5. SHIELDING ANALYSIS

5.1 Discussion and Results

5.1.1 Applicable Regulatory Criteria

Packages such as the modified BMI-1 cask which are transported in a vehicle assigned for the sole use of the consignor, and unloaded by the consignee from the transport vehicle in which they are originally loaded have radiation dose-rate limits [§173.393(j) of 49 CFR] under normal conditions of:

(1) 1,000 millirem per hour at 3 feet from the external surface of the package (closed transport vehicle only)

(2) 200 millirem per hour at any point on the external surface of the car or vehicle (closed transport vehicle only)

(3) 10 millirem per hour at 6 feet from the external surface of the car or vehicle and

(4) 2 millirem per hour in any normally occupied position in the car or vehicle (except in the case of private motor carriers).

Packages such as the modified BMI-1 cask which are transported by a commercial carrier have radiation dose-rate limits [§173.393(i) of 49 CFR] under normal conditions of:

(1) 200 mr/hr at any point on the external surface of the package and

(2) The transport index does not exceed 10 (i.e., 10 mr/hr at 3 feet from the surface of the package).

In addition to the above radiation limits for normal shipping conditions, the package must meet the shielding requirement for hypothetical accident conditions which is listed in §71.36(a)(1) of 10-CFR-71 as:

"The reduction of shielding would not be sufficient to increase the external dose rate to more than 1,000 milliroentgens per hour or equivalent at 3 feet from the external surface of the package."

5.1.2 Design Features

Lead shielding is provided in the BMI-1 cask to limit the surface dose to less than 200 mr/hr and the dose at 3 feet from the surface of the cask to less than 10 mr/hr. As required by the regulations, the dose will be measured before shipment to verify that these limits are not exceeded. The shielding for the sides of the cask is provided by 8 inches of lead and 0.875 inch of steel, where 0.25 inch of steel is in the inside cylinder and 0.025 inch of steel is in the outside cylinder and fire shield. The bottom is shielded by 7.75 inches of lead and 1.875 inches steel, where 0.75 inch of steel is in the inside plate and 1.125 inches is in the outside plate. The top is also shielded by 7.75 inches of lead and 1.875 inches of steel, where 0.75 inch of steel is in the inside plate and 1.125 inches is in the outside plate and fire shield.

5.2 Source Specification

5.2.1 Description of Radiation Sources

Types and quantities of radioactive materials analyzed for shipment in the modified BMI-1 cask fall within the transport groups and corresponding curie limits indicated in Tables 5.2 and 5.3. These analyses were carried out to determine the maximum quantity of radioactive materials in each transport group which, when transported in the BMI-1 container, result in compliance with the regulatory dose-rate limits.

5.2.2 Source Radiation Type and Intensity

Table 5.4 lists the radiation characteristics of the radionuclides which control the curie limit of each of the transport groups listed in Tables 5.2 and 5.3.

5.3 Model Specification

5.3.1 Source Geometry

All radiation sources were assumed to be right-circular cylinders 15.5 in. in diameter by 54 in. high with an average density equivalent to that of aluminum* (i.e., 2.7 g/cm^3).

5.3.2 Description of Shield

Figure 5.1 illustrates the shield configurations utilized in the dose-rate calculations for the modified BMI-1 cask under normal and hypothetical accident conditions. For purposes of analysis, the shield was assumed to consist of an annular lead cylinder 8.50** in. thick with end plates of 8.75-in. thickness.

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* This material was used for purposes of simulating the average packing density of approximately 150 lb/ft^3 (2.4 g/cm^3) which is based on past shipping experience with the NECO-3 cask.

** For photon energies above 1.5 Mev and below 4 Mev, the 1-in. steel thickness is equivalent to about 0.75 in. of lead.

TABLE 5.1. SUMMARY OF MAXIMUM DOSE RATES
(mR/hr)

	Package Surface			3 Feet from Surface of Package		
	Side	Top	Bottom	Side	Top	Bottom
Normal Conditions						
Gamma					10	
Neutron						
Total						
Hypothetical Accident Conditions						
Gamma					<1000	
Neutron						
Total						
10 CFR Part 71 Limit	--	--	--	1000	1000	1000

TABLE 5.2 RADIONUCLIDES AND ASSOCIATED CURIE LIMITS PLANNED FOR TRANSPORT IN MODIFIED BMI-1 CASK (SOLE USE OF VEHICLE)

Transport Group*	Quantity (in curies)
I.	1,000
II.	8,120
General mixed fission products . . .	Unlimited**
III.	4,960
IV.	11,070
V.	8,120
VI and VII	800,000

* As defined in §173.390 of 49 CFT and Appendix C of 10-CFR-71.

** Limit will be imposed by dose-rate limits specified in §173.393(j) of 49 CFR.

TABLE 5.3 RADIONUCLIDES AND ASSOCIATED CURIE LIMITS PLANNED FOR TRANSPORT IN MODIFIED BMI-1 CASK (SHIPMENTS BY COMMERCIAL CARRIER)

Transport Group*	Quantity (in curies)
I.	1,000
II.	2,520
General mixed fission products . . .	Unlimited**
III.	1,540
IV.	3,440
V.	2,520
VI and VII	800,000

* As defined in §173.390 of 49 CFR and Appendix C of 10-CFR-71.

** Limit will be imposed by dose-rate limits specified in §173.393(i) of 49 CFR.

TABLE 5.4 RADIATION CHARACTERISTICS OF LIMITING RADIONUCLIDES

Transport Group	Limiting Radionuclide*	Radiation Type	Radiation Energy and Intensity	Limiting Radiation
I	Ac-228	β⁻, γ	β: 2.18 (10%), 1.85 (9.6%), 1.7 (6.7%), 1.11 (53%), 0.64 (7.6%), 0.45 (13%) γ: 1.64, 1.59, 1.095, 1.035, 0.965, 0.907, 0.458, 0.410, 0.336, 0.232, 0.184, 0.174, 0.1275, 0.113, 0.0978, 0.0781, 0.0568	1.64 and 1.59 Mev
II	Kr-87	β⁻, γ	β: 3.8 (70%), 3.3 (5%), 1.3 (25%) γ: 2.57 (25%), 0.85 (10%), 0.403 (50%)	2.57 γ's
III	Co-56	β⁺, γ	β+: 1.5 (18%) γ: 3.47 (1%), 3.25 (12%), 2.99 (1%), 2.6 (16%), 2.02 (11%), 1.75 (18%), 1.36 (5%), 1.24 (71%), 1.03 (16%), 0.845 (100%), 0.51 (36%)	3.47 - 1.36 Mev γ's
IV	Cl-38	β⁻, γ	β: 4.81 (53%), 2.77 (16%), 1.11 (31%) γ: 2.15 (50%), 1.6 (40%)	2.15, 1.6 γ's
V	Kr-87**	β⁻, γ	β: 3.8 (70%), 3.3 (5%), 1.3 (25%) γ: 2.57 (25%), 0.85 (10%), 0.403 (50%)	2.57 Mev γ's
VI and VII	Kr-85	β⁻, γ	β: 0.672 (100%) γ: 0.517 (0.7%)	0.672 β's

* Radionuclide which results in the highest external dose rate for a given curie level.

** Uncompressed.

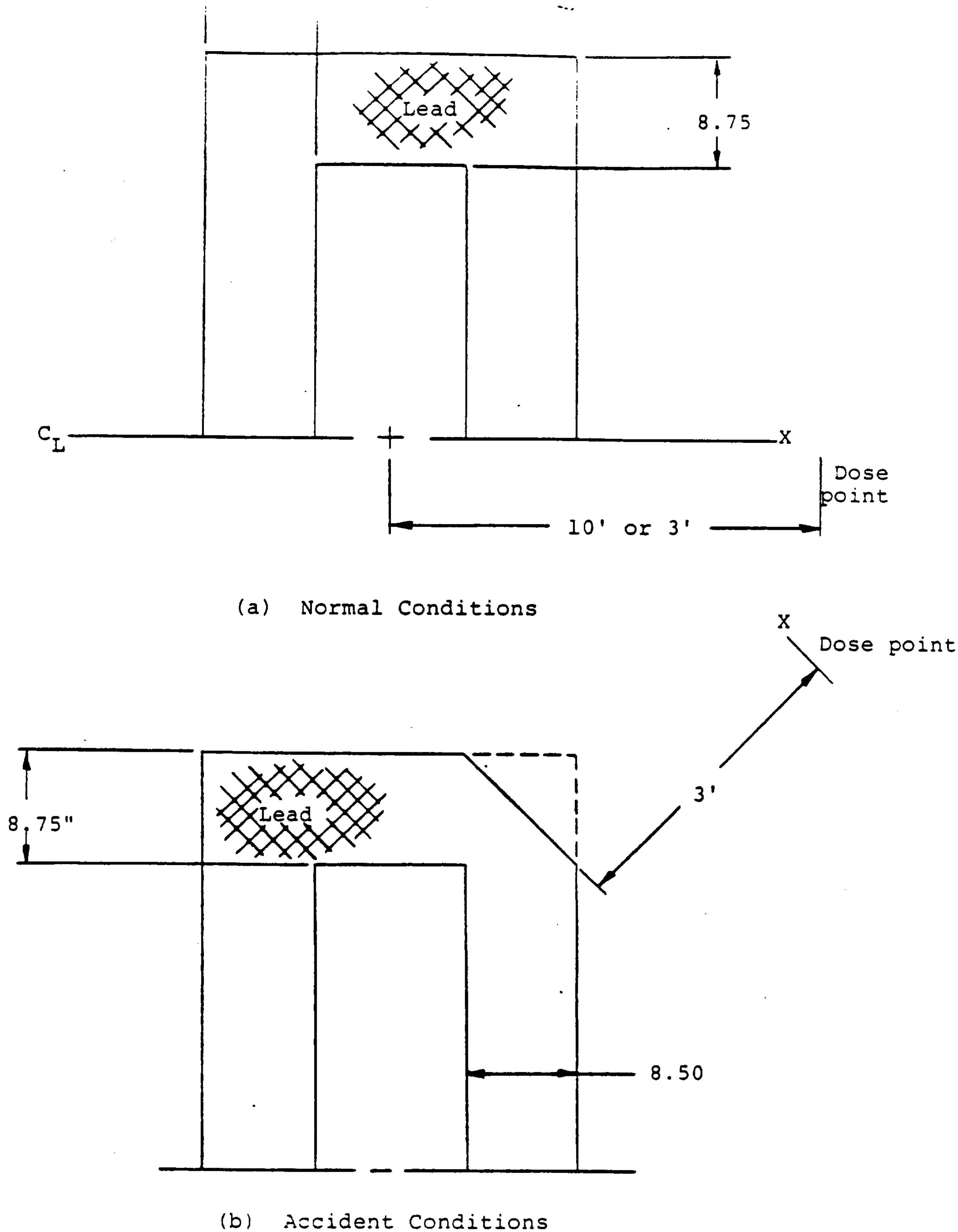

FIGURE 5.1 SHIELD CONFIGURATIONS UTILIZED IN THE DOSE RATE CALCULATIONS FOR THE MODIFIED BMI-1 CASK

5.4 Shielding Evaluation

5.4.1 Dose Rate Under Normal Conditions

5.4.1.1 General Contents

Point kernel integration techniques were utilized to calculate the gamma dose rates at 6 ft (the 10 mr/hr value at this distance is the most stringent of the aforementioned limits for sole use of vehicle transport) from the transport vehicle and 3 ft from the cask (the 10 mr/hr value at this distance is the most stringent of the aforementioned limits for shipment by commercial carrier) for various quantities of radionuclides. The basic equation employed was:

$$D_\gamma = \frac{K(E)B(E,\mu t)\ S_\gamma\ e^{\mu(E)t}}{4\pi r^2}$$

where:

D_γ = gamma dose rate at distance r from source, mrem/hr

K(E) = flux-to-dose conversion factor as a function of photon energy, $\frac{\text{mrem/hr}}{\text{photon/cm}^2\text{-sec}}$

B(E),μt) = dose buildup factor as a function of photon energy relaxation length in the shield

S_γ = source strength of photons with energy E, photons/ cm^3-sec

μ(E) = linear attenuation coefficient for photons of energy E, cm^{-1}

t = shield thickness, cm

r = source-to-dose point separation distance, cm.

Parameters used in the above equation for the shield and source materials are listed in Table 5.5. The curie limits for each of the transport groups listed in Tables 5.2 and 5.3 were obtained by varying the source quantity and calculating the dose rate at 6 ft from the vehicle and at 3 ft from the cask until a maximum dose rate of 10 mr/hr was reached. Exceptions to this procedure were made for Transport Group I as well as for VI and VII in which cases curie limits of 1000 Ci and 800,000 Ci, respectively, were considered sufficiently high although the corresponding dose rates at the aforementioned distance were well below the allowable limit.

TABLE 5.5 LINEAR ATTENUATION COEFFICIENT OF THE SOURCE AND SHIELD MATERIALS

Group	Energy, Mev	μ_{source}, cm^{-1}	μ_{Pb}, cm^{-1}
1	4.0	0.071	0.474
2	3.0	0.081	0.476
3	2.0	0.099	0.500
4	1.55	0.110	0.585
5	1.0	0.140	0.729

Actually, the curie limit for Transport Groups VI and VII was established by the restrictions on internal heat load. That is, 800,000 Ci of Kr-85 corresponds to about 1500 watts of decay heat which is consistent with the value utilized for heat-transfer analyses (see thermal analysis section).

In the case where the shipper wishes to ship a mixture of radionuclides classified under various transport groups, the following conditional equation may be used to determine the shipping limits:

$$\sum_{K=1}^{7} \frac{Ci_K}{Limit_K} \leq 1 \quad ,$$

where:

Ci_K = number of curies in Transport Group K to be shipped

$Limit_K$ = curie limit (maximum shipment) for the K^{th} transport group

K = an index number assigned to each transport group (i.e., K=1 implies Transport Group I, etc.).

It should be noted that the curie limits referenced above are the "large-quantity" limits listed in Tables 5.2 and 5.3. These limits are based on a safety-related shielding analysis and, in several cases, involve inert gases* which will not be shipped); nevertheless, these limits will be maintained though overly restricted in the case of nongaseous elements.

5.4.1.2 Specific Contents

(A) Co-60

Gamma dose rates at the cask surface and a position 3 feet from the surface in the axial midplane were estimated to be 356 and 35.6 mrem/hour, respectively, for a 40,000-curie Co-60 source. These estimates were based on an assumed point isotropic source with no self-absorption. If the copper basket is placed around the Co-60 source, these dose rates are decreased to 12.2 and 1.22 mrem/hour, respectively.

* See Table 5.4 - Transport Groups II, V, VI, and VII.

(B) Fermi Fuel

(paragraph deleted)

(C) TRIGA Fuel

The stainless steel elements were measured at 1.6 R/hr at 10 ft in air in the summer of 1970, after approximately 6 months' cooling time. Assuming the Way-Wigner relation for fission product decay to be valid for this case indicates a cooling factor of 3.4, or that the current (December, 1971) activity of the SS elements is about 0.47 R/hr at 10 ft. The Al-clad elements have been measured at 0.30 R/hr at 10 ft in air (5-day cooling period). Using 0.5 R/hr as the maximum activity for one element at 10 ft, 38 elements in air would be 19.0 R/hr at 10 ft. (This assumes no self-attenuation). The BMI-1 cask has 8 in. of Pb shielding which is more than five 10th-value thicknesses. (The tenth value thickness of 1.5 MEV gamma is 1.5 in.). The 38 fuel assemblies surrounded by 8 in. of lead would have an activity of ∿0.2 mr/hr at 10 feet or ∿0.6 mr/hr at 3 ft. These values are well within the regulations.

(1) References to Section 5. are found in Section 5.5.1.

(Paragraph deleted)

(D) EPRI Crack Arrest Capsules

The activity of the capsules was determined with the computer code ORIGEN using the material quantities given in Table 1.2 and the irradiation parameters given in Table 5.6.

The results indicated that shortly after discharge ($\sim$30 to 60 min), the activity is about 5200 Ci due entirely to isotopes in Transport Group IV. Since the present license for the BMI-1 cask permits up to 11,000 Ci of activity of materials in Transport Group IV the activity is within permissible levels.

TABLE 5.6. IRRADIATION PARAMETERS FOR EPRI CRACK ARREST CAPSULES

Target Fluence (E>1.0 Mev)	- (10^{19}) n/cm^2
Fast Flux (E>1.0 Mev)	2.2 (10^{12}) n/cm^2-sec
Thermal Flux	1.8 (10^{12}) n/cm^2-sec
Fission in Fission Monitors	
U^{238}	1.2 (10^{14}) f/dosimeter
Np^{237}	1.5 (10^{15}) f/dosimeter

5.4.2. Dose Rate Under Accident Conditions

Accident conditions which can significantly alter the dose rate external to the cask are (a) the fire condition, (b) the 30-ft corner drop, and (c) the side drop in which case gross displacement of the lead occurs. Each of these cases is discussed in the following paragraphs.

5.4.2.1 Standard Fire

Since the amount of lead which could escape from the cask due to a fire is less than (i.e., 3 to 4 percent* of the melt or 574 in.^{3} would melt) the amount displaced in the 30-ft corner drop, the resulting dose rate at 3 ft from the surface of the cask would be below 1 R/hr.

* Due to expansion of the lead upon melting.

A drop in the lead level of 3 inches at the corner of the cask, will not increase the gamma dose above tolerable levels. Consider only the 2.18 Mev gamma dose from Pr-144 (which accounts for 70 percent of the dose) where μ_{Pb} = 0.493 cm and $\mu_{Fe} = 0.307\ cm^{-1}$. For the most weakly shielded ray, the lead thickness is 3.25 in. and the steel thickness is 2.25 in. Compared to the side of the cask, this is 4.75 in. less lead and 1.5 in. more steel. The increase in dose along the most weakly shielded ray is proportional to

$$e^{4.75 \times 2.54 \times 0.493}\, e^{-1.5 \times 2.54 \times 0.307} = e^{4.78} = 119.$$

The increase in dose also depends on the angle subtended at one meter by the weakly shielded region. The ratio of the angles subtended by the weakly shielded region to the total source is about 1:7. Under normal conditions, the maximum dose one meter from the side of the cask is 13 mr/hr. Considering only the reduction in shielding, the dose at the upper corner after a fire would be less than 119 x 13 = 1.55 r/hr. When the angles subtended by the radiation sources are considered, the dose will be less than 1 r/hr at one meter after a fire.

5.4.2.2 Corner Drop

As shown in the section on structural integrity analysis, the maximum deformation of the lead shield at the corner resulting from the 30-ft drop onto an unyielding surface is 5.63 in. which results in a residual lead thickness of 4.53 in. at the point closest to the source region. This deformation results in a maximum dose rate of less than 1 R/hr at 3 ft from the surface of the cask for the curie levels listed in Tables 5.2 and 5.3. The analysis was conducted by numerically integrating* the previously defined point kernel over the source volume while incorporating the effects of the irregular shield geometry.

5.4.2.3 Side Drop

The maximum impact load on the side of the cask will cause 1.44 inches deformation of the lead. Since this is less than the 1.65 inches lead melted during the fire accident, which was previously demonstrated to be safe, the dose rate 3 feet from the surface after impact will be less than 1.0 rem/hour.

5.5 Appendix

5.5.1 References

(1) Perkins, J. F., and King, R. W., Nuclear Science and Engineering, Vol. 3, p 725, 1953.

(2) Rockwell, T., "Reactor Shielding Design Manual", TID-7004, 1956.

(3) Private communication from Dr. Martin N. Haas, Associate Director, Nuclear Science and Technology Facility, State University of New York at Buffalo to Dr. C. E. Williams, Manager, Idaho Operations Office, U.S. ERDA, 550 Second Street, Idaho Falls, Idaho 83401.

* By utilizing a simple three-dimensional code under development at Battelle.

5.5.2 Shielding Evaluation of BMI-1 Cask with Eight MURR Fuel Elements

The Shielding Analysis from the Safety Analysis Report (SAR), Revision H, for the Model BMI-1 Shipping Cask is attached. MURR spent fuel is shipped in an exclusive-use vehicle. Of the two regulatory limits for this case, (1) 200 millirem/hour at the cask surface, and (2) 10 millirem/hour at any point two meters from the exclusive-use vehicle outer surface, per 10 CFR 71.47(a)(c), the SAR considers (in Section 5.4.1.1) that limit (2) is more stringent. The calculation shows (see the paragraph above SAR Table 5.5) that with a source of 800,000 curies (Ci) in the cask, limit (2) is not exceeded.

The most recent ten spent fuel shipments from MURR, in 1986-87-89, had activities ranging from 98,000 Ci to 268,000 Ci with an average of 184,000 Ci. (These were calculated using the ORIGEN-2 code with a model that included the cycling of elements in and out of twenty 1-week cores over 1.5-2. years). Future shipments will be just like these. In particular, the activity of any shipment will be less than 400,000 Ci. Therefore, based on the BMI-1 SAR calculations, the 10 CFR 71.47 limits will not be exceeded.

Support for this conclusion is available from BMI-1 use experience. Mr. Robert A. Strack, Nuclear Project Engineer for Cintichem, Inc., owner of the BMI-1 cask, states that with sources of the order of 500 watts, dose rates two meters from the cask did not get above background.

The external dose rate requirement of 10 CFR 71.51(a)(2) for accident conditions is less than one rem per hour at one meter from the external surface of the package. When lead melt 1.65 inches from the outside Pb surface and loss of coolant are included, there is no change in previously calculated external dose rates. Melted lead is not lost (SAR Section 3.5.4.1) so continues to serve its shielding function. See Item 6 of our Thermal Evaluation. Coolant has negligible shielding effect. Loss of coolant could only affect dose rates if fission products were in the coolant. We have explained in the Structural Evaluation that escape of fission products from fuel plate cladding is not credible in an accident. SAR Section 5.4.2.1 explains that in the standard fire, the dose rate will be less than 1 rem/hr at one meter from the cask.

In a hypothetical fire accident, cask containment integrity is maintained. SAR Section 5.4.2.1 explains that the dose rate at 1 meter from the cask surface remains below 1 rem/hr under accident conditions. Thus the dose rate requirement of 10 CFR 71.51(a)(2) is met.

5.5.3 Shielding Evaluation of BMI-1 Cask with Eight Spent MITR Fuel Elements

Theory

For a finite, cylindrical, homogenous and volumetric radioactive source stored in a cylindrical shielding cask, the gamma flux at a point P on the surface of the shielding cask can be calculated as equivalent to flux contribution from a vertical line source inside the cask. The line source is located at a "self-absorption" distance, d, within the surface of the source material (see Figure 1). For ease of understanding, simplicity of calculations, and also because of the geometrical symmetry, the cylindrical shielding is always approximated by an infinite slab. This approximation tends to generate conservatism in the results but it is reasonably acceptable [1]. The following figure shows the relation of d, P and other parameters that will be used in the calculation.

