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An Analysis of Shielding Design of TRIGA Mark-Ⅱ Reactor

—After Power Upgrading by 2.5 Folds—

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Abstract

Korea's TRIGA Mark-II reactor was primarily designed in 1950's and was constructed in 1962 for 100 kw thermal output, but it was upgraded to 250 kw in July 1969. Nevertheless, the shield remains unchanged, although the radiation level has increased. The result of computation in this paper shows that, with the existing shield, it is safe for the fast neutrons even after the power upgrading by 2.5 times. It is, however, somewhat dangerous for the gamma rays which are comprised of primary and secondary. For the analysis of the reactor shielding design, an attempt is made for the computation toward the horizontal direction. From theoretical point of view, it can be concluded that some layer of additional shield must be reinforced to the existing concrete in order to be radiologically safe in the reactor hall.

요 약

1950년대의 미국 General Atomic 사에서 열출력 100 kw 로 설계, 제작하여 1962 년 3 월에 건조완료한 TRIGA Mark-Ⅱ 원자로는 1969년 7월에 250 kw 로 출력 증강되었으나 방사선차폐는 보강되지 않았다. 본 논문에서의 계산에 의하면 출력 증강후 현재의 차폐물로도 중성자에 대하여는 확실히 안전하지만 Gamma 선에 대해서는 위험하다는 것이 판명되었다.

원자로의 구조와 출입인 및 실험종사자들의 위치로 보아 차폐물의 안전도 검토는 수평방향에 한하였고, 또 정확을 기하기 위하여 중성자와 Gamma 선의 투과문제를 나누어 검토하였다. 이를 근거로 하여 이론적인 측면에서 본 콘크리트의 보강을 요하는 두께도 산출하였다.

I. Introduction

One cannot operate a nuclear reactor without providing sufficient shields against radiation, since its shielding is not only vital for the protection of reactor operating crew and associated personnel but also unique and different from the conventional facilities. The inherently hazardous radiations emitted from a nuclear reactor are usually alpha particles, beta particles, gamma rays, thermal and fast neutrons.

Of these radiations, alpha and beta particles are easily stopped by a solid substance having thickness less than one centimeter, whereas thermal neutrons are rapidly absorbed by a layer of matter, owing to the fact that the cross-section of most of the shielding materials vary as 1/v for the nuclear capture of the thermal neutrons. Hence, only two kinds of radiation, namely, gamma rays and fast neutrons, fall under the category of our concern in the treatment of nuclear shielding problem of our reactor.

The production of fast neutrons and gamma rays in a nuclear reactor is proportional to the rate of fission and hence to the power under the condition that the reactor is operated at a constant power level.

As of the 23rd of July 1969, the power of TRIGA Mark-II reactor was upgraded by 2.5 folds, which has naturally resulted in increasing the emission of fast neutrons and gamma rays of about 2.5 times than ever.

On this occasion, the shielding of TRIGA Mark-II reactor has been analysed and reviewed, and then computation has taken place for the safety analysis of reactor shield, and finally we have been driven to the impression that the shield of our reactor is not sufficiently thick enough to protect personnel from the emitting radiation. Since the reactor designer's data for the concrete was not available and the density of the concrete used in the shielding was not measurable, the author was obliged to refer to the data in the Nuclear Engineering Handbook[1).

Our analysis and calculation lead us to the suggestion that the shield of this reactor is unable to meet the operating power of 250 kw, as far as the gamma rays are concerned. Therefore, should the shielding be the absolutely requisite factor in design and operation of reactor, the TRIGA Mark-II reactor must be reinforced with 22.9cm of additional concrete layer at the horizontal direction, owing to the fact that the calculated gamma ray intensity outside the concrete shield 3.95 times more than the maximum permissible dose. It is, however, our understanding that reactors are usually designed and constructed from the viewpoint of its usability, taking into account of the economic factor as the prime importance.

II. Theoretical Background

For the computation of the shielding property of TRIGA Mark-II reactor, we have referred to the internationally accepted data and theory with regard to the nature of fast neutrons and gamma rays together with the attenuation of these two kinds of radiation.

II-1). Fast Neutrons

In order to clarify the effect of fast neutrons in the reflector, two group diffusion equation is employed herein[2). First of all, let us consider the diffusion equations for the fast and thermal neutrons in the core, respectively.

$$\text{Fast Group: } -D_1\nabla^2\phi_1(\mathbf{r})+\Sigma_R^{(1)}\phi_1(\mathbf{r}) = \nu\Sigma_f\phi_2(\mathbf{r}) \qquad (1)$$

$$\text{Thermal Group: } -D_2\nabla^2\phi_2(\mathbf{r})+\Sigma_a^{(2)}\phi_2(\mathbf{r}) = p_c\Sigma_R^{(1)}\phi_1(\mathbf{r}) \qquad (2)$$

The quantity $\Sigma_R^{(1)}$ in Eq. (1) is to be chosen so as to let $\Sigma_R^{(1)}\phi_1(\mathbf{r})$ be a good estimate of the slowing down density of fast neutrons out of the fast group at **r**.

It is assumed here that the source of fast neutrons. $\nu\Sigma_f\phi_2(\mathbf{r})$ is resulted only from the nuclear fission reaction due to the absorption of the thermal neutrons, while D_2 and $\Sigma_a^{(2)}$ are the thermal diffusion coefficient in the core and thermal absorption cross-section, respectively.

The source term for the thermal neutron equation (2) is taken as the number of thermal neutrons which have been slowed down of the fast group, reduced by the numbers lost in actual absorption during the slowing down process, where p_c is given by the resonance-escape probability in the core.

