Journal of the Korean Nuclear Society
Volume 10, Number 1, March 1978

<Technical Report>

Slab Thickness Calculations on Hot Cell

Yung Joon Ha
Korea Nuclear Fuel Development Institute
Seong Yun Kim and Dong Hoon Kim
Korea Atomic Energy Research Institute

(Received January 30, 1978)

Abstract

Numerical computations of radioactivities and decay energies in a spent fuel have been carried out for designing of a hot cell. Optimum wall and window thicknesses that can preserve spent fuel rods for experimental purposes are estimated with burnup rate of 33,000 MWD/T(U) which is nearly maximum from a pressurized water reactor such as the Go-Ri Unit 1. Before putting the spent fuels into a hot cell, it is assumed for thickness estimates of shield materials that they are cooled in a storage bay for several time intervals. Considered are various types of shield materials through which changing the distances from a source to an observation point is also made.

요 약

Hot cell의 설계를 위하여 기사용 연료에서의 방사능과 붕괴 에너지의 수치적 계산을 하였다. 고리 1호기와 같은 경수로에서 거의 최대 연소율인 33,000MWD/T(U)으로 태워진 연료봉 시험을 위하여 보관할 수 있는 최적의 벽과 창 두께가 추정되었다. 기사용 연료를 hot cell에 넣기 전에 차폐물질의 두께 추정을 위해 그 연료를 여러 시간 간격동안 저장용기 속에서 냉각시켰다는 가정을 했다. 여러 종류의 차폐물질이 고려되었으며 방사선원과 관측점과의 거리도 변화시켜 보았다.

I. Introduction

Numerical calculations for determining hot cell wall thicknesses have been performed to guide the engineering design in the design of the facilities for storage of radioactive materials, with special emphasis on the shielding aspects. For hot cell construction, many materials of relatively high density are generally available for shielding against radiations. Various types of materials, such as concrete, lead, water and/or glass, are some of the more readily available ones. Before making selection of shield materials, thorough examinations of radiations from spent fuel and the associated material thicknesses must be carried out since the selected material should be reasonalble in cost and have sufficient density so that the wall will not be undesirably thick. The primary concern in determining slab thickness of a hot cell is to estimate the radioactivities and decay energies from hot materials (usually spent fuels) with respect to radiations which are approximately in proportion to the relative density of the material of construction. The spent fuels considered are assumed to be cooled for 3, 6, 9 and 12 months in a spent fuel storage bay. The physical configurations and specifications of the spent fuel are depicted in Fig. 1 and Table 1, respectively. CINDER[1] computer code is used for obtaining radioactivities and decay energies in terms of irradiation time, and the input data for the program running of CINDER code are obtained from LEOPARD[2] In computing optimum shield thicknesses, CASOS computer code has been independently programmed for interesting shield materials which are classified into two categories: Wall materials for a hot cell wall and window materials for viewing. Presented

Table 1. Fuel Specifications

Fuel Pellet		
1.	Material	UO_2 Sintered
2.	Density (% of Theoretical)	95
3.	Diameter, in.	0.3659
4.	Length, in.	0.600
5.	U-235 Enrichment (%)	3.2
Fuel Rod		
1.	Outside Diameter, in.	0.422
2.	Diametral Gap, in., Regions 1, 2 and 3	0.0075
3.	Clad Thickness, in.	0.0243
4.	Clad Material	Zircaloy-4
5.	Fuel Rod Length, in.	151.83
6.	Fuel Column Length, in.	144.0
7.	Spring Length, in.	6.45

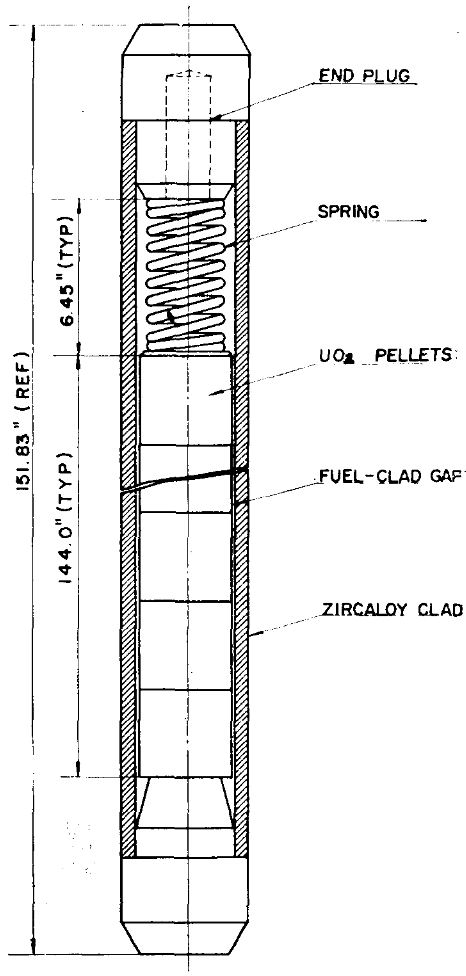

Fig. 1. Schematic Fuel Rod.

in Table 2 are the density and mass attenuation coefficients of selected shield materials, and gamma rays are only considered throughout this study because gamma radiation is a major contributor to shielding calcuations.