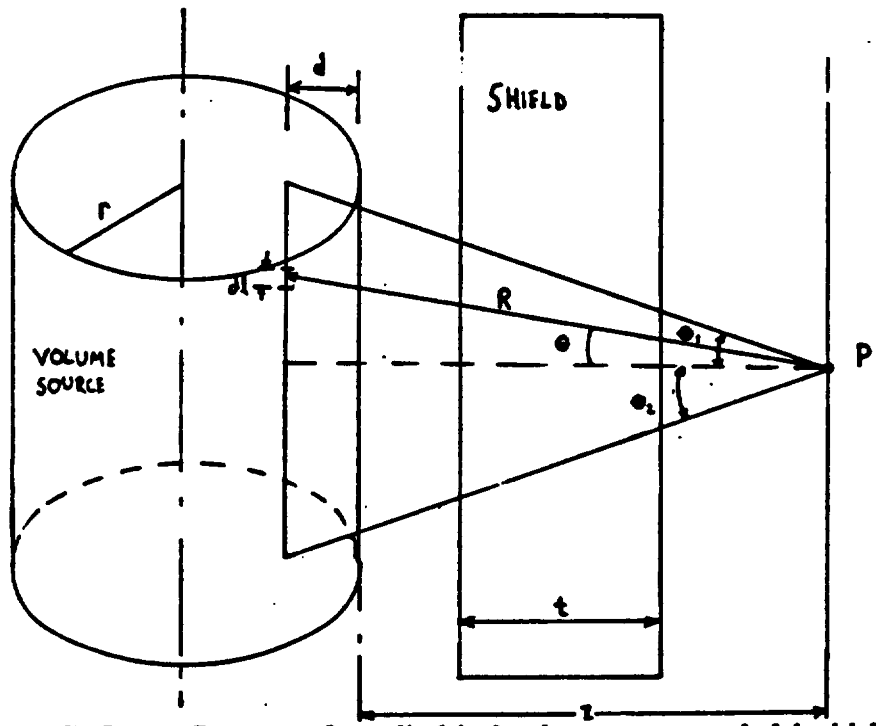

FIG. 1: Geometry for cylindrical volume source and slab shield at side for calculation of detector response from a shielded cylindrical source.

The response of the detector at P due to the infinitesimal source element dl is given by

$$\Phi_d\text{, flux due to element dl} = Sv\, G(R)B(\mu R)\pi r^2 dl \qquad (1)$$

where Sv is the volume source strength [particles/cm^2-s] ;

$$G(R)\text{, the Point-Kernel} = \frac{e^{-\mu R}}{4\pi R^2}\ ;$$

and $B(\mu R)$ is the Buildup Factor.

Thus equation (1) can be written as:

$$\Phi_{d\ell} = \frac{S_v B(\mu R)\pi r^2 e^{-\mu R}}{4\pi R^2}dl \qquad \text{(II)}$$

As illustrated in Figure 2,

$$R = d\sec\theta + z\sec\theta\ ;$$

$$\mu R = \mu_v d\sec\theta + \mu t\sec\theta;$$

(μ_v is the linear attenuation coefficient of the volume source material. μ is the linear attenuation coefficient of the slab shield material.)

and
$$d\ell = R\sec\theta\, d\theta$$
$$= (d+z)\sec^2\theta\, d\theta$$

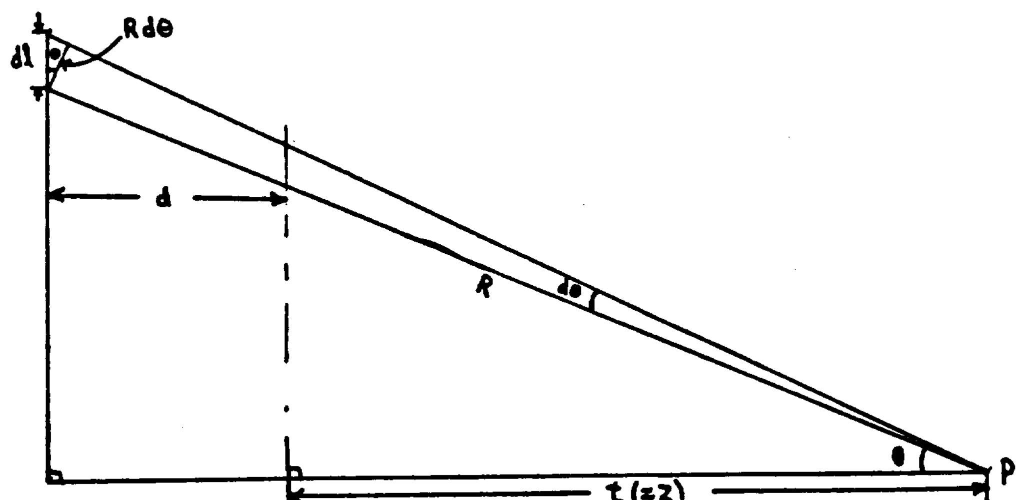

Fig. 2. Calculation of R, distance between the detector and the source element dl.

Thus, Equation (II) can be written as:

$$\Phi_{d\ell} = \left(\frac{S_v r^2}{4(d+z)}B(\mu R)e^{-}(\mu_v d + \mu t)\sec\theta\right)d\theta \qquad \text{(III)}$$

In calculations of doses from source arrays, it is convenient to approximate the build-up factor, $B(\mu R)$, by an analytic function. According to Etherington et.al. [2], build-up factor for γ rays can be well approximated by the Taylor form as a sum of two exponential functions, as follows:

$$B(\mu R) = A_1 e^{-\alpha\mu R} + A_2 e^{-\beta\mu R}$$

where A_1, $A_2(=1-A_1)$, α and β are constants.

Thus, $B(\mu R) = A_1 e^{-\alpha(\mu_v d\sec\theta + \mu t\sec\theta)} + A_2 e^{-\beta(\mu_v d\sec\theta + \mu t\sec\theta)}$

$$= A_1 e^{-\alpha(\mu_v d + \mu t)\sec\theta} + A_2 e^{-\beta(\mu_v d + \mu t)\sec\theta}$$

and Equation (III) can be expressed as:

$$\Phi_{d\ell} = \frac{S_v r^2}{4(d+z)} [A_1 e^{-(1+\alpha)(\mu v d + \mu t)\sec\theta} + A_2 e^{-(1+\beta)(\mu v d + \mu t)\sec\theta}] \qquad \text{(IV)}$$

The gamma flux at point P from the equivalent line source of the volume source is obtained by integration from $\theta = 0$ to $\theta = \theta_1$ and from $\theta = 0$ to $\theta = \theta_2$; thus, the response of the detector to the total flux is:

$$\Phi_\ell = \frac{S_v r^2}{4(d+z)} \left\{ A_1 \left[\int_0^{\theta_1} e^{-(1+\alpha)(\mu_v d + \mu t)\sec\theta} d\theta + \int_0^{\theta_2} e^{-(1+\alpha)(\mu_v d + \mu t)\sec\theta} d\theta \right] + \right.$$

$$\left. A_2 \left[\int_0^{\theta_1} e^{-(1+\beta)(\mu_v d + \mu t)\sec\theta} d\theta + \int_0^{\theta_2} e^{-(1+\beta)(\mu_v d + \mu t)\sec\theta} d\theta \right] \right\} \qquad \text{(V)}$$

where Φ_ℓ is γ flux at point P due to the line source of length ℓ behind the slab shield.

Equation (V) can also be expressed in terms of the Sivert's Integral as follows:

$$\Phi_\ell = \frac{S_v r^2}{4(d+z)} \{ \quad A_1[F(\theta_1, (1+\alpha)(\mu_v d + \mu t)) + F(\theta_2, (1+\alpha)(\mu_v d + \mu t))]$$

$$+ A_2[F(\theta_1, (1+\beta)(\mu_v d + \mu t)) + F(\theta_2, (1+\beta)(\mu_v d + \mu t))]\} \qquad \text{(VI)}$$

In the following dose calculations, point P is chosen to be at a point on the surface of the fuel cask such that $\theta_1 = \theta_2$. Equation (VI) then becomes:

$$\Phi_{\ell} = \frac{S_v r^2}{2(d+z)} \left[A_1 F(\theta,(1+\alpha)(\mu_v d+\mu t)) + A_2 F(\theta,(1+\beta)(\mu_v d+\mu t))\right] \quad \text{(VII)}$$

The conversion between gamma flux and exposure dose rate is given by the following relation [3]:

$$\dot{D}[R/hr] = k(E) E \Phi_{\ell}(E) \quad \text{(VIII)}$$

where $k(E)$ is the conversion factor from gamma flux to exposure dose rate;

and E is gamma energy in MeV.

The following sections will show all the calculation steps in obtaining the γ dose at the vertical outer surface of the BMI-1 shipping cask.

Calculation Procedures

The geometry of the cylindrical cask is listed as follows (also see Figure 1):

ℓ = 86.875 cm

r = 19.375 cm

$z \cong t = 20$ cm

Several parameters are predetermined so that Equation VII can be used to calculate the gamma flux at the surface of the BMI-1 cask. The calculations for these parameters are presented in the following sections:

(A) d, the self-absorption distance in the source material;

(B) S_V, the activity per unit volume of the source.

(C) The constant coefficients of Equation VII: A_1, A_2, α, β and the Sivert's Integral, F_s.

The gamma flux, Φ_{ℓ}, could then be determined after these parameters were obtained.

A. Determination of d, the Self-Absorption Distance in the Source Material

Table Z4 of Reference [3] was used to determine the self-absorption distance. Several parameters were required for the use of the Table. These included the Z/r-ratio, the total optical thickness of the shield b_1, and μ_s, the linear attenuation coefficient of the source material.

The Z/r-ratio is equal to 1.032. The total optical thickness of the shield, b_1 ($=\mu_{lead} * t$), is equal to 15.649. The linear attenuation coefficient of the source material is determined as follows.

The source material in this case is not an homogenous mixture of radioactive material. It is rather made up of the stainless steel University of Missouri basket with eight radial slots into which a total of eight spent fuel elements can be fitted. The stainless steel basket has the same vertical height as the inner portion of the BMI-1 cask. The spent fuel elements will only occupy the lower 68% of the length of the basket. The space in between is filled with air at atmospheric pressure. A homogenization of the cask contents would facilitate the calculation here and the approximation will not underestimate the outcome. A homogenized mixture of the basket and the spent fuel elements is composed mainly of stainless steel, aluminum and a relatively small amount of fissionable materials. The total linear attenuation coefficient of radiation source in a homogenized mixture of M materials is given as:

$$\mu_s = \sum_{m=1}^{M} \mu_m \left(\frac{\rho_m}{\rho_m^*}\right)$$

where μ_1 is the linear attenuation coefficient for stainless steel, μ_2 is for aluminum; and μ_3 for fissionable materials; ρ_m^* is the normal density of material m; ρ_m is the density of material m in the homogenized mixture.

The total inner volume of the BMI cask is 102453.79 cm^3. The volume of the stainless steel of the University of Missouri basket is estimated to be 14728.027 cm^3. The normal density of stainless steel is 7.978 g/cm^3. Thus ρ_1 is calculated to be 1.147 g/cm^3. For 1 MeV gamma radiation, μ_1 is equal to 0.5 cm^{-1}.

Aluminum in the source material, from the eight spent fuel elements, amounts to a total volume of 11348.414 cm^3. The normal density of aluminum is 2.699 g/cm^3. Thus ρ_2 is equal to 0.299 g/cm^3. For 1 MeV gamma radiation attenuation, μ_2 is equal to 0.166 cm^{-1}.

The fissionable materials, contained in the eight spent fuel elements, amount to a total volume of 3297.833 cm^3. The theoretical density of uranium dioxide is used as the normal density of the fissionable material. It is 10.958 g/cm^3. Thus ρ_3 is 0.328 g/cm^3. For 1 MeV gamma ray attenuation, μ_3 is equal to 1.41 cm^{-1}.

Therefore, the total linear attenuation coefficient, μ_s is given as:

$$\mu_s = \mu_{ss} \times \left(\frac{\rho_{ss}}{\rho_{ss}^*}\right) + \mu_{al} \times \left(\frac{\rho_{al}}{\rho_{al}^*}\right) + \mu_u \times \left(\frac{\rho_u}{\rho_u^*}\right) = 0.1325\ cm^{-1}.$$

Using the calculated quantities of the Z/r-ratio, b_1, μ_s and referring to Table Z4 [3], the self absorption distance in the source material, d, is determined to be 14.339 cm.

B. <u>Determination of Sv, the Activity per Unit Volume of the Source</u>

The fission product activity A(t) in Curies after t days due to one fission is given as [4]:

$$A(t) = 1.027 \times 10^{-16} t^{-1.2}$$

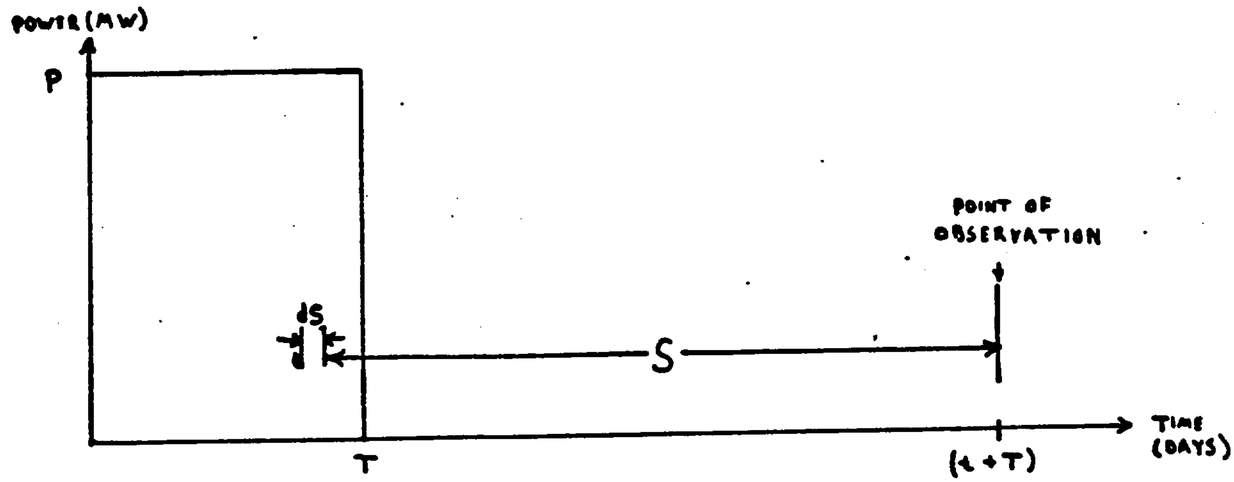

Figure 3: Fuel Element Operating History for the Determination of Fission Product Activity.

The total fission product activity due to all twenty-five fuel elements in the reactor core operated at constant power level P MW for T days and thereafter shutdown and cooled for t days (see Figure 3) can be calculated as:

$$A_T(t) = 3\times10^{10}\left[\tfrac{\#fissions}{Watt}\right] \times 8.64\times10^{4}\left[\tfrac{sec}{day}\right] \times 10^{6}\left[\tfrac{W}{MW}\right] \times 1.027\times10^{-16} \times P \int_t^{t+T} S^{-1.2}\,dS$$

$$\therefore A_T(t) = 1.33\times10^{-6} \times P\left[t^{-0.2} - (t+T)^{-0.2}\right] \qquad \text{(IX)}$$

Of all the spent fuel elements to be loaded into the BMI cask, the element 4M12 has the longest operating history (99198.31 MWH). Hence it is the most reactive 4M element that is to be loaded. For conservatism, all spent fuel elements under consideration are assumed to have similar operating history that of 4M12, that is, $T = 826.65$ days and $t = 1317$ days. Thus, the total fission product activity of the reactor core, according to Equation IX, is given as:

$$A_T(t) = 1.33 \times 10^{6} \times 5\,[MW] \times [(1317)^{-0.2} - (1317 + 826.65)^{-0.2}]$$

$$= 1.474 \times 10^{5}\,[Ci]$$

For a core of 25 fuel elements, the fission product activity for one fuel element is then equal to 5856.36 Ci. The inner volume of the BMI cask is calculated to be $1.0245 \times 10^5 cm^3$. The activity due to eight fuel elements loaded in the cask per unit volume, Sv, is thus equal to 1.69×10^{10} #/$cm^3 \cdot s$.

C. Determination of A_1, A_2, α, β and the Sivert's Integrals

Using Table 10.3 of reference [1], for gamma rays at 1 MeV energy, we have:

α = -0.035

β = 0.135

A_1 = 2.98

and A_2 = $(1-A_1)$ = -1.98

The Sivert's Integrals are $F_1 = F(\theta_1, (1+\alpha)(\mu_v d+\mu t))$ and $F_2 = F(\theta_1, (1+\beta)(\mu_v d+\mu t))$, where

$$\theta_1 = 51.7°$$
$$\mu_v = \mu_s = 0.1325\ cm^{-1}.$$
$$\mu = \mu_{lead} = 0.7825\ cm^{-1}.$$
$$d = 14.339\ cm.$$
and
$$t = 20\ cm.$$

According to Chapter 10 of reference [1], the Sivert's Integrals can be given by the following approximation:

$$F(\theta_1\ \alpha) = 1.2\, e^{-x}\chi^{-0.5}$$
if $\chi > 4$
and $\theta > 22.5°$

From our calculations, we have

$$F_1(\theta, \chi) \Rightarrow F_1(51.7°, 16.94) = 6.011 \times 10^{-10}$$
and
$$F_2(\theta, \chi) \Rightarrow F_2(51.7°, 19.92) = 1.288 \times 10^{-8}$$

Results

Having determined all the required parameters, Equation (VII) is used to determine the gamma flux at the surface of the BMI-1 cask:

$$\Phi_\ell = 1.69\text{x}10^{10}[\#/cm^3 \cdot sec]\text{x}(19.375)^2[cm^2]\frac{[2.98\text{x}1.288\text{x}10^{-8} + (-1.98)\text{x}6.011\text{x}10^{-10}]}{[2\text{x}(14.399[cm]+20[cm])]}$$

$$= 3435.616\,[\#/cm^2 \cdot sec]$$

Thus the gamma flux, Φ_ℓ (P) at point P on the outer surface of the BMI-1 cask is equal to 3435.616[#/cm$^2\cdot$s].

The conversion of gamma flux to exposure dose rate is given by Equation (VIII). For 1 MeV gamma energy, we have

$$k(E) = 1.91 \times 10^{-6}\left[\frac{R}{hr} \bullet \frac{cm^2 \cdot sec}{MeV}\right].$$ Thus,

$$\dot{D}[R/hr] = 1.91\text{x}10^{-6}\text{x}1.0\text{x}3435.616$$
$$= 0.00656[R/hr] \quad \text{or} \quad 6.562[mR/hr].$$

The surface dose of the BMI-1 shipping cask is estimated not to be in excess of 10 mR/hr.

References

[1] S. Glasstone and A. Sesonske, "Nuclear Reactor Engineering," Van Nostrand Reinhold Co., 1981, 3rd Edition, Chap. 10.

[2] H. Etherington, "Nuclear Engineering Handbook," McGraw-Hill Book Company, Inc., 1958, 1st Edition, Sec. 7-3.

[3] A. Foderaro, "The Photon Shielding Manual," The Pennsylvania State University, 1978, 2nd Edition, Section C-1.

[4] J. Lamarsh, "Introduction to Nuclear Engineering," Addison-Wesley Publishing Company, 1983, 2nd Edition, Chapter 3.

5.5.4 Shielding Analysis of Model BMI-1 Package with HFBR Fuel

The ORIGEN2 case described in Section 3.5.2 was also used to define the source term for shielding analysis. The MICROSHIELD computer program was used to analyze the BMI-1 cask with a payload of 20 HFBR fuel assemblies with an average decay time of 470 days. The MICROSHIELD code input and output parameters are given in Reference 22.

Table 5.7 summarizes the ORIGEN2 generated source term used as input to he MICROSHIELD program. The ORIGEN2 results present the source term after 470 days of decay. The results in the table are adjusted to account for a 20 fuel assembly payload in the BMI-1 package as opposed to the 28 assembly core loading. As shown, the fission product inventory is about 256,000 curies.

This source term of 256,000 curies was used as input to MICROSHIELD. As shown in Table 5.7, the major contributors are Zr-95, Nb-95, Ru-103, Ce-144 and Pr-144.

The MICROSHIELD results indicate that the contact and 2 meter dose rates will be 18 mRem/hr and 2 mRem/hr respectively.

BMI CASK RADIATION SHIELDING SOURCE TERM

Nuclide	ORIGEN2 Activity Ci 28 Assy.	MSHIELD Input Ci 20 Assy.
Ba–137m	1.064E+04	7.600E+03
Ce–141	9.348E+01	6.677E+01
Ce–144	1.171E+05	8.364E+04
Cs–134	6.826E+03	4.876E+03
Cs–137	1.125E+04	8.036E+03
Kr–85	1.320E+03	9.429E+02
Nb–95	2.082E+04	1.487E+04
Pm–147	2.337E+04	1.669E+04
Pr–144	1.171E+05	8.364E+04
Pr–144m	1.405E+03	1.004E+03
Rh–103m	2.284E+02	1.631E+02
Ru–103	2.534E+02	1.810E+02
Ru–106	9.259E+03	6.614E+03
Sr–89	2.116E+03	1.511E+03
Sr–90	1.083E+04	7.736E+03
Te–129	1.446E+00	1.033E+00
Y–90	1.083E+04	7.736E+03
Y–91	5.683E+03	4.059E+03
Zr–95	9.383E+03	6.702E+03
Totals	3.585E+05	2.561E+05

6. CRITICALITY EVALUATION

6.1 Discussion and Results

6.1.1 Applicable Regulatory Criteria

Regulatory criteria pertaining to criticality which are applicable to Fissile Class I and II shipments are delineated in §71.37, §71.38, and §71.39 of 10-CFR-Part 71. These sections are summarized as follows:

§71.37 Evaluation of an array of packages of fissile material

(a) An array of packages shall be evaluated by subjecting a sample package to the conditions specified in 71.38, 71.39, or 71.40.

(b) In determining whether the standards of the Class I, II, and III sections are met, it will be assumed that, consistent with the condition of the package:

(1) The fissile material is in the most reactive credible configuration.

(2) Water moderation occurs to the most reactive extent.

§71.38 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:

(a) 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

(b) Two-hundred-fifty such packages would be subcritical in any arrangement, if each package were subjected to the hypothetical accident conditions.

§71.39 Specific standards for Fissile Class II package.

(a) For a Fissile Class II package; designed, constructed, loaded, and number limited so that:

  (1) Five times the number of undamaged packages would be subcritical in any arrangement with close water reflection.

  (2) Two times the number of packages damaged by the Appendix "B" tests would be subcritical in any arrangement with close water reflection and optimum interspersed moderation, or built-in moderation if it is greater.

(b) The number of radiation units to be fifty divided by the allowable number of packages, rounded up to next higher tenth.

6.1.2 Determination of Allowable Number of Packages

6.1.2.1 Fissile Class I

For Fissile Class I shipments a 1000 x 1000 x 1000 array was analyzed.

6.1.2.2 Fissile Class II

The number of allowable packages that can be shipped under Fissile Class II was calculated from the equation:

$$\frac{50}{N} = I_t$$

where N is the allowable number of packages and I_t is the transport index or minimum number of radiation units. For the case of sole use of vehicle shipments, § 173.396(f) of 49 CFR states that for nuclear criticality control purposes, the transport index must not exceed 10. Thus, using this transport index, the number of casks that could be shipped was calculated from the above equation and found to be 5.

On the basis of the allowable number of packages determined above, § 71.39 of 10 CFR Part 71 specifies that 5N or 25 packages remain subcritical under normal conditions and that 2N or 10 packages remain subcritical under accident conditions.

