient isolation is provided by water or neutron absorbing materials between the assemblies.

To investigate how far supercritical four assemblies would be, several one-dimensional two group diffusion theory calculations were made by TGAN with parameters generated by MGBS. The fuel region was 167.6 cm (66 in.) high and, with the assemblies in contact, 43.784 cm (26×0.663 in.) thick in each of the other two directions. Calculations were made to determine extrapolation distances in the vertical direction and in the two (identical) horizontal directions, with the assemblies in contact and at several separations. The extrapolation distances together with the dimensions gave geometric bucklings and hence values of k_{eff} for the 2×2 array as a function of separation. Results are given in table 2, and when consideration is given to the correlation with experiment indicate that the array can be supercritical by nearly as much as 10%.

An insert is being designed (figures 2 and 3) for the cask that will contain a central steel clad cross fabricated from 1/4-in.-thick Boral sheet to provide sufficient neutron absorption to keep the cask subcritical. The Boral sheet has a 0.210-in. Boral core, having a density of 2.53 g/cm^3 and containing 35% B_4C by weight, with 0.020-in.-thick aluminum cladding. The steel cladding for the Boral cross is 11 gage (0.120-in. thick). Steel walls (11 gage) enclose the cross forming two 9×9 in. (internal dimensions) compartments and two $9\text{-}7/8 \times 9$ in. compartments. One of the larger compartments has spacers restricting the fuel assembly to a 9×9 -in. space adjacent to the arms of the cross. The other larger compartment is for accommodating fuel assemblies with special external structure. The arms of the cross extend essentially the full extent of the compartments. The four compartments are enclosed by a 2 ft 5-3/4 in. OD cylindrical steel shell (11 gage). Braces maintain the position of the compartments within the shell but occupy little volume. The gap between the compartments and the shell will be filled with water when the cask is flooded. On the average, the gap is 3.5 in. thick, if the shell is assumed to be rectangular while enclosing the same volume.

TGAN calculations, similar to those reported in table 2, were made for water reflected compartments containing each type of fuel assembly. Reactivity was greatest with the fuel assemblies adjacent to the steel clad Boral, least with them adjacent to the outer wall of the compartment, but differences were small (~ 0.004 in. k_{eff}). For type 1 fuel the maximum k_{eff} was calculated to be 0.879 and for type 2, 0.897. The reactivity, however, is strongly dependent on the boron cross section chosen for the higher energy group. The cross section selected was derived from an appropriate 4-group set of cross sections, but confirmation of these results by other methods was considered necessary.

Two methods were selected. In one, the fuel cell was homogenized, and Hansen-Roach cross sections [6] were used in ANISN [7] and KENO [8] calculations to determine the effectiveness of the Boral. In the other, HAMMER [9] calculations (B_1 , Nordheim resonance treatment for ^{238}U , iterated until $k_{\text{eff}} \approx 1$ @ input buckling) for a cell containing Boral sheets and homogenized fuel were matched by SLICE calculations in which Boral cross sections were adjusted to give agreement. These cross sections were then used in SLICE and KENO calculations together with average HAMMER cross sections generated for the heterogeneous fuel cell. These calculations were restricted to type 2 fuel because it is the more reactive.

Critical bucklings calculated by various methods for a cell containing type 2 fuel are compared in table 3. Homogenization reduces the reactivity. The extent of the reduction can be seen in table 4 where reactivities are given for the 14.70-in.-dia, 66-in.-tall water reflected cylinder containing 386 fuel rods that was estimated from the correlation of MGBS-TGAN with experiment to be critical. In the calculation of extrapolation distances from the end and lateral surfaces by TGAN and in the KENO calculation, both of which used HAMMER generated group constants, transport cross sections were required. These were generated for each of the four groups in the HAMMER few group scheme by equating the diffusion constant to the transport theory expression for isotropic scattering. For use in TGAN the 4-group cross sections were collapsed to two groups and adjusted to preserve the critical buckling and the asymptotic spectrum when the diffusion constant is set equal to $1/3 \Sigma_{\text{tr}}$ as it is in diffusion theory. Calculations involving the

Hansen-Roach cross sections were made by ANISN(S_4). Extrapolation distances were computed from the bucklings calculated to correspond to the values of k_{eff} computed for a 14.70-in. water-reflected cylinder of infinite height and for an infinite 66-in.-thick water-reflected slab. The values of k_{eff} in table 4 are those that should be considered critical for the particular method employed in the cask calculations.

The effectiveness of Boral sheets of various thicknesses clad in 0.020-in.-thick aluminum was investigated for infinite lattices of sheets alternating with 4.0-in.-thick regions of homogenized type 2 fuel at the 0.663-in.-square-pitch in water. ANISN calculations with Hansen-Roach cross sections are reported in table 5 and HAMMER calculations in table 6. The HAMMER calculations indicate a greater reduction in reactivity than the ANISN calculations. Although HAMMER employs 84 groups compared with only 16 for the Hansen-Roach cross sections and hence on this basis should be more accurate, there are a couple of features about HAMMER that may introduce error: (1) the source feeding the 30 thermal groups is assumed to be spatially flat and (2) the angular distribution of neutrons at all interfaces between mesh intervals in the epithermal groups is assumed to be proportional to the cosine of the angle made with the normal to the interface. To attempt to account for these shortcomings and to develop a set of 4-group cross sections for Boral suitable for use with the average parameters for the heterogeneous fuel cell, a search was made for cross sections that with the cosine current option of SLICE would duplicate the relative absorptions in 1/4-in. Boral sheet calculated by HAMMER for the three upper groups. It is apparent from table 6 that Boral of this thickness is essentially black to thermal neutrons and hence that as long as the thermal cross section is sufficiently large, its exact value is unimportant. Table 7 lists the cross sections that were adopted for the Boral core. Cross sections for the aluminum cladding were taken from the INCYCE [10] library. The relative Boral absorptions agree well with the HAMMER results except in the thermal group where the higher HAMMER result undoubtedly is due to the assumption of a flat source. These cross sections when used with the collision probability option in SLICE give a somewhat larger value of k_{eff} for the lattice of Boral sheets. Values of

$\Delta k/k$ calculated by HAMMER, by SLICE with the cosine current option, and by SLICE with the collision probability option are, respectively, -0.4271, -0.3672, and -0.3534. The ANISN result from table 5 is -0.3750.

The cross sections of table 7 together with average cell cross sections computed by HAMMER for the heterogeneous type 2 fuel cell and cross sections for Al, Fe, Cr, Ni, and H₂O taken from the INCYCE library were employed to calculate the reactivity of water reflected compartments loaded with type 2 fuel assemblies. The assemblies were placed against the arms of the Boral cross because TGAN calculations for a 2 × 2 array and SLICE cell calculations with and without a water gap between fuel and Boral indicate this to be the most reactive location. On the basis of 1-dimensional SLICE calculations k_{eff} for the water reflected compartments is 0.9196. KENO monte carlo calculations give a k_{eff} of 0.8980 ± 0.0043 when the thickness of water reflector outside the compartment walls is 10 cm, 0.9005 ± 0.0039 when it is 15 cm. Absorption in the compartment walls is important. The SLICE calculations indicate an increase in k_{eff} from 0.9196 to 0.9343 when the walls are omitted and KENO gives a value of 0.9179 ± 0.0050 when the steel walls are omitted and the thickness of water reflector is 15 cm. On the other hand, SLICE indicates that steel cladding for the Boral cross increases k_{eff} .

The insert will be reflected by the walls of the cask, but the 3.5 in. of water (on the average) separating the compartment walls from the outer shell of the insert reduces the effect of the lead walls. The calculations giving the effectiveness of the lead were made by KENO with Hansen-Roach cross sections. Cross sections for lead are not included in the original set, however. Two sets for lead were used: one supplied by G. E. Whitesides of ORNL with the KENO code and a set labeled Lazarus obtained from D. R. Smith of LASL. The former set in ANISN calculations overestimates the effect of lead on the reactivity of the HFIR core [11]. The basic KENO calculation for compartments loaded with type 2 fuel and reflected by 6-in. of water gave a k_{eff} of 0.8619 ± 0.0038 , which is subcritical by $0.9510 - 0.8619 = 0.0891$ compared with $0.9915 - 0.9005 = 0.0920$ in the HAMMER-KENO calculation. Enclosing the compartments by the cask walls and then by water

gave k_{eff} of 0.8591 ± 0.0059 with the ORNL and 0.8692 ± 0.0052 with the Lazarus cross section sets for lead. The cask walls at a separation of 3.5 in. from the compartment walls thus are not significantly different from water in their effect on reactivity. With no reliance on burnup, the cask loaded with type 2 fuel and filled and surrounded by water is subcritical by a margin of about 0.09 in. k_{eff} .

Table 1. Fuel Rod Characteristics

	Type 1	Type 2
Stainless steel clad OD, in.	0.500	0.500
Stainless steel clad ID, in.	.430	.430
Oxide OD, in.	0.430	0.430
Average density of oxide, g/cm ³	9.808	9.769
²³⁵ U enrichment, wt %	4.194	4.596
O, wt %	11.85 ^a	11.85 ^a
Active length, in.	66	66
Square lattice pitch, in.	0.663	0.663
Volume ratio of water to uranium metal with density 18.9 g/cm ³	3.66	3.68

^a Assumed.

Table 2. Reactivity of 2 x 2 Arrays in Water

Water Gap Between Assemblies, cm	k _{eff}	
	Type 1	Type 2
0	1.049	1.069
0.25	1.059	1.079
0.5	1.065	1.085
1.0	1.063	1.084
1.5	1.046	1.067
2.0	1.020	1.041

Table 3. Critical Parameters

Method	B ² , cm ⁻² x 10 ⁶	M ² , cm ²	k
MGBS (heterogeneous)	8554.8	36.34	1.3109
HAMMER (heterogeneous)	8617.1	36.50	1.3145
HAMMER (homogenized)	7549.6	35.46	1.2677
Ransen-Roach (homogenized)	6702.9	38.28	1.2566

Table 4. Reactivity of 14.70-in.-dia, 66-in.-tall Cylinder of Type 2 Rods

Method	k _{eff}
MGBS-TGAN	0.9872
HAMMER (heterogeneous) - TGAN	.9915
HAMMER (heterogeneous) - KENO	.9910 ± 0.0042
HAMMER (homogenized)	.9723
ANISN (homogenized)	0.9510

Table 5. Infinite Lattice of Type 2 Fuel Cells and Boral Sheets
(ANISN - S₁₆)

Boral Core Thickness, ^a in.	Number of Mesh Intervals	k _{eff}	Δk/k
0.085	7	0.9794	-0.2830
0.210	1	0.9170	-0.3703
	3	0.9137	-0.3753
	7	0.9139	-0.3750
	15	0.9140	-0.3748
0.335	7	0.8797	-0.428

^a Homogenized fuel region was 4-in. thick, with 32 mesh intervals; aluminum cladding was 0.02-in. thick, with 1 mesh interval.

Table 6. Infinite Lattice of Type 2 Fuel Cells and Boral Sheets
(HAMMER)

Boral Core Thickness, ^a in.	Number of Mesh Points	k _{eff}	Δk/k	Ratio of Boral ^b to Fuel Absorptions in Group			
				1	2	3	4
0.021	4	1.0799	-0.1739	0.00397	0.03275	0.1694	0.1598
.042	8	1.0197	-0.2432	.00649	.06463	.2335	.1934
.105	16	0.9415	-0.3465	.01416	.1600	.4231	.2159
.210	16	.8883	-0.4271	.02690	.3181	.5915	.2196
0.420	16	.8373	-0.5140	.05223	0.7635	.7675	.2201
1.05	16	.7686	-0.6494	.1262	1.5443	0.9834	.2205
2.10	16	0.7128	-0.7785	0.2398	2.9963	1.1064	0.2209

^a Homogenized fuel region was 4-in. thick with 22 mesh points; aluminum cladding was 0.02-in. thick with 1 mesh point.

^b Includes cladding.

Table 7. Four-Group Cross Sections for 1/4-in. Boral Sheet

Group	Σ	Σ _a	Σ _r	Boral/Fuel Absorptions	
				Cosine Current	Collision Probability
1	0.1531	0.0019	0.0202	0.0261	0.0252
2	.25876	.016	.00376	.3184	.3154
3	0.77138	0.56	0.00438	.5915	.5411
4	11.24	11.0		0.1923	0.1969

APPENDIX 2

Normal Conditions of Transport

A. Mechanical Analysis of Insert

Spent fuel from NS Savannah reactor is to be shipped in the Piqua-Elk River cask with a new insert with poison cross so that there will be no possibility of a criticality accident.

The mechanical analysis included as part of Appendix 3 demonstrates that the insert will perform its primary safety function; viz. maintain separation of the fuel in four compartments or quadrants. Deformation of the walls of the compartments is permissible; however, no fuel can be allowed to move from one compartment to another.

Figures 2 and 3 show the design of the insert. The Knapp Mills Final Design Report [1] and Thermal Test Report (COO-102802, May 29, 1964) describe the design, fabrication, and thermal testing of the Piqua-Elk River cask.

B. Package Standards

Inasmuch as the insert will be used inside the cask, the features covered by "Package Standards" are features of the cask itself. The weight of the insert, 1750 lb, plus the weight of the four fuel assemblies, 3000 lb, equals 4750 lb. This is about 1800 lb less than the weight of the Piqua reactor fuel and insert on which the cask design was based. Thus the loads on the lifting devices and holddown devices are slightly less than those currently authorized, and the factors of safety are slightly increased. Similarly, there is an increase in the factor of safety for loading the package as a simple beam. Resistance of the cask to external pressure is not affected by the change of cask contents proposed in this supplementary report.

C. Normal Conditions of Transport

1. Heat

The insert is an open vessel and vents into the cask cavity. The potential for pressure buildup in the cask cavity from heat generation is discussed in Appendix 3.

2. Cold

The materials of which the insert is constructed maintain reasonable ductility at -40°F .

3. Pressure

The insert, being open, is not subject to differential pressure.

4. Vibration

No damage from vibration in transit has been experienced with numerous other inserts of generally similar design.

5. Water Spray

Applicable to the cask rather than the insert.

6. Free Drop

In Appendix 3, the insert while in the cask is shown to be able to withstand a 30-ft drop. Damage from the 1-ft drop specified for normal transport would be negligible.

7. Carrier Drop

Not applicable.

8. Penetration

Applicable to the cask rather than to the insert.

9. Compression

Applicable to the cask rather than to the insert.

APPENDIX 3

Hypothetical Accident Conditions

The new cask insert is provided on the premise that a 30-ft fall of the Piqua-Elk River cask, and exposure to the hypothetical 30-minute fire, would impair the leaktightness of the cask but

- 1) would not permit fuel assemblies to form a critical array if the cask were flooded after the fall and fire
- 2) would not impair the integrity of the stainless cladding on NS Savannah fuel or permit escape of more than the allowable amount of radioactivity from the cask.