For the reflector we assume the following set of equations:

$$\text{Fast Group: } -D_3\nabla^2\phi_3(\mathbf{r})+\Sigma_R^{(3)}\phi_3(\mathbf{r})=0 \tag{3}$$

$$\text{Thermal Group: } -D_4\nabla^2\phi_4(\mathbf{r}) +\Sigma_a^{(4)}\phi_4(\mathbf{r})=p_R\Sigma_R^{(3)}\phi_3(\mathbf{r}) \tag{4}$$

where $\Sigma_R^{(3)}$ corresponds to $\Sigma_R^{(1)}$ for the core, which is called the fast removal cross-section.

Since the fluxes, ϕ_1 and ϕ_2, at the core-reflector boundary are available from Eqs. (1), (2), (3) and (4), an attempt is made here only for the calculation of the fast neutron flux, ϕ_3, in order to get rid of duplication and complication. In accordance with the design, the reflector is made of graphite, so

$$\Sigma_R^{(3)}=D_G\kappa_G^2 \tag{5}$$

Therefore, Eq. (3) becomes

$$\frac{1}{\mathrm{r}}\frac{\partial}{\partial \mathrm{r}}[\mathrm{r}\frac{\partial}{\partial \mathrm{r}}\phi_3(\mathbf{r})]-\kappa_G^2\phi_3(\mathbf{r})=0 \tag{6}$$

If we define a new variable $x\equiv \mathbf{r}\kappa_G$, the equation takes the standard form of the modified Bessel equation,

$$\phi_3''(x)+\frac{1}{x}\phi_3'(x)-\phi_3(x)=0 \tag{7}$$

The general solution is

$$\phi_3(x)=AI_o(x)+BK_o(x) \tag{8}$$

Putting the boundary condition, $\phi_3(\tilde{R})=0$, at the extrapolated radius $\tilde{R}$, we get

$$\phi(\mathbf{r})=S[I_o(\kappa_G R)K_o(\kappa_G\tilde{R}) -I_o(\kappa_G\tilde{R})K_o(\kappa_G R)] \tag{9}$$

By definition, the attenuation coefficient is the fractional rate of loss in intensity of incident beam of radiation as it traverses the absorbing media, and the attenuation is generally described by Beer's law.

$$I=I_o e^{-\mu x} \tag{10}$$

where I: intensity of the radiation,

x: thickness of material,

μ: attenuation coefficient.

The attenuation coefficient, μ, is related to the cross-section σ per atom for the attenuation of the particles[3].

$$\mu=n_o\sigma=N_o\rho\Sigma\frac{\sigma}{A}=N_o\rho\frac{\Sigma a\sigma}{\Sigma aA} \tag{11}$$

where n_o: the number of atoms per unit volume of the material,

N_o: Avogadro's number, 6.023×10^{23}

ρ: density,

A: the atomic weight of the material,

a: the atomic fraction of each neutron-absorbing medium

The two main shielding materials used in TRIGA Mark-II reactor are water and concrete; and the latter is believed to have enough water crystallization to be qualified as a hydrogen-containing material.

For hydrogen[3], the cross-section is

$$\sigma_H=\frac{11}{E+1.65} \tag{12}$$

where σ_H: the cross-section of hydrogen in barns.

E: the neutron energy in millions of electron volts.

For the heavier materials, the so-called "geometrical" cross-section is used

$$\sigma=2\pi R^2=0.12A^{2/3} \tag{13}$$

Hence, the attenuation coefficient for the removal of fast neutrons in this approximation is obtained from Eq. (11) by inserting Eqs. (12) and (13) in it.

$$\mu_{H_2O}(E)=0.05+\frac{0.7}{E+1.65} \tag{14}$$

$$\mu_{\text{concrete}}(E)=0.05+\frac{0.1}{E+1.65} \tag{15}$$

$$\mu_{\text{others}}=N_o\rho\frac{\Sigma a(0.12A^{2/3})}{\Sigma aA} \tag{16}$$

From Eqs. (14) and (15), we get

$$\frac{I}{I_o}=\int_o^\infty \phi(E)e^{-\mu(E)x}dE \tag{17}$$

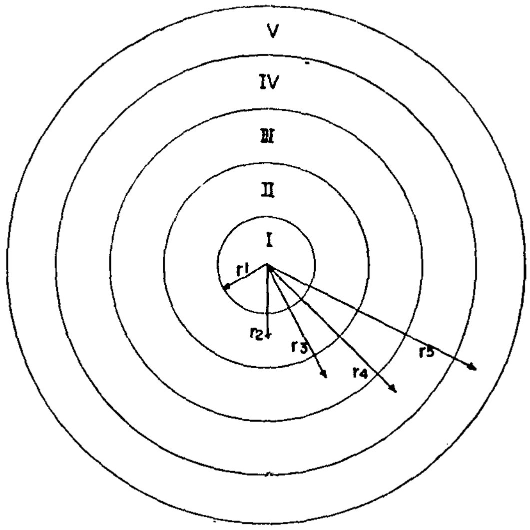

Fig. 1. Schematic Diagram of Core Plane

Annulus	I	II	III	IV	V
r(cm)	2.75	5.375	10.00	14.00	17.25
$\phi(10^{12}$n's/cm^2-sec)	8.657	5.434	3.389	3.623	3.850

This integral has been performed numerically[3], and the results can be approximated by the sum of the following two exponential terms:

For water: $$\frac{I}{I_o}=0.98e^{-0.19x}+0.02e^{-0.10x} \tag{18}$$

For concrete: $$\frac{I}{I_o}=0.96e^{-0.08x}+0.04e^{-0.06x} \tag{19}$$

II-2). Gamma rays

For the investigation of gamma ray energy attenuation in the core, the core is divided into a number of annuli so that annulus is assumed to be a gamma ray source.