The container is shown to undergo little dimensional change in the accident conditions; thus, shipment of 25 packages will be the most restrictive case.

6.1.3 Contents Evaluated

The following contents are evaluated for shipment in the BMI-1 cask.

- General contents
- BRR Fuel Elements
- MTR Fuel Elements
- TRIGA Fuel Elements
- PULSTAR Fuel Elements
- MURR Fuel Elements
- MITR Fuel Elements
- HFBR Fuel Elements

6.2 Criticality Evaluation for General Contents

6.2.1 Package Fuel Loading

A criticality evaluation was made of the modified BMI-1 cask using the KENO[1] computer code. The analysis was made for mutually exclusive shipments of U-235 or Pu-239. Two separate criticality analyses were performed for shipment with and without an inner container. In addition, a check calculation was made for the case of a critical solution of $U(30.45)O_2F_2$ in spherical geometry to verify the accuracy of the Hansen and Roach[2] cross sections used in the aforementioned analysis.

6.2.2 Shipment Without Inner Container

6.2.2.1 Calculational Model

The configurational model of the modified BMI-1 cask used for the criticality analysis is shown in Figure 6.1. The fissile material is assumed to contain the concentration of water required to make the material most reactive (i.e., optimum H/X atomic ratio) and is located in a spherical volume at the geometric center of the container. The remainder of the inner cavity of the container is assumed to be void.*

The criticality calculations (to determine X_{eff}) for Fissile Class II shipments were performed on a 3 x 3 x 3** array

---

* This represents the most reactive condition.

** For simplicity of the calculational model, 27 casks were used in lieu of 25.

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(1) References for Section 6. are found in Section 6.8.1.

--•.*12"*

Stainless steel

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 6.1. CALCULATION MODEL UTILIZED IN CRITICALITY EVALUATION

of BMI-1 casks. The array (shown in Figure 6.2) has a cubic lattice with a pitch of 33.0 inches, and the array is completely surrounded by a 10-inch water reflector.

The criticality calculations for Fissile Class I shipments were performed on a 1,000 x 1,000 x 1,000 array of BMI-1 casks. This array was assumed to be infinite for calculational purposes.

6.2.2.2 Results

The results of the KENO calculations for shipment without an inner container are given in Table 6.1. These results show that an infinite array (i.e., 1 billion units) of BMI-1 casks, each of which contains 500 g of U-235 or 280 g of Pu-239, remains subcritical. Since Pu-239 is more reactive on a mass basis than U-233, 280 g of U-233 in an infinite array would also be subcritical.

The difference in K_{eff} for 27 (i.e., 3 x 3 x 3 array) and 1 billion (i.e., 1,000 x 1,000 x 1,000) casks each containing 500 g of U-235 was found to be only 0.01. Thus, the lattice type was found to be unimportant in the case of the foregoing calculations.

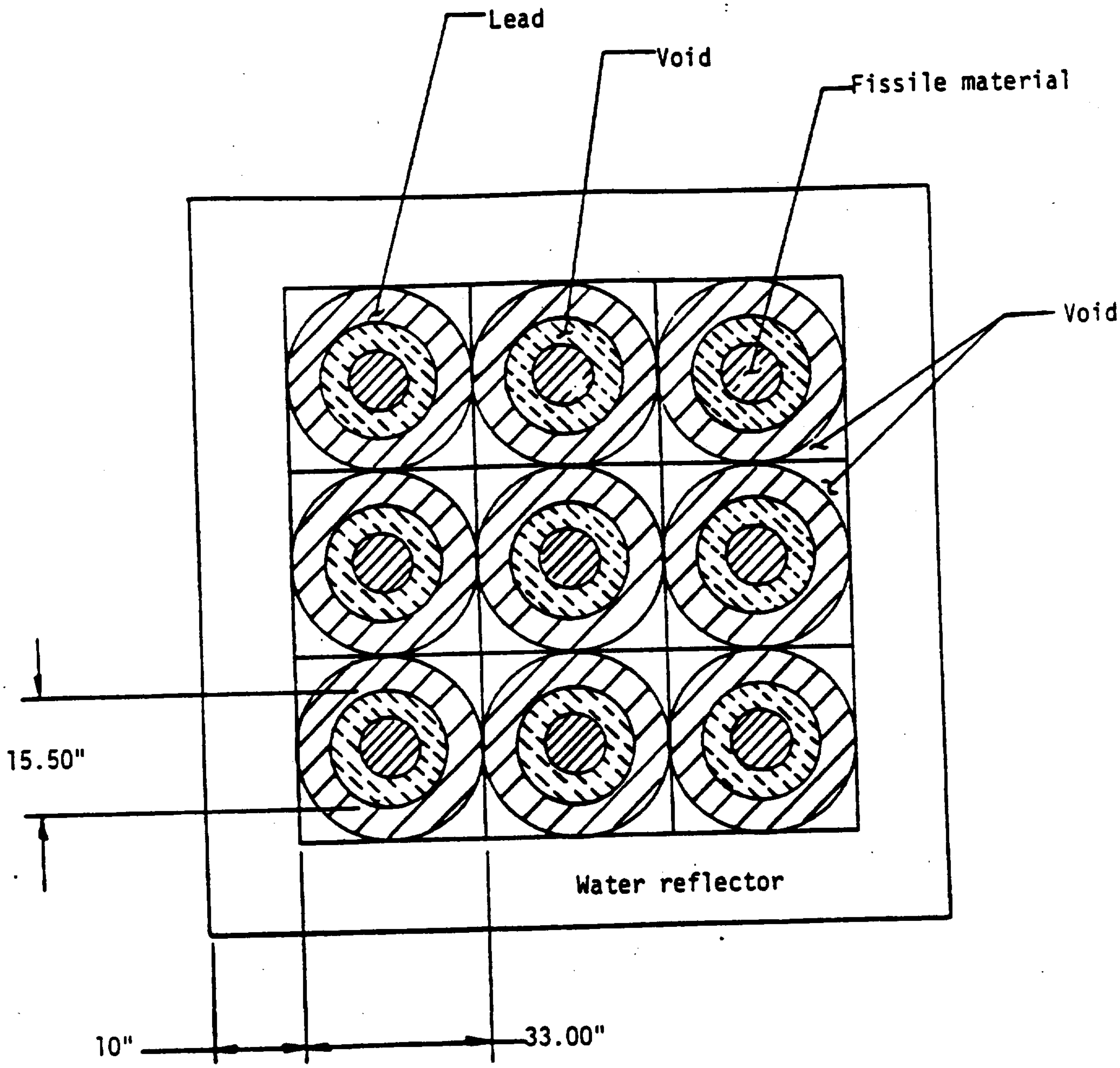

FIGURE 6.2. CROSS SECTION OF 3x3x3 ARRAY OF CASKS

TABLE 6.1. RESULTS OF THE KENO CODE CALCULATIONS OF K_{EFF} FOR SHIPMENT WITHOUT AN INNER CONTAINER

Run No.	Description	Fissile Material	Mass Fissile Material (grams)	H/X[a] Atomic Ratio	Calculated K_{eff}
1	Critical sphere test case[b]	U-235	991	573	1.01 ± 0.01
2	3 x 3 x 3 array - H_2O reflected (cubic lattice)	U-235	500	500	0.93 ± 0.02
3	1,000 x 1,000 x 1,000 array (cubic lattice)	U-235	500	500	0.94 ± 0.025
4	1,000 x 1,000 x 1,000 array (cubic lattice)	Pu-239	280	800	0.93 ± 0.02

(a) Reference 3, pages 12, 14, and 35.

(b) Reference 3, page 12.

In the case where the shipper wishes to ship a mixture of the fissile isotopes considered here, the following conditional equation may be used to determine the mass limit for shipment:

$$\sum_{k=1}^{3} \frac{M_K}{\text{Limit}_K} \leq 1 \quad , \qquad (1)$$

where:

M_K = mass (grams) of $K^{\underline{th}}$ fissile isotope to be shipped

Limit_K = mass limit (grams) (maximum shipment) for the $K^{\underline{th}}$ fissile isotope

K = an index number assigned to each fissile isotope (i.e., K=1 for U^{235}; K=2 for Pu^{239}; and K=3 for U^{233}).

6.2.3. Shipment with Inner Container

6.2.3.1 Calculational Model

The configurational model of the modified BMI-1 cask used for the criticality analysis is shown in Figure 6.3. The fissile material is assumed to contain an H/X ratio not greater than 20 and is located in a spherical volume at the geometric center of the container. The remainder of the inner cavity of the container is assumed to be void.

The criticality calculations for Fissile Class I shipments were performed on a 1,000 x 1,000 x 1,000 array of BMI-1 casks. This array was assumed to be infinite for calculational purposes.

Stainless steel

FIGURE WITHHELD UNDER 10 CFR 2.390

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FIGURE 6.3. CILCULATION MODEL UTILIZED IN CRITICALITY EVALUATION

6.2.3.2 Results

The results of the KENO calculations are given in Table 6.2. These results show that an infinite array (i.e., 1 billion units) of BMI-1 casks, using an inner container each of which contains 1,000 grams U-235 or 600 grams Pu-239, remains subcritical. Since Pu-239 is more reactive on a mass basis than U-233, 600 grams of U-233 in an infinite array would also be subcritical.

Since the lattice extent was found to be unimportant for the arrays of casks without inner containers, it is obvious that a 3 x 3 x 3 cubic array of casks (Fissile Class II) with inner containers and with 1,000 grams U-235 or 600 grams Pu-239 would also be subcritical.

In this case where the shipper wishes to ship a mixture of the fissile isotopes considered here, Equation (1) may be used to determine the mass limit for shipment.

TABLE 6.2. RESULTS OF KENO CODE CALCULATIONS OF K_{EFF} FOR SHIPMENT WITH AN INNER CONTAINER

Run No.	Description	Fissile Material	Mass Fissile Material (grams)	H/X Atomic Ratio	K_{eff}
1	1,000 x 1,000 x 1,000 array (cubic lattice)	U-235	1,000	20	0.83 ± 0.02
2	1,000 x 1,000 x 1,000 array (cubic lattice)	Pu-239	600	20	0.81 ± 0.02

6.3 Criticality Evaluation for BRR Fuel Elements

6.3.1 Package Fuel Loading

The modified BMI-1 shipping cask is a cylindrical, double-walled stainless-steel vessel, in which the space between the inner and outer shells is occupied by lead shielding. Fuel assemblies are positioned within the central cavity by two identical stainless-steel clad boral plates acting as center dividers as shown in Drawing 0004, Rev. B.

For this analysis, BRR fuel elements with 200 g of U-235 were considered. Each is 3.16 x 3.00 x 23.25 in., fueled length. A description of a standard fuel assembly for Battelle's Research Reactor is given in Figure 6.4. Each fuel assembly is a heterogeneous mixture of Al, H_2O, U-235, and U-238. The composition of a BRR fuel element is given in Table 6.3.

TABLE 6.3. COMPOSITION OF BRR'S FUEL ASSEMBLY

Material	Weight, gm	Atoms or Molecules per cc (Volume Homogenized)
H_2O	2725	2.5253×10^{22}
Al	2780	1.7188×10^{22}
U-235	200	1.41×10^{20}
U-238	15	1.05×10^{19}

A cross section of BMI-1 shipping cask's fuel basket is shown graphically in Figure 6.5 and in detail in Drawing 0004. This is the fuel basket used to ship the fuel element assemblies. The dimensions of each of the 12 cavities are 3.34 x 3.34 in. The fuel assemblies are shipped into these cavities.

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 6.4. STANDARD FUEL ASSEMBLY FOR BATTELLE RESEARCH REACTOR

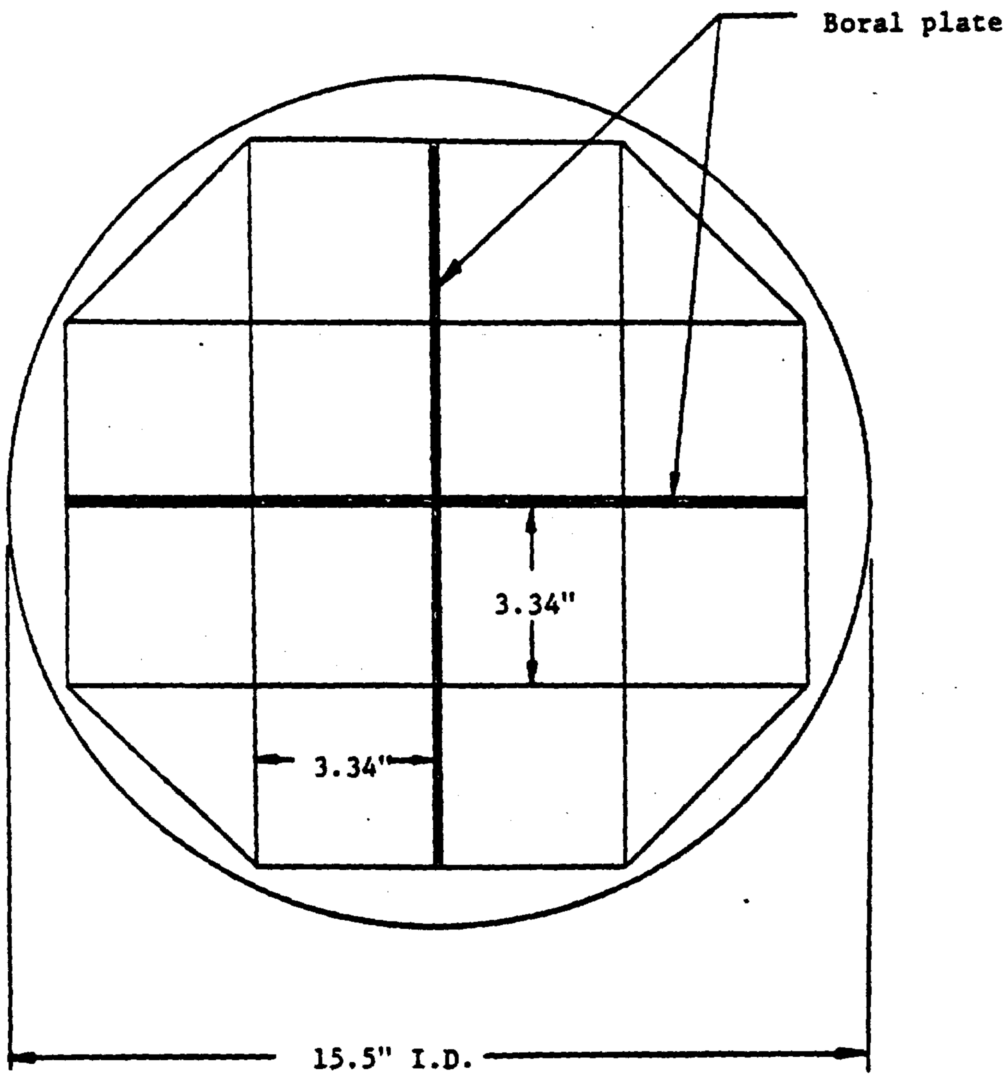

FIGURE 6.5. TOP VIEW OF SHIPPING CASK FUEL BASKET

6.3.2 Calculational Model

The fuel assembly was volume-homogenized for calculational purposes. Since the thermal flux is depressed in the fuel and peaks in the water of the cell, this is a conservative approximation, i.e., one that would tend to give relatively greater importance to the fuel and overpredict the value of K_{eff}.
The effect is known to be small.

In order to properly account for transverse leakage effects, Fe-Pb-Fe end plugs for the BMI-1 cask were incorporated into the two cask systems as shown in Figure 6.6.

A void region was also defined between the casks in order to increase the effect of neutron interaction between casks (see Figure 6.7 for details). The remainder of the two cask systems was immersed in a water reflector.

A 16-group set of cross sections was used. This modified Hansen-Roach set has been shown[4] to predict, accurately, values of K_{eff} when used in conjunction with the KENO program for systems such as the one described here.

6.3.3. Package Regional Densities

Table 6.4 gives the number densities for a basket cavity with a fuel assembly in it (volume fraction of fuel cell, 0.85, volume fraction of water, not in the cell, 0.15).

TABLE 6.4 NUMBER OF ATOMS PER CC IN THE HOMOGENIZED FUEL BASKET

Element	$N \times 10^{24}$
H	0.052981
O	0.0264905
Al	0.0146098
U-235	0.0001199
U-238	0.0000089

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 6.6. AXIAL REPRESENTATION OF THE SYSTEM
(SYSTEM IMMERSED IN WATER)

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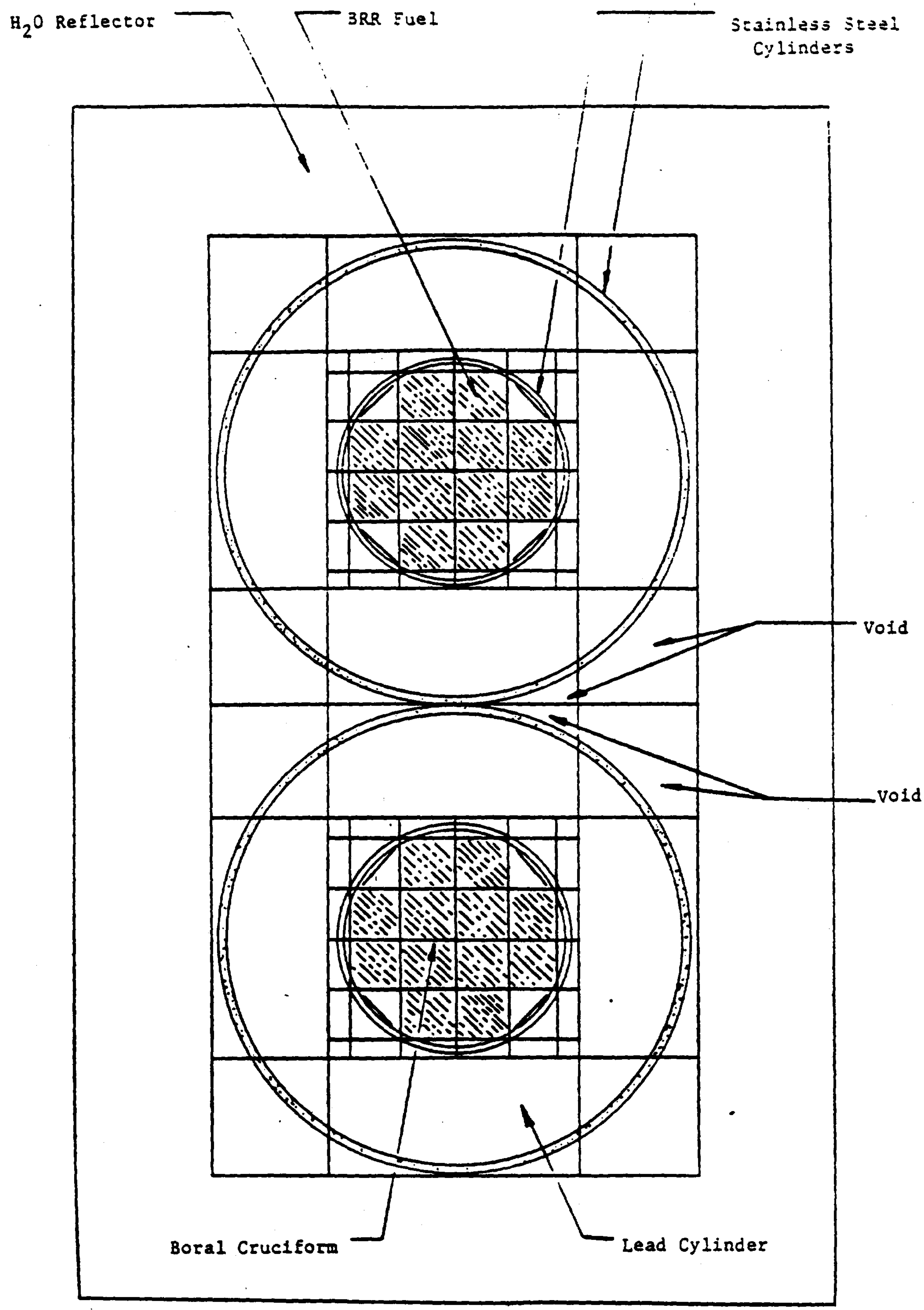


FIGURE 6.7. KENO CROSS-SECTIONAL REPRESENTATION OF

When considering the cruciform boral plates a simple volume homogenization is not adequate since it does not account for self-shielding effects, and thus would tend to over-estimate the boron absorption.

The total reaction rate in the homogeneous mixture of absorber (boral plates) and diluent (Al-cladding and SS), $V_T\Sigma\Phi$, can be related to the reaction rate of each of the components by:

$$V_T\overline{\Sigma}\ \overline{\Phi} = V_a\overline{\Sigma}_a\overline{\Phi}_a + V_d\overline{\Sigma}_d\overline{\Phi}_d \qquad (2)$$

where V_a, V_d, V_T, refer to the volumes of the absorber, the diluent, and the total, respectively. Equation (2) can be interpreted as a defining equation for proper homogenization, with the average flux defined by:

$$V_T\overline{\Phi} = V_a\overline{\Phi}_a + V_d\overline{\Phi}_d \quad . \qquad (3)$$

Using Equations (2) and (3), the properly homogenized number densities, N, are then found to be:

$$N \text{ (absorber)} = \frac{V_a^f N^o \text{ (absorber)}}{V_a^f + V_d^f \frac{\Phi_s}{\Phi_a}} \quad , \qquad (4)$$

and

$$N \text{ (diluent)} = \frac{V_d^f N^o \text{ (diluent)}}{V_a^f \frac{\overline{\Phi}_a}{\Phi_s} + V_d^f} \quad . \qquad (5)$$

where f indicates volume fractions and N^o are the true number densities of absorber and diluent materials. A ratio of the surface ($\bar{\Phi}_d = \bar{\Phi}_s$) to average flux in the absorber $\bar{\Phi}_s/\bar{\Phi}_a$ is equivalent to volume-homogenized number densities. The effect of using Equations (4) and (5) is to decrease the simple volume weighted number density of the absorber and to increase that of the cladding, as expected.

The boral plate, 62 mils thick, is composed of a o.103 cm thick B_4C-Al plate containing 35wt/o B_4C in the B_4C-Al mixture (65 w/o Al, 27.4 w/o B, 7.6 wt/o C), and clad with aluminum. There are also two outer SS plates, each 31 mils thick. The volume fractions of diluent Al, diluent SS, and absorber B_4C-Al mixture are, 17.3 percent, 50 percent, and 32.7 percent, respectively.