The overall satisfactory performance of this type of insert in a 30-ft drop of the cask is illustrated by tests of mockups for the Bismuth Cask (DPSPU 71-124-5) and Paducah Demonstration Cask (DPSPU 71-124-4A). The only structural feature of the insert for NS Savannah fuel considered to require further analysis is the ability to retain fuel assemblies within the compartments so that assemblies will remain isolated from one another by intervening poison. Stress calculations for this purpose are subsequently presented. As indicated in the thermal analysis (Appendix 4), the temperatures of those stressed members involved are below the temperature at which a significant reduction in strength occurs, so normal stress values for carbon steel apply. Also, temperatures of the steel-clad poison cross and Boral poison sandwiched within this cladding remain within acceptable limits.

1. Free Drop (30-ft drop in position of maximum damage)

A fall in which the axis of the cask is perpendicular to the impact surface produces no force component tending to collapse fuel assembly supports or to drive fuel assemblies from one compartment to another. A fall with the axis horizontal, however, produces maximum forces on the compartment boundaries; for this reason, horizontal impact is selected as the case for analysis.

In drop tests of the Paducah cask (KY-552), which appears to provide no more cushioning than the Piqua-Elk River cask, a deceleration of 200 g was measured on the dummy fuel load. Allowing a 25% margin for

conservatism, a decelerating force on the insert of 250 g is assumed for the following calculations.

The insert (1750 lb) and the fuel assemblies (3000 lb) weigh about 4750 lb.

a. Impact at S

Consider first a worst case where no insert members are stressed in tension and all impact loads are absorbed by compressive stresses in support members. This requires the attitude of fall in which a quadrant divider is vertical, as with impact at S as shown at right.

Assume that the entire impact force is absorbed by the five bottom support members acting in compression with load divided equally. Check the weakest (i.e. longest and thinnest) of these ("X") for the critical load for buckling. [12]

The slenderness ratio, or ratio of length to radius of gyration is

$$\frac{L}{k} = 4.5 \div \left(\frac{0.120}{\sqrt{12}} \right) = 130$$

Since this exceeds 110, Euler's column formula is applicable.

$$\text{Impact force per support} = \frac{4750 \text{ lb} \times 250}{5 \text{ supports}} = 238,000 \text{ lb}$$

L = height of support "X" = 4.5 in.

d = thickness = 0.120 in.

length = 70.5 in.

cross section area = $0.120 (70.5) = 8.45 \text{ sq in.}$

unit area loading = $\frac{238,000 \text{ lb}}{8.45 \text{ sq in.}} = 28,200 \text{ psi (compression)}$

unit length loading = $\frac{238,000 \text{ lb}}{70.5 \text{ in.}} = 3380 \text{ lb/in.}$

$$\text{Critical load for buckling} = P = \frac{4\pi EI}{L^2} \text{ lb/in.}$$

where: $I = (1/12)(bd^3) = (1/12)(1 \text{ in.} \times 0.120^3) = 1.44 \times 10^{-4} (\text{in.})^4$

$E = 30 \times 10^6 \text{ psi/(in./in.)}$ Young's modulus

$$P = \frac{4(3.14)^2 (30 \times 10^6)(1.44 \times 10^{-4})}{(4.5)^2} = 8450 \text{ lb/in.}$$

The actual loading of 3380 lb/in. load is less than the critical load for buckling, and compressive stress is within the elastic range; therefore, no damage to the insert occurs.

b. Impact at Intermediate Locations

The sketch shows that in addition to the five compression members, which are capable of carrying the entire impact load, the vertical extension of the center member ("X") and the two side members of the insert are available for carrying load when serving as tension members. These members are anchored to the outer shell of the insert with full penetration welds. It is evident that with insert impact at W or E, or at any intermediate point, the total impact load can be absorbed without deformation of the insert by combinations of compression in some insert support members and tension in others, and that the maximum stress in any one support member cannot exceed the previously calculated maximum stress of 28,200 psi, which is acceptable.

c. Effect of Impact on Fuel Cladding

Each fuel assembly consists of four subassemblies of 41 fuel rods each, a total of 164 rods. The outside diameter of the fuel rod is 0.5 in. (0.035-in. type 304 SS cladding). The length of the fuel rod assembly (figure 4) is 76.5 in. Total weight of each assembly is about 750 lb.

In each assembly, the fuel rods are fastened at the top and bottom by the upper and lower support frames (figure 4). These frames in turn are locked together by the central shaft support spine. Within the assembly, all fuel rods are maintained at 0.66 in.-square-pitch spacing by a series of main and peripheral spacing ferrules positioned at intermediate support zones within the assembly. The assembly thus constitutes a strong and durable framework which provides protection to the individual fuel rods grouped within the structure.

The magnitude of shock and extent of distortion which fuel pins of this type can withstand have been demonstrated by tests performed by Jersey Nuclear. [13]

The rods dropped individually onto an unyielding surface were 158 in. long, 144-in. fuel column of UO₂ pellets, 0.57 in. OD, 0.036 in. clad thickness, cladding of unannealed Zircaloy-2, unirradiated. Rupture of the unprotected fuel rods did not occur from end drop, 45° drop, or side drop from 30 ft.

The properties of the Zircaloy-2 cladding used on the drop specimens and of the type 304 SS cladding on NS Savannah fuel are similar, as shown in the following table. The low radiation exposure of the NS Savannah fuel does not adversely affect the comparison.

	Properties at 70°F Zircaloy-2 [14]	Type 304 SS [15]
Tensile, psi	76,000	73-85,000
Yield, psi	49,000	35-45,000
Reduction in area, %	28	45-60

Fuel pins restrained in an assembly would suffer less distortion but would be subjected to other stresses. Those of particular concern for the NS Savannah fuel pins are considered to be (1) in a side drop, the shear and tensile stress in the cladding for a section acting as a fixed-end beam between two spacing ferrules, and (2) in an end drop, the tensile stress in the cladding due to the column of fuel pellets impacting on the end fitting. Drops at other angles would have intermediate effects.

Casks of the type under consideration, with fins and other appurtenances to absorb energy, have been found to decelerate at a maximum rate of about 200 g from a 30-ft drop. This is taken also to be the deceleration of the contents, although some attenuation by the slight distortion of the insert would be expected. Calculated stresses (shown below) even if multiplied by two because of the shock nature of the loading, are less than the tensile strength of ~50,000 psi for the type 304 stainless cladding at the maximum fuel temperature.

In view of the various foregoing considerations, it is concluded that the fuel pins have adequate strength and ductility to resist failure from a 30-ft drop of the assembly.

1) Side Drop

The ferrules are pieces of tubing about 1 in. long, between and brazed to the fuel rods so as to maintain the 0.500-in.-dia rods on a 0.663-in. rectangular pitch. The maximum spacing is about 8 in. The fuel rod is treated as a fixed-end beam between ferrules:

$$\begin{aligned}\text{Rod OD} &= 0.500 \text{ in.} \\ \text{Wall thickness} &= 0.035 \text{ in.} \\ \text{Moment of inertia} &= 0.00139 \text{ in}^4 \\ \text{Clad cross-sectional area} &= 0.051 \text{ in}^2 \\ \text{Fuel wt per in. of length} &= 0.051 \text{ lb} \\ \text{Clad wt per in. of length} &= 0.015 \text{ lb}\end{aligned}$$

For 8-in.-long fixed-end beam loaded uniformly at 200 (0.051 + 0.015) lb/in., with other dimensions as given above, the calculated stresses and deflection are:

$$\begin{aligned}\text{Shear stress at supports} &= 2100 \text{ psi} \\ \text{Max tensile stress (at supports)} &= 12,700 \text{ psi} \\ \text{Deflection at midpoint} &= 0.0034 \text{ in.}\end{aligned}$$

2) Side Drop

The weight of the fuel column is 3.37 lb. Tensile stress in cladding at 200 g internal force against end fitting:

$$\frac{200 \times 3.37}{0.051} = 13,200 \text{ psi}$$

2. Puncture (40-in. drop onto a 6-in.-dia bar)

This is a function of the cask and not the insert. [1]

3. Thermal (30-minute exposure to radiant heat source at 1475°F)

The temperatures of the various elements of the insert structure and of the hottest fuel have been calculated in the thermal analysis for conditions prevailing during the hypothetical fire (Appendix 4).

a. Effect on Cask

It is expected that the fire will impair the leaktightness of the cask but have no serious effect on cask structure or shielding ability. [1]

If the leaktightness of the dry cask were not impaired by the fire, the pressure rise in the cask as a result of thermal expansion of the contained air would not exceed 5 psig:

Average temperature of cask contents

During normal transport = 480°F = 940°R

During the fire = 770°F = 1230°R

Pressure in cask

During normal transport = 0 psig = 14.7 psia

During the fire = 14.7 psia $\left(\frac{1230^{\circ}\text{R}}{940^{\circ}\text{R}} \right) = 19.2 \text{ psia}$

Maximum possible pressure = 19.2 - 14.7 = 4.5 psi

This pressure, which is below the design pressure of 100 psig and below the rupture disk relief setting of 75 psig, is acceptable.

b. Effect on Insert

- 1) As indicated in the thermal analysis (Appendix 4), the maximum temperature on the steel cross plates that serve as support members for the fuel assemblies amounts to only 400°F during normal transport when stresses might be maximum as a result of a 30-ft drop of the cask. At this temperature, there is no significant loss of strength of carbon steel (and only a 10% loss of strength at 750°F, Table UCS-23, Sect VIII, ASME Boiler Code).

- 2) The steel-sheathed Boral cross that serves as nuclear poison and as divider for the four fuel assemblies supported by the insert structure can attain a temperature in its central zone approaching that of the hottest fuel element, which during a fire has been calculated conservatively to be less than 1070°F. At this temperature, there is no interaction between Boral and the steel sheathing, and no melting of the aluminum matrix of the Boral plate (mp of aluminum is 1220°F); hence, no damage to or loss of nuclear poison can occur.

c. Effect on Fuel

The temperature attained by the hottest fuel element carried by the insert is shown in the thermal analysis (Appendix 4) to be less than 1030°F during normal transport and less than 1070°F during the hypothetical fire. At these elevated temperatures, the helium sealed in the fuel as thermal bonding agent together with fission gases released from the fuel undergo thermal expansion with resultant increase in pressure within the fuel element, and increase in stress on the 0.035-in.-thick stainless steel cladding. As shown below, this increase in pressure and resultant stress on cladding is within the strength capability of the cladding and will not result in failure or release of fission products to cask or environment:

Maximum reactor heat release (NS Savannah) = 74,000 KW

Active fuel length in reactor = 26,200 ft

32 assemblies (164 pins/assembly)(5 ft/pin)

Heat release per foot, $74,000/26,200 = 2.8 \text{ KW/ft}$

% of fission gases released = <0.5 [16]

Fission gas generated = 1.45 g-atoms/1000 MWD [17]

Uranium irradiated $(1.360 \text{ kg } ^{235}\text{U/pin})(32)(164) = 7150 \text{ kg } ^{235}\text{U}$
 $= 7.15 \text{ MTU}$

Maximum irradiation (NS Savannah) = 9250 MWD/MTU

Gas generated = $(1.45 \text{ g-atoms/1000 MWD})(9250 \text{ MWD/MTU}) = 13.4 \text{ g-atoms/MTU}$

$$= \frac{13.4 \text{ g-atoms/MTU}}{\text{fuel pin}/1.36 \times 10^{-3} \text{ MTU}} = 0.018 \text{ g-atoms/fuel pin}$$

$$= 0.018 \text{ g-atoms} \times \frac{22,400 \text{ cm}^3}{\text{g-atoms}} = 404 \text{ cm}^3$$

$$\text{Gas released} - \frac{0.5\% (404 \text{ cm}^3)}{100} = 2.02 \text{ cm}^3 \times \frac{530^\circ\text{R}}{492^\circ\text{R}} = 2.18 \text{ cm}^3 @ 70^\circ\text{F}$$

Initial free gas space in fuel pin, which is filled with helium at 1 atm abs, $70^\circ\text{F} = 8 \text{ cm}^3$

Total gas in fuel pin at 1 atm abs ($\sim 70^\circ\text{F}$, 530°R) = 10.2 cm^3

Minimum gas space after allowance for fuel growth (based on information from Todd Shipyards) = 0.77 cm^3

Temperature of fuel pin during fire = 1070°F , 1530°R

$$\begin{aligned} \text{Pressure in fuel pin} &= 1 \text{ atm abs} \frac{(10.2 \text{ cm}^3)(1530^\circ\text{R})}{(0.77 \text{ cm}^3)(530^\circ\text{R})} = 38.2 \text{ atm/abs} \\ &= 561 \text{ psia} = 546 \text{ psig} \end{aligned}$$

The hoop stress in the fuel pin cladding

$$S = \frac{P \times 0.5 \text{ in.}}{2 \times 0.035} = \frac{546 (0.5)}{2 \times 0.035} = 3900 \text{ psi}$$

The allowable stress at 1200°F for rupture in 1000 hours is 17,000 psi, which is about 4.3 times the actual stress. Under these conditions there should be no failure of fuel cladding. [18]

4. Water Immersion (8 hours at depth of at least 3 ft)

It is assumed that one consequence of the 30-ft drop or fire may be impairment of the cask seal and introduction of water into the cask. As indicated in the criticality analysis (Appendix 1), the contents of the cask remain subcritical when flooded.

With respect to water coming in contact with the hot fuel elements, no mechanism has been found that would permit entry of water in such a way as to generate high steam pressure or to damage the fuel by thermal shock. Any increase in pressure in the cask above the rupture disk relief setting of 75 psig would blow this disk. The internal design pressure is 100 psig, and structural integrity would be maintained at much higher pressures.

The external surfaces of fuel assemblies after draining the cask may be contaminated, primarily with Group III and Group IV radionuclides for which the escape of 10 curies from the cask is permitted. Experience with other shipments of relatively fresh fuel has shown that the

expected release to water corresponds to activity of less than 10^{-6} Ci/ml in a water-filled cask. The volume of the Piqua-Elk River cask is about 10^6 cm³. If this entire volume contained 10^{-6} Ci/ml, the total would be 1 curie. In view of the long underwater storage of the NS Savannah fuel, the expected that actual release to air or water in the accident would be much less than 1 curie.

APPENDIX 4

Thermal Analysis

FOREWORD

The thermal analysis for the dry shipment of NS Savannah fuel elements from Galveston, Texas, to the Savannah River Plant in the Piqua-Elk River cask was prepared to determine a conservatively-high maximum temperature that might be reached by the hottest (innermost) fuel pin in the closely spaced fuel pin array that comprises the NS Savannah fuel assembly, both during normal transport and following the hypothetical fire. The calculated temperatures are on the high side because the heat release used for the analysis is 2000 Btu/hr per assembly, or a total of 8000 Btu/hr for the loaded cask, whereas the actual heat release as of May 1, 1972, is now calculated as 1250 Btu/hr per assembly or 5000 Btu/hr per cask, which is only 63% of the heat release assumed. A small offset to this conservative heat release is the use of a symmetrical 169 fuel pin array per assembly rather than the actual 164 pin array, which results in a heat release per pin that is about 3% low; another slight offset results from the use of an 11/16-in. fuel-pin pitch (0.686-in.) instead of the 0.663-in. pitch used in the actual fuel assembly. The effect of nonuniform reactor axial flux distribution and heat release has been offset in the calculations by the assumption that all of the decay heat released from fuel is released over a central 60-in.-long section of the 66.25-in. active length.