In carrying out computations of buildup factors in various shield materials, there are a number of formulae to analytically estimate buildup factors. However, except single elements such as aluminum, iron, and lead, they cannot be easily evaluated for composite meterials of heavy concrete and lead glasses. Buildup factors are, the-

Table 2. Shield Materials

Materials	Density (g/cm^3)	Mass Attenuation Coefficient (cm^2/g) 0.5 Mev	1.0 Mev
Aluminum	2.699	8.4000×10^{-2}	6.1400×10^{-2}
Iron	7.87	8.2800×10^{-2}	5.9500×10^{-2}
Concrete-04 (ordinary)	2.35	8.6809×10^{-2}	6.3362×10^{-2}
Concrete-BAa (heavy)	3.50	9.0514×10^{-2}	6.0120×10^{-2}
Lead	11.34	1.4500×10^{-1}	6.8400×10^{-2}
Glass-8365 (Crown Glass)	2.67	8.6517×10^{-2}	6.0674×10^{-2}
Glass-8362 (Lead Glass)	3.27	1.0489×10^{-1}	6.3609×10^{-2}
Glass-8363 (Lead Glass)	6.22	1.3360×10^{-1}	6.6667×10^{-2}
RS-360(Lead Glass)	3.60	1.1111×10^{-1}	6.7742×10^{-2}
RS-620(Lead Glass)	6.20	1.3065×10^{-1}	6.7742×10^{-2}

refore, computed from Taylor, Capo, and a linear form by using the appropriate coefficients of the functions. Since any published data of buildup factors are not available for the calculated values, comparison among the numerical quantities from the functions is made for the buildup factor of the shield materials as a function of gamma energy and a proper form is selected for the respective material.

II. Governing Equations

A. Activity and Decay Energy

The radioactive decay of one species in a sample is described by a well-known expression:

$$\frac{dN}{dt}=-\lambda N, \tag{1}$$

where $\frac{dN}{dt}$ is the decay rate of the nuclei and λ the characteristic decay constant of the nuclei. Sometimes the quantity λN is also known to be "activity" of the sample. Thus the decay energy can be defined by

$$E=f\lambda N, \tag{2}$$

in which f is the energy per decay.

When a nucleus undergoes fission process, a number of fission products are formed. In addition to the fission fragment, γ and β rays are appeared along with neutrons either at the instant of the fission or sometime later as the fission fragments undergo radioactive decay.

A nuclide can be transformed by several modes of radioactive decay or by neutron absorption. A complete schematic of all coupling mechanisms between the fed precursors and each transformed nuclide usually appears to be very complicated. If a nuclide concentration is independent of the concentration of its progeny, it is always possible to resolve the coupling so as to obtain nuclei fed by a single parent. Such chains will be referred to line chains, implying only that the coupling nuclides can be schematically represented in a straight line with no branch point.

In terms of a few energy group structure, the concentration of the i-th nuclide in a chain is determined by the following sequence of coupled equations:

$$\frac{dN_i}{dt}=\sum_k Y_{ik}S_k-(\lambda_i+A_i)N_i+\gamma_{i-1}N_{i-1}, \tag{3}$$

where Y_{ik}=yield fraction from fissile nuclide k,

S_k=fission rate of the k-th fissile nuclide,

λ_i=radioactive decay constant.

The absorption term in the i-th nuclide is given as

$$A_i=\sum_j g_{ij}\delta_{ij}\phi_j, \tag{4}$$

in which ϕ_{ij} is the total (n, γ) absorption cross section of the i-th nuclide in the j-th energy group, g_{ij} a non-1/v or shielding factor, and ϕ_j the neutron flux. The notation γ_{i-1} is either λ_{i-1} or A_{i-1} depending on coupling from precursor i-1; here A_{i-1} is not necessarily the total absorption rate, but it is only the protion of the (n, γ) reaction rate leading to N_i. This set of equations can equally well describe the nuclide transmutations in fission process. For instance, the representative of the depletion chains is the case of $Y_i=0$ for all i and of no contribution from the fission product chains.