It has been shown[5] that conventional calculations of the absorption cross-section of boral do not consider channeling of neutrons due to B_4C lumping. Experimentally, it has been verified[6] that B_4C lumping in the boral plates reduces the thermal neutron absorption cross-section about 20 to 30 percent. This implies that the number density (above) of B should be reduced by that factor for computation of the absorption cross sections. It can be shown that:

$$\frac{\Phi_s}{\bar{\Phi}_a} = \frac{2(2 - \beta)}{\beta} \Sigma_a \frac{V}{S} \quad ,$$

where β is the capture fraction, and V and S refer to the volume and surface of the absorber. For a slab,

$$\frac{\Phi_s}{\bar{\Phi}_a} = \frac{(2 - \beta)}{\beta} \Sigma_a \ell \quad .$$

For heavy absorbers, $\beta \doteq 1$, we have

$$\frac{\Phi_s}{\Phi_a} = \Sigma_a \ell \quad .$$

Where Σ_a is the absorption cross section of B_4C-Al mixture and ℓ is the thickness of the plate. $\Sigma_a \doteq 17.664 \text{ cm}^{-1}$, $\ell = 0.103$ cm, therefore,

$$\frac{\Phi_s}{\Phi_a} = 1.82 \quad .$$

The properly homogenized number densities are then,

$$N_d \text{ (Al)} = \frac{(0.173)\ (0.0602)\ \times 10^{24}}{\frac{(0.327)}{(1.82)} + (0.173 + 0.5)} = 0.0122 \times 10^{24} \quad ,$$

$$N_a \text{ (Al)} = \frac{(0.327)\ (0.0392)\ \times 0.7 \times 10^{24}}{(0.327 + (0.673)\ (1.82)} = 0.00578 \times 10^{24} \quad ,$$

6.3.4 Results

The k_{eff} for two BMI-1 shipping casks loaded with 24 BRR fuel elements (200 g of U-235 per element), having the inner cavity filled with water, a void between the casks where they are in contact, and the cask systems surrounded by a water reflector was calculated to be:

$$k_{eff} = 0.934$$

with a standard deviation of $\pm$ 0.0099.

6.4 Criticality Evaluation for MTR Fuel Elements

The MTR fuel assemblies to be shipped comply in type and form to 5(b) (1) (i) of the present BMI-1 container license. The modified basket maintains the identical geometry for the active fuel portion of the MTR assemblies as the licensed package containing 24 assemblies. The criticality analysis for shipment of 24 assemblies is detailed in Section 6.3. The present shipment represents the same geometry as analyzed in Amendment 5 except the length of the active fuel is reduced from 48 inches to 24 inches. This is a less critical loading and geometry than analyzed in Section 6.3 Thus, the shipment of 12 MTR assemblies in the modified basket will remain subcritical in the most reactive credible configuration.

6.5 Criticality Evaluation for Fermi Fuel Elements

(Paragraph deleted)

6.6 Criticalicty Evaluation for TRIGA Fuel Elements

6.6.1 Package Fuel Loading

Irradiated TRIGA fuel assemblies containing not more than 55 grams U-235 (unpoisoned) or 135 grams U-235 (poisoned) per assembly prior to irradiation. Uranium may be enriched to a maximum 93.2% in the U-235 isotope. Active fuel length unirradiated shall be nominally 15 inches or less.

Up to 38 fuel assemblies as contained in product containers specified in 1.2.1.2(d).

6.6.2 Results

Experiments have shown that in an optimally moderated array, criticality is achieved with 60 TRIGA fuel elements. Information on these experiments is contained in "Torrey Pines TRIGA MARK III Reactor Startup and Post Critical Tests," GAMD-7445. Only 38 elements will be shipped in the BMI-1 cask and the geometry for shipment is highly non-ideal compared to the optimally moderated critical core loadings described below. Structural analysis shows the basket maintains its integrity during the accident sequence. Thus the shipment of 38 elements will remain subcritical under both the normal and accident conditions for transport.

6.6.3 Criticality Measurements

Following are the results of the loading-to-critical experiment in the University of Arizona TRIGA. The measured value of k for the 38-element loading in the most reactive configuration (i.e. graphite moderated optimally spaced array) is seen to be 0.83; however, by curve fitting as shown in Figure 6.8 the best value would be a k of 0.80. These experimental results are compared with other critical loadings for other fuel types. The comparisons show that 38 elements of any TRIGA fuel type is substantially subcritical even without the added disadvantage factors of displaced reflector and a large void in the center of the array.

Other Criticality Determinations:

Using actual critical loading for the several varieties or types of TRIGA fuel elements with experimentally determined corrections for configuration differences gives the following values for critical loading of a water reflected core with a central fuel element:

(1)	14" long fuel - aluminum clad - 8% U - 20% enriched -	59 elements
(2)	15" long fuel - aluminum clad - 8% U - 20% enriched -	55 elements
(3)	15" long fuel - stainless steel clad - - 8.5% U - 20% enriched -	60 elements
(4)	15" long FLIP fuel - stainless steel clad new - 8.5% U, 70% enriched, 1.48% Er -	58 elements
(5)	15" long FLIP fuel - stainless steel clad After 5 mw-yrs. burnup (max. reactivity worth) - 8.5% U, 70% enriched, 1.48% Er	46 elements
(6)	15" long fuel - 8.5%, 93% enriched (graphite reflector)	19* elements

*Projected critical loading based on worth of six elements substituted in an operating reactor. Only six elements of this type have been manufactured.

For the purposes of establishing a maximum allowable loading for cask shipment, we propose using an ideal critical loading curve (1/M vs # of fuel elements) to select a max. $k_{eff} = 0.8$ (i.e. 80% of critical loading). Review of available critical loading curves indicates a large variation in the curvature of the 1/M vs. # of elements and thus a very large variance in the number of elements needed to achieve $k_{eff} = 0.8$. The variations ranged from 20 out of 58 for the Arizona data to as few as 3 or 4 out of 60 for some Mk. III data. The variation in curve shape is generally influenced most strongly by the relative location of the neutron source, detectors, and fuel, and for application here causes unacceptable uncertainties at the 0.8 k_{eff} point. Actual criticality for a standard water moderated optimum spaced close packed array yields a more informative result. Using a fixed value of 80% of the number of elements for criticality will provide adequate safety margin and consistency.

No credit has been taken for the large (8.25 in. dia.) central hole in the BMI cask with TRIGA "basket" even though the cask SAR indicates the basket remains intact after the calculated maximum accident. Experimental data is available for critical loading values for a TRIGA core with a 9" (across flats) hexagonal hole in the center of the core.

15" long fuel - stainless clad - 8.5% U (20% enriched)		
9" hex w/.25" ss liner	air filled	170 elements
	water filled	193 elements

The presence of a hole very close to that in the cask produced an increase in the number of elements for a critical loading by approximately a factor of 3. This indicates that the BMI cask loaded as suggested above for TRIGA elements is a very conservative arrangement.

This demonstrates and assures that continued subcriticality exists during the cask unloading. Further, there cannot be a radial water reflector when the elements are in the cask.

6.6.4 Data Relative to the Criticality Determination

Core loading data is available for TRIGA cores with graphite reflector (Arizona, TRIGA Mark I), with water reflector (DOFL, Mk. F, Mk. III), with no central fuel element (Mk. I, Mk. F, DOFL), and with a central fuel element (Mk III, Mk III FLIP) using fuel of either 14" or 15" length, aluminum or stainless steel clad, 8% or 8.5% U content and 20% enriched, 70% enriched with Erbium burnable poison (FLIP) and a few 93% enriched with no poison. In order to establish a basis for specifying a maximum cask content we first established a standard core configuration - water reflected core with a central fuel element - i.e. the most compact optimum moderated core possible. Adjustment to actual critical loading values for several cores was made using the following experimental data.

(1) Graphite reflected core vs water reflected core. The original submission by U. of Arizona indicated a 58 element critical loading for 14" fuel length aluminum clad elements. This was for a core with a 30 cm thick graphite reflector. Data for a like core with water reflector (Mk. F, TRIGA at GA) gave a critical loading of 74 elements or ~1.25 times the graphite reflected critical loading. This agrees with calculated values indicating ~25% change in critical mass if the graphite reflector is removed. Since the BMI cask contains no graphite it would seem reasonable to utilize water reflected critical mass values.

(2) 14" long fuel vs 15" long fuel. The effect upon critical loading of an additional inch of fuel for the aluminum clad core is determined by comparing the critical loading of TRIGA Mark F (1960) using 14" meats aluminum-clad, 8% U with critical mass of 74 elements to the DOFL TRIGA with 15" meats aluminum-clad, 8% U with critical mass of 69 elements. Neither core had a central fuel element and was entirely water reflected.

(3) Critical loading of water reflected core with central (A-1) fuel element vs core with no central element. The 8.5% U - 15% long meat - stainless steel clad fuel was designed and demonstrated to have reactivity worth equal to the older 14" long 8% U, aluminum-clad fuel. The original Mk. III reactor achieved criticality with 60 elements (8.5 % U, 15" ss clad), using a central (A-1) fuel element. This compared to the critical loading without the central element (using 14", 8% U, aluminum-clad) of 74 elements. Thus the central element was found to be worth approximately 14 elements in the critical loading and the ratio of the critical masses was 0.818. For simplicity we use a value of 0.8 as the ratio of critical masses for cores without a central element to cores with a central element. Applying this correction to the 14" aluminum-clad 8% U (74 elements) and the 15" aluminum-clad 8% U (69 elements) gives a critical loading with central element of 59 and 55 elements respectively.

(4) & (5) FLIP fuel - critical loading with new fuel was 58 elements using 8.5% U, 15" long fuel, 70% enriched w/1.48% Erbium, stainless steel clad. Calculated data for the reactvity effect of burnup during core life shows for an initial core loading for 6% reactivity the "cold clean" excess rises to 7.5% (a factor of 1.25) as a maximum after a 5Mw years whole core history. The value does not take into account poison buildup as a result of operation. Thus the minimum mass of a FLIP core - water reflected with central element is 46 elements.

References and Sources of Data

Critical Loading - DCFL TRIGA - GA 2995, DCFL TRIGA Reactor Acceptance Report - March 1962.

Critical Loading - Torrey Pines TRIGA Mark III, 8% U ss clad - TRIGA Mark III Logbook - 01-19-1966.

Critical Loading - Mark III FLIP - Startup Report for Mk. III FLIP-TRIGA reactor - June 8-August 8, 1971 - GAA-10878.

Critical Loading - Mark F TRIGA, 8% U, 15" ss fuel - Experiments with Mark F Reactor - 26 July 1961, GAMD-2430.

FLIP Fuel Burnup - SAR for the Torrey Pines TRIGA Mk. III (FLIP) core.

TABLE 6.5. MEASURED RESULTS DURING LOADING TO CRITICAL IN TRIGA AT THE UNIVERSITY OF ARIZONA

Fuel Elements	CPM	$\frac{1}{M} = 1 - k$
0	188	1
6	188	1
16	317	0.76
25	495	0.38
33	678	0.28
38	1135	0.17
42	1543	0.12
46	2111	0.09
50	3722	0.05
58	Critical	0

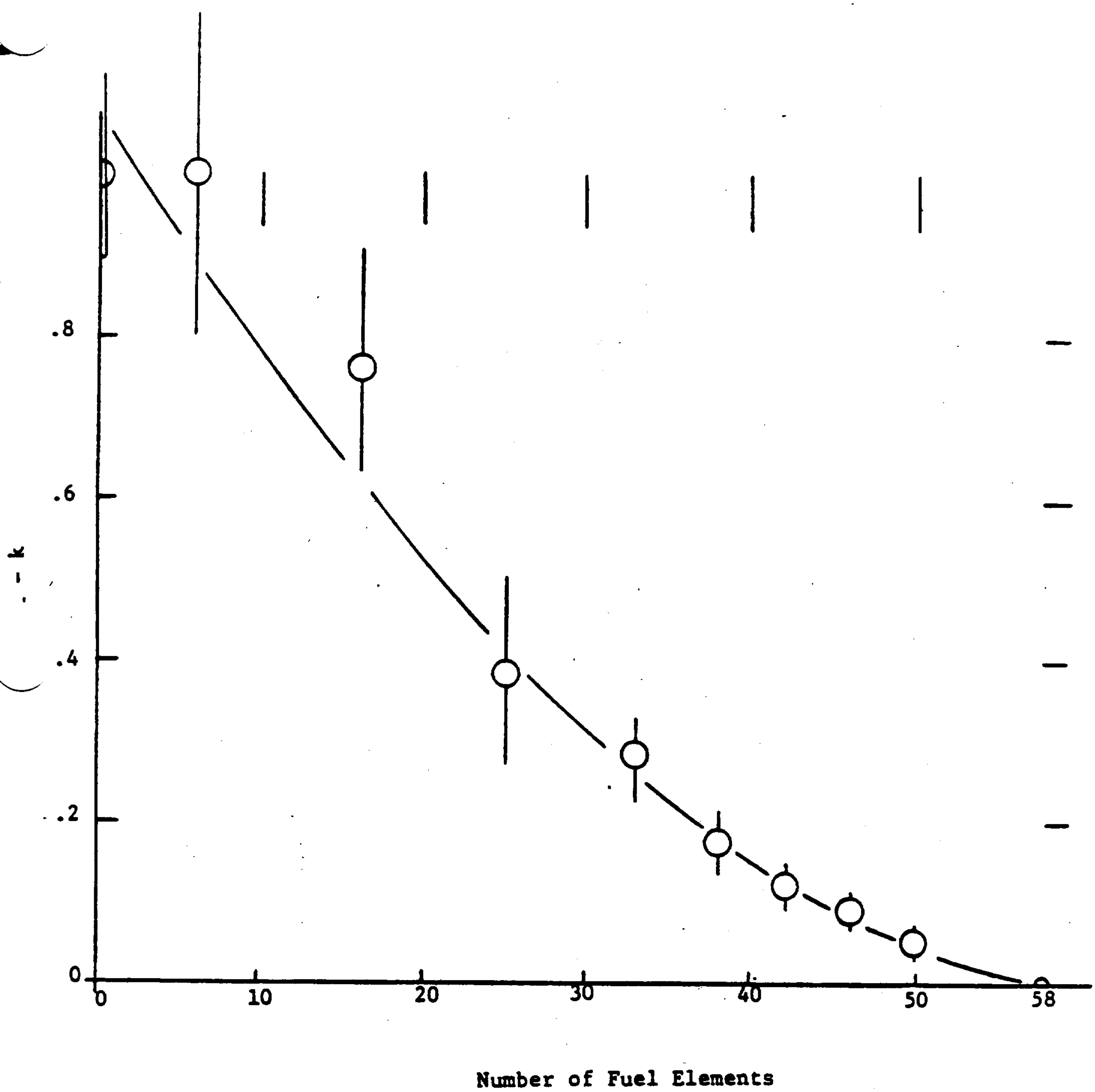


FIGURE 6.8. LOADING TO CRITICAL RESULTS IN TRIGA USING ALUMINUM-CLAD FUEL ELEMENTS.

6.7 Criticality Evaluation for Union Carbide Process Uranium Oxide

6.7.1 Package Fuel Loading

The process uranium oxide is formed by pyrolyses within the process container. The containers are nominally 2.50-inches I.D. and 11.75-inches internal length. They are made entirely of 6061-T6 aluminum alloy and sealed dry. The oxide may be in flake or powder form. Due to the manner in which it is formed directly in the container its distribution is random, i.e., although the major portion will be in a bed at the bottom of the container, some powder will adhere to the walls of the container.

The product may include a mixture of oxides of uranium. For purposes of analysis it was assumed that the oxide is in the form of UO_2 which would have the greatest percentage of uranium per unit weight of oxide. Analyses were done on the basis of 400 grams of UO_2 powder which for the 93 percent enrichment represents 352 grams of U^{235}.

6.7.2 Normal Conditions

The shipments are to be made dry. The total mass of U-235 in 24 process uranium oxide containers, each containing 400 grams of $U(93)O_2$ is 9.088 kg. The minimum critical mass of fully reflected U-235 is 22.8 kg. Thus, even for two dry packages in contact and reflected on all sides by water, $k_{eff} < 1$.

In the case where some or all of the containers are replaced by MTR fuel elements the total mass of U-235 is smaller than in the above case since each fuel element contains only 200 grams of U-235. Thus, shipments of containers with 400 grams of $U(93)O_2$ interspersed among MTR fuel elements and fully reflected by water will have $k_{eff} < 1$.

6.7.3 Accident Conditions

6.7.3.1 Calculational Model (Process Uranium Oxide Only)

Under accident conditions for Fissile Class III materials, one shipment of packages is to remain subcritical with optimum hydrogenous moderation and close reflection by water.

Consider first the transport of 24 Union Carbide process uranium oxide containers carrying equal amounts (from 200 grams to 400 grams) of $U(93)O_2$ powder. To determine when optimum moderation occurs KENO calculations were done for the cases where each container carries 200, 300, and 400 grams of UO_2 powder and where, in each case, the remainder of the container is filled with water. Also, KENO calculations were done for the cases where each container carries either 200 grams or 400 grams of UO_2 powder and the containers are filled to approximately 7/10 of their capacity with water. All of these calculations were done using the 123 group neutron structure available with the AMPX-1 modular code system. This consists of the 93 GAM-II groups combined with a 30 group THERMOS structure below 1.89 ev. Although the amount of U-238 in these loadings was very small, NITAWL runs were made to correct for resonance self-shielding in each of the cases. The KENO calculation was done using the mixed-box option of KENO geometry. The reflective plane capabilities of KENO were used so that only one quadrant of the geometry had to be modelled, i.e., reflective planes were used at tye x-z plane, at the x-y plane, and at the y-z plane. Figure 6.13 shows a horizontal cross-section of the loaded BMI-1 shipping cask fully reflected by water. Figure 6.14 shows a vertical cross-section of box types 1, 2, and 3. In these cases the fuel basket and the cask are void.

6.7.3.2 Package Regional Densities

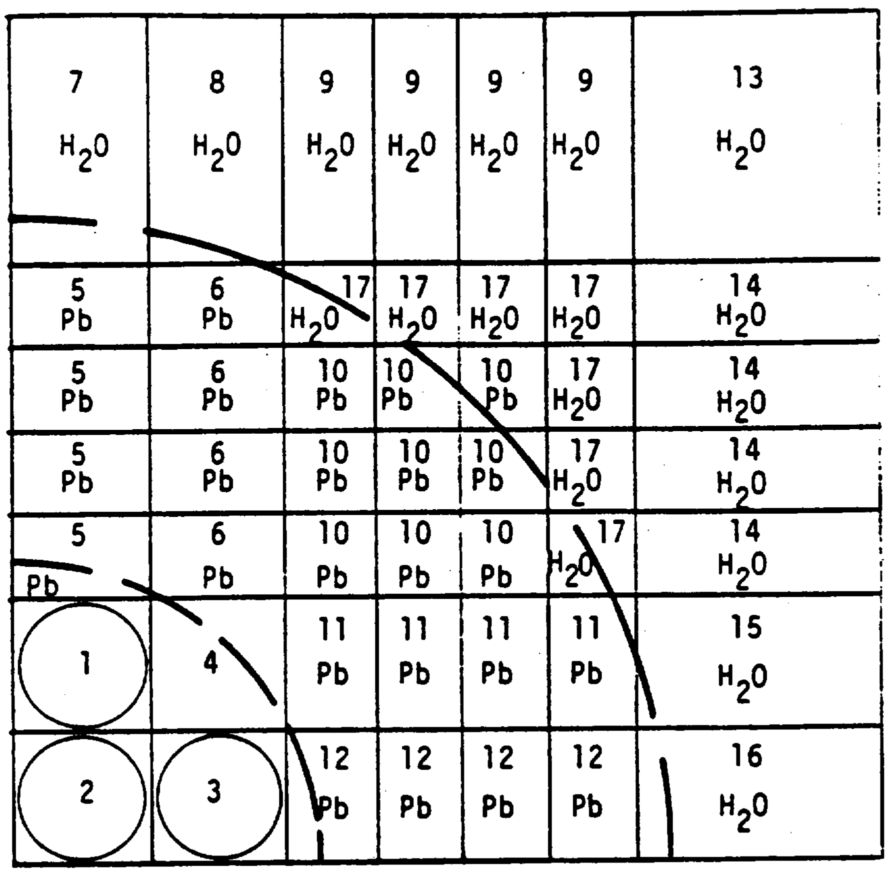
The KENO calculation requires as input the number densities of six mixtures. These are the homogenized UO_2-H_2O mixture, stainless steel, aluminum, the boral poison plates, the lead shield, and the water moderator and reflector.

The UO_2 powder was assumed to have a density of 7.56, i.e., about 0.7 times that of normal UO_2. The molecular weight of 93 percent enriched uranium was taken to be 235.21 and that of $U(93)0_2$ was taken to be 267.21. The number densities for the aqueous solutions of water for the 5 cases considered above are given in Table 6.10. Also in the table are shown the H/U235 atomic ratios for the cases.

TABLE 6.10. NUMBER OF ATOMS PER CC IN THE AQUEOUS SOLUTIONS OF UO_2

Case	200 g Container[a]	200 g Container[b]	300 g Container[a]	400 g Container[a]	400 g Container[b]
H/U Atomic ratio	134	96	88	65	46
Element					
U-235	0.0004853	0.0006664	0.0007279	0.0009705	0.0013328
U-238	0.0000361	0:0000495	0.0000541	0.0000721	0.0000990
H	0.0648160	0.0640520	0.0637940	0.0627700	0.061640
O	0.0334507	0.0334580	0.0334610	0.0334703	0.0334960

(a) Water filled.
(b) 0.73 water filled.



(Numbers Refer to Box-Type)

Figure 6.13. Horizontal Cross-Section of Loaded Cask

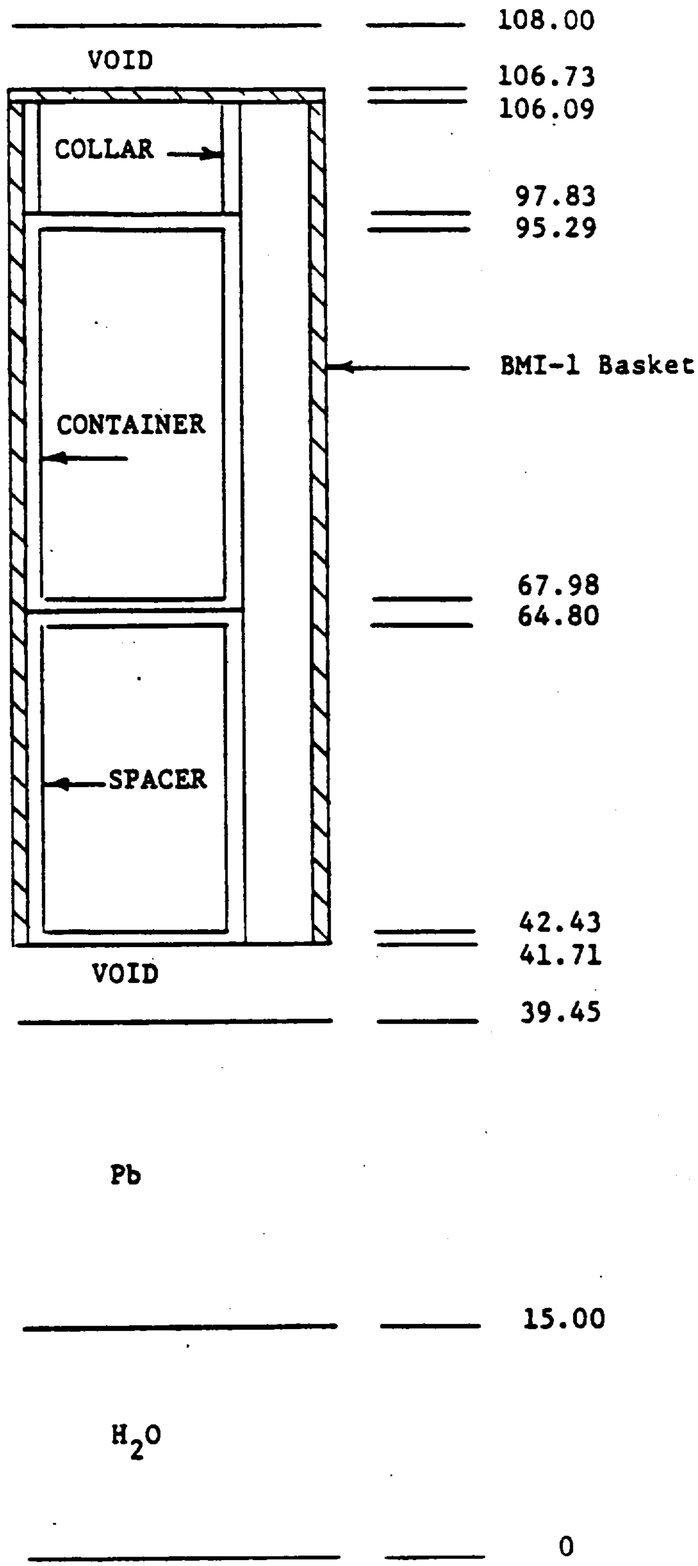


Figure 6.14 Vertical Cross-Section of Loaded Cask Box Types 1, 2, and 3 in a Void Cask

The stainless steel is a mixture of 3.0 percent silicon, 19.0 percent chromium, 2.0 percent manganese, 67.0 percent iron, and 9.0 percent nickel with a density of 7.92 grams/cc. The resultant number densities are given in Table 6.11.