The calculated maximum temperature of the fuel is likewise on the high side because the calculations take no account of convection within the cask or of conductive heat flow out from the center of the insert. Thus the maximum temperature of the poison cross, which contains Boral, is less than the melting point of aluminum, 1220°F; the margin of safety exceeds 150°F and probably exceeds 250°F.

As indicated in the structural analysis (Appendix 3), the calculated maximum temperature of 655°F for the outer shell of the insert and the associated structure that supports the fuel assemblies (item 4 of table 9 that follows) is well within the temperature range where no significant decrease in the structural strength of carbon steel occurs.

SUMMARY

A thermal analysis has been made covering the dry shipment of the NS Savannah fuel elements from Galveston, Texas, to the SRP in the Piqua-Elk River cask. The analysis includes normal transport conditions followed by exposure to a 1475°F fire for 30 minutes.

It was assumed that during normal transport the cask would be exposed to sun load and ambient still air at 130°F. Under these conditions, a maximum fuel element temperature of 1032°F can be reached.

Calculations indicate that a maximum fuel element temperature of about 1070°F will be reached 12 hours after the fire. Some of the lead shield will melt but this was not taken into account. This is considered a conservative approach. Maximum temperatures of the various cask components are summarized in table 9.

OVERALL CALCULATION PROCEDURE

- 1) Steady-state temperatures of cask components 1-4 (exhibit A) were established for normal shipment in 130°F ambient still air with sun loading. Table 8 summarizes the temperatures. Detailed calculations are presented starting on page 32.
- 2) Steady-state heat transfer from fuel element array was calculated by using a computer program (see exhibits B and C). Convection heat transfer between elements was neglected.
- 3) This steady-state temperature profile was taken as the initial condition for the fire exposure. See exhibit C for first 90 minutes after fire.

It was assumed that during the fire

Fire temperature	= 1475°F
Surface emissivity	= 0.9
Fire duration	= 30 minutes
Outside convection coefficient	= 3 Btu/(hr)(°F)(ft ²)
(based on unfinned area)	

Melting of the lead was not taken into account. This is conservative.

- 4) It was assumed that following the fire

Ambient temp drops to 100°F

Surface emissivity = 0.6

Outside convection coefficient = 0 Btu/(hr)(°F)(ft²)
(assumes cask on side)

Melting and solidification of
the lead was not considered.

- 5) Maximum temperatures were calculated and summarized in table 9.
- 6) Physical properties and cask dimensions are summarized in table 10.

EXHIBIT A

VERTICAL HEIGHT OVER WHICH HEAT IS DISSIPATED = 5 FT

CASK COMPONENT

1. Outer Steel Shell
2. Inner Stainless Steel
3. Steel Liner
4. Steel Cross Plates

EXHIBIT B

View Factors for Radiation

Element FA1 = 0.09
to
Element FA2 = 0.092
Element
to FA3 = 0.2
Plate

4
Steel Cross
Plates

Fuel Element

Assembly (Four per Cask)
13 x 13 elements = 169 total per assembly
Heat release per assembly = 2000 Btu/hr
Heat release per cask = 8000 Btu/hr
Emissivity element to element = 0.5
Emissivity element to plate = 0.5

Table 8. Steady State Temperature Profile

Basis: Normal shipping with sun loading in still air
 Ambient air at 130°F
 Cask heat release from fuel = 8000 Btu/hr

Cask Component (Exhibit A)	Steady-State Temp, °F
1. Outer steel shell	198
2. Inner stainless steel shell	205
3. Steel liner	277
4. Steel cross plates	397
5. Central most fuel element	1032
6. Average of items 2, 3, 4, & 5	480

Table 9. Maximum Temperatures Resulting from Fire Exposure

Cask Component (Exhibit A)	Maximum Temp, °F	Max Temp Occurs, hours after start of fire
1. Outer steel shell	897	0.5
2. Inner stainless steel shell	715	0.84
3. Steel liner	645	1.50
4. Steel cross plates	655	2.67
5. Central most fuel element	1070	11.7
6. Average of items 2, 3, 4, & 5	771	

Table 10. Physical Properties of Cask Materials and Components

	Steel	Fuel	Lead	SS
Density, lb/ft ³	490	-	687	490
Specific heat, Btu/(lb)(°F)	0.11	0.08	0.033	0.11
Thermal conductivity, Btu/(hr)(°F)(ft ² /ft)	25	∞	18.6 solid	9
Cask Component	Dia, in.	Mass, lb	Area, ft ²	
1. Outer steel shell	48	3500	60	
2. Inner SS shell	31	400	40	
3. Steel liner	31	400	40	
4. Steel cross plates	20	692	34	
5. Fuel element	0.5	3.25 ea.	-	

DETAILED CALCULATIONS

Normal Transport — Steady-State Temperatures

Basis: Fuel heat release per cask = 8000 Btu/hr

Ambient conditions: 130°F still air

Cask vertical on uncovered truck

1. Sun Loading

Total outside cask surface area including fins exposed to sun rays =
7.35 ft² per foot of height

$$\text{Sun flux} = q_s/A = 225 \text{ Btu}/(\text{hr})(\text{ft}^2)$$

East-facing vertical surface at 7-9 a.m. or west-facing
vertical surface in the afternoon. [19]

Solar heat input to surface per foot = (225)(7.35) = 1660 Btu/(hr)(ft)
Use 1700 Btu/(hr)(ft).

For 5 ft heat dissipating height

$$q_{\text{solar}} = (1700)(5) = 8500 \text{ Btu/hr}$$

$$q_{\text{fuel}} = 8000 \text{ Btu/hr}$$

Total heat to dissipate from surface = 16,500 Btu/hr

2. Temperature Drop Cask Surface to Ambient

Fin efficiency

Assume $h_c + h_r = 2$

$$\text{Fin factor} = X_f \sqrt{\frac{(h_c + h_r)}{k Y_o}}$$

X_f = fin height = 3 in. = 0.25 ft

Y_o = half thickness fin = 0.125 in. = 0.0104 ft

k = fin thermal conductivity = 25 Btu/(hr)(°F/ft)(ft²)

Fin factor = 0.693

Fin efficiency = 0.86 [20]

Area for convective heat transfer = A_c

$$A_c = (\pi D_o - N_f Y_o) L + \eta N_f (2 \ell L)$$

$$D_o = \text{dia outer surface} = 4 \text{ ft}$$

$$N_f = \text{number of fins} = 76$$

$$Y_o = \text{fin thickness} = \frac{0.25}{12} = 0.0208 \text{ ft}$$

$$L = \text{height} = 5 \text{ ft}$$

$$\eta = \text{fin efficiency} = 0.86$$

$$\ell = \text{fin height} = 0.25 \text{ ft}$$

$$A_c = 218 \text{ ft}^2$$

Area for radiation transfer = A_r

$$A_r = \pi (D_o + 2\ell) L$$

$$A_r = (3.14)(4 + 2 \times 0.25)(5) = 70.6 \text{ ft}^2$$

$$q_{\text{convection}} = h_c A_c (T_{\text{surface}} - T_a)$$

$$q_{\text{radiation}} = h_r A_r (T_{\text{surface}} - T_a)$$

$$q_c + q_r = 16,500 \text{ Btu/hr}$$

Assume $T_{\text{surface}} - T_{\text{ambient}} = 68^\circ\text{F} = \Delta T$

$$h_c = 0.19 (\Delta T)^{1/3} = 0.19 (68)^{1/3} = 0.776 \text{ Btu/(hr)}(^{\circ}\text{F})(\text{ft}^2)$$

$$h_r = 1 \text{ Btu/(hr)}(^{\circ}\text{F})(\text{ft}^2) \text{ for } \epsilon = 0.6 \quad [19]$$

$$q_c = \frac{h_c A_c \Delta T}{(0.776)(220)(68)} = 11,600 \text{ Btu/hr}$$

$$q_r = \frac{h_r A_r \Delta T}{(1)(70.6)(68)} = 4,800 \text{ Btu/hr}$$

$$q_c + q_r = 16,400 \text{ Btu/hr} \quad \sim 16,500 \text{ Btu/hr}$$

$$\text{Use } \Delta T = 68^\circ\text{F}$$

$$T_{\text{surface}} = 130^\circ + 68^\circ = 198^\circ\text{F}$$

3. Temperature Drop Across Outer Steel Shell

Thickness = 1 in.

$$h = \frac{k}{X} = \frac{25}{(1/12)} = 300 \text{ Btu/(hr)(}^\circ\text{F)(ft}^2\text{)}$$

$$q = q_{\text{fuel}} = 8000 \text{ Btu/hr}$$

$$A = \pi DL = (3.14)(4)(5) = 62.8 \text{ ft}^2$$

$$\Delta T = \frac{q}{hA} = \frac{8000}{(300)(62.8)} = 0.425^\circ\text{F, say } 0.5^\circ\text{F}$$

$$\text{Temp inside steel shell} = 198.5^\circ\text{F}$$

4. Temperature Drop Across Lead

Assume lead well bonded to wall

$$q_{\text{fuel}} = \frac{k_m 2\pi L \Delta T}{\text{hr} (D_2/D_1)}$$

$$D_2 = 46.5 \text{ in.}$$

$$D_1 = 31 \text{ in.}$$

$$\Delta T = \frac{q_f \text{ hr} (D_2/D_1)}{2\pi L k_m}$$

$$k_m = 18.6 \frac{\text{Btu}}{\text{hr} \frac{^\circ\text{F}}{\text{ft}}} \text{ Btu/(hr)(}^\circ\text{F/ft)(ft}^2\text{)}$$

$$L = 5 \text{ ft}$$

$$\Delta T = 5.57^\circ\text{F}$$

$$\text{Temp inside of lead} = 198.5 + 5.6 = 204.1^\circ\text{F}$$

5. Temp Drop Across Inner Stainless Steel Shell

Thickness = 0.5 in. stainless steel

$$h = \frac{k}{X} = \frac{9}{(0.5/12)} = 216 \text{ Btu/(hr)(}^\circ\text{F)(ft}^2\text{)}$$

$$\Delta T = q/hA = \frac{8000}{216 \times 40} = 0.925^\circ\text{F, say } 1^\circ\text{F}$$

$$A = 3.14 \left(\frac{30.5}{12} \right) (5 \text{ ft}) = 40 \text{ ft}^2$$

$$\text{Temp of inner SS shell} = 205.1^\circ\text{F}$$

6. Temp Drop Across Air Gap (SS Shell to Liner)

Air gap = 1/8 in.

Heat transfer by conduction and radiation

Conduction

$$h_c = \frac{k_{\text{air}}}{X} = \frac{0.019}{\left(\frac{0.125}{12}\right)} = 1.82 \text{ Btu/(hr)}(^{\circ}\text{F})(\text{ft}^2)$$

Radiation

$$h_r = \frac{0.173 F_a F_e \left[\left(\frac{T_h + 460}{100} \right)^4 - \left(\frac{T_c + 460}{100} \right)^4 \right]}{(T_h - T_c)}$$

Assume $\Delta T = 70^{\circ}\text{F}$

$$T_h = 205 + 70 = 275^{\circ}\text{F}$$

$$T_c = 205^{\circ}\text{F}$$

$$F_a = 1$$

$$F_e = \frac{1}{\frac{1}{\epsilon_1} + \frac{1}{\epsilon_2} - 1} = 0.43$$

$$\epsilon_1 = \epsilon_2 = 0.6$$

$$h_r = 1.02 \text{ Btu/(hr)}(^{\circ}\text{F})(\text{ft}^2)$$

$$h_c + h_r = 1.82 + 1.02 = 2.84 \text{ Btu/(hr)}(^{\circ}\text{F})(\text{ft}^2)$$

$$A = 3.14 \left(\frac{30}{12} \right) (5 \text{ ft}) = 39.2 \text{ ft}^2$$

$$q = 8000 \text{ Btu/hr}$$

$$\Delta T = \frac{q}{(h_c + h_r)A} = \frac{8000}{(2.84)(39.2)} = 71.8^{\circ}\text{F} \quad \begin{array}{l} \text{(checks assumed} \\ \Delta T = 70^{\circ}\text{F)} \end{array}$$

Use $\Delta T = 72^{\circ}\text{F}$

$$\text{Temp of liner} = 205 + 72 = 277^{\circ}\text{F}$$

7. Temperature Drop Liner to Cross Plates

For simplicity, assume $A_o = A_i = 36 \text{ ft}^2$

Assume cross plate temp = 397°F

$$h_{ci} = h_{co} = 0.19 (\Delta T)^{1/3}$$

Convection

$$\Delta T = (400 - T_A) = T_A - 277 = \frac{397 - 277}{2} = 60^\circ\text{F}$$

$$h_{ci} = h_{co} = (0.19)(60)^{1/3} = 0.744 \text{ Btu}/(\text{hr})(^\circ\text{F})(\text{ft}^2)$$

$$\frac{1}{U_c} = \frac{1}{h_{ci}} + \frac{1}{h_{co}}$$

$$U_c = 0.37 \text{ Btu}/(\text{hr})(^\circ\text{F})(\text{ft}^2)$$

Radiation

$$h_r = \frac{0.173 F_a F_e \left[\left(\frac{T_i + 460}{100} \right)^4 - \left(\frac{T_o + 460}{100} \right)^4 \right]}{(T_i - T_o)}$$

$$h_r = 1.52 \text{ Btu}/(\text{hr})(^\circ\text{F})(\text{ft}^2)$$

$$F_a = 1$$

$$F_e = 0.43$$

$$(U_c + h_r) = 0.37 + 1.52 = 1.89 \text{ Btu}/(\text{hr})(^\circ\text{F})(\text{ft}^2)$$

$$q = (U_c + h_r)(A)(\Delta T)$$

$$q = (1.89)(36)(120) = 8160 \text{ Btu/hr} \quad (\text{Checks } q_{\text{fuel}}.)$$

Temp of cross plates = 397°F

EXHIBIT C

Computer Calculations

The following tables, through page 41, are the computer calculations for the thermal analysis of the cask.