The gamma ray energy, J_k, in the annulus k is

$$J_k=\sum_j\sum_i J_{ijk} \tag{20}$$

$$J_{ijk}=\chi_i V_j N\sigma_f \Delta r_k E_r \tag{21}$$

where x_i is the discrete gamma ray energy spectrum,

V_j is the volume fraction of composite material in the core,

Δr_k is the width of annulus k.

The image gamma ray source, S_{ijk}, from the central point 0 is

$$S_{ijk}=2\pi r_k J_{ijk} e^{\mu_{ij} r_k} \tag{22}$$

When the attenuation of this image source takes place up to the core-reflector boundary, the gamma ray energy at the boundary, $(J_c)_{ijk}$ is

$$(J_c)_{ijk}=\frac{S_{ijk}\,e^{-\mu_{ij}r_c}}{2\pi r_c}$$
$$=\frac{r_k}{r_c}J_{ijk}\,e^{-\mu_{ij}(r_c-r_k)} \tag{23}$$

Since the real source and the imaginary source are identical each other, the behavior of the image gamma ray source in the annulus 1 is

$$S_{ij1}=\pi r_1^2 x_i V_j N\sigma_f E_r \tag{24}$$

Then, total gamma ray energy of each gamma ray spectrum at the boundary of core-reflector is

$$(J_c)_{ij}=x_i V_j \pi r_1^2 N\sigma_f E_r + x_i V_j N\sigma_f E_r \sum_{k=2}\frac{r_k}{r_c}\Delta r_k e^{-\mu_{ij}(r_c-r_k)}$$
$$=N\sigma_f E_r x_i V_j \left[\pi r_1^2+\sum_{k=2}\frac{r_k}{r_c}\Delta r_k e^{-\mu_{ij}(r_c-r_k)}\right] \tag{25}$$

$$(J_c)_i=\sum_j (J_c)_{ij}$$

For the consideration of gamma ray energy attenuation of each spectrum in the reflector, similar calculation is conducted for the image source, S_i',

$$S_i'=2\pi r_c (J_c)_i e^{(\mu_G)_i r_c} \tag{26}$$

where $(\mu_G)_i$ is the attenuation coefficient of graphite of ith gamma ray spectrum.

Then, the energy of gamma ray, $(J_G)_i$, at the boundary of graphite-water is

$$(J_G)_i=\frac{S_i' e^{-(\mu_G)_i r_G}}{2\pi r_G}=\frac{r_c (J_c)_i e^{-(\mu_G)_i (r_G-r_c)}}{r_G} \tag{27}$$

at the water-aluminum boundary,

$$(J_W)_i = \frac{r_G(J_G)_i e^{-(\mu_W)_i(r_W - r_G)}}{r_W} \tag{28}$$

at the aluminum-gunite boundary,

$$(J_A)_i = \frac{r_W(J_W)_i e^{-(\mu_A)_i(r_A - r_W)}}{r_A} \tag{29}$$

at the gunite-steel boundary,

$$(J_B)_i = \frac{r_A(J_A)_i e^{-(\mu_B)_i(r_B - r_A)}}{r_B} \tag{30}$$

at the steell-concrete boundary,

$$(J_S)_i = \frac{r_B(J_B)_i e^{-(\mu_S)_i(r_S - r_B)}}{r_S} \tag{31}$$

at the edge of concrete shielding,

$$(J_E)_i = \frac{r_S(J_S)_i e^{-(\mu_E)_i(r_E - r_S)}}{r_E} \tag{32}$$

The summation of all the above components is

$$\begin{aligned}(J_E)_i = \frac{r_C(J_C)_i}{r_E} \exp[&-(\mu_G)_i(r_G - r_C)\\ &-(\mu_W)_i(r_W - r_G) - (\mu_A)_i(r_A - r_W)\\ &-(\mu_B)_i(r_B - r_A) - (\mu_S)_i(r_S - r_B)\\ &-(\mu_E)_i(r_E - r_S)]\end{aligned} \tag{33}$$

$$J_E = \sum_i (J_E)_i$$

The secondary Compton-scattered radiation is included in the attenuation formula, Eq. (10), by writing

$$\frac{J}{J_o} = (1+B)e^{-\mu x} \tag{34}$$

where the "build-up factor", B, is a function of the distance, x, 'through the medium, being zero at $x=0$ and increasing continuously with x. Extensive calculations of the build-up factor are available for a number of materials[5]. If the build-up factor is written in the form of

$$(1+B) = e^{\Delta\mu x} \tag{35}$$

Eq. (34) then becomes

$$\frac{J}{J_o} = e^{-(\mu - \Delta\mu)x} = e^{-\mu_B x} \tag{36}$$

Strictly speaking, $\Delta\mu$ and hence μ_B are functions of x, but over moderate range of x, it is sufficient to approximate μ_B by a constant. Approximate values of μ_B are plotted in Fig. (2) for lead, iron, water, and concrete (concrete is considered to be equivalent to aluminum for gamma ray shielding).

All the values contained in Fig. (2) were obtained from a combined effort of both theoretical and experimental approach. The curves in the figure show a marked dependence of μ_B on the energy of the incident gamma radiation.

Another method of finding the values of μ_B is denoted in Fig. (3).