TABLE 6.11. NUMBER OF ATOMS PER CC IN STAINLESS STEEL

Element	N x 10^{24}
Si	0.005100
Cr	0.017426
Mn	0.001737
Fe	0.057226
Ni	0.007315

Aluminum has a density of 2.7 g/cc and a molecular weight of 27 resulting in a number density of 0.06023 atoms/cc x 10^{24}.

Number densities for poison boral plates, lead, and water have previously been listed on Pages 6.36 and 6.37.

The results of these calculations are shown in Table 6.12. As can be seen from these results, the most reactive loading occurs for the 400 grams/container (water filled) case. These calculations are conservative because they assume that the containers in the top basket were misloaded so that the containers are in the bottom of the basket with the spacers above, whereas in the bottom basket the containers are properly loaded at the top with the spacers beneath. This places the two groups of twelve containers in closer proximity than for a normal loading condition.

The results of flooding the inside of the shipping cask must also be determined. Therefore, KENO calculations were made for the case where all void regions inside the cask are replaced with water. Only the two more reactive of the previous loadings were considered, i.e., 300 grams/container (water filled) and 400 grams/container (water filled). These results are also given in Table 6.12. As seen from these results the desired loadings will at all times be subcritical.

TABLE 6.12. KENO RESULTS FOR VARIOUS BMI-1 SHIPPING CASK LOADINGS

Case	H/U235	K_{eff}
24 - 200 gram/container (water filled) - void cask	134	0.681 ± 0.013
24 - 200 gram/container (0.73 water filled) - void cask	96	0.632 ± 0.010
24 - 300 gram/container (water filled) - void cask	88	0.738 ± 0.014
24 - 400 gram/container (water filled) - void cask	65	0.762 ± 0.008
24 - 400 gram/container (0.73 water filled) - void cask	46	0.694 ± 0.009
24 - 300 gram/container (water filled) - flooded cask		0.833 ± 0.011
24 - 400 gram/container (water filled) - flooded cask		0.825 ± 0.010
16 - 400 gram/container (water filled) 8 - MTR fuel elements (water filled) - flooded cask		0.810 ± 0.010
24 - MTR fuel elements (water filled) - flooded cask		0.862 ± 0.008

6.7.3.3 Calculational Model (Process Uranium Oxide Containers with Interspersed MTR Fuel Elements)

Some shipping cask loadings will have process uranium oxide containers with interspersed MTR fuel loadings. Therefore, KENO calculations of such cases have also been made. The number density of the homogenized fuel element (flooded with water) and occupying the available area in the BMI-1 shipping fuel basket has already been given in Table 6.4. A vertical representation of a box containing a fuel element is shown in Figure 6.15. The KENO calculations were done for the flooded cask case. Results for the cases of a partial loading of MTR elements -- partial loading of 400 grams waste containers and for the case of 24 MTR elements are given in Table 6.12. As seen from the results mixed loadings will also be subcritical.

6.8 Criticality Evaluation for Union Carbide Special Form Capsule

6.8.1 Package Fuel Loading

The special form capsules are nominally 1.25 inches in diameter and 18.0 inches long. They are made entirely of 300 Series stainless steel. Up to 100 grams of U^{235} may be contained in each capsule in oxide form. The uranium oxide is sealed dry within the capsules.

6.8.2 Normal Conditions

The shipments are to be made dry. The total mass of U-235 in twenty-four (24) special form capsules is 2.4 kg. The minimum critical mass of fully reflected U-235 is 22.8 kg. Therefore, even for two packages in contact and reflected on all sides by water, $k_{eff} < 1$.

6.8.3 Accident Conditions

Under accident conditions for fissile Class III materials, one shipment of packages is to remain subcritical with optimum hydrogenous moderation and close reflection by water. In Section 6.8.3 it was shown

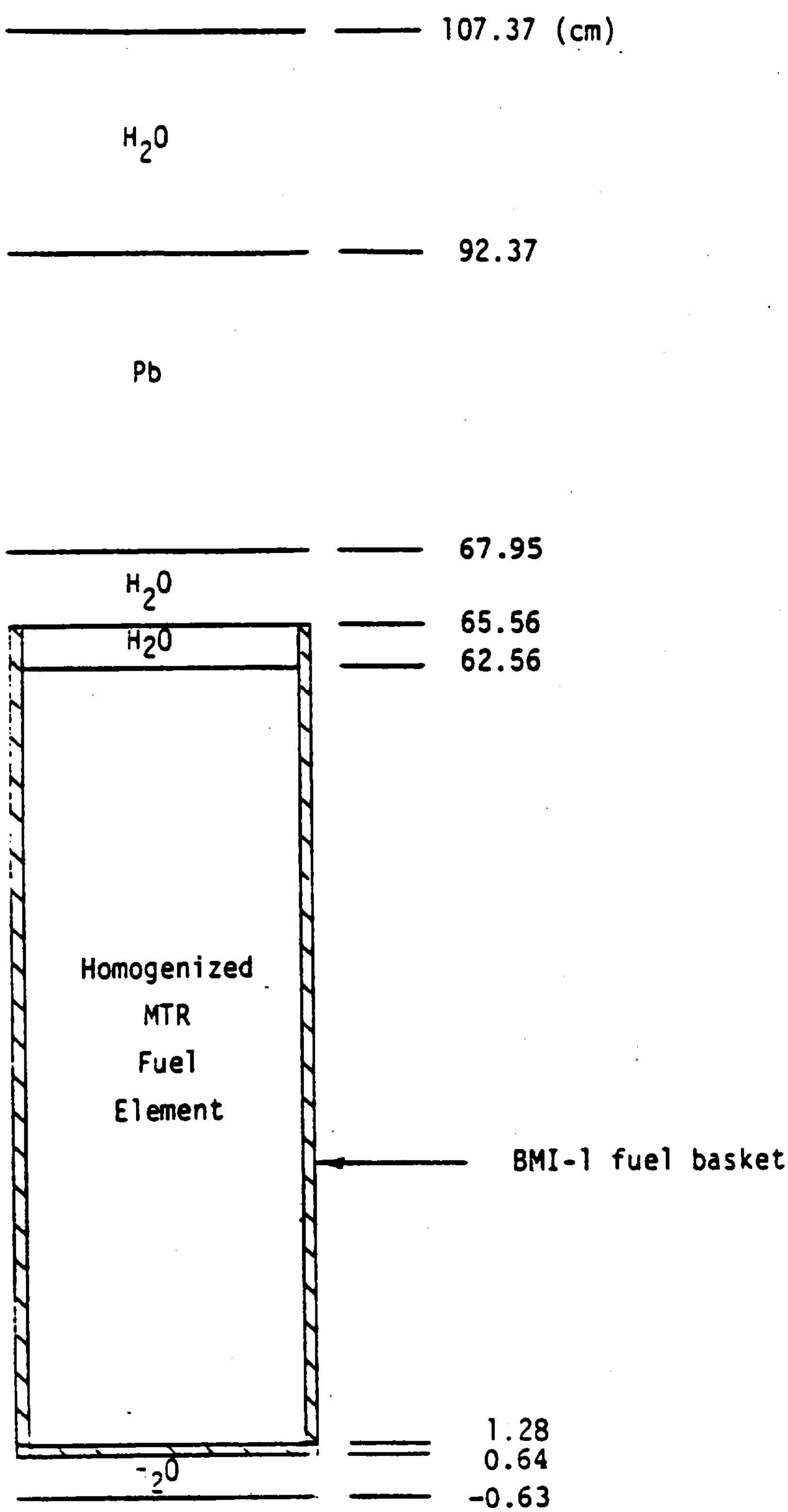


Figure 6.15 Vertical Cross-Section of Box Type Carrying MTR Element in Flooded Cask

that up to 400 grams of uranium oxide fully enriched in U-235 was subcritical for various combinations of cask flooding and pressure of water within process uranium oxide containers. Since the maximum quantity of U-235 contained in the special form capsules is significantly less than for the process oxide containers, by reference to the analytical results presented in Section 6.8.3 (specifically Table 6.11) the shipment of twenty-four (24) Union Carbide special form capsules is considered to be subcritical for all accident conditions.

6.9 Criticality Evaluation for MURR Fuel Assemblies

A KENO-V model of eight MURR fuel assemblies in the MURR fuel basket was used for criticality calculations. Twenty four runs were performed. The average calculated $K_{effective}$ value for this model is 0.70746 ± 0.005 for one cask and 0.71040 ± for two casks in close contact. See Appendix 6.12.2.

6.10 Criticality Evaluation for MITR Fuel Elements

The MITR-II fuel elements to be shipped are very close in form to 5(b) (1) (xii) (MURR Fuel Assemblies) of the present BMI-1 container license. The basket used to contain the MITR fuel elements is the same as is used for the MURR fuel. A KENO-V model of eight MITR-II elements inside the BMI-1 cask was made. Ten KENO-V runs were made to analyze this model for criticality. The results of the runs was that the average $K_{effective}$ value is 0.73159 with a standard deviation of 0.00498. To test meeting requirements of 10 CFR 71.61, two BMI-1 casks together were modeled using the above technique. The resulting $K_{effective}$ value is 0.73148 with a standard deviation of 0.00465. See Appendix 6.12.3.

In addition, a calculation was made to test the conservatism involved in the homogenizing of the MITR-II fuel elements. The results of a heterogeneous model showed $K_{effective}$ to be 0.65088, with a standard deviation of 0.02987.

Document 22

6.11 Criticality Evaluation for HFBR Spent Fuel Assemblies

In its application to NRC in support of shipping HFBR Spent Fuel Assemblies in the BMI-1, Brookhaven National Laboratory cited a criticality evaluation prepared by General Electric (GE) to support shipment of the HFBR fuel in the GE-700 package (formerly Certificate of Compliance 5942) in the same array as in the BMI-1. (Document 22, Appendix D). In its approval for the Certificate of Compliance, dated September 18, 1985, the NRC noted that it had performed independent criticality calculations of 20 HFBR assembly loadings in the GE-700 package. Both analyses confirmed that the 20 HFBR assembly payload does not affect the ability of the GE 700 package to meet the requirements of 10 CFR Part 71.61. (Document 22, Appendix D-4) The differences between the array analyzed in the GE and NRC evaluations and the array proposed for the BMI-1 package are twofold; (1) the BMI-1 inside package diameter is 15.5 inches while that of the GE-700 was 15 inches, and (2) the BMI-1 cavity length is 54 inches as compared with 54.75 in the GE-700. Brookhaven, in its analysis concluded that the dimensional variances were not material to the results of the earlier criticality evaluations, and that "if anything, the additional water gap at the basket periphery due to the larger BMI-1 package internal diameter will improve the absorber characteristics of the stainless steel basket and therefore reduce K_{eff}." See Document 22, p. 9.

6.12 APPENDIX

6.12.1 References

(1) Whitesides, G. E. "Adjoint Biasing in Monte Carlo Criticality Calculations," Trans. Am. Nuc. Soc., 11, 159, (1968).

(2) Bell, G. E. et. al., "Los Alamos Group-Averaged Cross Sections," LAMS-2941 (September 1963).

(3) TID 7028

(4) Whitesides, G. E., Private Communication.

(5) Burrus, W. R., "How Channeling Between Chunks Raises Neutron Transmission Through Boral," Nucleonics, 16, 91-94 (January 1958)

(6) BMI - Internal Memo from R. O. Wooten to E. C. Lusk (April 9, 1964).

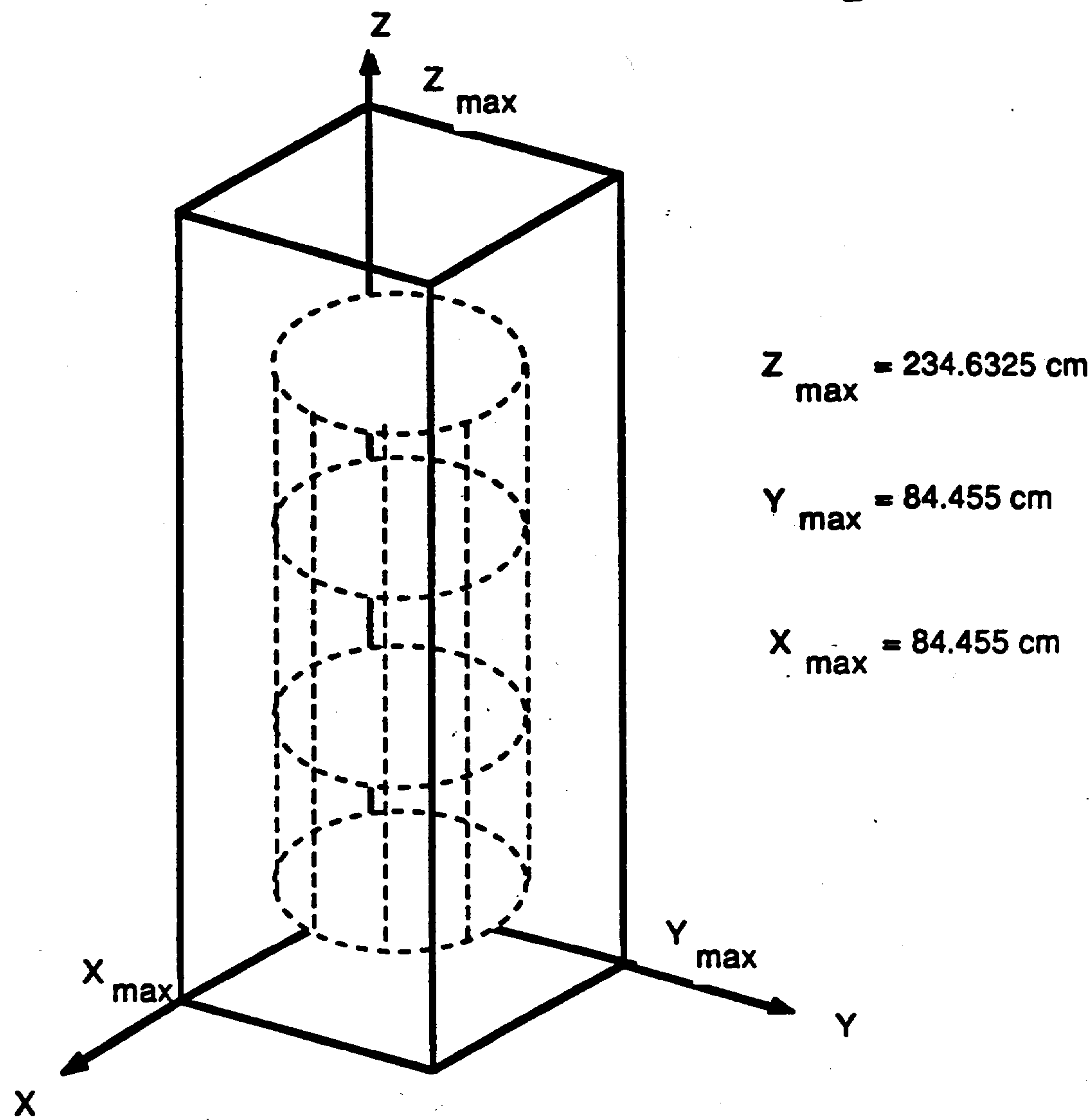
(7) Nuclear Safety Guide, TID-7016, Rev. 1, Goodyear Atomic Corporation (1961).

(8) Private communication from Martin N. Haas, Associate Director, Nuclear Science and Technology Facility, State University of New York at Buffalo to Dr. Richard Denning, Battelle Memorial Institute, Columbus, Ohio 43201 (March 15, 1977).

(9) RSIC Computer Code Collection, ORNL-TM-3076, AMPX-1, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830.

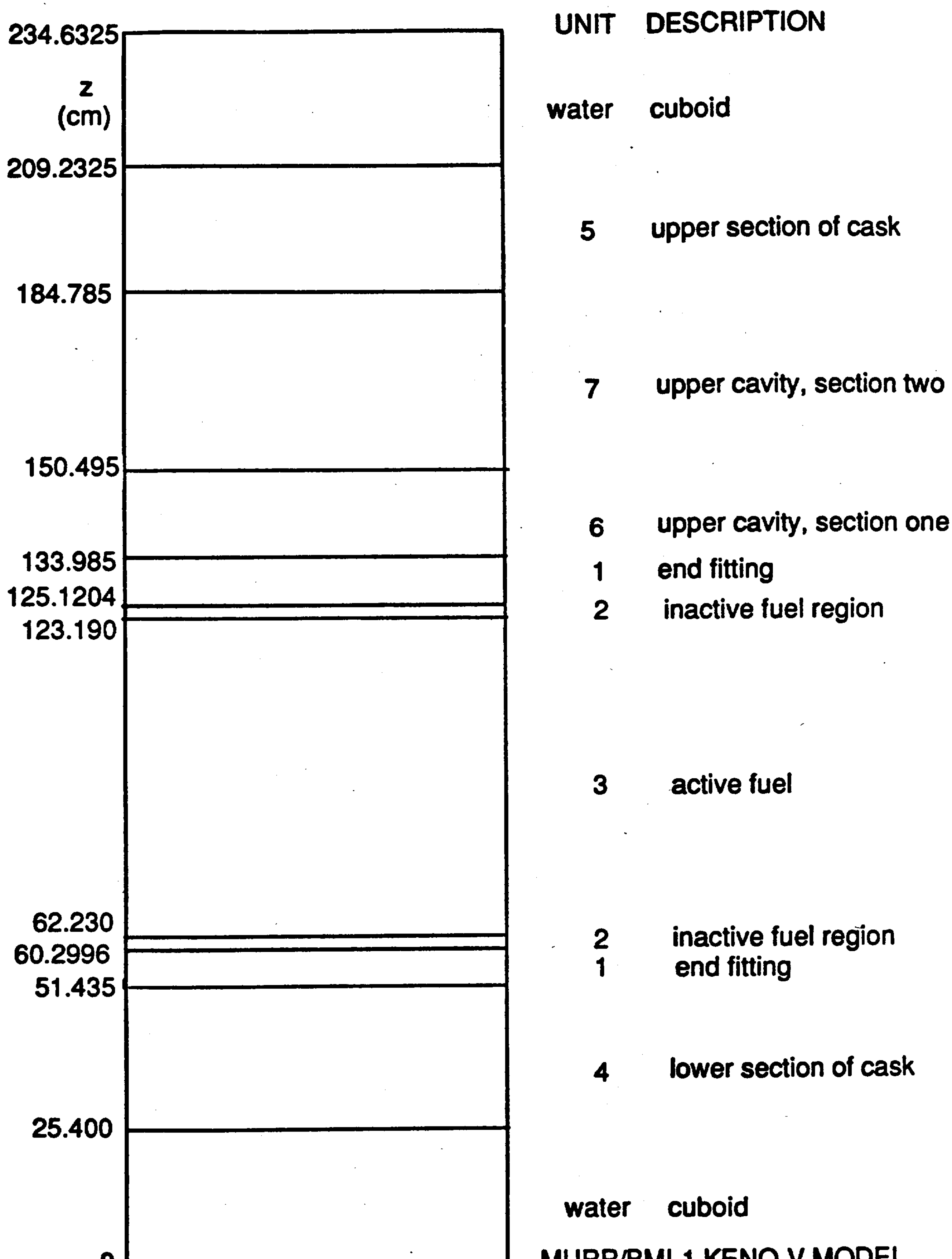
(10) Petrie, L. M., and Cross, N. F., "KENO-IV, An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830.

6.12.2 MURR Fuel Criticality Analysis



The rectangular parallelepiped of dimensions 84 cm x 84 cm x 234 cm defines the outermost region referenced in the KENO-V Manual NUREG/CR-0200 VOL. 2 Section F11.4.6 ALBEDO DATA. The parallelepiped is formed from water cuboids in the UNIT definitions. The enclosed cylinder is formed by the stack of UNITS in the sequence 4 1 2 3 2 1 6 7 5 from Z = 0 to Z = Zmax. Water cuboids on UNITS 4 and 5 add 25.4 cm of water above and below the cylinder. The cylinder sides are tangent to the parallelepiped vertical sides. The cylinder has radius 42.2275 cm, and its axis passes through the point (x,y) = (42.2275 cm, 42.2275 cm).

MURR/BMI-1 KENO-V MODEL ARRAY GEOMETRY



MURR/BMI-1 KENO-V MODEL
UNIT BLOCK DIAGRAM

MURR/BMI-1 KENO-V MODEL MIXTURES AND UNITS

1. Inner cylinder of fuel element end fitting section. Includes cask water, liner 304SS plates, and fuel element aluminum homogenized in cylinder.

2. Pure 304SS. Use Hansen Roach mixture ID 200.

3. Pure water, density 1 g/cm^3.

4. Pure lead, density 11.35 g/cm^3.

5. Coolant channel, including fuel element side plate of 6061 aluminum and liner 304SS intersected if coolant channel swept through 2 p radians. Homogenized throughout the sum of 23 different effective volumes $\pi\,(R_{out}^2 - R_{in}^2)$.

6. Fuel plate cladding in inactive fuel section (3/4" from end of fuel to end of fuel plate) of 6061 aluminum including fuel element side plate of 6061 aluminum and liner 304SS intersected if coolant channel swept through 2 p radians. Homogenized throughout the sum of 48 different effective volumes $\pi\,(R_{out}^2 - R_{in}^2)$.

7. Like 6., except in active fuel region.

8-31. Fuel, including U-235, U-238, and aluminum in UAl_x and aluminum powder. Intersected aluminum side plate and 304SS liner radial walls disregarded. Homogenized individually in 24 different effective volumes $\pi\,(R_{out}^2 - R_{in}^2)$.