FOR CASK
UNIVAC 1108 FORTRAN V LEVEL 2206 0023
THIS COMPILATION WAS DONE ON 04 MAY 72 AT 10:22:28

MAIN PROGRAM

STORAGE USED (BLOCK, NAME, LENGTH)

0001	*CODE	001473
0000	*DATA	002130
0002	*BLANK	000000

EXTERNAL REFERENCES (BLOCK, NAME)

0003	NRDUS
0004	NI02S
0005	NRDUS
0006	SQRT
0007	COS
0010	NI01S
0011	NEXP6S
0012	NSTOPS

STORAGE ASSIGNMENT FOR VARIABLES (BLOCK, TYPE, RELATIVE LOCATION, NAME)

0001	000163	160G	0001	000170	164G	0001	000317	200L	0001	001366	210L	0000	002010	212F
0000	002026	220F	0001	000235	220G	0001	001325	230L	0001	000251	232G	0001	000256	236G
0001	000304	253G	0001	000447	264G	0001	000450	267G	0000	001775	300F	0000	001777	302F
0001	000516	303G	0000	002000	304F	0000	002006	306F	0001	000572	320G	0001	000612	324G
0001	000720	346G	0001	001024	370G	0001	001230	435G	0001	001340	464G	0001	001341	467G
0001	001352	476G	0001	001411	516G	0001	001416	522G	0001	001444	540G	0000	R 001726	A
0000	R 001642	BT	0000	R 001712	BTI	0000	R 001661	BTI	0000	R 001706	CP	0000	R 001700	D
0000	R 001721	DELT	0000	R 001752	DT	0000	R 001735	F	0000	R 001727	FA1	0000	R 001731	FA2
0000	R 001703	FA3	0000	R 001734	FD	0000	R 001705	FE	0000	R 001715	FEO	0000	R 001733	FP
0000	R 001716	HCO	0000	R 001753	HCP	0000	R 001762	HRL	0000	R 001772	HRO	0000	R 001756	HRP
0000	I 001736	I	0000	I 001737	J	0000	I 001751	K	0000	I 001724	KK	0000	I 001770	KKK
0000	I 001740	L	0000	I 001723	LIMT	0000	I 001746	M	0000	I 001722	N	0000	I 001741	NN
0000	R 001702	PT	0000	R 001243	Q	0000	R 001750	QD	0000	R 001717	GF	0000	R 001604	QIN
0000	R 001771	QINO	0000	R 001763	QLIN	0000	R 001745	QNET	0000	R 001760	QNETL	0000	R 001623	QOUT
0000	R 001773	QOUTO	0000	R 001747	QP	0000	R 001742	QTOL	0000	R 001757	QTPTL	0000	R 001743	QI
0000	R 001744	QZ	0000	R 000000	T	0000	R 001755	TC	0000	R 001714	TF	0000	R 001754	TH
0000	R 001725	TIME	0000	R 001720	TIME	0000	R 001711	TTNS	0000	R 001710	TL	0000	R 001713	TO
0000	P 001704	TP	0000	R 000702	TR	0000	R 000341	TT	0000	R 001767	TTINS	0000	R 001764	TTL
0000	R 001774	TYO	0000	R 001761	TTY	0000	R 001765	XKK	0000	R 001701	XL	0000	R 001707	XMAS
0000	R 001732	XN	0000	R 001766	XT	0000	R 001730	XX						

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00101 1* DIMENSION T(15,15),TT(15,15),TR(15,15),Q(15,15)
00103 2* DIMENSION QIN(15),QOUT(15),BT(15),BTT(15)
00104 3* READ(5,300) D, XL, PT, FA3, TP, FE, CP, XMAS
00116 4* READ(5,300) TL, TINS, BT1, TO, TF, FEO, HCO
00127 5* READ(5,300) QF, TIML, DELT
00134 6* READ(5,302) N, LMT, KK
00141 7* 300 FORMAT(8F10.2)
00142 8* 302 FORMAT(3I10)
00143 9* WRITE(6,304)
00145 10* 304 FORMAT(11I1,15X,'ELK RIVER CASK',//)
00146 11* TIME = 0.
00147 12* A = 3.14*(D/12.)*XL/12.
00150 13* FA1 = .09
00151 14* XX = 1.+(PT*SQRT(2.)*D)/D)
00152 15* FA2 = (2./3.14)*(SQRT(XX**2.-1.))-XX*3.14/2.-1./COS(1./XX)
00153 16* XH=N
00154 17* FP=FA1*FE*.173*A
00155 18* FD=FA2*FE*.173*A
00156 19* F=FA3*FE*.173*A
00157 20* DO 360 I = 1,N
00162 21* READ(5,300)(T(I,J),J=1,N)
00170 22* 360 CONTINUE
00172 23* L=1
00173 24* WRITE(6,306)D,XL,PT,FA3,TP,FE,CP,XMAS
00205 25* WRITE(6,306)QF,TIML,DELT,FA1,FA2,FEO,HCO
00216 26* 306 FORMAT(1H0,8F12.3)
00217 27* DO 350 NN = 1,KK
00222 28* BT(NN) = BT1
00223 29* 350 CONTINUE
00225 30* WRITE(6,212)TIME,QTOL
00231 31* DO 308 I=1,N
00234 32* WRITE(6,220)(T(I,J),J=1,N)
00242 33* 308 CONTINUE
00244 34* WRITE(6,220)TP, TL, TINS
00251 35* WRITE(6,220)(BT(NN),NN =1,KK)
00257 36* WRITE(6,220) TO, TF
00263 37* 200 DO 14 J=1,N
00266 38* DO 14 I=1,N
00271 39* TR(I,J)=(T(I,J)+460.)/100.)*.4.
00272 40* 14 CONTINUE
00275 41* Q1=2.*FP*(TR(I,1)-TR(2,1))
00276 42* Q2=FD*(TR(I,1)-TR(2,2))
00277 43* QNET=(Q1+Q2-QF)*DELT/60.
00300 44* TT(I,1)=T(I,1)-QNET/(XMAS*CP)
00301 45* M=M+1
00302 46* DO 16 I=2,M
00305 47* J=1
00306 48* QP=3.*FP*(3.*TR(I,J)-TR(I-1,J)-TR(I,J+1)-TR(I+1,J))/3.
00307 49* QD = 4.*FD*(4.*TR(I,J)-TR(I-1,J)-TR(I+1,J)-TR(I,J+1)-TR(I+1,J)
00307 50* ) /4.
00310 51* QNET=(QP+QD-QF)*DELT/60.
00311 52* TT(I,J)=T(I,J)-QNET/(XMAS*CP)
00312 53* TT(J,1)=TT(I,J)
00313 54* 16 CONTINUE
00315 55* K=1
00316 56* M=N-1
00317 57* DO 18 J=2,M
00322 58* K=K+1
00323 59* DO 18 I=K,4
00326 60* QP=4.*FP*(4.*T(I,J)-T(I-1,J)-TR(I,J+1)-TR(I+1,J)-TR(I,J-1))/4.
00327 61* QD=4.*FD*(4.*TR(I,J)-TR(I-1,J+1)-TR(I+1,J+1)-TR(I+1,J-1)-
00327 62* TR(I-1,J-1))/4.
00330 63* QNET=(QP-QF+QD)*DELT/60.
00331 64* TT(I,J)=T(I,J)-QNET/(XMAS*CP)
00332 65* TT(J,1)=TT(I,J)
00333 66* 18 CONTINUE
00336 67* QP=2.*FP*(2.*TR(N,1)-TR(N-1,1)-TR(N,2)) /2.
00337 68* QD = 2.*FD*(2.*TR(N,1)-TR(N-1,2)-TR(N-1,1)) /2.
00340 69* Q(N,1)=F*(TR(N,1)-((TP+460.)/100.)*.4.)*.7
00341 70* QNET=(QP-QF+QD+Q(N,1))*DELT/60.
00342 71* TT(N,1)=T(N,1)-QNET/(XMAS*CP)
00343 72* TT(1,N)=TT(N,1)
00344 73* M=N-1
00345 74* DO 20 J=2,M
00350 75* QP=3.*FP*(3.*TR(N,J)-TR(N-1,J)-TR(N,J+1)-TR(N,J-1)) /3.
00351 76* QD=2.*FD*(2.*TR(N,J)-TR(N-1,J+1)-TR(N-1,J-1)) /2.
00352 77* Q(N,J)=F*(TR(N,J)-((TP+460.)/100.)*.4.)*.7
00353 78* QNET=(QP-QF+QD+Q(N,J))*DELT/60.
00354 79* TT(N,J)=T(N,J)-QNET/(XMAS*CP)
00355 80* TT(J,N)=TT(N,J)
00356 81* 20 CONTINUE
00360 82* QTOL=D.
00361 83* QP=2.*FP*(2.*TR(N,N)-TR(N-1,N)-TR(N,N-1)) /2.
00362 84* QD=FD*(TR(N,N)-TR(N-1,N-1))
00363 85* Q(N,N)=2.*F*(TR(N,N)-((TP+460.)/100.)*.4.)*.7
00364 86* QNET=(QP+QD+Q(N,N)-QF)*DELT/60.
00365 87* TT(N,N)=T(N,N)-QNET/(XMAS*CP)
00366 88* M=N-1
00367 89* DO 22 J=1,M
00372 90* QTOL=QTOL+2.*Q(N,J)
00373 91* 22 CONTINUE
00375 92* QTOL=QTOL+Q(N,N)
00376 93* TIME = TIME + DELT
00377 94* IF(TIME.GT.30.) HCO = 0.
00401 95* IF(TIME.GT. 30.) TF = 100.
00403 96* IF(TIME.GT. 30.) FEO = .6.
00405 97* DT = ABS(TP - TL)

