SHIELDING CALCULATIONS

FOR THE V. P. I. UTR-10 REACTOR AT 10KW*

by

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I. INTRODUCTION

The Virginia Polytechnic Institute Training and Research Reactor (UTR-10) is a highly enriched U-235, light water cooled and moderated, graphite reflected reactor. The UTR-10 was designed by Advanced Technology Laboratories division of the American-Standard based on the Argonne National Laboratories' "Argonaut" type (1). A similar but not identical design has been marketed by American Machine and Foundry.

The core consists of two nearly identical fuel regions (5-27/32" x 19-15/16" x 57-1/4" high) embedded in a stack of graphite (3'8" x 4'8" x 4'0" high). The two fuel regions are parallel and separated by 18 inches of graphite so as to form a high flux region at the center of the reactor (see Figure 1). The design power level is 10 kw thermal.

An increase in power to 100 kw is planned to increase the neutron flux available for experiments. A 1-3/4 inch lead shield is to be placed on two sides of the core. On one of the remaining sides of the reactor is a graphite thermal column and on the other is a shield tank.

This work was undertaken to determine the radiation level around and on top of the reactor at the increased power level of 100 kw thermal.

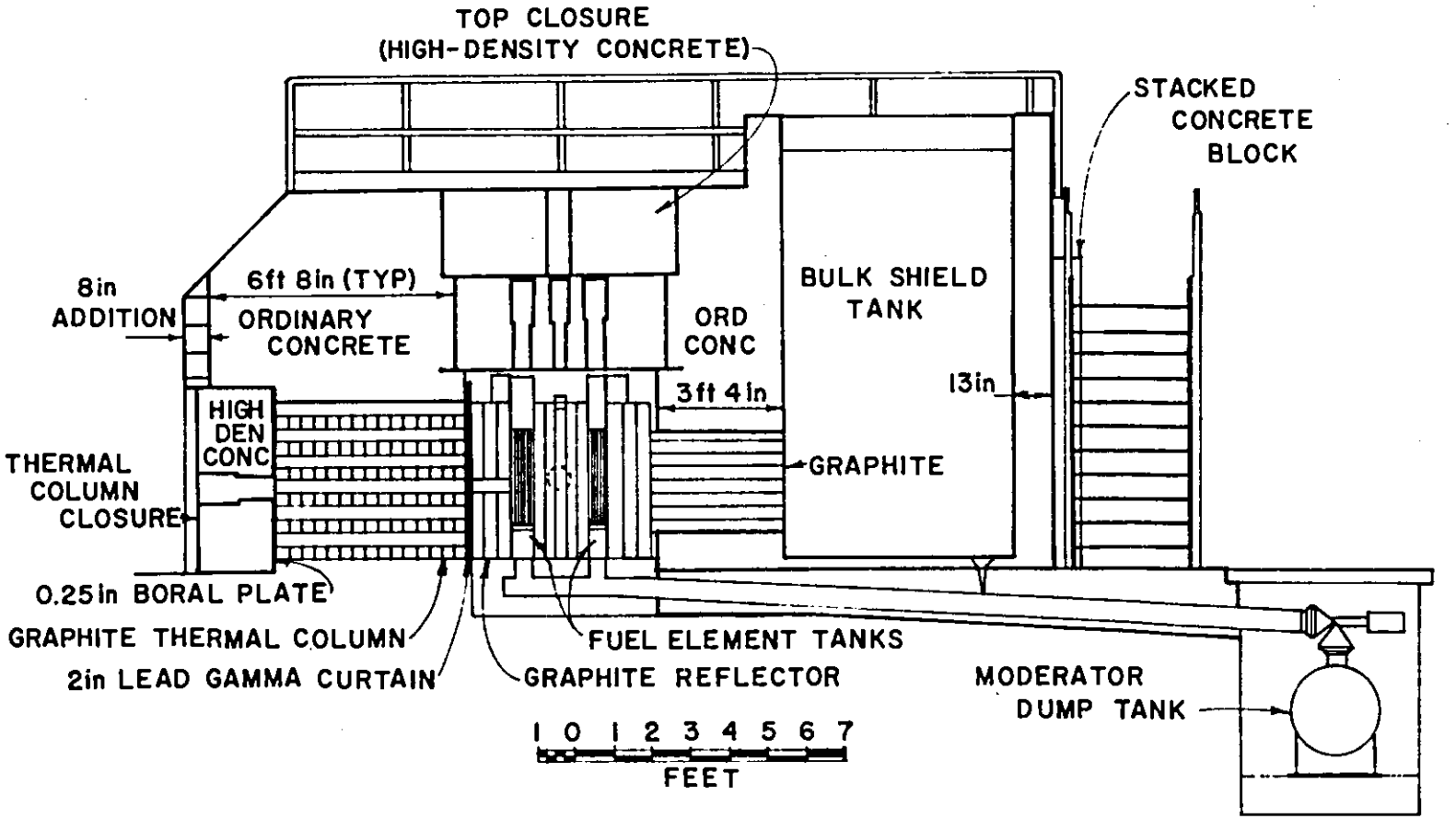

FIG. 1. PLAN AND SECTION OF THE VPI UTR-10 REACTOR SHIELD

II. LITERATURE REVIEW

A complete description of the UTR-10 reactor, including the original shield design, is to be found in the technical literature issued by Advanced Technology Laboratories (2). Detailed measurements of the performance characteristics of the reactor have been made by Stam (3) and by Sears (6). The radiation level around the reactor at a power of 10 kw has been measured by Blosser and Freestone of the Neutron Physics Division of Oak Ridge National Laboratories (7). The ORNL survey indicated that the existing reactor shield is a thoroughly effective neutron shield. The gamma shield, however, did not completely fulfill the design requirements. In addition, the Radiation Protection Guides now recommended by the Atomic Energy Commission have lowered the recommended dose rate since the original shield design (13).

The basic fundamentals of the interaction of gamma rays with matter are covered in detail by Evans (8). There are many general references on reactor shield design (10,11,12). Reactor shielding is also covered as a topic in most reactor engineering texts (13). The treatment followed in this work will be that of Glasstone and Sesonske (14) and Etherington (23). These follow closely

the method set forth in the "Reactor Shielding Design Manual" (27).

Nuclear data has been taken from ANL 5800 ("Reactor Physics Constants"), to insure a consistent set of data for the calculations (32). A thorough listing of references on reactor shielding can also be found in the latter publication.

III. RADIATION PROTECTIONS GUIDES

Allowed radiation levels, formerly called maximum permissible exposures, have been recommended by the National Committee on Radiation Protection.

A summary of these Radiation Protection Guides follows (15).

1.) The maximum permissible dose (MPD), shall not exceed 3 rem per 13 weeks, preferably spread out over this time period.

2.) The MPD shall not exceed 5 times the age minus 18 rem during a person's lifetime, i.e., 5(age-18).