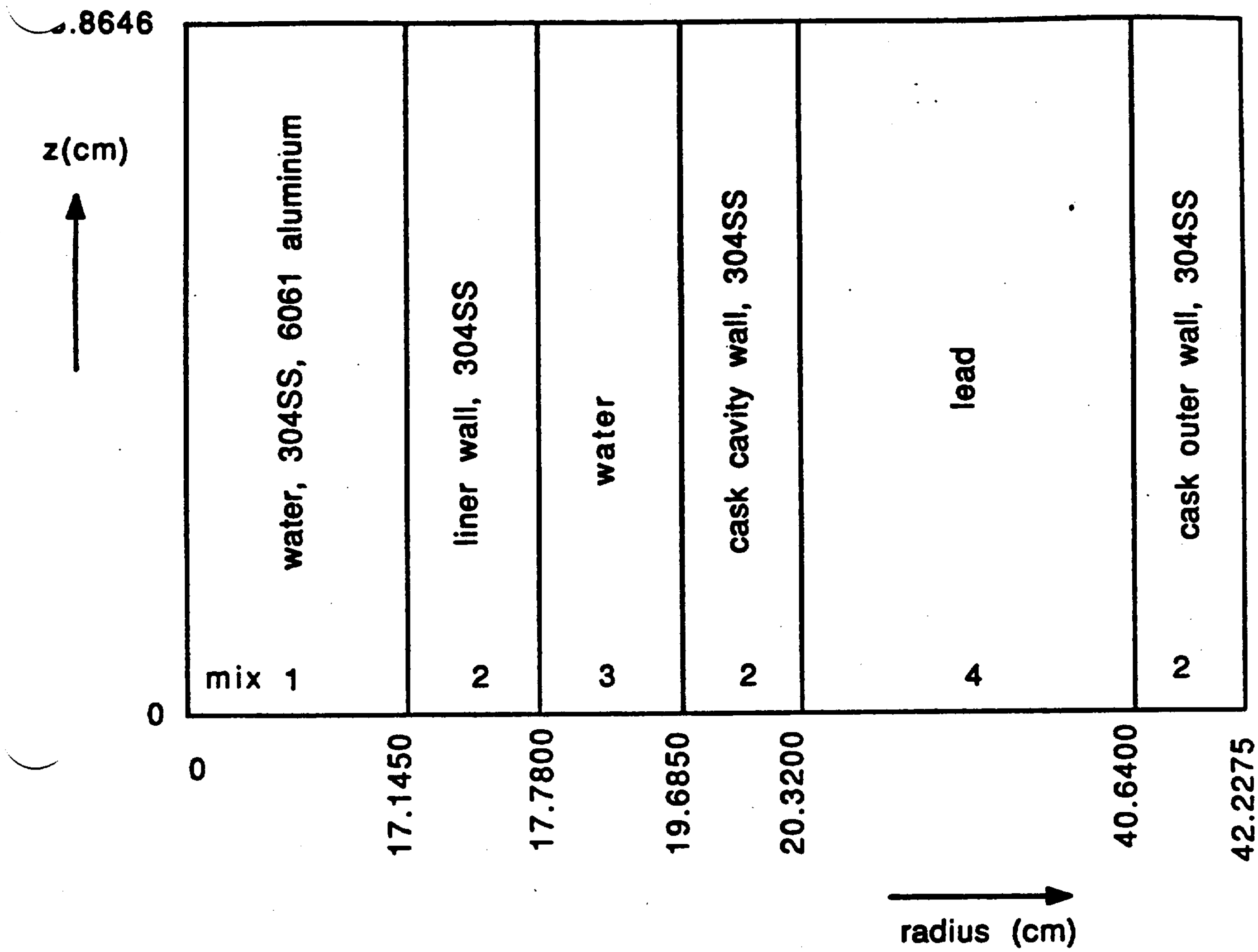
Total eight element mass of U-235 is 6200.0 grams and of U-238 is 455.94 grams; 93.15 w/o.

NOTE: Unirradiated fuel used. No credit taken for U-235 burnup or for existence of fission products as neutron absorbers.

32. Inner cylinder of water and liner radial walls 304SS in active and inactive fuel section.

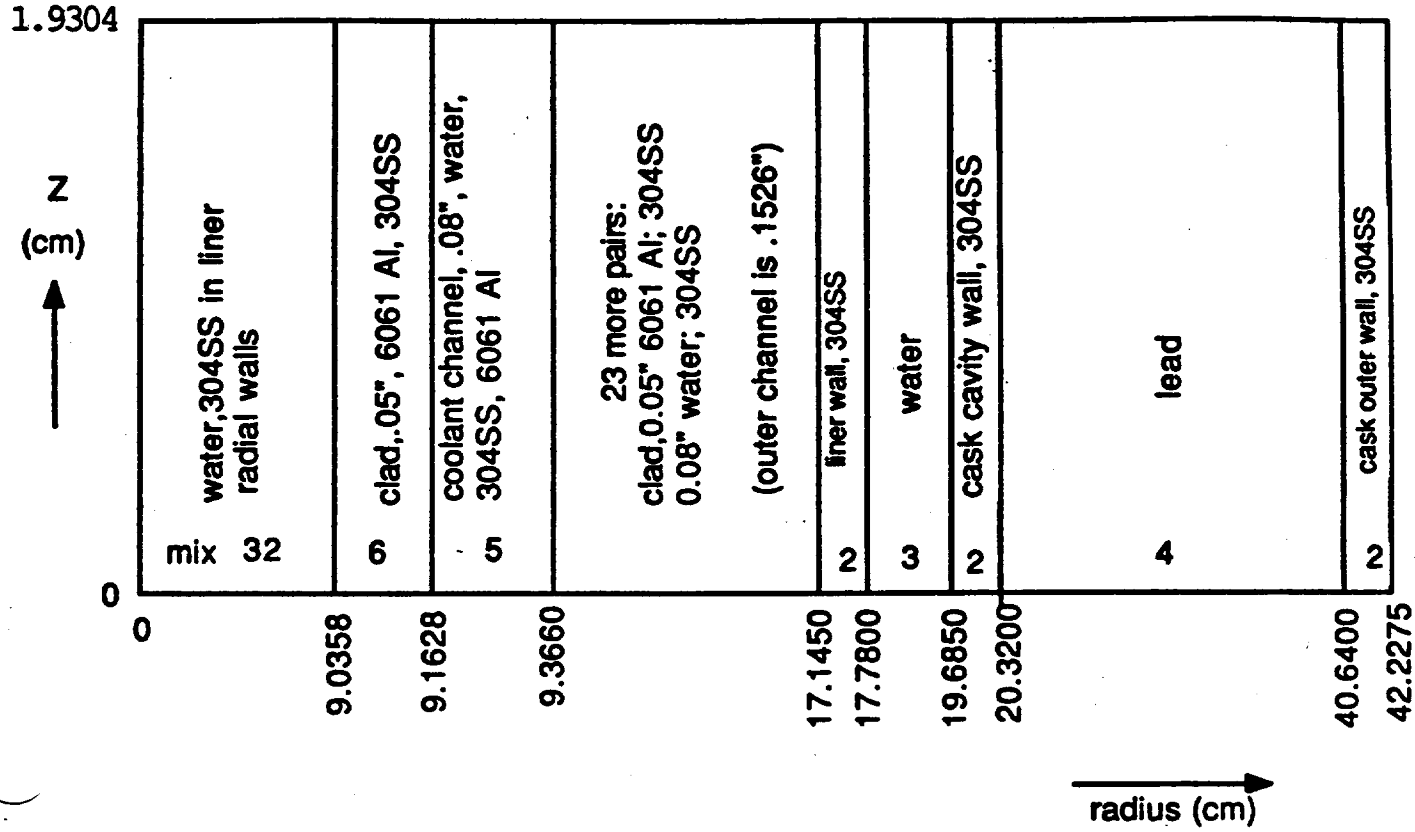
33. 1/4" 304SS plates at liner bottom (unit 4) and spacer top (unit 7) and bottom (unit 6). Water in holes in plates.

34. Water and 304SS in 4" pipes in main volume of spacer. Homogenized throughout cylinders of radius 17.1450 cm.



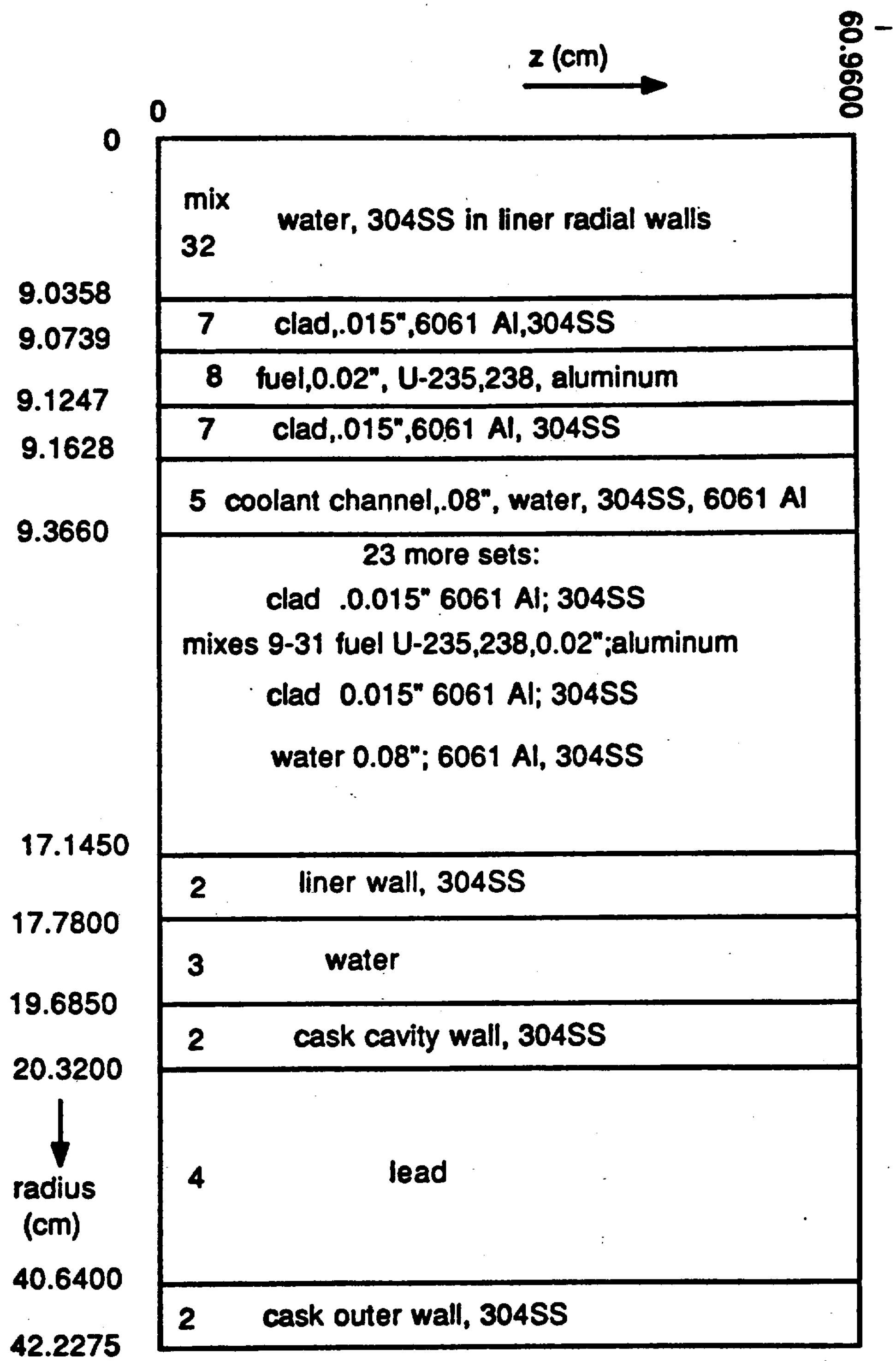
MURR/BMI-1 KENO-V MODEL
UNIT 1 : END FITTING SECTION

Rev. I, February 28, 1995

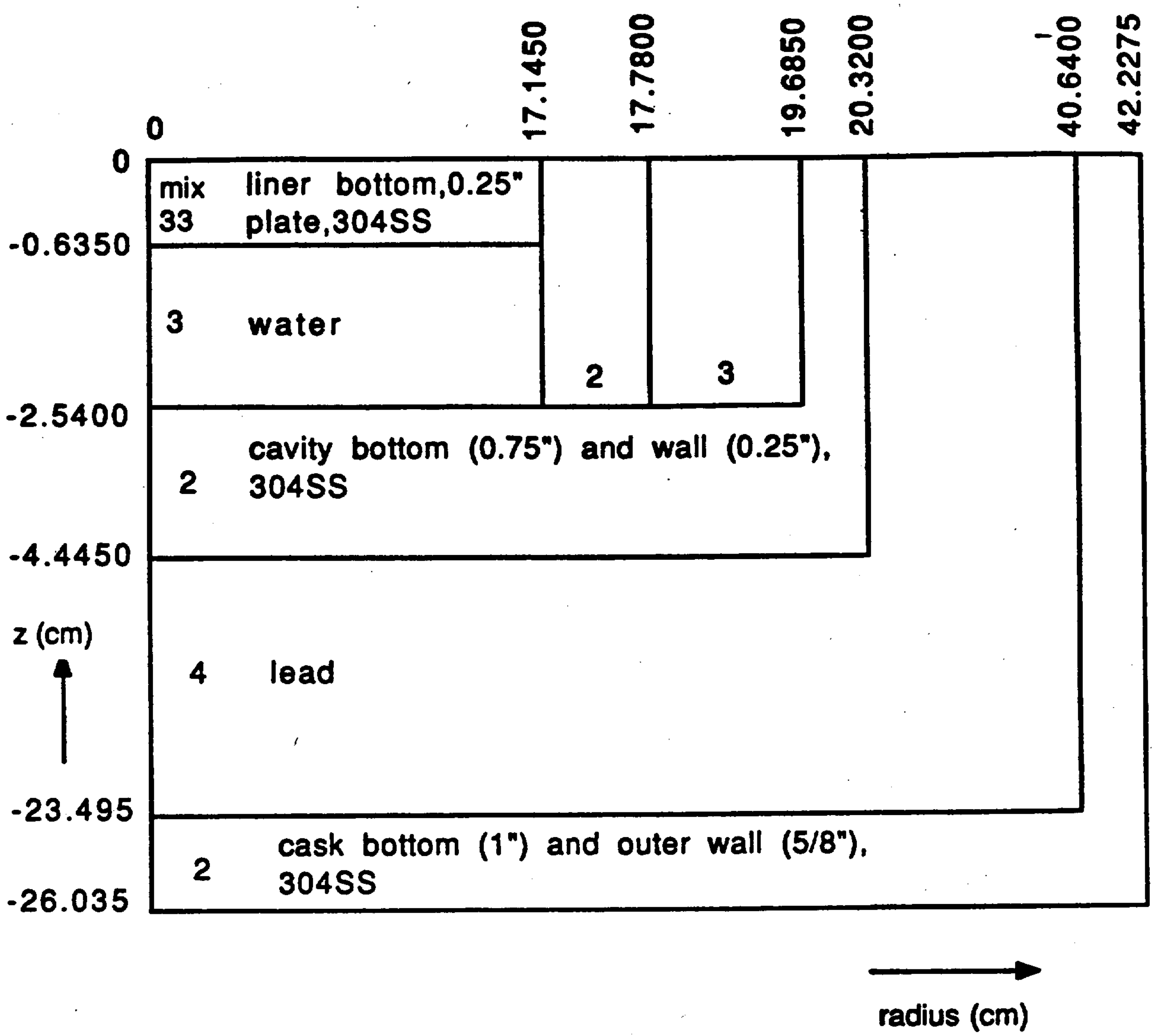


MURR/BMI-1 KENO-V MODEL
UNIT 2 : INACTIVE FUEL SECTION

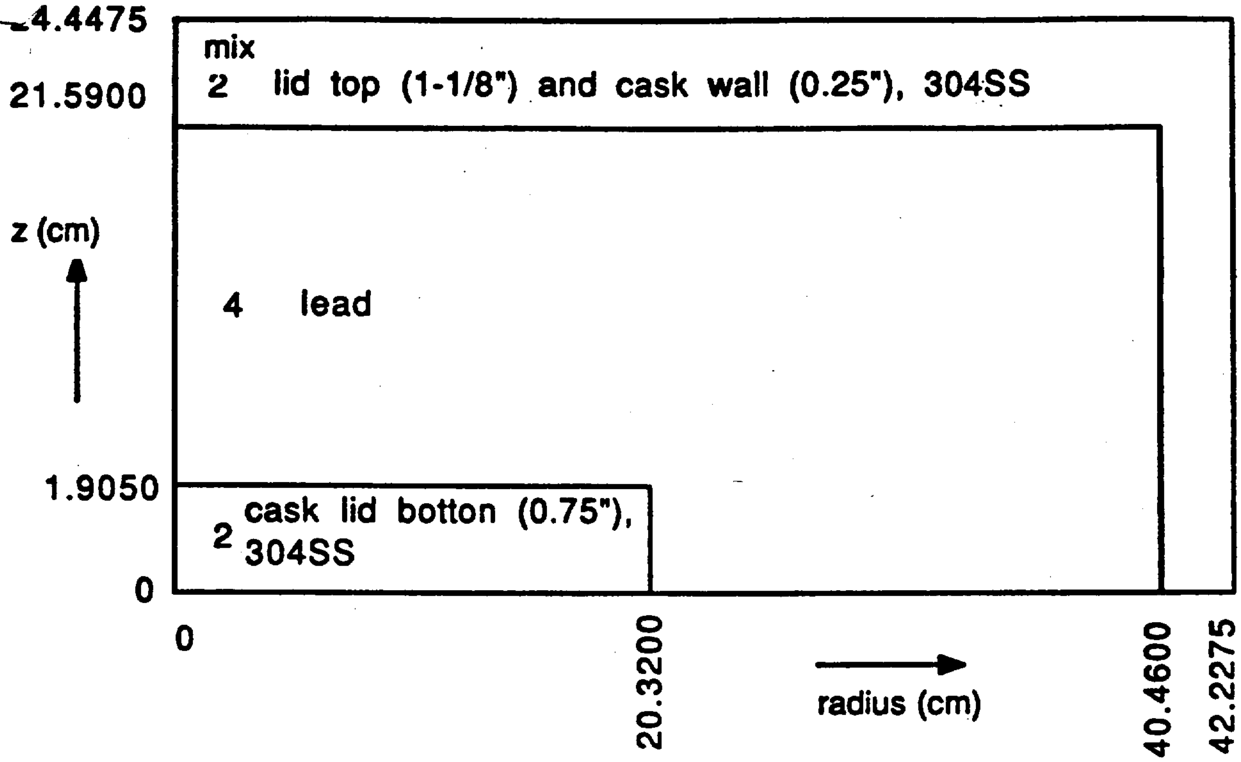
Rev. I, February 28, 1995



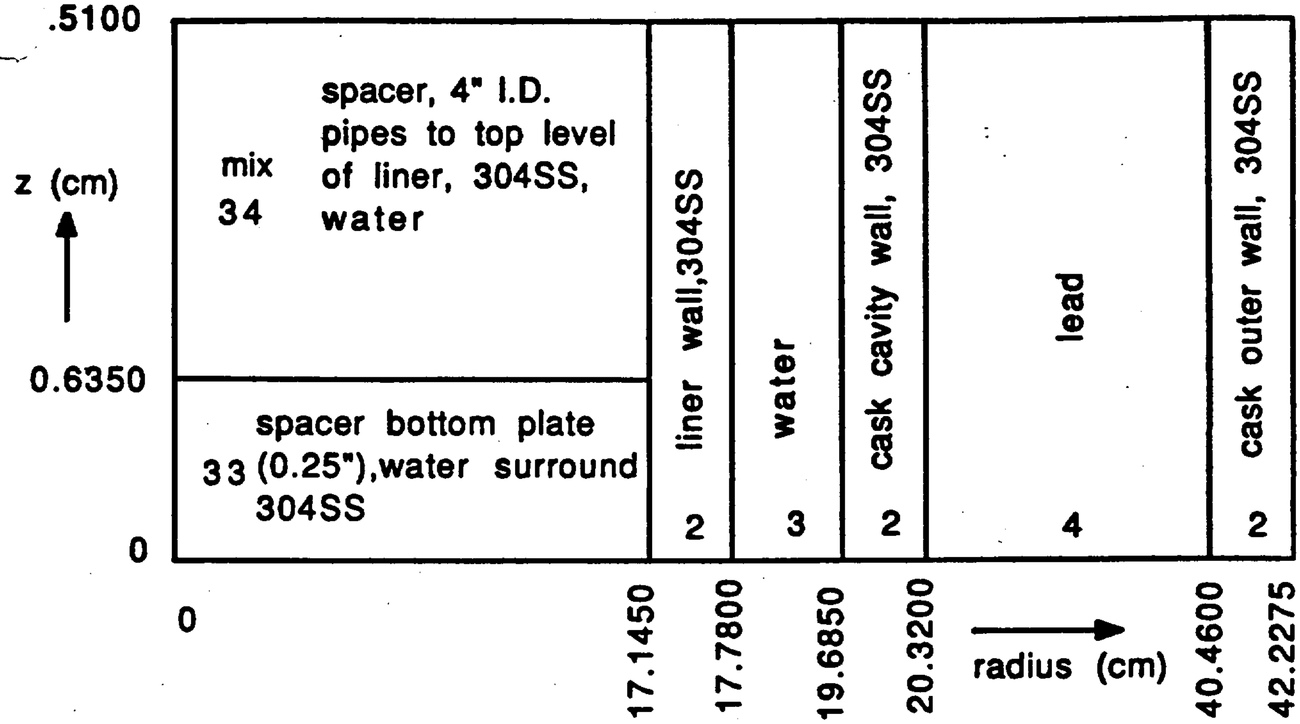
MURR/BMI-1 KENO -V MODEL
UNIT 3 : ACTIVE FUEL SECTION



MURR/BMI-1 KENO-V MODEL
UNIT 4 : LOWER SECTION OF CASK



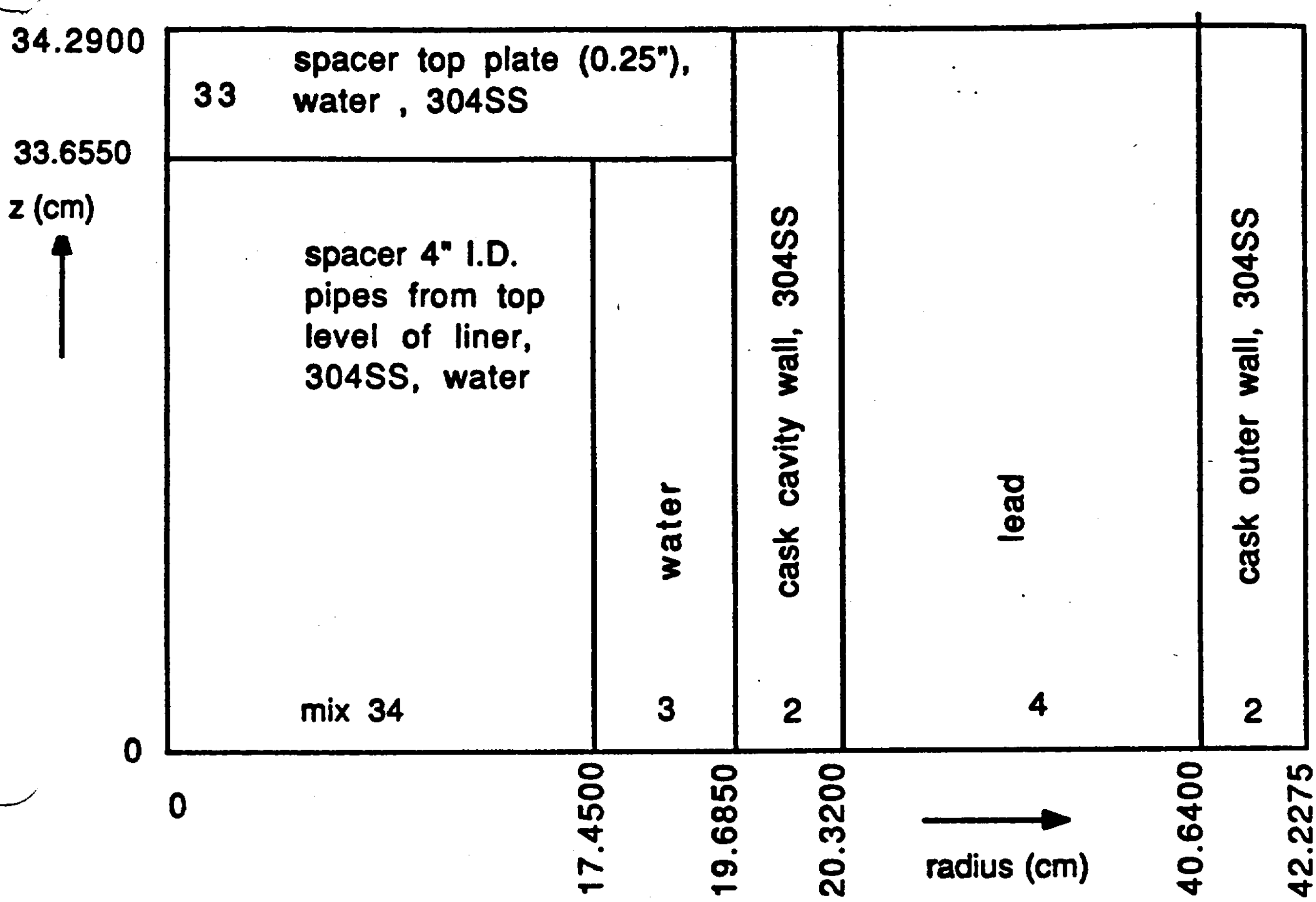
MURR/BMI-1 KENO-V MODEL
UNIT 5 : UPPER SECTION OF CASK



MURR/BMI-1 KENO-V MODEL

UNIT 6 : UPPER CAVITY, SECTION ONE

Rev. I, February 28, 1995



MURR/BMI-1 KENO-V MODEL
UNIT 7 : UPPER CAVITY, SECTION TWO

Rev. I, February 28, 1995

TABULATION OF $K_{EFFECTIVE}$ VALUES FROM KENO-V CODE FOR MURR/BMI-1 SPENT FUEL MODEL

$k_{effective}$	$k_{effective}$
0.69698	0.70077
0.71528	0.71105
0.70904	0.70113
0.70887	0.71468
0.71708	0.71325
0.70083	0.70443
0.70491	0.71444
0.70438	0.70333
0.70681	0.70787
0.70468	0.71460
0.70704	0.71363
0.70384	0.70006

Average $k_{effective}$ = 0.70746

The above 24 values have a standard deviation of 0.0056. This agrees with the KENO-generated standard deviations which were all in the range 0.004 - 0.005.