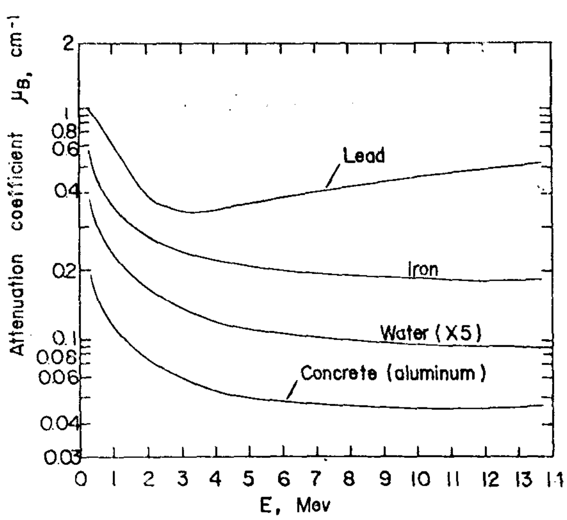

Fig. 2. Effective γ-ray attenuation coefficient

Fig. (3) shows the mass attenuation coefficient $\mu_B/\rho = \mu_B'$ against the atomic number Z for the energies of distribution. It must be noted that the correction for the build-up included in the curves of Fig. (3) is very much approximate, since they are obtained from the interpolation of μ_B with the assumption that $\Delta\mu$ in Eq. (35) is a constant. The build-up effect is more pronounced for the light elements, further accentuation the relative effectiveness of heavy elements for gamma ray attenuation.

Last of all, let us consider the secondary gamma rays. About half of the gamma radiation produced by a reactor is associated with the capture of neutrons in non-fissionable materials. A certain fraction of this neutron

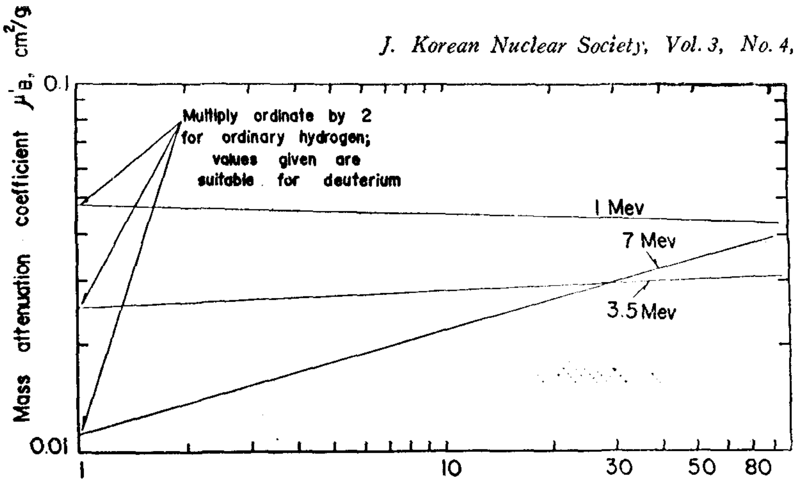

Fig. 3. Mass attenuation coefficients

capture must occur in the shield, thus resulting in a source of gamma radiation. Approximately, half of this radiation escapes outward through the shield, whereas the other half returns to the interior region and enters the shield on the opposite side of the reflector.

The outward-escaping half of the radiation does not have to penetrate the full thickness of the shield, so that additional shielding will be required over that calculated for a gamma source entirely interior to the shield. To caculate this secondary effect, Beer's law will be assumed to hold with the constant attenuation coefficients μ_n and μ_r for the fast neutrons and gamma rays.

The gamma ray flux that escapes a shield of thickness x_o from a production point at a depth x in the shield is

$$dJ=\frac{1}{2}EdIe^{-\mu_r(x_o-x)} \qquad (37)$$

where $\frac{1}{2}$: factor for outgoing radiation,

E: average gamma energy emitted per fast neutron captured.

Futhermore,

$-dI=\mu_n I_o e^{\mu_n x}dx$=rate of fast neutron absorption at depth x.

The total gamma flux emergent at the thickness x_0 is

$$J=\int_o^{x_o}dJ=\frac{E}{2}\mu_n e^{-\mu_r x_o}I_o\int_o^{x_o}e^{(\mu_r-\mu_n)x}dx$$

$$=\frac{1}{2}EI_o\frac{\mu_n}{\mu_r-\mu_n}(e^{-\mu_n x_o}-e^{-\mu_r x_o}) \qquad (38)$$

III. Computation

As is seen in Fig. 4, the entire layer of shield from the center line of the reactor core to the outer edge of the concrete shield amounts to the about 326.91 cm which is comprised of 18.5 cm of core, 35.176 cm of graphite reflector, 38.1 cm of water, 0.635 cm of aluminum plate, 4.445 cm of gunite, 0.635 cm of steel plate, and 228.6 cm of concrete[6].

This shielding calculation is performed being based on the experimental data of criticality test which was conducted right after the power upgrading of this reactor[7]. The reactor core is divided into a number of annuli so as to utilize the measured value of each neutron flux in each corresponding annuls(Fig. 1).

Hence it is assumed that the measured value of each neutron flux at respective position is taken as the realistic neutron flux during the normal operation of the reactor at full power and that the fast neutrons which are actually contributed to the attenuation through the shielding materials are only originated from the extreme outer layer of annulus.

On the other hand, in order to carry out the gamma ray computation, we assume gamma ray energy source as the energy resulted from the thermal fission of each annulus of the core.