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00406 98* HCP = .5* .19 * DT**333
00407 99* TH = ((TP+460.)/100.)*.4.
00410 100* TC = ((TL+460.)/100.)*.4.
00411 101* HRP = (.173 * .43 * (TH - TC))/(TP - TL)
00412 102* QTPIL = (HCP + HRP) * (TP - TL) * 8.5
00413 103* QNETL = (QTOL - QTPIL) * DELT / 60.
00414 104* TTP = TP + QNETL / (.173 * .43)
00415 105* TH = TC
00416 106* TC = ((TINS + 460.)/100.)*.4.
00417 107* HRL = ((.173*.43*(TH - TC)) / (TL - TINS)
00420 108* CLIN = (.182 + HRL) * 10. * (TL - TINS)
00421 109* QNET = (QTPIL - CLIN) * DELT / 60.
00422 110* TTL = TL + QNET / (100. * .43)
00423 111* XKK = KK
00424 112* XT = 7.75 / (.12 * XKK)
00425 113* QIN(1) = ((18.6 * 2.) / XT) * 12.6 * (TINS - BT(1))
00426 114* QNET = (CLIN - QIN(1)) * DELT / 60.
00427 115* TTINS = TINS + QNET / (100. * .43)
00430 116* QOUT(1) = (18.6 / XT) * 12.6 * (BT(1) - BT(2))
00431 117* QNET = (QIN(1) - QOUT(1)) * DELT / 60.
00432 118* BT(1) = BT(1) + QNET / (12.6 * XT * 687. * .033)
00433 119* KKK = KK - 1
00434 120* DO 26 NN = 2, KKK
00437 121* QIN(NN) = QOUT(NN-1)
00440 122* QOUT(NN) = (18.6 / XT) * 12.6 * (BT(NN) - BT(NN+1))
00441 123* QNET = (QIN(NN) - QOUT(NN)) * DELT / 60.
00442 124* BT(NN) = BT(NN) + QNET / (12.6 * XT * 687. * .033)
00443 125* 26 CONTINUE
00445 126* QIN(KK) = QOUT(KK - 1)
00446 127* QOUT(KK) = ((18.6 * 2.) / XT) * 12.6 * (BT(KK) - TO)
00447 128* QNET = (QIN(KK) - QOUT(KK)) * DELT / 60.
00450 129* BT(KK) = BT(KK) + QNET / (12.6 * XT * 687. * .033)
00451 130* QINO = QOUT(KK)
00452 131* TC = ((TF + 460.)/100.)*.4.
00453 132* TH = ((TO + 460.)/100.)*.4.
00454 133* HRO = ((.173 * FEQ * (TH - TC)) / (TO - TF)
00455 134* QOUTO = (HCO + HRO) * 15. * (TO - TF)
00456 135* QNET = (QINO - QOUTO) * DELT / 60.
00457 136* TTO = TO + QNET / (.875 * .43)
00460 137* IF (L.EQ.LIMT) GO TO 210
00462 138* L = L+1
00463 139* DO 24 J=1,N
00466 140* DO 24 I=1,N
00471 141* TT(I,J) = TT(I,J)
00472 142* 24 CONTINUE
00475 143* DO 28 NN = 1, KKK
00500 144* BT(NN) = BT(NN)
00501 145* 28 CONTINUE
00503 146* TO = TTO
00504 147* TINS = TTINS
00505 148* TL = TTL
00506 149* TP = TTP
00507 150* GO TO 200
00510 151* 210 WRITE(6,212) TIME, QTOL
00514 152* 212 FORMAT(1H0,3X,'TIME=',F8.2,3X,'MIN.',/4X,'HEAT RELEASE =',F20.2,
00514 153* 13X,'BTU PER HR.',/)
00515 154* DO 214 I = 1, N
00520 155* WRITE (6,220) TT(I,J), J=1,N
00526 156* 214 CONTINUE
00530 157* 220 FORMAT(15F8.2)
00531 158* WRITE (6,220) TTP, TTL, TTINS
00536 159* WRITE (6,220) BT(NN), NN=1, KKK
00544 160* WRITE (6,220) TTO, TF
00550 161* L = 1
00551 162* IF (TIME.GT.TIML) STOP
00553 163* GO TO 230
00554 164* END

```

END OF UNIVAC 1108 FORTRAN V COMPILATION. D *DIAGNOSTIC* MESSAGE(S)

ELK RIVER CASK

TIME	HEAT RELEASE	BTU PER HR
.500	60.000	.688
11.800	480.000	.200
.200	.090	.092
.500	.080	3.250
.900	3.000	

TIME = .00 MIN.		.00 BTU PER HR													NORMAL SHIPPING TEMP OF FUEL ELEMENT ARRAY	
CENTER ELEM.	HEAT RELEASE	1032.00	1027.00	1020.00	1010.00	998.00	981.00	960.00	934.00	901.00	861.00	810.00	745.00	664.00	STEEL CROSS PLATE FUEL ELEMENT TEMP. OF ONE ASSY OF 13X13 ELEMENTS	
		1027.00	1024.00	1017.00	1008.00	995.00	978.00	957.00	931.00	899.00	858.00	807.00	741.00	644.00		
		1020.00	1017.00	1011.00	1002.00	989.00	972.00	952.00	926.00	894.00	853.00	802.00	734.00	638.00		
		1011.00	1008.00	1002.00	993.00	980.00	964.00	943.00	918.00	886.00	846.00	795.00	727.00	632.00		
		998.00	995.00	989.00	980.00	968.00	952.00	932.00	907.00	876.00	836.00	785.00	719.00	626.00		
		981.00	978.00	972.00	964.00	952.00	936.00	917.00	893.00	862.00	823.00	775.00	709.00	618.00		
		960.00	957.00	951.00	944.00	932.00	917.00	899.00	875.00	845.00	808.00	760.00	697.00	609.00		
		934.00	931.00	926.00	918.00	907.00	893.00	875.00	852.00	824.00	788.00	743.00	682.00	598.00		
		901.00	898.00	894.00	886.00	876.00	862.00	845.00	824.00	798.00	764.00	721.00	664.00	585.00		
		860.00	858.00	853.00	846.00	836.00	824.00	808.00	788.00	764.00	733.00	693.00	641.00	568.00		
		810.00	807.00	802.00	795.00	785.00	774.00	760.00	743.00	721.00	693.00	658.00	611.00	547.00		
		745.00	740.00	734.00	727.00	719.00	709.00	697.00	682.00	664.00	640.00	611.00	572.00	518.00		
		665.00	645.00	638.00	632.00	626.00	618.00	609.00	598.00	565.00	568.00	546.00	518.00	473.00		
		400.00	280.00	208.00											CROSS PLATE	
		205.00	205.00	205.00	205.00											
		200.00	1475.00													
TIME = 10.00 MIN.		2002.61 BTU PER HR													HEAT FLOW OUT OF FUEL ELEMENT ASSEMBLY	

1031.83	1027.13	1020.33	1010.69	997.73	980.97	959.87	933.62	901.03	860.52	809.81	745.22	664.68
1027.13	1023.76	1017.24	1007.72	994.83	978.15	957.15	931.01	898.52	858.07	806.94	740.53	644.64
1020.33	1017.24	1011.07	1001.74	989.01	972.51	951.71	925.82	893.58	853.16	801.76	733.73	637.69
1010.69	1007.72	1001.74	992.63	980.13	963.93	943.48	917.91	885.99	845.89	794.63	726.87	631.93
997.73	994.83	989.01	980.13	967.96	952.12	932.09	907.01	875.66	836.11	785.49	718.82	625.72
980.97	978.15	972.51	963.93	952.12	936.72	917.25	892.84	862.21	823.63	774.17	708.99	618.15
959.87	957.15	951.71	943.48	932.09	917.25	898.47	874.84	845.23	807.93	760.10	697.00	609.12
933.62	931.01	925.82	917.91	907.01	892.84	874.84	852.27	824.02	788.33	742.67	682.22	598.16
901.03	898.52	893.58	885.99	875.66	862.21	845.23	824.02	797.48	763.90	720.80	663.89	584.77
860.52	858.07	853.16	845.89	836.11	823.63	807.93	788.33	763.90	732.97	693.17	640.59	567.86
809.81	806.94	801.76	794.63	785.49	774.17	760.10	742.67	720.80	693.17	657.80	611.05	546.22
745.22	740.53	733.73	726.87	718.82	708.99	697.00	682.22	663.89	640.59	611.05	571.66	517.91
664.68	644.64	637.69	631.93	625.72	618.15	609.12	598.16	584.77	567.86	546.22	517.91	472.84
399.62	285.57	268.28										
271.22	303.79	367.53	468.72									
540.99	1475.00											

TIME FROM START OF FIRE

TIME = 20.00 MIN.
HEAT RELEASE =

1983.94 BTU PER HR

1031.80	1027.14	1020.32	1010.64	997.67	980.93	959.85	933.61	901.08	860.63	809.87	745.28	664.97
1027.14	1023.75	1017.22	1007.67	994.80	978.16	957.17	931.03	898.58	858.11	806.96	740.58	644.67
1020.32	1017.22	1011.03	1001.70	989.00	972.55	951.76	925.83	893.56	853.15	801.71	733.65	637.70
1010.64	1007.67	1001.70	992.60	980.15	963.96	943.49	917.89	885.96	845.84	794.55	726.79	631.96
997.67	994.80	989.00	980.15	967.99	952.17	932.11	907.00	875.61	836.09	785.51	718.78	625.75
980.93	978.16	972.55	963.96	952.17	936.79	917.28	892.82	862.20	823.61	774.19	709.02	618.31
959.85	957.17	951.76	943.49	932.11	917.28	898.46	874.83	845.24	807.93	760.12	697.04	609.32
933.61	931.03	925.83	917.89	907.00	892.82	874.83	852.27	824.01	788.35	742.63	682.26	598.36
901.08	898.56	893.56	885.96	875.61	862.20	845.24	824.01	797.42	763.86	720.78	663.89	584.87
860.63	858.11	853.15	845.84	836.09	823.61	807.93	788.35	763.86	732.93	693.21	640.74	567.98
809.87	806.96	801.71	794.55	785.51	774.19	760.12	742.63	720.78	693.21	657.81	611.10	546.45
745.28	740.58	733.65	726.79	718.78	709.02	697.04	682.26	663.89	640.74	611.10	571.59	518.01
664.97	644.67	637.70	631.96	625.75	618.31	609.32	598.36	584.87	567.98	546.45	518.01	473.01
402.66	331.91	430.30										
435.46	479.10	556.70	668.59									
741.48	1475.00											

END OF FIRE

TIME = 30.00 MIN.
HEAT RELEASE =

1909.66 BTU PER HR

1031.78	1027.12	1020.30	1010.61	997.64	980.92	959.85	933.63	901.11	860.68	809.94	745.45	665.72
1027.12	1023.73	1017.20	1007.65	994.78	978.15	957.18	931.05	898.60	858.13	807.01	740.74	645.62
1020.30	1017.20	1011.01	1001.68	988.99	972.55	951.77	925.84	893.56	853.15	801.72	733.77	638.65
1010.61	1007.65	1001.68	992.59	980.15	963.98	943.50	917.89	885.95	845.83	794.55	726.91	632.92
997.64	994.78	988.99	980.15	968.01	952.19	932.13	907.00	875.59	836.07	785.52	718.92	626.72
980.92	978.15	972.55	963.98	952.19	936.82	917.30	892.82	862.19	823.60	774.22	709.19	619.35
959.85	957.18	951.77	943.50	932.13	917.30	898.46	874.83	845.24	807.93	760.15	697.23	610.39
933.63	931.05	925.84	917.89	907.00	892.82	874.83	852.27	824.00	788.35	742.65	682.45	599.43
901.11	898.60	893.56	885.95	875.59	862.19	845.24	824.00	797.41	763.85	720.81	664.07	585.93
860.68	858.13	853.15	845.83	836.07	823.60	807.93	788.35	763.85	732.93	693.26	640.95	569.06
809.94	807.01	801.72	794.55	785.52	774.22	760.15	742.65	720.81	693.26	657.88	611.30	547.59
745.45	740.74	733.77	726.91	718.92	709.19	697.23	682.45	664.07	640.95	611.30	571.86	519.18
665.72	645.62	638.65	632.92	626.72	619.35	610.39	599.43	585.93	569.06	547.59	519.18	475.13
414.65	427.23	600.09										
606.53	650.66	725.80	830.92									
897.53	1475.00											

TIME = 40.00 MIN.
HEAT RELEASE =

1722.63 BTU PER HR

1031.77	1027.10	1020.28	1010.59	997.63	980.91	959.85	933.64	901.14	860.74	810.12	746.10	668.18
1027.10	1023.71	1017.18	1007.63	994.77	978.15	957.18	931.05	898.62	858.18	807.18	741.42	648.79
1020.28	1017.18	1011.00	1001.67	988.99	972.55	951.78	925.84	893.57	853.19	801.88	734.48	641.89
1010.59	1007.63	1001.67	992.59	980.15	963.98	943.50	917.90	885.95	845.86	794.70	727.61	636.18
997.63	994.77	988.99	980.15	968.02	952.20	932.14	907.00	875.60	836.10	785.68	719.63	630.01
980.91	978.15	972.55	963.98	952.20	936.83	917.30	892.82	862.19	823.64	774.38	709.92	622.69
959.85	957.18	951.78	943.50	932.14	917.30	898.47	874.83	845.24	807.96	760.31	697.96	613.77
933.64	931.05	925.84	917.90	907.00	892.82	874.83	852.27	824.01	788.38	742.82	683.18	602.85
901.14	898.62	893.57	885.95	875.60	862.19	845.24	824.01	797.42	763.89	720.97	664.80	589.38
860.74	858.18	853.19	845.86	836.10	823.64	807.96	788.38	763.89	732.99	693.44	641.70	572.58
810.12	807.18	801.88	794.70	785.68	774.38	760.31	742.82	720.97	693.44	658.16	612.12	551.24
746.10	741.42	734.48	727.61	719.63	709.92	697.96	683.18	664.80	641.70	612.12	573.15	523.24
668.18	648.79	641.89	636.18	630.01	622.69	613.77	602.85	589.38	572.58	551.24	523.24	482.17
442.87	544.32	704.25										
708.16	722.20	738.43	750.90									
752.25	1000.00											

AMBIENT TEMP. = 100°F

TIME = 50.00 MIN.
HEAT RELEASE =

1452.78 BTU PER HR

1031.75	1027.08	1020.26	1010.58	997.62	980.91	959.86	933.66	901.20	860.91	810.66	747.84	673.75
1027.08	1023.70	1017.16	1007.62	994.76	978.14	957.18	931.07	898.67	858.35	807.73	743.27	655.85
1020.26	1017.16	1010.98	1001.66	988.98	972.55	951.78	925.86	893.63	853.36	802.44	736.44	649.14
1010.58	1007.62	1001.66	992.58	980.15	963.98	943.51	917.91	886.00	846.02	795.26	729.59	643.50
997.62	994.76	988.98	980.15	968.02	952.20	932.14	907.02	875.64	836.26	786.23	721.62	637.39
980.91	978.14	972.55	963.98	952.20	936.83	917.31	892.84	862.24	823.79	774.94	711.91	630.14
959.86	957.18	951.78	943.51	932.14	917.31	898.48	874.85	845.29	808.12	760.87	699.96	621.31
933.66	931.07	925.86	917.91	907.02	892.84	874.85	852.29	824.05	788.53	743.36	685.18	610.48
901.20	898.67	893.63	886.00	875.64	862.24	845.29	824.05	797.49	764.06	721.52	666.81	597.12
860.91	858.35	853.36	846.02	836.26	823.79	808.12	788.53	764.06	733.25	694.05	643.75	580.50
810.66	807.73	802.44	795.26	786.23	774.94	760.87	743.36	721.52	694.05	659.09	614.44	559.51
747.84	743.27	736.44	729.59	721.62	711.91	699.96	685.18	666.81	643.75	614.44	576.80	532.69
673.75	655.85	649.14	643.50	637.39	630.14	621.31	610.48	597.12	580.50	559.51	532.69	497.88
485.92	610.02	713.74										
715.76	718.42	718.70	715.34									
711.24	1000.00											

TIME = 60.00 MIN.
HEAT RELEASE =

1179.03 BTU PER HR

1031.73	1027.06	1020.25	1010.57	997.61	980.91	959.87	933.71	901.35	861.35	811.88	751.24	682.62
1027.06	1023.68	1017.15	1007.61	994.75	978.15	957.20	931.13	898.83	858.80	808.99	746.87	666.72
1020.25	1017.15	1010.97	1001.65	988.98	972.55	951.80	925.92	893.78	853.81	803.74	740.25	660.37
1010.57	1007.61	1001.65	992.58	980.15	963.99	943.53	917.97	886.16	846.47	796.58	733.46	654.67
997.61	994.75	988.98	980.15	960.02	952.21	932.16	907.07	875.79	836.71	787.55	725.50	648.87
980.91	978.15	972.55	963.99	952.21	936.84	917.33	892.89	862.39	824.23	776.24	715.81	641.73
959.87	957.20	951.80	943.53	932.16	917.33	898.51	874.90	845.43	808.55	762.16	703.87	633.03
933.71	931.13	925.92	917.97	907.07	892.89	874.90	852.37	824.22	788.97	744.65	689.11	622.38
901.35	898.83	893.78	886.16	875.79	862.39	845.43	824.22	797.72	764.55	722.85	670.79	609.25
861.35	858.80	853.81	846.47	836.71	824.23	808.55	788.97	764.55	733.96	695.56	647.91	592.99
811.88	808.99	803.74	796.58	787.55	776.24	762.16	744.65	722.85	695.56	661.32	619.26	572.77
751.24	746.87	740.25	733.46	725.50	715.81	703.87	689.11	670.79	647.91	619.26	584.15	548.18
682.62	666.72	660.37	654.67	648.87	641.73	633.03	622.38	609.25	592.99	572.77	548.18	521.98
529.19	633.62	703.05										
704.21	703.85	700.53	693.93									
688.86	100.00											

TIME= 70.00 MIN.

HEAT RELEASE = 982.17 BTU PER HR

1031.71	1027.05	1020.24	1010.56	997.61	980.93	959.93	933.85	901.71	862.23	814.03	756.33	693.61
1027.05	1023.67	1017.14	1007.61	994.76	978.16	957.26	931.27	899.19	859.69	811.19	752.20	679.69
1020.24	1017.14	1010.97	1001.65	988.98	972.57	951.85	926.06	894.14	854.71	806.00	745.87	673.80
1010.56	1007.61	1001.65	992.58	980.16	964.01	943.58	918.10	886.51	847.38	798.88	739.18	668.50
997.61	994.76	988.98	980.16	968.03	952.23	932.22	907.21	876.15	837.62	789.86	731.27	662.65
980.93	978.16	972.57	964.01	952.23	936.88	917.40	893.03	862.74	825.13	778.55	721.62	655.67
959.93	957.26	951.85	943.58	932.22	917.40	898.59	875.06	845.79	809.45	764.47	709.73	647.17
933.85	931.27	926.06	918.10	907.21	893.03	875.06	852.59	824.62	789.90	746.98	695.04	636.76
901.71	899.19	894.14	886.51	876.15	862.74	845.79	824.62	798.29	765.61	725.29	676.88	623.98
862.23	859.69	854.71	847.38	837.62	825.13	809.45	789.90	765.61	735.47	698.40	654.42	608.31
814.03	811.19	806.00	798.88	789.86	778.55	764.47	746.98	725.29	698.40	665.43	626.98	589.32
756.33	752.20	745.87	739.18	731.27	721.62	709.73	695.04	676.88	654.42	626.98	595.46	567.82
693.61	679.69	673.80	668.50	662.65	655.67	647.17	636.76	623.98	608.31	589.32	567.82	550.02
564.93	641.57	689.14										
649.81	688.41	684.13	676.87									
671.73	100.00											

TIME= 80.00 MIN.

HEAT RELEASE = 872.16 BTU PER HR

1031.70	1027.04	1020.23	1010.56	997.63	980.97	960.05	934.14	902.36	863.66	817.11	762.66	705.32
1027.04	1023.66	1017.14	1007.61	994.77	978.21	957.38	931.56	899.84	861.14	814.34	758.80	692.99
1020.23	1017.14	1010.96	1001.66	989.00	972.62	951.97	926.34	894.79	856.18	809.23	752.77	687.56
1010.56	1007.61	1001.66	992.59	980.18	964.06	943.71	918.39	887.17	848.87	802.18	746.23	682.51
997.63	994.77	989.00	980.18	968.07	952.30	932.35	907.50	876.80	839.11	793.18	738.41	676.83
980.97	978.21	972.62	964.06	952.30	936.96	917.54	893.33	863.40	826.62	781.89	728.84	670.04
960.05	957.38	951.97	943.71	932.35	917.54	898.79	875.41	846.48	810.96	767.84	717.05	661.77
934.14	931.56	926.34	918.39	907.50	893.33	875.41	853.06	825.42	791.49	750.44	702.52	651.66
912.36	899.84	894.79	887.17	876.80	863.40	846.48	825.42	799.38	767.46	728.99	684.67	639.34
863.66	861.14	856.18	848.87	839.11	826.62	810.96	791.49	767.46	738.04	702.78	662.92	624.46
817.11	814.34	809.23	802.18	793.18	781.89	767.84	750.44	728.99	702.78	671.59	637.23	607.07
762.66	758.80	752.77	746.23	738.41	728.84	717.05	702.52	684.67	662.92	637.23	609.76	588.92
705.32	692.99	687.56	682.51	676.83	670.04	661.77	651.66	639.34	624.46	607.07	588.92	577.37
592.25	644.13	675.38										
675.73	673.80	669.20	661.80									
656.86	100.00											

TIME= 90.00 MIN.

HEAT RELEASE = 832.42 BTU PER HR

1031.70	1027.03	1020.24	1010.58	997.67	981.08	960.28	934.62	903.35	865.66	820.95	769.67	716.68
1027.03	1023.66	1017.14	1007.63	994.82	978.31	957.60	932.04	900.84	863.15	818.25	766.03	705.46
1020.24	1017.14	1010.97	1001.68	989.05	972.73	952.20	926.83	895.80	858.23	813.25	760.29	700.43
1010.58	1007.63	1001.68	992.62	980.24	964.18	943.94	918.88	888.19	850.94	806.27	753.92	695.62
997.67	994.82	989.05	980.24	968.15	952.43	932.60	908.00	877.83	841.20	797.33	746.23	690.14
981.08	978.31	972.73	964.18	952.43	937.14	917.83	893.86	864.45	828.74	786.09	736.78	683.55
960.28	957.60	952.20	943.94	932.60	917.83	899.18	876.02	847.60	813.14	772.12	725.15	675.52
934.62	932.04	926.83	918.88	908.00	893.86	876.02	853.88	826.72	793.84	754.91	710.88	665.76
903.35	900.84	895.80	888.19	877.83	864.45	847.60	826.72	801.13	770.23	733.88	693.51	653.97
865.66	863.15	858.23	850.94	841.20	828.74	813.14	793.84	770.23	741.79	708.64	672.77	640.03
820.95	818.25	813.25	806.27	797.33	786.09	772.12	754.91	733.88	708.64	679.55	649.16	624.39
769.67	766.03	760.29	753.92	746.23	736.78	725.15	710.88	693.51	672.77	649.16	625.53	609.29