6.12.3 MITR Criticality Studies of BMI-1 Shipping Container

A KENO-V model of eight MITR-II fuel elements inside the BMI-1 cask was made by J.A. Carvajal and is described in the attached report. Ten KENO-V runs were made to analyze this model for criticality.

The results of these runs are shown below. The average $K_{effective}$ value is 0.73159 with a standard deviation of 0.00498. This agrees well with the KENO generated deviation which were all within the range of 0.0035 and 0.0055.

$K_{effective}$ values from KENO-V Code for BMI-1 Cask

Run	Keff	Standard Deviation
1	.72875	.00443
2	.72154	.00440
3	.72679	.00397
4	.73567	.00523
5	.72977	.00470
6	.73499	.00480
7	.73513	.00393
8	.73290	.00474
9	.73831	.00479
10	.73202	.00456

To test meeting the requirements of 10 CFR 71.61, two BMI-1 casks together were modeled using the above technique. The resulting $K_{effective}$ value was 0.73148 with a standard deviation of 0.00465.

In addition, a calculation was made to test the conservatisms involved in the homogenizing of the MITR-II fuel elements in the Carvajal report. The results of a heterogeneous model showed $K_{effective}$ to be 0.65088, with a standard deviation of 0.02987.

7 OPERATING PROCEDURES

7.1 Summary of the Procedures

The procedures outlined in this section are designed to be generally applicable for handling the BMI-1 cask during loading and unloading operations. The procedures are arranged chronologically. Each sequence of procedures is intended to address the needs of a specific type of user or group of users. The order of the procedures is:

(1) Preparation before the Cask Arrives
(2) Cask Receipt and Inspection
(3) Loading the Cask
(4) Unloading the Cask
(5) Closing the Cask
(6) Preshipment Testing and Preparation
(7) Preparation of the Cask for Transportation
(8) Leak and Pressure Testing

7.2 Preparations before the Cask Arrives

7.2.1 Record Keeping

7.2.1.1 For each shipment using the BMI-1, records specified in these procedures shall be kept on file for three years following the completion of the shipment.

7.2.1.2 For convenience in maintaining records, and as an aid to completion of all procedure requirements, each user should have a complete set of procedures, specific to its location, organized as checklists or travellers. Results of tests, comments, and signoffs shall be provided for on the checklist.

7.2.1.3 All record entries called for in the procedures shall be initialled by the operator or other authorized person.

7.2.1.4 Each user of the BMI-1 cask is required to have a copy of the BMI-1 SARP and the latest version of the NRC and DOE Certificates of Compliance.

7.2.2 Preloading Operations

Prior to each shipment of radioactive material in the cask, the following shall be verified as being in compliance with USNRC Certificate of Compliance Permit Number USA/5957/B( )F:

7.2.2.1 Permit is current and applicable;

7.2.2.2 Shipper is registered as a user under 10 CFR 71.12 (b);

7.2.2.3 Assure contents to be shipped qualify as to material type, form, and maximum quantity. This shall include meeting the limits for decay heat, fissile quantities and external radiation limits;

7.2.2.4 Requirement for an internal containment canister or basket has been met;

7.2.2.5 The maximum gross weight of the cavity contents do not exceed 1800 lbs;

7.2.2.6 The consignee is properly licensed to receive the material and consignee's license is current;

7.2.2.7 The annual and biennial inspections have been performed within the prior 12 and 24 months, respectively.

7.2.3 Materials

It is recommended that the following materials be on hand before starting to work with the cask.

(1) Procedures and checklists specific to local site
(2) Plastic bags
(3) Blotter paper or clean newsprint
(4) Gloves, coveralls, shoe covers
(5) Portable survey instruments
(6) Swipe survey equipment
(7) Teflon® tape
(8) Socket wrenches
(9) Torque wrenches
(10) Allen keys
(11) Neo-Lube® lubricant
(12) Snoop® or other bubble test liquid
(13) Pressurized air or inert gas source
(14) Ladder

7.3 Cask Receipt Inspection and Opening

7.3.1 Cask Arrival

7.3.1.1 Move the transport vehicle to the unloading area. Secure the trailer prior to removing the tractor (if applicable).

7.3.1.2 All external components of the cask and skid should be inspected for damage. If significant damage is evident, NRC and DOE/IPD must be notified; the equipment may not be used until IPD approval is granted. (Ref. 10 CFR 71.95)

7.3.1.2.1 The cask shall be examined for any obvious deformities or flaws, especially in weld areas, which would impair the safety of the cask. The safety plugs should be inspected for any flaws.

7.3.1.2.2 The skid shall be examined to assure that it is not twisted, bent, or otherwise damaged such that it would impair the safety of the package. If the skid is not attached to the cask, the bolt holes in the skid shall be inspected to assure that they are free of foreign material and that the threads are not damaged; and the bolts shall be examined to assure that they are not damaged.

7.3.1.3 Before proceeding with any unloading activities, the trailer, cask, skid, and tool box shall be surveyed for radiation and contamination. Results of the survey shall include radiation levels at the locations specified in 49 CFR 173.441 and the contamination levels specified in 49 CFR 173.443. Survey results shall be recorded on the checklist. If any of the levels exceed those specified in 49 CFR 173.441 or 49 CFR 173.443, notify supervision and perform a complete survey and decontamination. (Ref. 10 CFR 20.205)

7.3.1.4 If the cask is carrying a payload, it will be found sealed. Verify that all three seals on the cask are unbroken. Record the cask seal numbers. Verify that the seal numbers correspond to those on the shipping papers.

If any seals are broken or missing or if the numbers do not correspond to those on the shipping papers, STOP and notify supervision. Do not proceed without permission from supervision.

7.3.1.5 Remove the seals.

7.3.1.6 The cask may be removed from the trailer with or without the skid. If the skid is to be removed with the cask, loosen and remove only the tiedown cables securing the cask to the trailer. Otherwise, go to Step 7.3.1.9.

7.3.1.7 Verify that the lifting yoke has been tested within 5 years. Note on checklist.

7.3.1.8 Remove the cask and skid from the trailer.

7.3.1.9 Loosen and detach the turnbuckle anchors if the cask is to be removed from the skid. Unbolt the cask from the skid only after detaching the turnbuckle anchors.

7.3.1.10 Mark the orientation of the lid in relation to the cask body and the orientation of the cask in relation to the skid.

7.3.1.11 Transfer the cask, as required, into the reactor building or other loading area.

7.3.1.12 Perform a swipe survey of the trailer bed underneath where the cask was located. Record the results.

7.3.1.12.1 The radiation dose rate due to contamination on any accessible surface of an empty transport vehicle shall not exceed 0.5 mRem/hr at the time it is released for public use.

7.3.2 Preparation for Opening the Cask

7.3.2.1 If the cask is flooded and contains fissile material, remove the plug on the cask lid and insert a thermometer or thermocouple. Otherwise go to Step 7.3.2.4

7.3.2.2 Record the temperature of the cask. **If the temperature exceeds 300° F (150° C), STOP. Do not proceed without permission from supervision.**

7.3.2.3 Remove the thermometer and replace the plug.

7.3.2.4 Remove the cover plates from the fill and relief valve assembly cover box (upper box) and the drain valve assembly cover box. Record the pressure on the pressure gauge. If the pressure is $>$ 0 psig, take appropriate measures to control any gas release during Step 7.3.2.7.

7.3.2.5 After verifying that the fill and drain valves are closed, remove the access plug from the upper cover box and the plug from the fill valve. Remove the drain valve plug.

7.3.2.6 Attach a gas filtration (HEPA) manifold to the cask fill line to prevent release of contaminated gas to the atmosphere before proceeding. See Figure 7.1

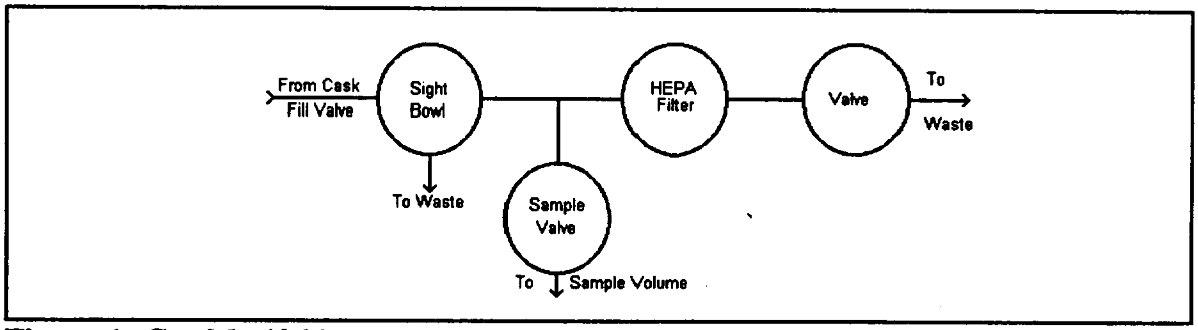


Figure 1 Gas Manifold

NOTE

In the event that liquid enters the manifold and passes through the filter, the in-line filter element shall be replaced.

7.3.2.7 Open the fill valve and slowly relieve the pressure in the cask.

7.3.2.8 Open the sampling valve and withdraw a sample of the gas inside the cask. Close the valve.

7.3.2.9 Analyze this gas and record the results of the analysis before proceeding. Verify that the indicated activity levels are within facility operating limits. **If not, STOP. Do not proceed without permission from supervision.**

7.3.2.10 If the cask is to be unloaded in a pool, go to Step 7.3.2.12. Otherwise, proceed to Step 7.3.2.11.

7.3.2.11 Dry cask

7.3.2.11.1 Connect a hose or tube from the drain valve to a liquid collection tank.

7.3.2.11.2 Open the drain valve and drain any residual liquid from the cask. Dispose of any water collected as hot waste.

7.3.2.11.3 Close the drain valve and disconnect the hose.

7.3.2.11.4 Attach a dry air or inert gas source (1-2 psig) to the drain valve.

7.3.2.11.5 Open the drain valve. Purge the cask through the HEPA filter with approximately 2-3 cask volumes of gas. Close the drain valve.

7.3.2.11.6 IF THE CASK IS EMPTY, perform Step 7.3.2.11.7. Otherwise, go to Step 7.3.3.

7.3.2.11.7 To help ensure that a leak is not found AFTER the cask is loaded, leak test the cask according to the BMI Leak Test Procedure, 7.10. If a leak is found, note it on the records. The leak must be repaired BEFORE loading the cask. Go to Step 7.3.3.

7.3.2.12 Flooding the Cask for Underwater Unloading

7.3.2.12.1 Verify by calculation or measurement that the cask contents permit the cask cavity to be safely flooded with water.

7.3.2.12.2 Connect a water line to the drain fitting. The water line shall contain a check valve to prevent water in the cask from flowing into the water supply. Open the water supply valve, and the drain valve and fill the cask with water. Note that the cask may already be full or nearly so with water.

7.3.2.12.3 Flush one additional liter of water through the cask, collecting it from the tap on the manifold attached to the fill valve. Shut the drain valve.

7.3.2.12.4 Collect a sample of the overflow water for counting. Discard the remainder to an appropriate hot waste container.

7.3.2.12.5 Verify that the indicated activity levels are within facility operating limits. **If not, STOP. Do not proceed without permission from supervision.** Shut the fill valve and remove the water supply and the gas manifold.

7.3.3 Opening the Cask

Proceed with Step 7.3.3.1 or Step 7.3.3.2, as appropriate.

7.3.3.1 Empty Cask or Dry, Loaded Cask

7.3.3.1.1 Remove the nuts from the cask lid.

7.3.3.1.2 Lift the lid carefully, using the two lifting plates on the lid, and set it aside. **DO NOT** use the ring in the lid center. Observe the O-ring under the lid as it is removed. Note whether the O-ring is damaged or out of place.

7.3.3.1.3 If the cask is to be unloaded, go to Step 7.5. If the cask is empty, go to Step 7.4.

7.3.3.2 Loaded Cask, Opening in Pool

7.3.3.2.1 Move the cask to a point above the pool.

7.3.3.2.2 Open the drain valve and lower the cask until it is nearly submerged. Open the fill valve. **DO NOT REMOVE THE FILL VALVE ASSEMBLY.**

7.3.3.2.3 Remove the nuts from the cask lid.

7.3.3.2.4 Lower the cask to the floor of the pool and remove the lifting yoke.

7.3.3.2.5 Lift the lid carefully, using the two lifting plates on the lid, and set it aside. **DO NOT** use the ring in the lid center. Observe the O-ring under the lid as it is removed. Note whether the O-ring is damaged or out of place. **EXERCISE CAUTION TO AVOID DISLODGING THE GASKET.**

7.3.3.2.6 Go to Step 7.5

7.4 Loading the Cask

7.4.1 Inspect the Contents

7.4.1.1 If the basket or inner can is to be replaced, remove the basket or can to permit installing a different basket or can. Decontaminate the removed basket, as required, and store it.

7.4.1.2 Before installing any basket or can, inspect it for any deformed areas. Inspect the bails or lifting eyes to assure that they are not frayed or otherwise damaged and that their attachments to the units are secure. Welds shall be inspected for cracks and flaws.

7.4.1.3 Verify that any required poison plates are intact. This may be done visually for baskets that were originally inspected by neutron measurements. Record the results. **If there is evidence of damage to the poison plates, STOP. Do not proceed without permission from supervision.**

7.4.1.4 Clean the basket or can as necessary to ensure that there is no foreign material present, and install it, if it is not already installed, in the cask cavity.

7.4.1.5 Pour one liter of water into the cask cavity and collect it from the drain valve to ensure the drain is clear.

7.4.1.6 Proceed with Step 7.4.2, 7.4.3, 7.4.4, or 7.4.5, as appropriate.

7.4.2 Spent Fuel - Pool Loading

7.4.2.1 Prepare for loading spent fuel.

7.4.2.1.1 Attach the cask lifting yoke to the crane and ensure all connections are completely engaged before moving the cask. Ensure all securing pins are properly positioned.

7.4.2.1.2 Transfer the cask to the pool. The drain valve may be opened as the cask is lowered into the pool to facilitate filling and draining the cask.

7.4.2.2 Loading the Cask - Pool.

Load the basket. Axial movement of fuel assemblies must be limited so that the active fuel region will remain correctly positioned with respect to the poisoned section of the basket. The contents must be securely confined in the cask cavity to minimize movement.

7.4.2.3 Closing the Cask - Pool

7.4.2.3.1 Inspect the O-ring according to Procedure 7.9.

7.4.2.3.2 Attach the lid to the crane using the lid lifting plates. With the pins and holes aligned and the lid centered above the cask, lower the lid on to the cask. **Exercise caution during this step, as the O-ring may be dislodged and fail to seat properly.**

7.4.2.3.3 Attach the cask lifting yoke to the cask and ensure proper engagement. Raise the top of the cask just above the surface of the pool.

7.4.2.3.4 Install two lid nuts. Neo-Lube or similar lubricant should be applied to all the stud threads before installing the nuts.

7.4.2.3.5 Remove the cask from the pool and install the remaining lid nuts.

7.4.2.3.6 If the cask is to be shipped "dry," drain the cask. Monitor for increased radiation levels. This water may be contaminated and should be handled accordingly.

7.4.2.3.7 If the cask is to be shipped "wet," attach a hose to the drain line valve. Open the drain valve (and the fill valve, if necessary) and extract between 4.33 and 5.0 gallons of water. This water may be contaminated and should be handled accordingly.

7.4.2.3.8 If the cask is to be shipped "wet," collect a sample of water for analysis.

7.4.2.3.9 Close the drain (and fill) valve.

7.4.2.3.10 Torque all twelve lid nuts, in 10 ft-lb increments to 50 ft-lbs, using a star pattern.

7.4.2.3.11 Wash and decontaminate all exterior surfaces of the cask.

7.4.2.3.12 If the cask is to be shipped "wet," collect a second sample of water from the drain valve no less than four (4) hours after the first sample. Install the drain valve plug, using Teflon tape. Analyze and record the results of the analysis. **If the two results of the two analyses differ by more than 10%, STOP. Notify supervision for instructions on proceeding.**

7.4.2.3.13 Go to Step 7.7.

7.4.3 <u>Spent Fuel - Dry Loading</u>

7.4.3.1 Load the basket. Axial movement of fuel assemblies must be limited so that the active fuel region will remain correctly positioned with respect to the poisoned section of the basket. The contents must be securely confined in the cask cavity to minimize movement.

7.4.3.2 Go to Step 7.6

7.4.4 <u>Non-Fissile Hardware or Material in Special Form</u>

7.4.4.1 Load the payload, ensuring that the payload is securely confined within the cask cavity to minimize movement.

7.4.4.2 Go to Step 7.6

7.4.5 Radioactive Material, not in Special Form, in the Inner Can

7.4.5.1 Load the payload, ensuring that the payload is securely confined within the inner can to minimize movement.

7.4.5.2 Inspect the inner can O-ring and sealing surfaces carefully for signs of damage. There must be no cracks, cuts, or permanent deformation of the O-ring. The sealing surfaces must be clean and free of scratches. Replace the O-ring if any damage is found or if the O-ring has not been replaced within the past twelve (12) months. Record the results of the inspection and whether the O-ring was replaced.

7.4.5.3 Replace the lid of the inner can and install the lid bolts. Neo-Lube or similar lubricant should be applied to the bolt threads before installing the nuts. Tighten nuts to a torque of 60 ± 10 inch pounds.

7.4.5.4 Go to Step 7.6.

7.5 Unloading the Cask

7.5.1 Unload the Cask

7.5.1.1 Remove the payload from the cask and store it.

7.5.1.2 Verify that the received material is identical to the contents specified on the shipping papers. Record this.

7.5.1.3 If the cask is dry, go to Step 7.6, otherwise, proceed with the next step.

7.5.2 Removing the Cask from the Pool

7.5.2.1 Lift the cask lid from the pool floor and move it to just above the cask. **EXERCISE CAUTION TO AVOID DISLODGING THE GASKET.**

7.5.2.2 Align the lid with the cask body and carefully set the lid on to the cask body.

7.5.2.3 Attach the lifting yoke to the cask and raise the cask to the surface of the pool.

7.5.2.4 Drain the cask.

7.5.2.5 Attach a dry air or inert gas source to the drain valve.

7.5.2.6 Purge the cask with 2-3 cask volumes of gas.

7.5.2.7 Rinse the exterior of the cask.

7.5.2.8 Attach a hoist to the lid lifting plates, remove the lid and set it aside.

7.5.2.9 Proceed with Step 7.6

7.6 Closing the Cask

NOTE: If the cask is loaded, go to Step 7.6.2.

7.6.1 Empty cask - Prepare for next Use

7.6.1.1 If a different basket or inner can is to be installed, remove the basket or inner can, to permit installing a different basket or can. Decontaminate the basket, as required, and store it.

7.6.1.2 Before installing any basket or can, inspect it for any deformed areas. Inspect the bails or lifting eyes to assure that they are not frayed or otherwise damaged and that their attachments to the units are secure. Welds shall be inspected for cracks and flaws. **If there is evidence of damage, STOP. Do not proceed without permission from supervision.**

7.6.1.3 Verify that any required poison plates are intact. This may be done visually for baskets that were originally inspected by neutron measurements. Record the results. **If there is evidence of damage to the poison plates, STOP. Do not proceed without permission from supervision.**

7.6.1.4 Clean the basket or can as necessary to ensure that there is no foreign material present, and install it in the cask cavity.

7.6.1.5 Inspect the O-ring according to Procedure 7.9.

7.6.2 Attach the Lid

7.6.2.1 Attach the lid to the crane using the lid lifting plates. With the pins and holes aligned and the lid centered above the cask, lower the lid on to the cask. **EXERCISE CAUTION DURING THIS STEP, AS THE O-RING MAY BE DISLODGED AND FAIL TO SEAL PROPERLY.**

7.6.2.2 Install the lid nuts. Neo-Lube or similar lubricant should be applied to the stud threads before installing the nuts. Torque all twelve lid nuts, in 10 ft-lb increments to 50 ft-lbs, using a star pattern.

7.6.2.3 If the cask has been immersed in a pool, go to Step 7.6.3. Otherwise, go to Step 7.6.4

7.6.3 Remove excess Water

In order to remove excess water from the cask cavity after loading or unloading in a pool, the cask should be purged with dry air.