Assuming a constant flux and an average fission rate during a time interval Δt, the general solution of Eq. (3) for the n-th nuclide is obtained[3]:

$$N_n(t+\Delta t)=\sum_{m=1}^{n}\frac{1}{\gamma n}\prod_{k=m}^{n}\gamma_k\Bigg\{\overline{Y_mS}\Bigg(\frac{1}{\prod_{l=m}^{n}(\lambda_l+A_l)}-\sum_{j=m}^{n}\frac{\exp[-\lambda_j+A_j)\Delta t]}{(\lambda_j+A_j)\prod_{i=m\neq j}^{n}(\lambda_i+A_i-\lambda_j-A_j)}\Bigg)$$
$$+N_m(t)\left(\sum_{j=m}^{n}\frac{\exp[-(\lambda_j+A_j)\Delta t]}{\prod_{i=m\neq j}^{n}(\lambda_i+A_i-\lambda_j-A_j)}\right), \tag{5}$$

where $\overline{Y_mS}=\sum_k\overline{Y_{mk}S_k}$ is the time averaged direct yield rate to nuclide m during a time interval Δt summed over all fissile nuclides. This solution is valid provided that $\lambda_i+A_i\neq\lambda_j+A_j$ for $i\neq j$ and that $\lambda_i+A_i=O$ in which i and j denote members of the chain and A_i involves the pure absorptions.

B. Slab Thickness for Gamma Ray Shielding

1. Gamma Ray Intensity and Dose Rate

Gamma ray intensity denoted by I can be obtained from the following equation[4]:

$$I=\frac{\dot{X}}{(0.0659)E(\mu_a/\rho)\text{air}}, \tag{6}$$

where E is the gamma ray energy in MeV, $\dot{X}$ the exposure rate in mR/hr and $(\mu_a/\rho)^{air}$ the mass absorption coefficient of air in units of cm^2/g.

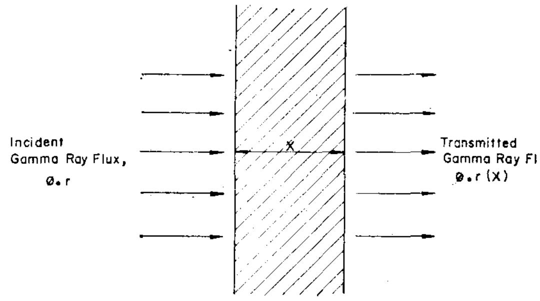

Fig. 2. Geometry For Gamma Ray Transmission.

The absorbed dose rate in a tissue which is subject to the exposure rate $\dot{X}$ is given by

$$\dot{D}=0.874\frac{(\mu_a/\rho)^{\text{tis}}}{(\mu_a/\rho)^{\text{air}}}\dot{X} \tag{7}$$

The dose rate $\dot{D}$ is gamma radiation absorption rate in units of mrad/hr and $(\mu_a/\rho)^{\text{tis}}$ mass absorption coefficient of concerning tissue.

2. Buildup Factor

When a monoenergetic and monodirectional gamma ray transmits through a slab as shown in Fig. 2, the uncollided flux at point x, $\phi_{\gamma,\text{ dir}}(x)$, is expressed in terms of the incident flux $\phi_{o\gamma}$ by

$$\phi_{\gamma,\text{ dir}}(x)=\phi_{o\gamma}e^{-\mu x}, \tag{8}$$

where μ is the linear gamma ray attenuation coefficient of shield material.

The total flux at point x is the sum of the direct and scattered fluxes:

$$\phi_{\gamma,\text{ tot}}(x)=\phi_{\gamma,\text{ dir}}(x)+\phi_{\gamma,\text{ scatt}}(x), \tag{9}$$

where the subscripts have the meanings of total, direct and scattering.

A buildup factor, which is the flux incremental factor due to gamma ray scattering with nucleus of shield material, should be taken account of gamma ray shielding as well. Thus the buildup factor $B(\mu x)$ is defined as the ratio of the total flux to the direct flux:

$$B(\mu x)=\frac{\phi_{\gamma,\text{ tot}}(x)}{\phi_{\gamma,\text{ dir}}(x)}=1+\frac{\phi_{\gamma,\text{ scatt}}(x)}{\phi_{\gamma,\text{ dir}}(x)} \tag{10}$$

The buildup factors can be computed with a certain accuracy by dividing $\phi_{r,\ tot}(x)$ obtained from solution of Boltzmann transport equation by $\phi_{r,\ dir}(x)$ calculated from Eq. (8).

The results for a given energy are then plotted and fitted to a simple exponential or a polynomial expression in order for $B(\mu x)$ to be described in a useful mathematical form for numerical calculations.