It should be realized that the above levels are based only on total dose over a stated period of time. They do not specify an allowable dose rate. The allowed dose rate can only be implied when the time a person will be in the radiation field is known. For industry, the allowed dose rate is usually based on a 40 hour work week with a dose rate of 2.5 mrem per hour for individuals whose work involves continuous exposure to radiation. A summary of radiation units can be found in Appendix I.

IV. GAMMA RAY SOURCES

Prompt Fission and Delayed Fission Source.

Approximately 7.83 Mev of prompt fission gammas are produced in a single fission event. These are distributed in energy according to the Gamma Ray Spectra given in Appendix II. The resulting decay of the excited fission products yields another 7.22 Mev of delayed fission gamma rays distributed in energy as given in Appendix II. As can be seen from a comparison of these gamma ray spectra, the prompt fission gamma rays are much more energetic and therefore more important than the delayed fission gamma rays in shielding of nuclear reactors. The total number of fissions per second taking place within the reactor is a direct function of the power level and is given by 3.1×10^{10} fissions/watt-sec of reactor power (16).

The measured dose rates at 10 kw were used to establish the source geometry. Good agreement between calculated and experimental dose rates was found with the source geometry consisting of two point sources, one centered in each of the two fuel regions. The strength of each point source is assumed to be one-half of the total source due to prompt and delayed fission gamma rays. A table of the gamma ray point source strengths can be found in Appendix III.

Capture Gamma Source.

The absorption of a neutron by a nucleus leaves the nucleus in an excited state. Among the competing modes of deexcitations is the emission of a capture gamma ray. The absorption of a thermal neutron leaves the nucleus in an excited level that corresponds to the separation energy of the last neutron, approximately 7 Mev (9). Thus the nucleus may go to the ground state by the emission of one gamma ray of substantial energy, depending on its neutron separation energy.

Due to the small volume fraction of aluminum and uranium in the core, the total capture gamma source in the core is assumed to be due only to captures in graphite and water. The capture gamma source in graphite is reduced from a volume distributed source to a point source at the center of the core. The same procedure is followed to reduce the distributed volume source in water to two point sources, one centered in each of the two fuel regions.

The source transformation proceeds as follows: It can be shown (17) that the total flux at the surface of an infinite slab source of thickness t(cm) can be written as

$$\text{Total flux at surface} = \frac{S_v}{2\mu_v}\left[1-E_2(\mu_v t)\right]$$

S_v represents the volume distributed source strength (photons/cm^3-sec) and μ_v is the total gamma attenuation coefficient for the volume material (cm^{-1}). $E_2(\mu_v t)$ is the exponential integral function (28).
For a thick slab $E_2(\mu_v t)$ is small and the flux can be written as

$$\text{Total Flux at surface} = \frac{S_v}{2\mu_v} .$$

Assuming an uniform neutron flux ϕ (neutrons/cm^2-sec) the volume source is given by

$$S_v = \phi \Sigma_c$$

where Σ_c is the macroscopic capture cross-section (cm^{-1}).