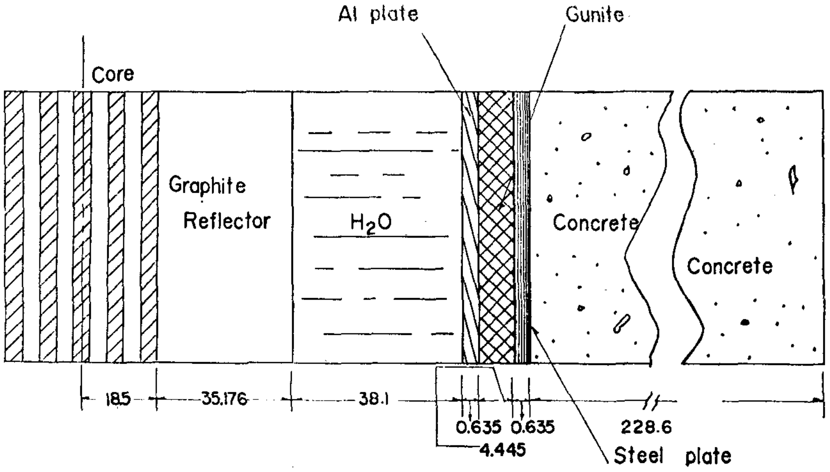

Fig. 4. Schematic Diagram of Shielding of TRIGA Mark-II Reactor at Horizontal Direction (Unit:cm)

III-1). The Attenuation of Fast Neutrons

Let us consider albedo, β, at the core-reflector boundary[4]. So far as the fast neutron flux shape is concerned, its slope at the boundary is presumed to be linear. With this assumption, the minimum possible albedo is chosen so that the maximum possible neutron flux may penetrate throughout the shielding materials.

$$\beta=\frac{J_-}{J_+}=\frac{1+\frac{2D}{\phi}\frac{d\phi}{dx}}{1-\frac{2D}{\phi}\frac{d\phi}{dx}}=0.639 \qquad (39)$$

Where J_+ and J_- are the outlet and inlet neutron current densities at the core-reflector boundary.

In this calculation, the following numerical values are employed (refer to Table 2),

$$\phi=4.214\times10^{12}\ \text{n's/cm}^2\cdot\text{sec}$$

$$\frac{d\phi}{dx}=-0.676\times10^{12}\ \text{n's/cm}^3\cdot\text{sec} \qquad (40)$$

$$D=1.016$$

Then, the fast neutron flux which can penetrate up to the reflector, $\phi_1'(r_c)$, from the core-reflector boundary from Eq. (3) is

$$\phi_1'(r_c)=\frac{1}{1+\beta}\phi_1(r_c)$$

$$=2.571\times10^{12}\ \text{n's/cm}^2\cdot\text{sec} \qquad (41)$$

Since extrapolated radius $\tilde{R}$ is 55.56 cm, S has the value of 6.107×10^{11} n's/cm²·sec.

From Eq. (9), we get

$$\phi_3(r_G)=2.07\times10^9\ \text{n's/cm}^2\cdot\text{sec} \qquad (42)$$

Table 1. Spatial Distribution of Thermal Flux in Core

($\times10^{12}$ n's/cm²-sec)

	a	b	c	d	e	C. T.	f	g	h	i	j
1	0.616	0.638	0.497	0.555	0.840	0.786	0.895	0.599	0.494	0.617	0.627
2	1.333	1.221	1.433	1.365	1.978	1.943	1.981	1.194	1.330	1.621	1.431
3	1.981	1.970	2.417	2.308	3.664	3.849	2.702	2.011	2.236	2.435	2.078
4	2.724	2.454	2.564	2.084	4.757	6.013	3.512	2.187	2.644	3.197	2.872
5	3.606	3.126	3.370	2.816	5.807	7.730	3.960	3.178	3.523	4.433	3.854
6	3.850	3.308	3.623	3.389	5.434	8.657	4.450	3.619	3.958	4.983	4.201
7	3.854	2.861	2.812	2.267	4.710	8.386	3.296	2.864	3.117	4.546	3.986
8	3.113	3.141	2.526	2.262	3.796	5.414	3.368	2.630	2.823	3.605	3.442
9	1.965	1.858	2.494	2.331	2.724	3.588	2.322	2.391	2.823	2.442	2.173

*a, b, ……, C. T. ……, j, indicate the positions of irradiation hole in the core.

1, 2, ……, 9 indicate the locations at the irradiation bar

Table 2. Spatial Distribution of Fast Flux in Core

($\times 10^{12}$ n's/cm^2—sec)

	a	b	c	d	e	C. T.	f	l	m	n	o
1	0.300	0.738	0.289	0.421	0.291	0.630	1.541	0.193	0.307	0.603	0.734
2	0.567	1.438	0.951	1.114	0.859	1.835	1.891	0.858	1.227	1.607	1.382
3	1.435	3.133	3.300	3.663	2.794	4.053	5.786	1.999	4.236	3.476	3.098
4	2.000	4.559	7.192	6.488	6.608	8.290	9.298	5.472	9.682	5.420	3.664
5	4.214	6.376	8.608	8.700	10.569	11.545	13.534	6.877	12.492	6.441	4.100
6	2.352	6.677	9.406	8.906	10.926	13.828	13.143	8.700	14.360	7.584	4.066
7	2.539	6.246	7.516	8.385	9.557	13.115	15.610	7.626	11.819	6.459	3.570
8	1.718	4.563	3.740	5.314	7.510	8.708	8.945	5.711	7.078	5.601	2.843
9	1.164	2.467	1.209	2.252	2.669	4.772	4.531	2.048	2.344	2.241	1.461

*a, b, ……, C. T., ……j, indicate the position of irradiation hole in the core.
1, 2, ……, 6 indicate the locations at the irradiation bar.

where r_G is the distance from origin to graphite-water boundary.

The actual plausible fast neutron flux which reaches the graphite-water boundary is 2.07×10^9 n's/cm$^2\cdot$sec.