The buildup flux ϕ_b now can be written together with Eq. (8) as

$$\phi_b(x) = \phi_{or} B(\mu x) \exp(-\mu x) \quad (11)$$

For calcuational purposes the value of $B(\mu x)$ must either be read from a tabulation of buildup factors or be used with one of the following analytic fits[5].

a) Linear formula:

$$B(\mu x) = 1 + k\mu x, \quad (12)$$

in which the value of k is readily derived from the original data by means of the formula

$$k = B(1) - 1 \quad (13)$$

b) Taylor's formula:

$$B(\mu x) = A_1 \exp(-\alpha_1 \mu x)$$
$$(1-A_1)\ \exp(-\alpha_2 \mu x), \quad (14)$$

where A_1, α_1 and α_2 are the functions of energy and the numerical values for the parameters can be found elsewhere[4].

c) Capo's formula:

$$B(\mu x) = \sum_{i=0}^{3} \sum_{j=0}^{4} C_{ij} (1/E)^j (\mu x)^i, \quad (15)$$

in which E is gamma ray energy and C_{ij} parameters depending on the gamma ray energy.

3. Buildup Fluxes

In calculating thebuildup flux at a certain point on the surface of shield material, the treatment of radioactive sources should be properly made for predicting a reasonable thickness of shielding. Since the prime objective of this study is to guide the adequate wall thickness of a hot cell for examining spent fuels as well as the selections of shield materials the radioactive source distribution can be assumed to be from a line source without loss of generality.

As shown in Fig. 3-(a), the gamma rays emitted from the element dz appear at the point P_1 as if they were emitted from a point source. If $\theta_1 = \theta_2$ in which the Maximum Dose Equivalent could be detected, the buildup factor is considered and the buildup flux at the point P_2 is obtained by[4]

$$\phi_b = B(\mu a) \frac{S_l}{2R} F(\theta_1,\ \mu a), \quad (16)$$

where S_l is the source intensity of gamma

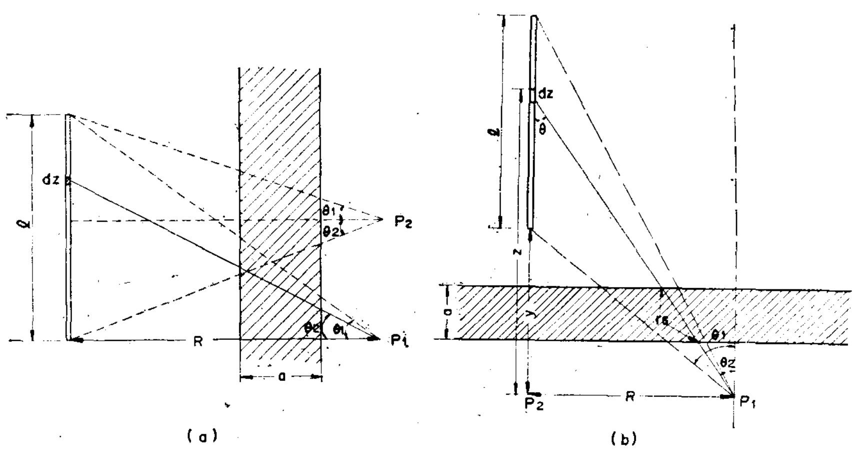

Fig. 3. Geometry For Line Source With Slab Shield In Various Arrangement.

ray in gammas/cm-sec and $F(\theta, \mu a)$ Sievert Function[6].

When the line source is in a vertical position to shielding wall shown in Fig. 3-(b), the buildup flux is given by

$$\phi_b = \frac{S_l}{4\pi R} B(\mu a)\,[F(\theta_1, \mu a) - F(\theta_2, \mu a)],$$

and approaching P_1 to P_2 yields

$$\phi_b \xrightarrow[R\to 0]{} B(\mu a)\,\frac{S_l l e^{-\mu a}}{4\pi y(l+y)} \qquad (17)$$

As indicated in Eq. (17), the buildup flux at P_2 can be described by assuming that there is a point source, and hence the buildup flux clearly depends on the distance between the point source and the observation point.

III. Numerical Computations

In computing shielding thickness of slab materials, radioactivities and decay energies from a spent fuel should be evaluated after a certain time of irradiation in a power reactor. Used for numerical computations is CINDER computer code which calculates the time-dependent concentrations of nuclides coupled in an arbitrary sequence of radioactive decays and neurton absorptions in a specified neutron flux spectrum. The radioactivities and decay energies are determined during a sequence of time steps.