7.6.4 Decontaminate the Cask

7.6.4.1 Wash and decontaminate all exterior surfaces of the cask.

7.6.4.2 Replace the protective plate over the lid bail.

7.7 Preshipment Testing and Preparation

7.7.1 Move the Cask

Move the cask, as required, to the final preparation area. If applicable, use Procedure 7.8.1

7.7.2 Leak Testing

7.7.2.1 Leak test the cask according to Procedure 7.10.

7.7.2.2 Verify that the fill and drain valves are shut.

7.7.2.3 Replace the fill and drain valve plugs, using Teflon tape to seal the threads.

7.7.2.4 Install the upper box access plug.

7.7.3 Thermal Testing the Loaded Cask

7.7.3.1 For shipments in a flooded cask, measure an equilibrium surface temperature.

7.7.3.1.1 Remove the plug from the thermocouple well on the cask lid and insert a thermocouple.

7.7.3.1.2 Record the temperature at hourly intervals until either:

(1) An asymptotic value for the maximum temperature can be determined, or

(2) The rate of temperature increase is less than 5 degrees F (5° F) in a one hour interval.

7.7.3.1.3 Remove the thermocouple and reinstall the plug, using Teflon tape.

7.7.3.2 For all shipments, verify that the temperature on any accessible surface does not exceed 122°F (non-exclusive use vehicle) or 180°F (exclusive use). [49 CFR 173.442(b)]

7.7.4 Contamination Testing

Survey the cask surface for radioactive contamination. Survey inside both the top and bottom valve assembly boxes. Surface contamination levels must be no higher than 2200 d.p.m./100 cm^2 of removable β/γ contamination or 220 d.p.m./100 cm^2 of α contamination (49 CFR 173.443). Record the results of the survey.

7.7.5 Sealing the Cask

7.7.5.1 Install the cover plates for both valve assembly boxes.

7.7.5.2 Apply lead security seals to the cask lid, drain line cover and upper pressure port cover. Record the seal numbers.

7.8 Preparation of the Cask for Transportation

7.8.1 Loading the Cask on to the Trailer

7.8.1.1 Using the lifting yoke, move the cask to a point above the skid.

7.8.1.2 Lower the cask to the skid in the same orientation as it was before it was removed.

7.8.1.3 Bolt the cask to the skid and secure the cask to the skid with the four (4) turnbuckle anchors.

7.8.1.4 If the skid was removed from the trailer, place the skid and cask on the trailer with the drain valve (lower) box facing the rear of the trailer.

7.8.1.5 Secure the cask to the trailer using four (4) sets of tiedown cables.

7.8.2 Radiation Survey

7.8.2.1 Perform a radiation survey of the cask and trailer. The maximum radiation levels must not exceed any of the following values (49 CFR 173.441):

(1) At any point on the outer surfaces of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load and on the lower external surface of the vehicle - 200 millirem per hour.

(2) At any point 2 meters (6.6 feet) from the outer lateral surfaces of the vehicle; or in the case of a flat-bed vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected from the outer edges of the vehicle (excluding the top and underside of the vehicle) - 10 millirem per hour.

(3) In any normally occupied space - 2 millirem per hour.

7.8.3 Labels and Documents

7.8.3.1 Label the Cask

7.8.3.1.1 Attach two address labels identifying both consignor and consignee and their addresses. Additional identification may be added as appropriate.

7.8.3.1.2 Fill out and attach labels, as applicable, in accordance with 49 CFR 172, Subpart E.

7.8.3.1.3 Verify that all container identification information is attached as required by the regulations.

7.8.3.2 Shipping Documents

7.8.3.2.1 Prepare and obtain required signatures on a Bill of Lading in accord with applicable regulations.

7.8.3.3 Review check sheets or travellers to verify all necessary procedures have been performed to ensure safe shipment. (49 CFR 173.475 and 10 CFR 71.87)

7.8.4 Package Transport

7.8.4.1 Apply "Radioactive" placards to all four sides of trailer in accordance with 49 CFR 172, Subpart F.

7.8.4.2 Obtain approval of driver and cognizant facility supervisor that the load and trailer are properly secured, marked, and sealed.

7.8.4.3 Give all applicable documents to the driver, including, but not limited to:
(1) Bill of Lading
(2) Emergency instructions

7.8.4.4 Release shipment to the driver.

7.9 O-ring Inspection

7.9.1 Inspect the O-ring under the cask lid and all the sealing surfaces carefully for signs of damage. There must be no cracks, cuts, or permanent deformation of the O-ring. The sealing surfaces must be clean and free of scratches. The O-ring must fit snugly in the recess at the top of the tapered portion of the lid.

7.9.2 Replace the O-ring if any damage is found or if the O-ring has not been replaced within the past twelve (12) months. Record the results of the inspection and whether the O-ring was replaced.

7.9.3 Do not remove the O-ring unless it needs to be replaced.

7.10 BMI-1 Cask Leak Test Procedures

The purpose of the leak test is to establish that the cask seal meets regulatory requirements. The maximum permissible leak rate permitted for the BMI-1 cask is 9.24×10^{-3} atm-cc/sec. The following procedure, with a sensitivity of 1×10^{-3} atm-cc.sec., meets the ANSI Standard N-14.5 requirement of a 4.62×10^{-3} atm-cc/sec. test to detect such a leak.

7.10.1 Leak Testing the Cask

7.10.1.1 Verify that the cask is already sealed and that drain and fill valves are closed.

7.10.1.2 Attach a pressurized air or inert gas line to the fill valve.

7.10.1.3 Open the fill valve and allow the pressure in the cask to rise to 50 psig. DO NOT LET THE PRESSURE EXCEED 55 PSIG. Hold this pressure for a minimum of fifteen (15) minutes.

7.10.1.4 Flood the seal region around the top flange, the seal under the fill valve, the seal under the drain valve, and the fill and drain ports with SNOOP® or similar bubble detection fluid.

7.10.1.5 Inspect all seal regions for leaks for a minimum of two (2) minutes, adding detection fluid as required to keep the seals thoroughly wet.

7.10.1.6 There must be no detectable bubble formation for the cask to be considered leak tight. If there are no detectable bubbles, record this and continue. If bubbles are detected skip to Step 7.10.2.

7.10.1.7 Shut the fill valve and remove the pressure line.

7.10.1.8 Open the fill valve and vent the cask. Shut the fill valve.

7.10.2 Repairing Leaks

7.10.2.1 If a leak is detected around the cask lid during the preloading inspection, the O-ring must be replaced. This can be done during the loading process, (Step 7.4) of the BMI-1 Loading Procedures.

7.10.2.2 If a leak is detected around the cask lid after the cask has been loaded, check that all the lid nuts are properly torqued (50 ft-lb). Repeat the test from Step 7.19. If the leak persists, the O-ring must be reseated or replaced. Return to Procedure 7.4.2, 7.4.3, or 7.4.4, as appropriate and repeat the test from Step 7.10.1.1

7.10.2.3 If a leak is detected around the O-ring seal under either the fill valve or the drain valve, carefully tighten the valve flange bolts and retest from Step 7.10.1.4. If the leak persists, depressurize the cask, replace the O-ring and record this.

NOTE

If the cask is loaded and contains water, it will be necessary to return the cask to the pool and unload it before attempting to repair the drain valve.

7.10.2.4 Repeat the test from Step 7.10.1.1. If the leak persists, record this and notify supervision for instructions.

7.10.2.5 If a leak is detected in any of the threaded connections, depressurize the cask, open the connection, clean the fittings, and remake the connection, using Teflon tape to seal the threads. Repeat the test from Step 7.10.1.1. If the leak still persists, record this and notify supervision for instructions.

7.10.2.6 If any other component of the valve assemblies is found to be defective, record this and notify supervision for instructions.

7.11 Relief Valve Test Procedure

7.11.1 Verify that the cask is already sealed and that drain and fill valves are closed.

7.11.2 Attach a pressurized air or inert gas line and calibrated pressure gauge to the pressure port.

7.11.3 Set the supply pressure between 75 and 80 lbs.

7.11.4 Open the fill valve and slowly raise the pressure in the cask to 75 lbs. Verify that the relief valve releases at between 65 and 75 lbs. Record the release pressure.

7.11.5 Shut the fill valve and remove the air supply.

7.11.6 Open the fill valve, release the pressure from the cask and shut the fill valve.

7.12 Periodic Inspection Procedures

This inspection shall be recorded and documented and filed with the packaging QA documents.

7.12.1 Annual Inspection

7.12.1.1 Visual Inspection

Visually inspect the body and lid surfaces. The cask shall meet the following standards.

7.12.1.1.1 All visible surfaces shall be free of corrosion, gouges, cracks, or other deformations which could impair the package's physical condition.

7.12.1.1.2 All sealing surfaces shall be free of pits, scratches, burrs, or other imperfections which could degrade the seal integrity.

7.12.1.1.3 All sliding surfaces shall be free of excessive wear or roughness not within design tolerances.

7.12.1.1.4 Painted surfaces shall be free of cracks, chips or blisters which could trap contamination.

7.12.1.1.5 Welds shall be free of cracks, tears, corrosion or other defects which could degrade their function.

7.12.1.1.6 Bolts and nuts shall conform to original design specifications and shall not be bent or otherwise deformed. Threads, including those in bolt holes shall be uniform and free of burrs. Provisions for installing seal wires shall be present and intact.

7.12.1.1.7 Reusable gaskets shall have no visible evidence of deterioration or damage and shall be replaced at least every 12 months in accordance with the certificate of compliance. They shall be retained in a position on one of the sealing surfaces in a manner which will not affect the seal.

7.12.1.1.8 Tiedowns shall be free of any apparent damage which could reduce their effectiveness.

7.12.1.2 Mechanical Inspection

Operate, move, or position mechanical devices as necessary to verify their proper function.

7.12.1.2.1 Guide pins, lugs, and the like shall not be deformed and shall function properly. Match marks shall be clearly visible.

7.12.1.2.2 Threaded fittings shall be free of galling or cross threading and shall function smoothly.

7.12.1.2.3 Lifting lugs, eyes, trunnions, guides, bearing surfaces and retainers shall indicate no misalignment, wear, or other deformation and shall function properly, without binding.

7.12.1.2.4 Valves shall be freely operable by hand pressure and shall function properly both open and closed. Lines shall be determined to be free of restrictions. Threaded connections shall be free of deformation or damage. Plugs or caps shall be provided for all line ends except those downstream of the pressure relief valve.

7.12.1.3 Leak Inspection

A leak test shall be performed in accordance with ANSI Standard N-14.5. Procedure 7.10 of this section is an acceptable test. The cask shall to demonstrate leaktightness at least to the degree specified in Procedure 7.10

7.12.1.4 Dryness Inspection

The expansion chamber shall be inspected for moisture according to Procedure 7.13, below.

7.12.2 Biennial Inspection

This inspection shall be recorded and documented and filed with the packaging QA documents.

7.12.2.1 Weld Inspection

Welds which are subject to stress and which must be sound for safe operation or compliance with regulations shall be inspected by dye penetrant, radiographic or ultrasonic methods. The results of the inspection shall be reviewed by a qualified inspector.

7.12.2.2 Pressure Gauge

The pressure gauge shall be tested and calibrated between 40 and 100 p.s.i.g. to verify its performance in accordance with specifications. The gauge may be compared with a calibrated gauge as an acceptable test.

7.12.2.3 Pressure Relief Valve

The pressure relief valve shall be tested in accordance with Procedure 7.11

7.13 Cask Dryness Check Procedure

This procedure shall be performed to inspect the BMI-1 lead cavities in the cask and lid for moisture.

7.13.1 Inspection

7.13.1.1NOTE: Both the cask body and the lid have lead filled cavities. Each must be inspected according to the following procedure. Items specific to the lid are noted in curly brackets, i.e., { }.

7.13.1.2 Remove the lead expansion chamber safety plug, near the bottom of the cask {in the top of the lid}. Check for moisture around the hole and on the plug.

7.13.1.3 Connect a vacuum system to the expansion chamber access hole. The vacuum system shall include a vacuum pump, gauge, filter, and a suitable trap (dry ice or liquid nitrogen cooled) for collecting water.

7.13.1.4 Evacuate the expansion chamber for 10 hours or more {2 hours for the lid}.

7.13.1.5 Note whether water is present in the cold trap. If water is present, go to Step 7.13.2. If not, the expansion chamber may be considered to be dry. Continue to the next step.

7.13.1.6 Bleed the expansion chamber to atmospheric pressure and disconnect the vacuum system.

7.13.1.7 Inspect the safety plug for damage. Replace the plug, if it is damaged, and install the plug, using Teflon tape or equivalent.

7.13.2 <u>Drying Procedure</u>

7.13.2.1 Install a heater in the cask cavity {around the lid} and insulate the cask {lid} to facilitate heating.

7.13.2.2 With the vacuum system running, heat the cask {lid} to approximately 150°F.

7.13.2.3 Continue evacuating the expansion chamber until water stops collecting in the trap.

7.13.2.4 Disconnect the heater, bleed the expansion chamber to atmospheric pressure and disconnect the vacuum system.

7.13.2.5 Inspect the safety plug for damage. Replace the plug, if it is damaged, and install the plug, using Teflon tape or equivalent.

8 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

The BMI-1 Shipping Cask has been in use in its present configuration since 1970. Tests of the package, therefore, properly are part of the maintenance program discussed in the next section.

8.2 Maintenance Program

8.2.1 References

10 CFR 71 Sub-part G	Operating Controls and Procedures
10 CFR 71 Sub-part H	Quality Assurance
49 CFR 173.441	Radiation Level Limitations
49 CFR 173.443	Contamination Control
49 CFR 173.475	Quality Control Requirements Prior to Each Shipment of Radioactive Materials.
ANSI-N 14.5	American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipments

8.2.2 Inspections

8.2.2.1 Types

The following types of inspections shall be performed and documented:

Periodic - Annual; Biennial
Preusage
Postaccident.

8.2.2.2 Frequency

(a) Periodic inspections shall be performed before initial use of the packaging and annually thereafter except during periods of prolonged storage. Items specified for biennial inspection may be included in the annual inspection on alternate years. No packaging shall be used which has not been inspected in the immediately preceding year for the items of Section 8.3.3.1 and the preceding two years for the items of Section 8.3.3.2.

(b) Preusage inspections shall be performed immediately prior to each shipment of the packaging.

(c) Post accident inspections shall be performed prior to use of the packaging after it has been subjected to unusual stress conditions, including fire, moving transportation accident, nuclear incident, free drop, internal or external explosion, pressurization above maximum design specification, exposure to materials corrosive to structural components, or freezing of liquid contents.

8.2.2.3 Inspecting Personnel

(a) Inspections shall be performed by personnel representing quality assurance organizations that are qualified in accordance with 10 CFR 71.101 through 71.137.

(b) Postaccident inspections shall be directed by a person in charge, designated by the responsible performing organization on the basis of engineering competence and familiarity with the Safety Analysis Report for Packaging for the BMI-1.

8.2.2.4 Records

(a) A corrosion-resistant metal tag indicating expiration date of most recent periodic inspection shall be firmly and securely affixed in a conspicuous location on the body of the packaging. A similar metal tag shall be affixed to any optional use components such as inserts. Where permanent attachment of a metal tag would damage the component, other durable methods of marking the expiration date, compatible with the material and service environment, shall be used.

(b) A record of each periodic inspection, including a description of defects and corrective actions which were taken, shall be prepared by the person in charge, and shall indicate his organizational title or responsibility. This record of inspection shall be placed in the Quality Assurance record file required in accordance with 10 CFR 71.91.

(c) A detailed description of all tests and inspections performed and the results thereof shall be prepared by the engineer responsible for performing a postaccident inspection. This record shall be placed in the Quality Assurance record file for the packaging. If the results of this inspection indicate the original design specifications are no longer applicable, then revisions of outstanding drawings, reports, specifications, licenses,and certificates of compliance reflecting the change shall be obtained.

(d) Handling and shipping procedures of the user of packaging shall contain appropriate instructions for preusage inspections and tests ensuring compliance with design and permit requirements. Record copies of these inspections shall be retained for at least 2 years after the shipment.

8.2.3 Periodic Inspections

8.2.3.1 Annual

An annual inspection together with the biennial inspection shall verify that the packaging continues to meet design specifications. Items for annual inspection shall include, but are not limited to the following:

(a) Surfaces. Surfaces shall be free of corrosion, gouges, cracks, or other deformations as determined by visual inspection. Painted surfaces shall be free of cracks, chips, or blisters.

(b) Sealing Surfaces. These surfaces shall be inspected for pits, scratches, burrs, corrosion and other imperfections.

(c) Sliding Surfaces. Surfaces which move relative to one another shall be inspected for excessive wear or roughness not within design tolerances.

(d) Welds. Welds shall be determined to be sound by visual inspection.

(e) Closure Devices. Bolts and nuts shall conform to original design specifications and shall not be bent or otherwise deformed. Threads shall be uniform and free of burrs. Provision for installing seal wires shall be present. Latches shall work smoothly and engage properly.

(f) Alignment Devices. Guide pins, lugs, etc., shall not be deformed and shall operate as intended. Match marks shall be clearly visible.

(g) Lifting Lugs, Eyes, and Trunnion. Visual inspection shall indicate no misalignment, wear, or other deformation which would significantly affect strength. Engagement guides and retainers shall be sound and operable as intended. Bearing surfaces shall be smooth to prevent binding.

(h) Valves, Lines, and Connections. Valves shall be freely operable by hand pressure. Lines shall be determined to be free of restrictions. Threaded connections shall be free of deformation or damage. "Quick-disconnect" type fittings shall operate freely. Plugs or caps shall be provided for all line ends except those downstream of pressure relief devices.

(i) Filters. Filters shall be replaced or inspected and tested to verify performance in accordance with design specifications.

(j) Reuseable Gaskets. These gaskets shall have no visible evidence of deterioration or damage. They shall be retained in position on one of the sealing surfaces in a manner which will not affect the seal. One-time-use gaskets are not inspected on periodic inspections.

(k) Tiedown. The tiedown shall comply with 49 CFR 173.475.

(l) Leaktightness. A test shall be performed to demonstrate leaktightness at least of the degree corresponding to the basis for the certificate of compliance, and in accordance with procedures in ANSI N 14.5. Procedure 7.9 of this SARP is an acceptable test.

(m) Lead Cavity Dryness. Dryness of the lead cavity shall be demonstrated by Procedure 7.12 of this SARP.

8.2.3.2 Biennial

Items to be inspected include the following:

(a) Welds. Welds which are subject to stress and which must be sound for safe operation or for compliance with regulations shall be inspected by dye penetrant, radiographic, or ultrasonic methods.

(b) Pressure Relief Valves and Instruments. Pressure relief valves and instruments such as pressure gages and thermocouples shall be bench-tested and calibrated as necessary to verify performance in accordance with specifications.

8.2.4 Preusage Inspections

Inspections shall be performed before each shipment of a loaded package or contaminated empty packaging to establish compliance with applicable regulations and standards. Items to be inspected include, but are not limited to, the following:

8.2.4.1 General Condition

Visual inspection shall be made for deformation of structural members, missing components, and other possible deficiencies. Any necessary maintenance shall be performed. Requirements may differ for loaded packages and empty packaging.

8.2.4.2 Closure

Reusable gaskets shall be inspected visually and determined to be sound before closure is made. Gaskets which are permanently deformed by normal use shall be replaced at each use. Rupture disks shall be visually inspected and determined to be free of imperfections, damage, or corrosion. All valves shall be closed, bolts and nuts shall be determined to be tightened within design torque specifications, and wire seals shall be installed.

After a loaded or internally contaminated package is closed, it shall be determined to be sealed by a leakage test of the sensitivity specified in ANSI N 14.5 to be used before each shipment, or by an alternative procedure approved by the USNRC or DOE Contracting Office having jurisdiction of the packaging if a leakage test of such sensitivity is impractical with the package loaded for shipment. When a potential leakage point is submerged in liquid on the upstream side and clearly visible on the downstream side, a hydrostatic test at 15 psig or more is acceptable, with any observed drops or seepage being cause for rejection. If a hydrostatic test is used with loss of pressure as a measure of leakage, account shall be taken of possible gas in the containment vessel, solubility of gas, temperature change, liquid compressibility, and stretch of the vessel.

8.2.4.3 Radiation and Contamination

Radiation and surface contamination surveys shall be performed on the loaded package. The results of these surveys shall fall within USDOT and USNRC standards.

8.2.4.4 Tiedown

The tiedown shall comply with design specifications and 49 CFR 173.475.

8.2.5 Postaccident Inspections

8.2.5.1 Purpose

Performance of postaccident inspection shall verify that the package complies with original design or permit specifications. Where characteristics have changed, but remain within permitted variances, these changes shall be noted in the Quality Assurance file.

8.2.5.2 Items of Inspection

Items of inspection shall include, but are not limited to the following:

(a) Dimensional stability. Measurements shall be taken to verify that all dimensions which affect performance are within specified tolerances. Particular attention shall be paid to flatness, straightness, or uniformity tolerances affecting fit between parts.

(b) Welds. Welds whose function is simple closure or joining shall be determined to be sound by visual inspection. Welds required to sustain stress shall be determined to be sound by nondestructive examination such as radiographic examination, liquid penetrant and/or ultrasonic means.

(c) Package Shielding. Where the nature of an accident is such that internal shielding material may have melted, deformed, or cracked, the shielding characteristics of the package shall be reevaluated. This shall be accomplished by enclosing a gamma source of known high energy level in the package cavity. Radiation measurements shall be taken on the surface of the package and shielding effectiveness shall be calculated. The effectiveness thus calculated shall not deviate significantly from that which would be expected from the original shielding design. Where neutron shielding is required, a similar test shall be performed using a neutron source.

(d) Heat Transmission. The heat transmission characteristics of the package shall be reevaluated by enclosing a heat source of design value in the package cavity and, after equilibrium with ambient conditions is achieved, measuring surface and internal temperature of the package. These transmission characteristics shall agree with the requirements of the certificate of compliance.

8.2.6 Preventive Maintenance

8.2.6.1 Definition

Preventive maintenance shall consist of repair, replacement, and adjustment during periods of usage or storage to ensure specified functioning and to avoid deterioration.

8.2.6.2 Frequency

Preventive maintenance shall be performed as follows:
(a) At the time of periodic inspection and at the time of inspection before each shipment, as considered necessary by the person in charge, to correct deficiencies which might lead to failure or damage.

(b) Before any period of protracted storage, to provide necessary protection against corrosion or other avoidable deterioration.

8.2.6.3 Records

All maintenance repairs shall be recorded and documented in accordance with NRC quality standards.

8.2.6.4 Items

Preventive maintenance is intended to be performed before failure or inoperability occurs. Such work shall include, but not be limited to, the following:

(a) Parts Replacement. Damaged, missing, deteriorated, or badly worn parts such as nuts, bolts, filters, gaskets, valves, cables, pressure relief devices and rupture disks shall be repaired or replaced. Such rework shall meet design specifications.

(b) Decontamination. Prior to shipment, the external surfaces of packages shall be decontaminated to levels within the limits specified by applicable regulations. Prior to storage, the external surfaces of packages shall be decontaminated to levels permitted in the specific storage area. The interior of packages shall be decontaminated as required to maintain consistency with the levels of contamination of items to be shipped therein. Decontamination shall remove contaminants. Application of paint, plastic film, or similar surface coatings to cover or shield contamination is not permitted.

8.2.7 Documentation

All inspections and preventative maintenance activities shall be recorded and documented in the quality assurance files of the organization performing these activities. Copies of these records shall be sent to the holder of the DOE Certificate of Compliance.