The fast neutron intensity which is presumed to contribute outside the concrete shielding is calculated by means of Eqs. (10), (18) and (19), as follows.

$$I = 1.1428 \times 10^{-2} \text{ n' s/cm}^2\cdot\text{sec} \tag{43}$$

III-2). Gamma rays

There are three principal sources of gamma radiation[11] in a nuclear reactor, namely:

a. About 7.5 Mev of the so-called "prompt" gamma rays accompanies the fission reaction;

b. In the steady operation of a nuclear reactor, the gamma decay of fission fragments produces 6.8 Mev or so per fission.

c. In a critical reactor employing U-235, 0.3454 neutron out of 2.5 neutrons produced per fission are eventually captured in non-fissionable materials, releasing 7 Mev of gamma radiation per capture. Note that, when the radiative capture macroscopic cross-section of U-238 is given as 0.1382 cm^{-1}, then the gamma energy comes out to be 2.42 Mev/fission in this reaction.

Each gamma ray is said to have a specific energy; however, there are so many different gamma ray energies that together they comprise a practically continuous distribution, so that its spectrum is normally represented by a small number of discrete lines. For the simplicity, these lines are assumed to be at 1, 3.5 and 7 Mev[3,8,9].

The prompt gamma rays resulted from nuclear fission is of relatively low energy and will be represented by a single line at 1 Mev range. On the other hand, the radiation from fission products decay will be equally divided between the 1 Mev and 3.5 Mev lines.

Last but certainly not the least of all, the gamma rays emitted from the captured neutrons will be divided between 3.5 Mev(40%)

Table 3. The Fractional Energy Distribution of Gamma rays

Energy Spectrum	1 Mev	3.5 Mev	7 Mev
Prompt gamma (7.5Mev)[10]	7.5 Mev (100%)		
Fission Fragments (6.8Mev)[8]	3.4 Mev (50%)	3.4 Mev (50%)	
Gamma from Radiative Capture (7 Mev)[11]		1.45 Mev (60%)	0.966 Mev (40%)
Total Energy	10.9 Mev	4.85 Mev	0.966 Mev
fr	65.21%	29.01%	5.78%

Table 4. Gamma energy Attenuation Coefficients

Material	Density (g/cm^3)	1 Mev		3.5Mev		7 Mev	
		$\mu_B'(cm^2/g)$	$\mu_B(cm^{-1})$	μ_B'	μ_B	μ_B'	μ_B
ZrH 1.0	6.5	0.1260	0.9190	0.0635	0.4127	0.0409	0.2658
H_2O	1	0.0707	0.0707	0.0368	0.0368	0.0258	0.0258
Al	2.694	0.0614	0.1654	0.0332	0.0862	0.0258	0.0681
Graphite	1.60	0.0636	0.1018	0.0330	0.0528	0.0229	0.0366
Gunite			0.1026		0.0445		0.0315
Steel	7.87	0.0595	0.4682	0.0346	0.2723	0.0300	0.0255
Concrete	2.35[7]	0.0635	0.1492	0.0340	0.0799	0.2361	0.0599

(*; It is assumed that gunite is made of water and graphite in equal volume fraction.)

Table 5. Gamma ray energy of each group

$(\times 10^{12}$ Mev/cm^2-sec)

Annulus	J_{11}	J_{12}	J_{13}	J_{21}	J_{22}	J_{23}	J_{31}	J_{32}	J_{33}
2	17.475	10.432	1.015	7.774	4.641	0.452	1.549	0.925	0.090
3	8.304	4.957	0.482	3.694	2.205	0.215	0.736	0.439	0.043
4	8.877	5.299	0.516	3.949	2.358	0.229	0.787	0.470	0.046
5	5.896	3.520	0.343	2.623	1.566	0.152	0.523	0.312	0.030

Table 6. Attenuated gamma energy at core-reflector boundary

$(\times 10^{10}$ Mev/cm^2-sec)

Annulus	$(J_c)_{11}$	$(J_c)_{12}$	$(J_c)_{13}$	$(J_c)_{21}$	$(J_c)_{22}$	$(J_c)_{23}$	$(J_c)_{31}$	$(J_c)_{32}$	$(J_c)_{33}$
1	0.000	0.175	0.003	0.000	0.146	0.006	0.000	0.004	0.002
2	0.010	118.174	3.285	0.948	82.352	4.176	1.343	19.001	1.056
3	2.074	203.195	10.701	16.053	112.625	7.938	8.497	24.139	1.781
4	16.794	291.748	6.088	46.622	151.150	11.771	17.983	31.637	2.546
5	197.364	300.373	25.963	146.063	139.443	12.760	34.986	28.166	2.600
$(J_c)_i$	1,175.767			732.053			173.738		

and 7 Mev(60%). As a summary of gamma rays, the fractional energy distribution, fr, is listed in Table 3.

To get more accurate result, the value from the Nuclear Engineering Handbook is referred to as the attenuation coefficient μ_B' in this calculation[1]. The μ_B' value is shown in Table 4.

J_{ijk} is calculated from Eq. (21), while $(J_c)_{ijk}$ is from Eq. (23), and the calculated values are in Tables 5 and 6.

It is taken as granted that most of the secondary gamma rays originated in shielding materials can be given consideration in graphite where the fast neutron intensity is rather high.

The fast neutrons which are escaped out of the core rapidly decrease in number and energy. Consequently, the actual fast neutron intensity at the graphite-water boundary is extremely weak.

Furthermore, the secondary gamma rays in graphite is almost negligible compared with gamma enegry from fission, owing to the fact that (n,γ) macroscopic cross-section in this medium is $0.00032cm^{-1}$.

Therefore, it is rather logical to assume that gamma rays measurable outside the shielding materials are mostly originated from the core itself[12].

IV. Discussion

In the computation of the shield thickness, it is necessary to specify a certain standard intensity that may be tolerated in the working space outside the shield. The conventional tolerance limit at present, based on a 40-hour work per week, is 2.5 mrem/hr[13], which used to be 7.5 mrem/hr up until the last part of 1950's.