In order to prepare the input data for the computer code CINDER, the LEOPARD code has been used to obtain the reasonable time interval in hours for indicated fuel burnup and decay. The thermal flux or power density in a fuel rod is computed by use of the code LEOPARD along with the thermal spectrum factor which multiplies thermal group cross sections. The input data are also prepared by the code WAPD-TM-333[3] which yields fission chains, total microscopic absorption cross sections of fission products, and decay constants of fission products accordingly. Decay energies of γ and β for fission products are fed by ENDF/B-IV[7]. The total elapsed time for the burnup rate of 33,000 MWD/T(U) is equivalent to 22,143 hour irradation with the thermal neutron flux level of 4.2×10^{13} neutrons/cm^2-sec and is divided by two large time intervals, i.e. burnup and decay time. Burnup time interval consists of 13 time steps, and decay time is separated by 5 steps for numerical analysis. Also used are two depletion chains of 8 nuclides and 19 fission chairs of 98 nuclides. The flux groups are divided into four energy ranges which are presented in Table 3.

The fission chains are only selected when the yield fraction of fission fragment is more than 1%.

A computer program CASOS has been made for slab thickness calculations along with taking source being treated as line source. The program is composed of one main program and four subroutines.

In the calculations the average gamma decay energy from source is taken to be 0.5 MeV and the calculations are also carried out in the case of 1 MeV to compare with the preceding results. Following conditions are considered to obtain the optimum and safe thickness:

The maximum permissible dose equivalent

Table 3. Energy Groups

Group	Energy Range, eV
1	$8.21\times10^6 \sim 10^7$
2	$5.53\times10^3 \sim 8.21\times10^5$
3	$0.625 \sim 5.53\times10^3$
4	$0 \sim 0.625$

Table 4. Cell Specifications from LEOPARD

Pitch, cm	1.4193
Pellet Out Radious, cm	0.4691
Clad Out Radious, cm	0.5370
Moderator Out Radius, cm	0.8008
Volume Fraction:	
Pellet	0.34315
Clad	0.10653
Moderator	0.55032

is taken to be 0.3 mrem/hr which is one-eighth of 2.5 mrem/hr that is required for a radiation worker with regard to the recommandations of the NCRP. This is due to assumption that a radiation worker will work with radioactive materials, mainly segments of irradiated fuel rod, for eight hours a day.

The observation point is located in the maximum dose equivalent point on the outside of the slab. As the wall material, aluminum, iron, ordinary concrete (concrete-04), heavy concrete (concrete-BAa), and lead are taken, while glass (8365), (8362), (8363), RS(360), (620) are selected to be window material.

The output of CINDER is the decay energy per unit volume and unit time, which will be converted into per unit length and time by using cell conditions from output of LEOPARD. The cell specifications are presented in Table 4.

IV. Results and Discussion

In computing shield wall thickness of a hot cell and window thickness for viewing, radioactivities and gamma decay energies from the spent fuel of the Go-Ri Unit 1 are evaluated with respect to various decay times in month and presented in Table 5.

Table 5. Activity and Gamma Decay Energy

Decay Time (Month)	Activity(Ci)	Gamma Decay Energy (1 Mev/cm-sec)
3	10165.9	3.23×10^{11}
6	5631.4	1.35×10^{11}
9	3802.0	6.06×10^{10}
12	2921.1	3.12×10^{10}

It should be pointed out from the table that the activity with one year cooling of the spent fuel is reduced by one-third of the activity with three month cooling.

Also considered are the buildup factors in terms of shield materials such as aluminum and ordinary concrete (concrete-04) in the thickness range of 100 cm. The analytical fits of the buildup factors are carried out by the Taylor, linear, and Capo formula in order to find the proper equation for the respective shield material. Fig. 4 illustrates the numerical results of the buildup factors from the three formulae for various materials. In the case of aluminum, the deviations of the formulae are appeared in the thicker range and the error is estimated to be$\pm$0.4 from the results of the Capo form with the aluminum thickness of 100 cm. The very similar behavior can be found for the buildup factors of iron but the descrepancies are rather small in the thicker range. The case of ordinary concrete is, however, very different from each formula so that use of any particular form is not able to make a proper estimate of buildup factors. It seems to be due to the different densities and chemical compositions of the material defined as ordinary concrete. In spite of large deviations among the analytical formulae for ordinary concrete, it is suggested[5] the Capo form for the buildup factors of heavy concrete. It should thus be noted that although some