716.68	705.46	700.43	695.62	690.14	683.55	675.52	665.76	653.97	640.03	624.39	609.29	601.37
612.43	644.29	662.43										
662.51	660.25	665.51	648.27									
643.40	100.00											

TIME= 100.00 MIN.

HEAT RELEASE = 841.78 BTU PER HR

1031.70	1027.04	1020.25	1010.62	997.76	981.26	960.64	935.34	904.71	868.14	825.30	776.83	727.09
1027.04	1023.67	1017.16	1007.67	994.91	978.50	957.97	932.76	902.20	865.66	822.67	773.37	716.54
1020.25	1017.16	1011.00	1001.73	989.15	972.92	952.58	927.55	897.18	860.77	817.76	767.87	711.84
1010.62	1007.67	1001.73	992.70	980.36	964.38	944.33	919.62	889.58	853.52	810.88	761.67	707.25
997.76	994.91	989.15	980.36	968.31	952.67	933.01	908.76	879.25	843.82	802.03	754.13	701.95
981.26	978.50	972.92	964.38	952.67	937.46	918.32	894.68	865.92	831.43	790.89	744.84	695.55
960.64	957.97	952.58	944.33	933.01	918.32	899.82	876.98	849.20	815.95	777.07	733.42	687.78
935.34	932.76	927.55	919.62	908.76	894.68	876.98	855.14	828.60	796.92	760.14	719.51	678.39
904.71	902.20	897.18	889.58	879.25	865.92	849.20	828.60	803.61	773.89	739.71	702.79	667.19
868.14	865.66	860.77	853.52	843.82	831.43	815.95	796.92	773.89	746.63	715.67	683.25	654.26
825.30	822.67	817.76	810.88	802.03	790.89	777.07	760.14	739.71	715.67	688.78	661.77	640.30
776.83	773.37	767.87	761.67	754.13	744.84	733.42	719.51	702.79	683.25	661.77	641.31	627.58
727.09	716.54	711.84	707.25	701.95	695.55	687.78	678.39	667.19	654.26	640.30	627.58	621.13
627.06	642.95	660.31										
650.19	647.69	642.88	635.75									
631.04	100.00											

TIME= 110.00 MIN.

HEAT RELEASE = 882.35 BTU PER HR

1031.71	1027.06	1020.29	1010.70	997.91	981.55	961.17	936.29	906.39	870.98	829.91	783.75	736.27
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FIGURE 1 c. PIQUA-ELK RIVER SHIPPING CASK - LID - ASSEMBLY (Knapp Mills dwg F-2056)

ITEM NO.	RECD	NAME	MATL	SPEC NO.	DWG NO.	DESCRIPTION
1	2	5" x 5" x 5" x 5" x 5"	Q2	T-1		1" x 5 1/2" x 5 1/2"
2	2	WELDED	Q2	T-1		2" x 6 1/2" x 6 1/2"
3	2	WELDED	Q2	T-1		1" x 6 1/2" x 6 1/2"
4	2	WELDED	Q2	T-1		4" x 8" x 12" x 1/2"
5	2	WELDED	Q2	T-1		5" x 6 1/2" x 30" x 1/2"
6	2	WELDED	Q2	T-1		4" x 5 1/2" x 6 1/2"
7	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
8	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
9	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
10	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
11	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
12	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
13	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
14	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"
15	2	WELDED	Q2	T-1		5" x 6 1/2" x 6 1/2"

- NOTES:
- 1- MAGNETIC PARTICLE TEST FOR ROOT & FINAL LAYERS PER PROC. 131.
 - 2- HEAT TREATED 150,000 PSI. MIN. TENSILE
 - 3- DELETED
 - 4- WIRE ROPES SLUNG TO BE TRU LOC OR EQUAL (AMERICAN CHAIN & CABLE CO.) 3/4" WIRE ROPES TO MIL C5424 (G113) WITH EYE END ON BOTH ENDS NO 200GB-18. FITTINGS PER MIL-T-731. LENGTH TO BE 42" CENTER TO CENTER OF HOLES.
 - 5- STRAIN HARDENED TO 125,000 PSI.
 - 6- LIQUID PENETRANT TEST FIRST & LAST WELD LAYERS PER MIL PROC. 127.
 - 7- PRINT PER NOTE 4 Dwg F-2096

FIGURE 1 e. PIQUA-ELK RIVER SHIPPING CASK - LIFTING YOKE - DETAILS (Knapp Mills dwg F-2102)

DETAIL (U) S5-2-5871
(LENGTH SPACER)
MATL: STEEL REQD: 4
SCALE: 6" = 1'-0"

DETAIL (W) S5-2-5871
(SHIELDING PLATE)
MATL: 1/2" THK 35% B2 BORAL PLATE REQD: 2

FIGURE 3. PIQUA-ELK RIVER CASK - INSERT DETAILS (Du Pont dwg S5-2-5872)

FIGURE 4. NS SAVANNAH FUEL ASSEMBLY (Savannah Tech Staff dwg 1248)

U.S. ATOMIC ENERGY COMMISSION
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AEC 0529-01
OS

Chapter 0529 SAFETY STANDARDS FOR THE PACKAGING OF RADIOACTIVE AND FISSILE MATERIALS

0529-01 POLICY

Fissile materials and other radioactive material shall be packaged and prepared for shipment in a manner that provides assurance of protection of the public health and safety during the transportation of such materials.

0529-02 OBJECTIVE

To establish standards for the packaging of fissile materials or large quantities of radioactive materials for transportation from facilities not subject to 10 CFR 71, and to establish responsibilities for issuing AEC letters of approval for such packages.

0529-03 RESPONSIBILITIES AND AUTHORITIES

031 Directors, Headquarters Divisions, provide guidance, instructions, standards, and criteria as described in chapter 0101-053b., consistent with this chapter, to assure the safe packaging of fissile and radioactive materials, including:

- a. directing cognizant managers of field offices to require modifications of equipment, procedures, or practices.
- b. imposing additional requirements for packaging standards.
- c. curtailing or suspending the use of specific packages when necessary.

032 Managers of Field Offices, Space Nuclear Propulsion Office, and the Directors, Divisions of Headquarters Services, Naval Reactors, and Technical Information, consistent with guidance, instructions, standards, and criteria issued pursuant to 031, above:

- a. grant AEC approval for packages to be used for the transportation of fissile materials or large quantities of radioactive materials which meet the standards contained in appendix 0529.

- b. establish standards and grant AEC approval for packages which are shipped under a "National Security" exemption provided to the AEC by any Federal transportation regulatory agency.

- c. grant such exemptions from the standards set forth in appendix 0529 as they determine will not endanger life or property or the common defense and security, and within 30 days after granting an exemption, provide the Director, Division of Operational Safety, a detailed report of the reasons for granting it.

- d. conduct annual appraisals of contractor operations to assure compliance with the requirements of this chapter.

(NOTE: Contractors shall not be permitted to exercise any of the above authorities.)

033 The Manager, Albuquerque Operations Office, in addition to the responsibilities and authorities assigned in 032, above, is authorized to establish safety standards for, and to transport, atomic weapons. This authority is not delegated to any other field office manager.

034 The Director, Division of Operational Safety:

- a. determines the need for, and develops new and revised safety standards to be applied in the preparation of, radioactive and fissile materials for transportation, and renders interpretations of this chapter.
- b. provides a central point of coordination with the Director of Regulation for developing and revising health and safety codes, regulations, and guides pertaining to safety in the transportation of radioactive and fissile materials

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which are intended for use in the AEC Regulatory program or by other Federal agencies.

- c. conducts periodic appraisals to determine the adequacy of the implementation of this chapter.

0529-04 DEFINITIONS

041 AEC contractor, for the purposes of this chapter, means a prime contractor or subcontractor of the Atomic Energy Commission who introduces radioactive and/or fissile material into commerce and who is exempt from the requirements of 10 CFR 71.

0529-05 BASIC REQUIREMENTS

051 Applicability. The provisions set forth in this chapter and its appendix apply to the Headquarters Offices, Field Offices, and AEC contractors.

052 Coverage. This chapter and its appendix cover AEC offices and AEC contractors responsible for the preparation of fissile materials and other radioactive materials for shipment outside the boundaries of AEC-controlled sites by common, contract, or private carriers, including Government vehicles.

053 Federal Regulations. Each shipment of fissile materials and other radioactive materials shall be in compliance with the safety regulations of the Department of Transportation (DOT) or the Post Office Department, depending on the mode of transportation, when offered to the carrier.

054 International Atomic Energy Agency Regulations. Each shipment of fissile materials and other radioactive materials consigned to a foreign country must meet the requirements set forth in IAEA Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials." Specifically, Part C, "Requirements for Packaging and for Delivery of Packages to Transport," must be met to be in compliance with this chapter.

055 Package Standards for Fissile and/or Large Quantities of Radioactive Materials:

- a. Packages which have a DOT permit are deemed to meet the standards of this chapter and need not be issued a new AEC letter of ap-

proval. The original DOT permit shall be revalidated with DOT when used by subsequent shippers.

- b. DOT specification containers are considered to meet the standards of this chapter, and no specific AEC approval is required for their use.

- c. Atomic weapons shall be packaged and transported in accordance with the standards issued by the Manager, Albuquerque Operations Office.

- d. Packages which are shipped in the interest of "National Security" shall be in compliance with the standards established by the responsible field office manager.

- e. All other packages for fissile materials and large quantities of radioactive materials shall be designed, constructed, and used in accordance with the standards contained in the attached appendix. Upon determination that a design does meet the standards, an AEC letter of approval shall be issued to the contractor. A copy of each AEC letter of approval shall be forwarded to the Division of Construction, Headquarters.

In making application to the DOT for a special permit for a package, the contractor shall provide the DOT with a copy of the AEC letter of approval for the package.

Where an AEC office is the actual shipper, rather than a contractor, appropriate internal procedures shall be established by the responsible field office manager to assure compliance with the standards contained in the attached appendix.

056 Existing Packaging. An existing packaging for fissile materials and/or large quantities of radioactive materials, must meet the standards of this chapter. However, loss of shielding resulting from subjecting the packaging to the "Puncture Test" followed by the "Thermal Test" will not be considered cause for disqualifying an existing packaging that otherwise meets the standards of this chapter.

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SAFETY STANDARDS FOR THE PACKAGING
OF RADIOACTIVE AND FISSILE MATERIALS

AEC 0529-057

057 References

a. Federal Regulations

49 CFR 171-178, DOT Regulations

14 CFR 103, DOT Regulations

39 CFR 124-125, Post Office Regulations

b. Chapter 0560, "AEC Atomic
Weapon Safety Program."

c. Chapter 2401, "Physical Protection of Classified Matter and Information," for security guidance.

d. Chapter 5201, "Transportation and Traffic Management," for guidance in general shipping requirements.

e. AEC Directory of Radioactive and Fissile Material Shipping Containers, for container design information.

0529-06 NATIONAL EMERGENCY
APPLICATION

During a national emergency, as defined in chapter 0601-04, the provisions of this chapter and appendix will continue in effect.

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I. DEFINITIONS AND EXEMPTIONS

A. Definitions (As used in this appendix)

1. Carrier means any person engaged in the transportation of passengers or property, as common, contract, or private carrier, or freight forwarder, as those terms are used in the Interstate Commerce Act, as amended, or the United States Post Office.
2. Close Reflection by Water means immediate contact by water of sufficient thickness to reflect a maximum number of neutrons.
3. Containment Vessel means the receptacle on which principal reliance is placed to retain the radioactive material during transport.
4. Fissile Classification means classification of a package or shipment of fissile materials according to the controls needed to provide nuclear criticality safety during transportation as follows:
 - a. Fissile Class I: Packages which may be transported in unlimited numbers and in any arrangement, and which require no nuclear criticality safety controls during transportation. For purposes of nuclear criticality safety control, a transport index is not assigned to Fissile Class I packages. However, the ex-

ternal radiation levels may require a transport index number.

- b. Fissile Class II: Packages which may be transported together in any arrangement but in numbers which do not exceed a transport index of 50. For purposes of nuclear criticality safety control, individual packages may have a transport index of not less than 0.1 and not more than 10. However, the external radiation levels may require a higher transport index number but not to exceed 10. Such shipments require no nuclear criticality safety control by the shipper during transportation.
- c. Fissile Class III: Shipments of packages which do not meet the requirements of Fissile Classes I or II and which are controlled in transportation by special arrangements between the shipper and the carrier to provide nuclear criticality safety.
5. Fissile Materials means uranium-233, uranium-235, plutonium-238, plutonium-239, and plutonium-241.
6. Large Quantity means a quantity of radioactive material, the aggregate radioactivity of which exceeds that specified in the following table for a transport group as defined in 16., below.

Radionuclide Identification	Transport Group						Special Form
	I	II	III	IV	V	VI-VII	
Radioactivity	20 curies	20 curies	200 curies	200 curies	5,000 curies	50,000 curies	5,000 curies

7. Low Specific Activity Material means any of the following:

- a. Uranium or thorium ores and physical or chemical concentrates of those ores;
- b. Unirradiated natural or depleted uranium or unirradiated natural thorium;
- c. Tritium oxide in aqueous solutions provided the concentra-

tion does not exceed 5.0 millicuries per milliliter;

- d. Material in which the activity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed:

(1) 0.0001 millicuries of Group I radionuclides; or

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- (2) 0.005 millicuries of Group II radionuclides; or
- (3) 0.3 millicuries of Groups III or IV radionuclides.

NOTE: This includes, but is not limited to, materials of low radioactivity concentration such as residues or solution from chemical processing; wastes such as building rubble, metal, wood, and fabric scrap, glassware, paper and cardboard; solid or liquid plant waste, sludges, and ashes.

- e. Objects of non-radioactive material externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over an area of one square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square centimeter of Group I radionuclides or 0.001 millicurie (2,200,000 disintegrations per minute) per square centimeter of other radionuclides.
- 8. Maximum Normal Operating Pressure means the maximum gauge pressure which is expected to develop in the containment vessel under the normal conditions of transport specified in annex 1, below, considered individually.
- 9. Moderator means a material used to reduce by scattering collisions, and without appreciable capture, the kinetic energy of neutrons.
- 10. Optimum Interspersed Hydrogenous Moderation means the occurrence of hydrogenous material between containment vessels to such an extent that the maximum nuclear reactivity results.
- 11. Package means packaging and its radioactive contents.
- 12. Packaging means one or more receptacles and wrappers and their contents, excluding fissile material and other radioactive material, but including absorbent material, spacing structures, thermal insulation, radiation shielding, devices for cooling and for absorbing mechanical shock, external fittings, neutron moderators, non-fissile neutron absorbers, and other supplementary equipment.
- 13. Primary Coolant means a gas, liquid, or solid, or combination of them, in contact with the radioactive material or, if the material is in special form, in contact with its capsule, and used to remove decay heat.
- 14. Sample Package means a package which is fabricated, packed, and closed to fairly represent the proposed package as it would be presented for transport, simulating the material to be transported, as to weight and physical and chemical form.