IV-1). Fast Neutrons

Fast neutrons in tissue lose energy predominantly through collisions with the nuclei of hydrogen atoms present in the tissue. The composition of tissue is known to be roughly that of water, in which there are $n_o=2N_o/18$ hydrogen atoms per cubic centimeter and then the cross-section of tissue will be replaced to that of hydrogen.

The rate H[3] of energy release per unit volume in the tissue by neutron intensity I is then

$$H=n_o I\int_o^\infty \frac{E}{e} f(E)\sigma_H dE=5.25\frac{n_o I}{e}\times 10^{24}\text{Mev/cm}^3 \qquad (44)$$

where $f(E)$ is theoretical reactor spectrum which depends on the neutron energy.

With this value, the tolerance dosage rate of 9 ergs/g becomes equivalent to premissible fast neutron intensity of

$I=30.3$ neutrons/cm$^2\cdot$sec

Compared with the result obtained from the previous computation, the fast neutron intensity through the shielding is quite lower than the accepted working level for horizontal directions. Therefore, it is quite satisfactory for the fast neutrons at any vertical position.

IV-2). Gamma rays

According to Eq. (36) the rate at which a gamma ray beam of energy E releases energy in tissue is given by

$$H=-\frac{dJ}{dx}=\mu(E)J \qquad (45)$$

For tissue the flatness of the curve of $\mu(E)$ in Fig. (3) over a wide range of E makes it suitable to assume that $\mu/\rho\approx 0.03\,\text{cm}^2/\text{g}$ roughly independent of E for E>0.1 Mev. If the tolerance limit for H is substituted in Eq. (45) the corresponding limit for J is

$$J=\frac{H\rho}{\mu}$$

From the above equation, H is then

$$H=\frac{\mu}{\rho}J$$

$$=\frac{(0.03\text{cm}^2/\text{g})\ (1.581\times10^2\text{Mev/g}\cdot\text{sec})\times(1.6\times10^{-6}\text{erg/Mev})\ (3600\text{sec/hr})}{(93\text{erg/g}\cdot\text{roentgen})}$$

$$=9.875\times10^{-3}\ \text{roentgen/hr}$$

$$=9.875\ \text{mrem/hr}$$

Since the relative biological effectiveness factor for gamma rays is 1, the above value is 9.875 mrem/hr.

Compared with the result obtained from this calculation, the gamma ray intensity through the shielding is higher than the accepted working level by 3.95 folds at the normal operation of the reactor.

In order to shield this extra gamma rays up to the accepted working level by concrete its computation is as below:

$$e^{-0.0599}x=\frac{1}{3.95}, \qquad x=22.9\,\text{cm}$$

In other words, 22.9 cm of additional concrete layer must be reinforced to the present shield in order to be radiologically safe from the viewpoint of the internationally tolerated standard.

IV-3). The Measurement of Gamma Radiation Level around the Reactor Shield

Since the reactor start-up in March 1962, the health physics group of this Institute has been regularly measuring the radiation level, both of neutrons and gamma rays, around the reactor shield at the intervals of one week or 10 days on the average, and the measured records are filed in with the Radiation Level Log Book[14], and some of them are shown in Table 7 for the reference. The instrument used for the measurement of gamma rays is GM survey meter by Messers. Nuclear Chicago (Model No. 2612), which is occasionally calibrated by 13.7 mci Ra-226 standard source. As is seen in the table, all the data were obtained after the date of reactor upgrading work completed in July 1969, and some of the radically high figures, such as those in the order of hundreds or thousands are interpreted as the radiation level mainly resulted from the leak-out radiation from beamports.

It may be helpful to know that the positions of A, D and G in Fig. 5 are located at 1-meter elevation from the ground level of the reactor hall, while those of B, E and H are at 1.5-meter elevation from the ground level, respectively. It is a marked fact that most of the measured radiation levels in these positions, where no gamma rays leakage could be detectable, are higher than 2.5 mrem/hr which is the maximum permissible dose rate.

Our actual survey at the point of T which is right on the top of water shield shows gamma radiation level between 30-50 mrem/hr, and such data are also recorded in Table 7 and elsewhere[7].

It can, therefore, be concluded that the thickness of the shield is insufficient to shield the gamma radiation, whereas it is considered to be satisfactory for the neutrons.

Table 7. Gamma Radiation Level around the Shielding Structure of TRIGA Mark-II Reactor at 250 kw

Position	Date of Measurement (unit:all in mrem/hr)						
	Jun. 24 1969	Aug. 20 1969	Mar. 24 1970	Jul. 16 1970	Nov. 26 1970	Jun. 24 1971	Jun. 30 1971
A	6.5	5.0	3.5	4.0	4.0	4.0	5.5
B	5.5	4.0	3.0	3.5	3.5	3.5	4.5
C	5.0	3.0	7.0	20.0<	20.0<	4.0	80
D	4.5	4.0	3.0	3.5	3.5	4.0	3.0
E	4.0	3.0	2.3	3.0	5.5	3.0	2.5
F	17	12.5	50.0	60	3.5	3.5	1.4
G	5.5	4.5	4.0	3.5	3.0	3.5	3.0
H	5.0	4.0	3.0	2.5	2.0	3.0	3.0
I	3.0	4.0	1.5	0.8	6.0	0.18	0.3
J	20	20	50	7.0	0.4	1.8	0.3
K	4.0	9.0	2.5	0.35	5.0	0.65	0.3
L	20<	3.0	34	5.0	4.0		0.3
M	20<	6.4	2.0	0.3	0.4	1.2	1.1
N	6.25	3.0	3.0	4.3	22	1500	800
O	1.3	0.5	0.7	0.2	20.0<	0.3	0.3
P	6.0	1.2	20.0<	20.0<	7.0	50	300
Q	450	1.5	8.0	100	1.0	350	400
R	7.5		8.0	9.0	10.0	13.5	15.0
S	10		6.5	7.5	8.0	20.0	18.0
T	20	40	20	20.0	20.0	14	19.5