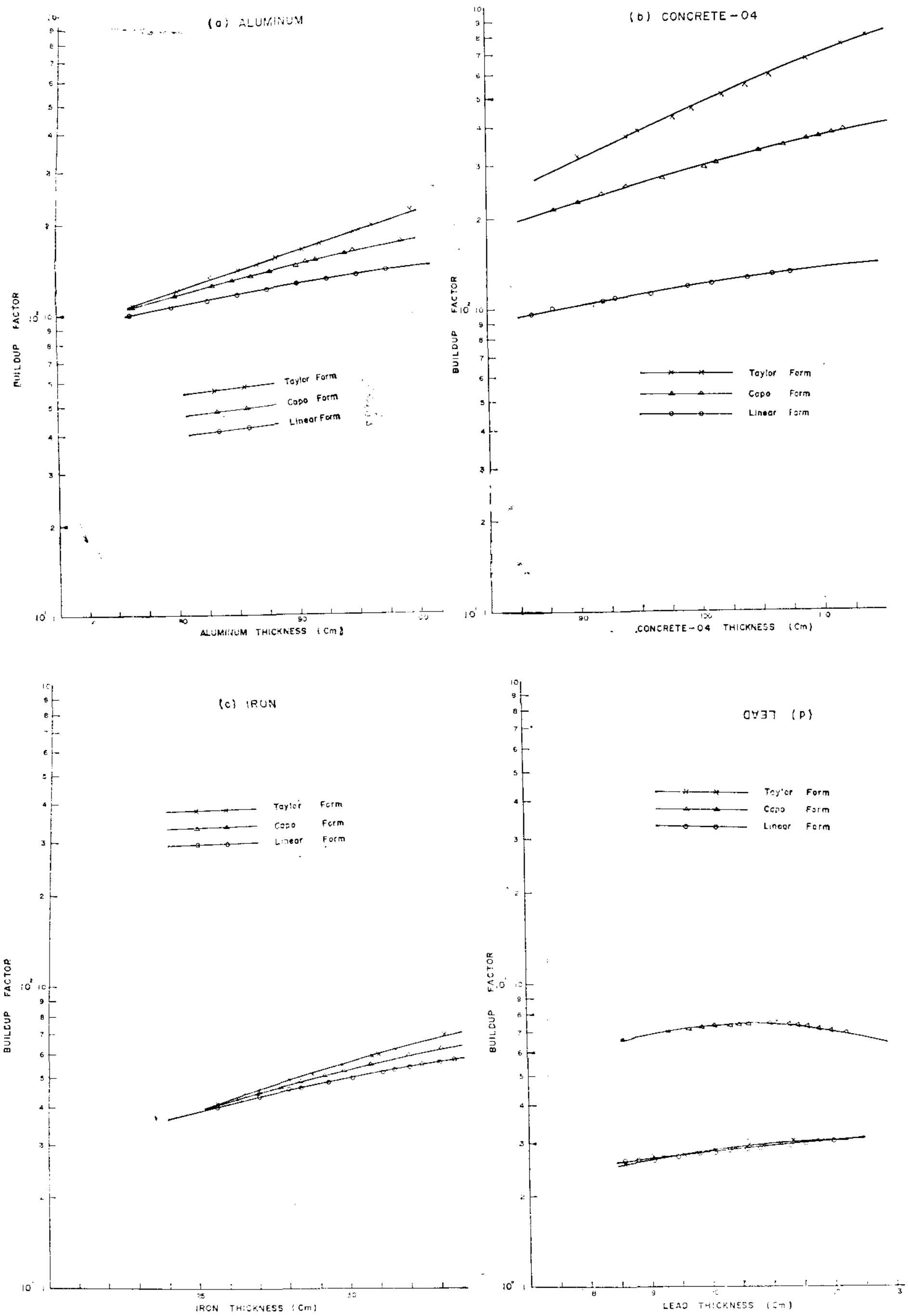

Fig. 4. Buildup Factors With Changes of Slab Shield Thickness In 0.5 MeV.

deviations of the buildup factors are detected in the formulae, the Capo form is found to be adequate because the results from the form seem to yield the average values among them. As shown in Fig. 4, the buildup factors of lead can be well predicted by the Taylor and linear form. Therefore, the Capo formula has been used