- 15. Special Form means any of the following physical forms of radioactive material of any transport group:
 - a. The material is in solid form having no dimension less than 0.5 millimeter or at least one dimension greater than 5 millimeters; does not melt, sublime, or ignite in air at a temperature of 1000° F; will not shatter or crumble if subjected to the percussion test described in annex 4, below; and is not dissolved or converted into dispersible form to the extent of more than 0.005 percent by weight by immersion for 1 week in water at 68° F or in air at 86° F; or
 - b. The material is securely contained in a capsule having no dimension less than 0.5 millimeter or at least one dimension greater than 5 millimeters, which will retain its contents if subjected to the tests prescribed in annex 4, below; and which is constructed of materials which do not melt, sublime, or ignite in air at 1475° F, and do not dissolve or convert into dispersible form to the extent of more than 0.005 percent by weight by immersion for 1 week in water at 68° F. or in air at 86° F.

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16. Transport Group means any one of seven groups into which radionuclides in normal form are classified, according to their toxicity and their relative potential hazard in transport, in annex 3, below.

- a. Any radionuclide not specifically listed in one of the groups in annex 3, below, shall be assigned to one of the groups in accordance with the following table:

Radionuclide	Radioactive Half-life		
	0 to 1000 days	1000 days to 10^6 years	Over 10^6 years
Atomic number 1-81	Group III	Group II	Group III
Atomic number 82 and over	Group I	Group I	Group III

- b. For mixtures of radionuclides the following shall apply:

- (1) If the identity and respective activity of each radionuclide are known, the permissible activity of each radionuclide shall be such that the sum, for all groups present, of the ratio between the total activity for each group to the permissible activity for each group will not be greater than unity.
- (2) If the groups of the radionuclides are known but the amount in each group cannot be reasonably determined, the mixture shall be assigned to the most restrictive group present.
- (3) If the identity of all or some of the radionuclides cannot be reasonably determined, each of those unidentified radionuclides shall be considered as belonging to the most restrictive group which cannot be positively excluded.
- (4) Mixtures consisting of a single radioactive decay chain where the radionuclides are in the naturally occurring proportions shall be considered as consisting of a single radionuclide. The group and activity shall be

that of the first member present in the chain, except that if a radionuclide "x" has a half-life longer than that of that first member and an activity greater than that of any other member, including the first, at any time during transportation, the transport group of the nuclide "x" and the activity of the mixture shall be the maximum activity of that nuclide "x" during transportation.

B. Exemptions

A shipper is exempt from all requirements of this appendix to the extent that he delivers to a carrier for transport packages each containing less than a large quantity of radioactive material, as defined in A.6., above, which may include one of the following:

1. Not more than 15 grams of fissile material; or
2. Thorium, or uranium containing not more than 0.72 percent by weight of fissile material; or
3. Uranium compounds, other than metal, (e.g., UF_4 , UF_6 , or uranium oxide in bulk form, not pelleted or fabricated into shapes) or aqueous solutions of uranium, in which the total amount of uranium-233 and plutonium present does not exceed

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- 1.0 percent by weight of the uranium-235 content, and the total fissile content does not exceed 1.00 percent by weight of the total uranium content; or
4. Homogeneous hydrogeneous solutions or mixtures containing not more than:
 - a. 500 grams of any fissile material, provided the atomic ratio of hydrogen to fissile material is greater than 7600; or
 - b. 800 grams of uranium-235, provided that the atomic ratio of hydrogen to fissile material is greater than 5,200, and the content of other fissile material is not more than 1 percent by weight of the total uranium-235 content; or
 - c. 500 grams of uranium-233 and uranium-235, provided that the atomic ratio of hydrogen to fissile material is greater than 5200, and the content of plutonium is not more than 1 percent by weight of the total uranium-233 and uranium-235 content; or
 5. Less than 350 grams of fissile material, provided that there is not more than 5 grams of fissile material in any cubic foot within the package.

II. PACKAGE STANDARDS

A. General Standards for All Packaging

1. Packaging shall be of such materials and construction that there will be no significant chemical, galvanic, or other reaction among the packaging components, or between the packaging components and the package contents.
2. Packaging shall be equipped with a positive closure which will prevent inadvertent opening.
3. Lifting Devices
 - a. If there is a system of lifting devices which is a structural part of the package, the system shall be capable of supporting 3 times the weight of the loaded

package without generating stress in any material of the packaging in excess of its yield strength.

- b. If there is a system of lifting devices which is a structural part only of the lid, the system shall be capable of supporting 3 times the weight of the lid and any attachments without generating stress in any material of the lid in excess of its yield strength.
- c. If there is a structural part of the package which could be employed to lift the package and which does not comply with a., above, the part shall be securely covered or locked during transport in such a manner as to prevent its use for that purpose.
- d. Each lifting device which is a structural part of the package shall be so designed that failure of the device under excessive load would not impair the containment or shielding properties of the package.

4. Tie-down Devices

- a. If there is a system of tie-down devices which is a structural part of the package, the system shall be capable of withstanding, without generating stress in any material of the package in excess of its yield strength, a static force applied to the center of gravity of the package having a vertical component of 2 times the weight of the package with its contents, a horizontal component along the direction in which the vehicle travels of 10 times the weight of the package with its contents, and a horizontal component in the transverse direction of 5 times the weight of the package with its contents.
- b. If there is a structural part of the package which could be employed to tie the package down and which does not comply with a., above, the part shall be securely covered or locked during transport in such a manner as to prevent its use for that purpose.

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- c. Each tie-down device which is a structural part of the package shall be so designed that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this section A.

B. Structural Standards for Large Quantity Packaging

Packaging used to ship a large quantity of radioactive material, as defined in I, A.6., above, shall be designed and constructed in compliance with the structural standards of this section. Standards different from those specified in this section may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

1. Load Resistance. Regarded as a simple beam supported at its ends along any major axis, packaging shall be capable of withstanding a static load, normal to and uniformly distributed along its length, equal to 5 times its fully loaded weight, without generating stress in any material of the packaging in excess of its yield strength.
2. External Pressure. Packaging shall be adequate to assure that the containment vessel will suffer no loss of contents if subjected to an external pressure of 25 pounds per square inch gauge.

C. Criticality Standards for Fissile Material Packages

1. A package used for the transport of fissile material shall be so designed and constructed and its contents so limited that it would be subcritical if it is assumed that water leaks into the containment vessel, and:
 - a. water moderation of the contents occurs to the most reactive credible extent consistent with the chemical and physical form of the contents; and
 - b. the containment vessel is fully reflected on all sides by water.

2. A package used for the transport of fissile material shall be so designed and constructed and its contents so limited that it would be subcritical if it is assumed that any contents of the package which are liquid during normal transport leak out of the containment vessel, and that the fissile material is then:

- a. in the most reactive credible configuration consistent with the chemical and physical form of the material;
 - b. moderated by water outside of the containment vessel to the most reactive credible extent; and
 - c. fully reflected on all sides by water.
3. The manager or other designated official may approve exceptions to the requirements of this section where the containment vessel incorporates special design features which would preclude leakage of liquids in spite of any single packaging error and appropriate measures are taken before each shipment to verify the leak tightness of each containment vessel.

D. Evaluation of a Single Package

1. The effect of the transport environment on the safety of any single package of radioactive material shall be evaluated as follows:
 - a. The ability of a package to withstand conditions likely to occur in normal transport shall be assessed by subjecting a sample package or scale model, by test or other assessment, to the normal conditions of transport as specified in E., below, and
 - b. The effect on a package of conditions likely to occur in an accident shall be assessed by subjecting a sample package or scale model, by test or other assessment, to the hypothetical accident conditions as specified in F., below.
2. Taking into account controls to be exercised by the shipper, the manager or other designated official

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may permit the shipment to be evaluated together with or without the transporting vehicle, for the purpose of one or more tests.

3. Normal conditions of transport and hypothetical accident conditions different from those specified in E. and F., below, may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

E. Standards for Normal Conditions of Transport for a Single Package

1. A package used for the shipment of fissile material or a large quantity of radioactive material, as defined in I, A.6., above, shall be so designed and constructed and its contents so limited that under the normal conditions of transport specified in annex 1, below:
 - a. there will be no release of radioactive materials from the containment vessel;
 - b. the effectiveness of the packaging will not be substantially reduced;
 - c. there will be no mixture of gases or vapors in the package which could, through any credible increase of pressure or an explosion, significantly reduce the effectiveness of the package;
 - d. radioactive contamination of the liquid or gaseous primary coolant will not exceed 10^{-7} curies of activity of Group I radio-nuclides per milliliter, 5×10^{-6} curies of activity of Group II radionuclides per milliliter; 3×10^{-4} curies of activity of Group III and Group IV radionuclides per milliliter; and
 - e. there will be no loss of coolant or loss of operation of any mechanical cooling device.
2. A package used for the shipment of fissile material shall be so designed and constructed and its contents so

limited that under normal conditions of transport, specified in annex 1, below, considered individually:

- a. the package will be subcritical;
- b. the geometric form of the package contents would not be substantially altered;
- c. there will be no leakage of water into the containment vessel. This requirement need not be met if, in the evaluation of undamaged packages under H.1., I.1.a., or J.1., below, it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and
- d. there will be no substantial reduction in the effectiveness of the packaging, including:
 - (1) reduction by more than 5 percent in the total effective volume of the packaging on which nuclear safety is assessed;
 - (2) reduction by more than 5 percent in the effective spacing on which nuclear safety is assessed, between the center of the containment vessel and the outer surface of the packaging; or
 - (3) occurrence of any aperture in the outer surface of the packaging large enough to permit the entry of a 4-inch cube.

3. A package used for the shipment of a large quantity of radioactive material as defined in I., A. 6., above, shall be so designed and constructed and its contents so limited that under the normal conditions of transport specified in annex 1, below, considered individually, the containment vessel would not be vented directly to the atmosphere.

F. Standards for Hypothetical Accident Conditions for a Single Package

1. A package used for the shipment of a large quantity of radioactive material, as defined in I, A.6., above,

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or the shipment of fissile material when the package will contain more than .001 curie of Group I radionuclides, .05 curie of Group II radionuclides, 3 curies of Group III radionuclides, 20 curies of Group IV or Group V radionuclides, or radionuclides in special form, or 1000 curies of Group VI or Group VII radionuclides shall be so designed and constructed and its contents so limited that if subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Puncture, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, it will meet the following conditions:

- a. The reduction of shielding would not be sufficient to increase the external radiation dose rate to more than 1000 millirems per hour at 3 feet from the external surface of the package.
- b. No radioactive material would be released from the package except for gases and contaminated coolant containing total radioactivity exceeding neither:
 - (1) 0.1 percent of the total radioactivity of the package contents; nor
 - (2) 0.01 curie of Group I radionuclides, 0.5 curie of Group II radionuclides, 10 curies of Group III radionuclides, 10 curies of Group IV radionuclides, and 1000 curies of inert gases irrespective of transport group.

A package need not satisfy the requirements of this paragraph if it contains only low specific activity materials, as defined in I, A.7., above, and is transported on a motor vehicle, railroad car, aircraft, inland water craft, or hold or deck of a seagoing vessel assigned for the sole use of the shipper.

2. A package used for the shipment of fissile material shall be so designed and constructed and its contents so limited that if subjected to the hypothetical accident conditions specified in annex 2, below,

as the Free Drop, Puncture, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, the package would be subcritical. In determining whether this standard is satisfied, it shall be assumed that:

- a. the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;
- b. water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and
- c. there is reflection by water on all sides and as close as is consistent with the damaged condition of the package.

G. Evaluation of an Array of Packages of Fissile Material

1. The effect of the transport environment on the nuclear criticality safety of an array of packages of fissile material shall be evaluated by subjecting a sample package or a scale model, by test or other assessment, to the hypothetical accident conditions specified in H., I., or J., below, for the proposed fissile class, and by assuming that each package in the array is damaged to the same extent as the sample package or scale model. In the case of a Fissile Class III shipment, the manager or other designated official may, taking into account controls to be exercised by the shipper, permit the shipment to be evaluated as a whole rather than as individual packages, and either with or without the transporting vehicle, for the purpose of one or more tests.