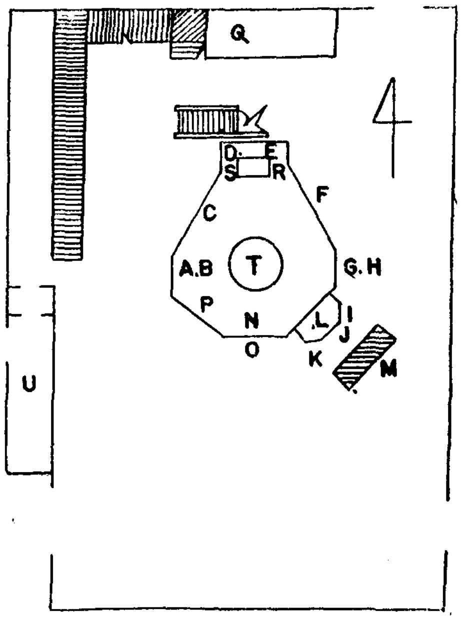

Fig. 5. Measuring Positions of γ-Radiation Level around the TRIGA Mark-Ⅱ Reactor (Horizontal Cross-section)

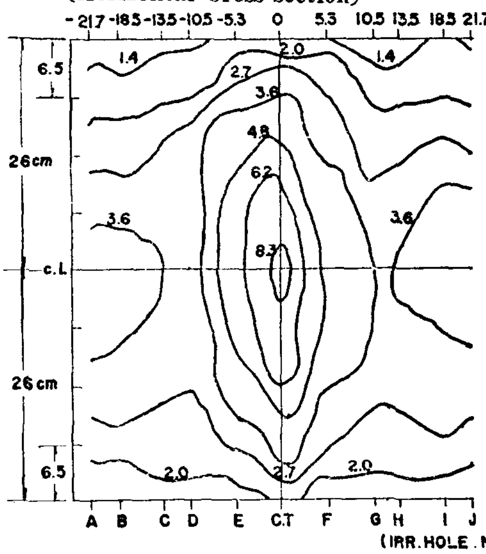

Fig. 6. Spatial Distribution of Thermal Flux of TRIGA Mark-Ⅱ Reactor Core at Power 250 KW(Unit: $\times 10^{12}$n' s/cm-sec)

V. Summary and Conclusion

It is our impression that the shielding design for the TRIGA Mark-Ⅱ reactor seems to have been carried out under the condition that the accepted working level of nuclear radiation was 300 mrem/week rather than 100 mrem/week which is the presently prevailing value.

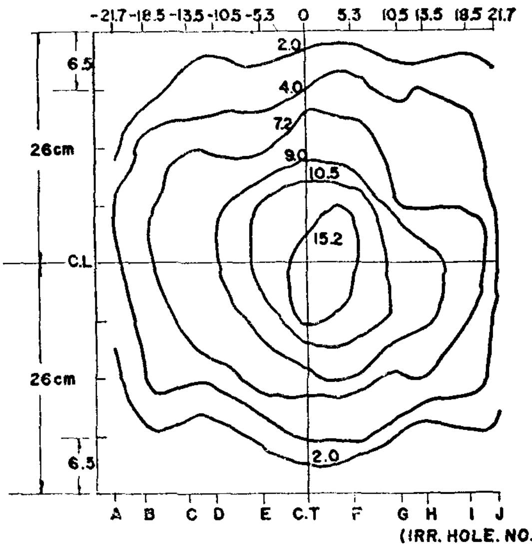

Fig. 7. Spatial Distribution of Fast Flux TRIGA Mark-Ⅱ Reactor Core at Power 250 KW (Unit: $\times 10^{12}$n' s/cm-sec)

On the contrary, the power level of this reactor was, as is well known, upgraded by 2.5 folds a couple of years ago without additional shielding reinforcement, which naturally causes a negative effect to the shielding problem.

As a conclusion, the summary of our computation in this paper can be listed as follows:

1. It is absolutely safe for both thermal and fast neutrons.

2. It is, however, not safe from gamma rays at horizontal direction.

Its magnitude is such that at the horizontal direction 22.9 cm of concrete wall must be reinforced to the present shield because the computed value comes out to be 3.95 times higher than the accepted working level, if and when the shield should meet the internationally recognized requirement.

The calculated value, for the gamma ray intensity outside the concrete shield, 9.875 mrem/hr, is a little bit higher than the measured figures in Table 7, such as 6.5, 5.5, 5.5 and 5.0 mrem/hr at the positions of A, B, G and H in Fig. 5 which are of the thinnest shield thickness around the reactor hall. Generally speaking, however, these two values fall under the same magnitude of order. It is presumed that this slight difference might have resulted from the self-absorption of the core itself, such as the absorption by the heavy nuclides.

3. It is, however, the present situation that the shielding requirement cannot be met competely unless additional shield is reinforced to the present shield or, at the worst case, the reactor building itself is torn down for the proper modification.

4. As a compromise, therefore, it is advised to the working personnel that their working hour on the ground floor of the reactor hall should be minimized so that they may not be exposed to radiation more than the maximum permissible dosage. It must be borne in mind that our people were not permissible to work on the reactor top for 40 hours per week due to the high level of radiation even at the reactor power of 100 kw. For the simplicity of computaion, gaseous radiation source, such as A-41 and N-16, which are generated in the reactor core, are not taken into account of computaion, although they are dominant on the reactor top.

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