Table 6. Slab Thickness

TYPE	DECAY TIME (MONTH)	GAMMA RAY ENERGY (MeV) / Fuel Rod THICKNESS (cm) / SHIELD MATERIAL	0.5				1.0			
			1		2		1		2	
			SIDE (R)	END (D)	SIED (R)	END (D)	SIDE (R)	END (D)	SIDE (R)	END (D)
WALL	3	ALUMINUM	82.7	84.3	85.8	87.5	105.7	107.3	109.9	111.5
		IRON	29.0	29.8	30.1	30.9	39.0	39.9	40.5	41.5
		CONCRETE(04)	94.7	96.5	98.3	100.1	118.0	119.5	122.7	124.3
		CONCRETE(BAa)	57.2	58.4	59.4	60.7	83.1	84.5	86.3	87.8
		LEAD	10.4	10.7	10.8	11.1	21.8	22.4	22.7	23.3
	6	ALUMINUM	78.7	80.2	81.9	83.4	100.5	101.9	104.6	106.2
		IRON	27.7	28.4	28.8	29.5	37.2	38.0	38.6	39.5
		CONCRETE(04)	90.2	91.9	93.8	95.5	112.0	113.5	116.7	118.3
		CONCRETE(BAa)	54.5	55.6	56.7	57.9	78.9	80.[illegible]	82.2	83.6
		LEAD	9.9	10.2	10.3	10.6	20.8	21.[illegible]	21.6	22.1
	9	ALUMINUM	75.1	76.6	78.3	79.8	95.7	97.1	99.8	101.3
		IRON	26.5	27.1	27.5	28.2	35.5	36.3	36.9	37.8
		CONCRETE(04)	86.1	87.7	89.7	91.3	106.6	108.0	111.3	112.8
		CONCRETE(BAa)	52.0	53.1	54.2	55.3	75.2	76.5	78.5	79.8
		LEAD	9.5	9.7	9.9	10.1	19.8	20.3	20.6	21.1
	12	ALUMINUN	72.1	73.5	75.3	76.7	91.7	93.0	95.8	97.2
		IRON	25.4	26.1	26.5	27.2	34.0	34.9	35.5	36.4
		CONCRETE(04)	82.7	84.2	86.3	87.9	102.2	103.5	106.3	108.2
		CONCRETE(BAa)	49.9	50.9	52.1	53.2	72.1	73.3	75.3	76.6
		LEAD	9.1	9.3	9.5	9.7	19.0	19.4	19.8	20.3
WINDOW	3	GLASS(8365)	82.1	83.7	85.1	86.7	107.9	109.4	112.1	113.7
		GLASS(8362)	55.8	57.0	57.8	59.0	83.9	85.4	87.2	88.7
		GLASS(8363)	23.3	23.9	24.1	24.7	41.6	42.5	43.2	44.2
		RS(360)	47.9	49.0	49.7	50.8	72.8	74.2	75.6	77.1
		RS(620)	23.7	24.3	24.5	25.2	41.5	42.4	43.1	44.1
	6	GLASS(8365)	78.4	79.8	81.4	82.9	102.6	104.0	106.8	108.3
		GLASS(8362)	53.2	54.4	55.2	56.5	79.7	81.1	83.0	84.5
		GLASS(8363)	22.2	22.8	23.0	23.7	39.5	40.4	41.1	42.1
		RS(360)	45.7	46.8	47.5	48.6	69.2	70.5	72.0	73.4
		RS(620)	22.6	23.2	23.5	24.1	39.4	40.3	41.0	42.0
	9	GLASS(8365)	74.9	76.3	77.9	79.4	97.7	99.1	101.9	103.4
		GLASS(8362)	50.9	52.0	52.9	54.1	76.0	77.3	79.3	80.6
		GLASS(8363)	21.2	21.8	22.1	22.7	37.7	38.5	39.3	40.2
		RS(360)	43.8	44.8	45.5	46.5	65.9	67.1	68.8	70.1
		RS(620)	21.6	22.2	22.5	23.1	37.6	38.4	39.2	40.1
	12	GLASS(8365)	72.0	73.4	75.0	76.4	93.6	94.9	97.9	99.2
		GLASS(8362)	49.0	50.0	51.0	52.1	72.8	74.0	76.1	77.4
		GLASS(8363)	20.4	21.0	21.3	21.8	36.1	36.9	37.7	38.6
		RS(360)	42.1	43.1	43.8	44.8	63.2	64.4	66.0	67.3
		RS(620)	20.8	21.3	21.7	22.2	36.0	36.8	37.6	38.5

throughout the numerical computations of the buildup factors except for the case of lead.

Table 6 presents the results of wall and window thicknesses for two different gamma energies with respect to decay time and number of spent fuel rods stored. Indication of the table is that change of number of spent fuel rods do not make much difference in shielding thickness but the gamma ray energy does. That is, addition of spent fuels requires only 4% of thickness change