2. In determining whether the standards of H.2., I.1.b., and J.2., below, are satisfied, it shall be assumed that:
 - a. the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package,

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the chemical and physical form of the contents, and controls exercised over the number of packages to be transported together; and

- b. water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.

H. Specific Standards for a Fissile Class I Package

A Fissile Class I package shall be so designed and constructed and its contents so limited that:

1. any number of such undamaged packages would be subcritical in any arrangement, and with optimum interspersed hydrogenous moderation unless there is a greater amount of interspersed moderation in the packaging, in which case that greater amount may be considered; and
2. two hundred and fifty such packages would be subcritical in any arrangement, if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, with close reflection by water on all sides of the array and with optimum interspersed moderation unless there is a greater amount of interspersed moderation in packaging, in which case that greater amount may be considered. The condition of the package shall be assumed to be as described in G., above.

I. Specific Standards for a Fissile Class II Package

1. A Fissile Class II package shall be so designed and constructed and its contents so limited, and the number of such packages which may be transported together so limited, that:
 - a. five times that number of such undamaged packages would be subcritical in any arrangement if closely reflected by water; and

- b. twice that number of such packages would be subcritical in any arrangement if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, with close reflection by water on all sides of the array and with optimum interspersed hydrogenous moderation unless there is a greater amount of interspersed moderation in the packaging, in which case that greater amount may be considered. The condition of the package shall be assumed to be as described in G., above.

2. The transport index for each Fissile Class II package is calculated by dividing the number 50 by the number of such Fissile Class II packages which may be transported together as determined under the limitations of 1., above. The calculated number shall be rounded up to the first decimal place.

J. Specific Standards for a Fissile Class III Shipment

A package for Fissile Class III shipment shall be so designed and constructed and its contents so limited, and the number of packages in a Fissile Class III shipment shall be so limited that:

1. the undamaged shipment would be subcritical with an identical shipment in contact with it and with the two shipments closely reflected on all sides by water; and
2. the shipment would be subcritical if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, with close reflection by water on all sides of the array and with the packages in the most reactive arrangement and with the most reactive degree of interspersed hydrogenous moderation which would be credible considering the controls to be exercised over the shipment. The condition of the package shall be

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assumed to be as described in G., above. Hypothetical accident conditions different from those specified in this subparagraph may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

III. OPERATING PROCEDURES

A. Establishment and Maintenance of Procedures

1. The shipper shall establish and maintain:
 - a. operating procedures adequate to assure that the determinations and controls required by this appendix are accomplished; and
 - b. regular and periodic inspection procedures adequate to assure that the shipper follows the procedures required by a., above.

B. Assumptions as to Unknown Properties

When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the shipper shall package the fissile material as if the unknown properties have such credible values as will cause the maximum nuclear reactivity.

C. Preliminary Determinations

1. Prior to the first use of any packaging for the shipment of a large quantity of radioactive material or fissile materials, such packaging shall be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce its effectiveness.
2. Prior to the first use of any packaging for the shipment of a large quantity of radioactive material or fissile materials, where the maximum normal operating pressure will exceed 5 pounds per square

inch gauge, the containment vessel shall be tested to assure that it will not leak at an internal pressure 50 percent higher than the maximum normal operating pressure.

3. Packaging shall be conspicuously and durably marked with its model number. Prior to applying the model number, an inspection shall be made to determine that the packaging has been fabricated in accordance with the approved design.

D. Routine Determinations

Prior to each use of a package for shipment of radioactive or fissile material, the shipper shall ascertain that the package with its contents satisfies the applicable requirements of part II of this appendix, including determinations that:

1. the packaging has not been significantly damaged.
2. any moderators and non-fissile neutron absorbers, if required, are as authorized.
3. the closure of the package and any sealing gaskets are present and are free from defects.
4. any valve through which primary coolant can flow is protected against tampering.
5. the internal gauge pressure of the package will not exceed, during the anticipated period of transport, the maximum normal operating pressure.
6. contamination of the primary coolant will not exceed, during the anticipated period of transport, the limits in II, E.1.d., above.

E. Records

The shipper shall maintain for a period of at least 2 years after its generation a record of each shipment of fissile material and of a large quantity of radioactive material, as defined in I, A.6., above, in a single package, showing where applicable:

1. identification of the packaging by model number;

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2. details of any significant defects in the packaging, with the means employed to repair the defects and prevent their recurrence;
3. volume and identification of coolant;
4. type and quantity of material in each package, and the total quantity in each shipment;
5. for each item of irradiated fissile material:
 - a. identification by model number;
 - b. irradiation and decay history to the extent appropriate to demonstrate that its nuclear and thermal characteristics comply with approved conditions;
 - c. any abnormal or unusual condition relevant to radiation safety.
6. date of the shipment;
7. for Fissile Class III, any special controls exercised;
8. name and address of the transferee;
9. address to which the shipment was made; and
10. results of the determinations required by C. and D., above.

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ANNEX 1**NORMAL CONDITIONS OF TRANSPORT**

1. Heat - Direct sunlight at an ambient temperature of 130° F in still air.
2. Cold - An ambient temperature of -40° F in still air and shade.
3. Pressure - Atmospheric pressure of 0.5 times standard atmospheric pressure.
4. Vibration - Vibration normally incident to transport.
5. 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 minutes.
6. Free Drop - Between 1 1/2 and 2 1/2 hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.
7. Corner Drop - 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 foot onto a flat essentially unyielding horizontal surface. This test applies only to packages which are constructed primarily of wood or fiberboard, and do not exceed 110 pounds gross weight, and to all Fissile Class II packagings.
8. Penetration - Impact of the hemispherical end of a vertical steel cylinder 1 1/4 inches in diameter and weighing 13 pounds, dropped from a height of 40 inches onto the exposed surface of the package which is expected to be most vulnerable to puncture.
9. Compression - For packages not exceeding 10,000 pounds in weight, a compressive load equal to either 5 times the weight of the package or 2 pounds per square inch multiplied by the maximum horizontal cross section of the package, whichever is greater. The load shall be applied during a period of 24 hours, uniformly against the top and bottom of the package in the position in which the package would normally be transported.

Free Fall Distance	
Package Weight (pounds)	Distance (feet)
Less than 10,000	4
10,000 to 20,000	3
20,000 to 30,000	2
More than 30,000	1

ANNEX 2**HYPOTHETICAL ACCIDENT CONDITIONS**

1. Free Drop - A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.
2. Puncture - A free drop through a distance of 40 inches striking, in a position maximum damage is expected, the top end of a vertical cylindrical mild steel bar mounted on an essentially unyielding horizontal surface. The bar shall be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than one-quarter inch, and of such a length as to cause maximum damage to the package, but not less than 8 inches long. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.
3. Thermal - Exposure to a thermal test in which the heat input to the package is not less than that which would result from exposure of the whole package to a radiation environment of 1475° F for 30 minutes with an emissivity coefficient of 0.9, assuming the surfaces of the package have an absorption coefficient of 0.8. The package shall not be cooled artificially until 3 hours 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 hours.
4. Water Immersion (fissile material packages only) - Immersion in water to the extent that all portions of the package to be tested are under at least 3 feet of water for a period of not less than 8 hours.

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ANNEX 3

TRANSPORT GROUPING OF RADIONUCLIDES

Element*	Radionuclides***	Group	Element*	Radionuclides***	Group
Actinium (89)	Ac 227	I	Cesium (55)	Cs 131	IV
	Ac 228	I		Cs 134 m	III
Americium (95)	Am 241	I		Cs 134	III
	Am 243	I		Cs 135	IV
Antimony (51)	Sb 122	IV		Cs 136	IV
	Sb 124	III		Cs 137	III
	Sb 125	III	Chlorine (17)	Cl 36	III
Argon (18)	Ar 37	VI		Cl 38	IV
	Ar 41	II	Chromium (24)	Cr 51	IV
	**Ar 41 (Uncompressed)	V	Cobalt (27)	Co 56	III
Arsenic (33)	As 73	IV		Co 57	IV
	As 74	IV		Co 58 m	IV
	As 76	IV		Co 58	IV
	As 77	IV		Co 60	III
Astatine (85)	At 211	III	Copper (29)	Cu 64	IV
Barium (56)	Ba 131	IV	Curium (96)	Cm 242	I
	Ba 133	II		Cm 243	I
	Ba 140	III		Cm 244	I
Berkelium (97)	Bk 249	I		Cm 245	I
Beryllium (4)	Be 7	IV		Cm 246	I
Bismuth (83)	Bi 206	IV	Dysprosium (66)	Dy 154	III
	Bi 207	III		Dy 165	IV
	Bi 210	II		Dy 166	IV
	Bi 212	III	Erbium (68)	Er 169	IV
Bromine (35)	Br 82	IV		Er 171	IV
Cadmium (48)	Cd 109	IV	Europium (63)	Eu 150	III
	Cd 115 m	III		Eu 152 m	IV
	Cd 115	IV		Eu 152	III
Calcium (20)	Ca 45	IV		Eu 154	II
	Ca 47	IV		Eu 155	IV
Californium (98)	Cf 249	I	Fluorine (9)	F 18	IV
	Cf 250	I	Gadolinium (64)	Gd 153	IV
	Cf 252	I		Gd 159	IV
Carbon (6)	C 14	IV	Gallium (31)	Ga 67	III
Cerium (58)	Ce 141	IV		Ga 72	IV
	Ce 143	IV	Germanium (32)	Ge 71	IV
	Ce 144	III	Gold (79)	Au 193	III
				Au 194	III
				Au 195	III
				Au 196	IV
				Au 198	IV
				Au 199	IV

See footnotes at end of table.

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TRANSPORT GROUPING OF RADIONUCLIDES--Continued

Element*	Radionuclide***	Group	Element*	Radionuclide***	Group
Hafnium (72)	Hf 181	IV	Molybdenum (42)	Mo 99	IV
Holmium (67)	Ho 166	IV	Neodymium (60)	Nd 147	IV
Hydrogen (1)	H 3 (see tritium)			Nd 149	IV
Indium (49)	In 113 m	IV	Neptunium (93)	Np 237	I
	In 114 m	III		Np 239	I
	In 115 m	IV	Nickel (28)	Ni 56	III
	In 115	IV		Ni 59	IV
				Ni 63	IV
				Ni 65	IV
Iodine (53)	I 124	III	Niobium (41)	Nb 93 m	IV
	I 125	III		Nb 95	IV
	I 126	III		Nb 97	IV
	I 129	III			
	I 131	III	Osmium (76)	Os 185	IV
	I 132	IV		Os 191 m	IV
	I 133	III		Os 191	IV
	I 134	IV		Os 193	IV
	I 135	IV			
Iridium (77)	Ir 190	IV	Palladium (46)	Pd 103	IV
	Ir 192	III		Pd 109	IV
	Ir 194	IV	Phosphorus (15)	P 32	IV
Iron (26)	Fe 55	IV	Platinum (78)	Pt 191	IV
	Fe 59	IV		Pt 193	IV
Krypton (36)	Kr 85 m	III		Pt 193 m	IV
	** Kr 85 m (uncompressed)	V		Pt 197 m	IV
	Kr 85	III		Pt 197	IV
	** Kr 85 (uncompressed)	VI	Plutonium (94)	Pu 238 (F)	I
	Kr 87	II		Pu 239 (F)	I
	** Kr 87 (uncompressed)	V		Pu 240	I
Lanthanum (57)	La 140	IV		Pu 241 (F)	I
				Pu 242	I
Lead (82)	Pb 203	IV	Polonium (84)	Po 210	I
	Pb 210	II			
	Pb 212	II	Potassium (19)	K 42	IV
Lutetium (71)	Lu 172	III		K 43	III
	Lu 177	IV			
Magnesium (12)	Mg 28	III	Praseodymium (59)	Pr 142	IV
				Pr 143	IV
Manganese (25)	Mn 52	IV	Promethium (61)	Pm 147	IV
	Mn 54	IV		Pm 149	IV
	Mn 56	IV			
Mercury (80)	Hg 197 m	IV	Protactinium (91)	Pa 230	I
	Hg 197	IV		Pa 231	I
	Hg 203	IV		Pa 233	II
Mixed Fission Products MFP		II			

See footnotes at end of table.

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SAFETY STANDARDS FOR THE PACKAGING
OF RADIOACTIVE AND FISSILE MATERIALS

AEC Appendix 0529
Annex 3

TRANSPORT GROUPING OF RADIONUCLIDES--Continued

Element*	Radionuclide***	Group	Element*	Radionuclide***	Group
Radium (88)	Ra 223	II	Tantalum (73)	Ta 182	III
	Ra 224	II			
	Ra 226	I	Technetium (43)	Tc 96 m	IV
	Ra 228	I		Tc 96	IV
Radon (86)	Rn 220	IV		Tc 97 m	IV
	Rn 222	II		Tc 97	IV
Rhenium (75)	Re 183	IV		Tc 99 m	IV
	Re 186	IV		Tc 99	IV
	Re 187	IV	Tellurium (52)	Te 125 m	IV
	Re 188	IV		Te 127 m	IV
	Re Natural	IV		Te 127	IV
Rhodium (45)	Rh 103 m	IV		Te 129 m	III
	Rh 105	IV		Te 129	IV
Rubidium (37)	Rb 86	IV	Terbium (65)	Te 131 m	III
	Rb 87	IV		Te 132	IV
	Rb Natural	IV			
Ruthenium (44)	Ru 97	IV		Tb 160	III
	Ru 103	IV	Thallium (81)	Tl 200	IV
	Ru 105	IV		Tl 201	IV
	Ru 106	III		Tl 202	IV
Samarium (62)	Sm 145	III		Tl 204	III
	Sm 147	III	Thorium (90)	Th 227	II
	Sm 151	IV		Th 228	I
	Sm 153	IV		Th 230	I
Scandium (21)	Sc 46	III		Th 231	I
	Sc 47	IV		Th 232	III
	Sc 48	IV		Th 234	II
Selenium (34)	Se 75	IV		Th Natural	III
Silicon (14)	Si 31	IV	Thulium (69)	Tm 168	III
				Tm 170	III
Silver (47)	Ag 105	IV		Tm 171	IV
	Ag 110 m	III	Tin (50)	Sn 113	IV
	Ag 111	IV		Sn 117	III
Sodium (11)	Na 22	III		Sn 121	III
	Na 24	IV		Sn 125	IV
Strontium (38)	Sr 85 m	IV	Tritium (1)	H 3	IV
	Sr 85	IV		H 3 (as gas, as luminous paint or absorbed on solid material)	VII
	Sr 89	III	Tungsten (74)	W 181	IV
	Sr 90	II		W 185	IV
	Sr 91	III		W 187	IV
	Sr 92	IV			
Sulphur (16)	S 35	IV			

See footnotes at end of table.

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TRANSPORT GROUPING OF RADIONUCLIDES--Continued

Element*	Radionuclide***	Group	Element	Radionuclide***	Group
Uranium (92)	U 230	II	Ytterbium (70)	Yb 175	IV
	U 232	I			
	U 233 (F)	II	Yttrium (39)	Y 88	III
	U 234	II		Y 90	IV
	U 235 (F)	III		Y 91 m	III
	U 236	II		Y 91	III
	U 238	III		Y 92	IV
	U Natural	III		Y 93	IV
	U Enriched (F)	III			
Vanadium (23)	U Depleted	III	Zinc (30)	Zn 65	IV
	V 48	IV		Zn 69 m	IV
	V 49	III		Zn 69	IV
Xenon (54)	Xe 125	III	Zirconium (40)	Zr 93	IV
	Xe 131 m	III		Zr 95	III
	** Xe 131 m (uncompressed)	V		Zr 97	IV
	Xe 133	III			
	** Xe 133 (uncompressed)	VI			
	Xe 135	II			
	** Xe 135 (uncompressed)	V			

* Atomic Number shown in parentheses.

** Uncompressed means at a pressure not exceeding one atmosphere.

*** Atomic weight shown after the radionuclide symbol.

m-Metastable state.

(F) Fissile Material.

ANNEX 4

TESTS FOR SPECIAL FORM MATERIAL

1. Free Drop - A free drop through a distance of 30 feet into a flat essentially unyielding horizontal surface, striking the surface in such a position as to suffer maximum damage.
2. Percussion - Impact of the flat circular end of a 1-inch diameter steel rod weighing 3 pounds, dropped through a distance of 40 inches. 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 1-inch thick, supported by a smooth essentially unyielding surface.
3. Heating - Heating in air to a temperature of 1475° F and remaining at that temperature for a period of 10 minutes.
4. Immersion - Immersion for 24 hours in water at room temperature. The water shall be at pH 6 - pH 8, with a maximum conductivity of 10 micromhos per centimeter.

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