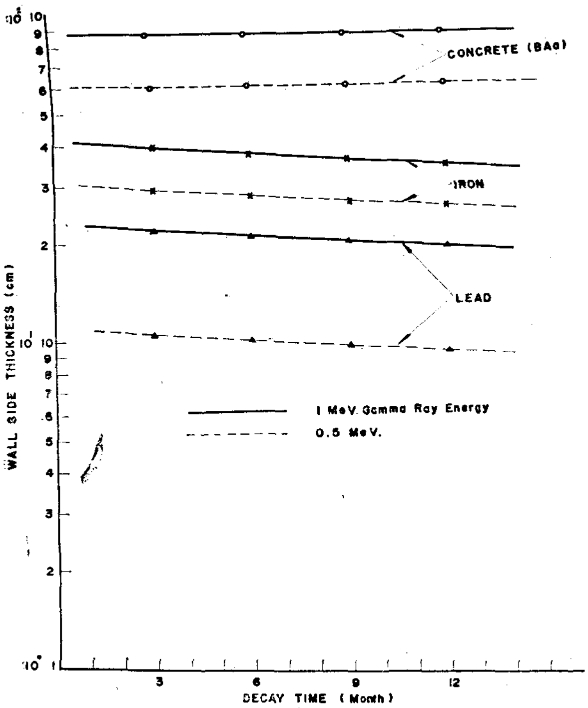

Fig. 5. Wall Side Thickness VS Decay Time.

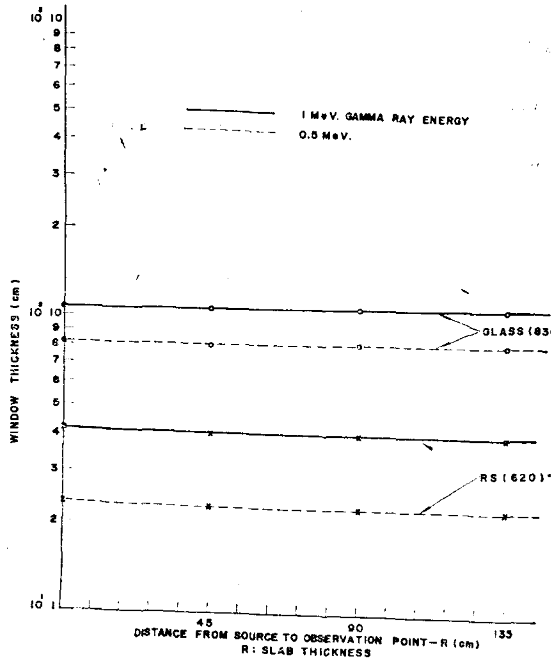

Fig. 6. Wall Side Thickness VS Distance From Source Te Observation Point-R At 3 Months Decay Time.

in aluminum, and the case of 1 MeV gamma energy however needs about 30% of additional aluminum shielding.

Also calculated are the thicknesses of the slab sides which do not differ from those of the end shields. Thus the table can provide the design parameters of a casket for transportation of spent fuels. In order to see the effect of cooling time on shielding, plotted in Fig. 5 is the wall side thickness of a hot cell versus decay time in month.

One can easily see from the figure that the shield thickness vary slowly with respect to decay time and the thickness reduce about 15% for one year cooling. This figure also shows the comparative thicknesses among the shield materials, and as would be expected, lead is found to be the practical material to reduce the dose rate from gamma rays. It is worth to note that the difference between two gamma ray energies, 0.5 MeV and 1 MeV, is largely estimated for the case of lead. It is ascribed to the photo-effects which is more predominant in lead than in other materials for high gamma energies ($\geq$1MeV). The window thicknesses for viewing are estimated for Crown Glass and Lead Glass (RS-620). As previously indicated, the physical densities of materials are the major factor in order to select the shielding material. While spent fuel rods are examined in hot cell, one would expect to be exposed by hot parts of the rods through the window. In this case, the effect of changing the distance from a source to an observation point is investigated, and the results are shown in Fig. 6 as a function of varying distances.

For the numerical calculations, the plain window thickness remains to be constant, and the change of observation point has almost no effect once a proper shield has been made.

Although the computed results are not able to be compared with other numerical values from experiments or analytical methods, the estimates would provide the order of magnitude in shielding thicknesses of some interesting materials and selections of shield materials in order to design a hot cell.

References

(1) T.R. England, WAPO-TM-384, CINDER-A One Point Depletion and Fission Product Program, August 1962

(2) R.F. Barry, LEOPARD-A Spectrum Dependent Non-Spatial Depletion Code, September 1963

(3) T.R. England, WAPO-TM-333, Time-Dependent Fission-Product Thermal and Resonance Absorption Cross Sections, November 1965

(4) J.R. Lamarsh, Introduction to Nuclear Engineering, Addison-Wesley Publishing Company, Reading Massachusetts, pp 421-445, 1975

(5) R.G. Jaeger et al., Engineering Compendium on Radiation Shielding, Vol., I, Springer-Verlag, New York, 1968

(6) Milton Abramowitz and Irene A. Stegun(Ed.), Handbook of Mathematical Functions, Dover Publications, Inc., pp 1000-1002, November, 1970

(7) T.R. England, ENDF-/B-IV, Fission-Product Files: Summary of Major Nuclide Data, pp 485-498, 1971