To estimate an equivalent isotropic point source we require that the flux at the surface at the nearest point to the point source be the same as that given by the distributed volume source. Thus we require that the point source have strength D_o photons/sec such that

$$\phi_\gamma = \frac{D_o \; e^{-\mu_v t/2}}{4\pi \; (t/2)^2} = \frac{S_v}{2\mu_v}$$

where t/2 is the half thickness of the slab source.

The calculated source strengths for both water and graphite are given in Appendix III.

The capture gamma sources in the non-multiplying parts of the reactor, i.e., the concrete shielding, thermal column and shield tank duct, are functions of the neutron distributions in these mediums. These sources are not conveniently handled by the previous method of source transformation. Instead we choose not to calculate a source strength as such but to consider the total problem of production and attenuation of these secondary gamma rays as a unit. This method will be demonstrated in the following section.

The sources that have been chosen yield good agreement when calculations are made and compared to the measured values of the gamma dose rates at a power level of 10 kw thermal.

V. ATTENUATION OF GAMMA RAYS

The attenuation of gamma rays is considered in two parts. The first is concerned with the attenuation from a point source. The second deals with both the production and attenuation of capture gamma rays due to a neutron flux in a medium. This entails the development of an expression for the neutron flux as a function of position within the shield, which in turn gives the distribution of gamma sources within the shield. All results are derived for a monoenergetic source of gamma rays. The total dose rate is obtained by adding the results of the calculations made for all energy groups in the gamma ray spectrum.

The energy flux of a monoenergetic source of gamma rays, ϕ_γ (Mev/cm^2-sec) is given for a single material by (18).

$$\phi_\gamma = S_p \, B(\mu x) \, \frac{e^{-\mu x}}{4\pi R^2}$$

Where

S_p = Gamma point source strength (Mev/sec)

μ = Total gamma ray attenuation coefficient for the material (cm^{-1})

x = Thickness of the material (cm)

R = Source to detector distance (cm)

$B(\mu x)$ = Build up factor which accounts for the effect of scattered radiation.

For a shield composed of various thicknesses of different materials, the equation for the flux can be written

$$\phi = S_p B_I\left(\sum_{i=1}^{I} \mu_i x_i\right) \frac{e^{-\sum_{i=1}^{I} \mu_i x_i}}{4\pi R^2}$$

where i denotes the ith material. The particular form chosen for the build-up factor will be discussed below.

The build-up factor for a point source in a single medium can be approximated by the sum of two exponentials of the form (20)

$$B(\mu t) = A\, e^{-\alpha \mu t} + (1-A) e^{-\beta \mu t}$$

where the constants A, α, and β are listed for concrete in Appendix II. The values of these constants for other materials can be found in Reference (33).

For multi-layer shields there are many different possible combinations of the build-up factors that have been used to estimate the total build-up factor for an entire shield (20). In this treatment we assume that the build-up factor is characteristic of the last thick medium, I,

and determined by the total number of mean free paths of all the materials between the source and the outer face of the reactor (24).

Since the gamma flux is a scalar function the total dose rate at any point is the sum of all point sources that contribute to the flux at the point.

To calculate the dose rate due to capture gammas outside the core the distribution of thermal neutrons in this medium must be calculated. Following the method of Glasstone and Sesonske (19) we first write the one velocity, steady state diffusion equation.

$$D_s \nabla^2 \phi_s (x) - \Sigma_{as} \phi_s(x) + S = 0$$

Where

$\phi_s(x)$ = Thermal neutron flux at a distance x(cm) in the material (neutrons/cm^2-sec)

Σ_{as} = Thermal neutron macroscopic absorption cross section (cm^{-1})

D_s = Thermal diffusion coefficient of the material (cm)

S = Thermal neutron source due to the slowing down of fast neutrons in the material (neutrons/cm^3-sec)

The thermal neutron source, S is assumed to be represented by $\Sigma_R \phi_f(x)$ where Σ_R is an effective macroscopic slowing down or removal cross section that is

determined for the material by experiment. The removal cross section represents the probability per cm that a fast neutron will be slowed down to the thermal group. The fast flux in the material is then assumed to be represented by

$$\phi_f(x) = \phi_f(0)\ e^{-\Sigma_R x}$$

where $\phi_f(0)$ is the fast flux at the shield-refector interface.

With these assumptions the diffusion equation becomes

$$D_s \nabla^2 \phi_s(x) - \Sigma_{as}\phi_s(x) + \Sigma_R \phi_f(0)\ e^{-\Sigma_R x} = 0.$$

The solution for an infinite slab shield can be written as

$$\phi_s(x) = Ae^{kx} + B\ e^{-kx} + Ce^{-\Sigma_R x}$$

where $k = \sqrt{\Sigma_{as}/D_s}$. For boundary conditions we assume that both $\phi_s(0)$ and $\phi_f(0)$ are known and that the flux must be finite for all values of x. The last boundary condition requires that $A \equiv 0$. From the knowledge of the fluxes $\phi_s(0)$ and $\phi_f(0)$ the constants B and C can be evaluated with the result that

$$\phi_s(x) = \left[\phi_s(0) - \frac{\Sigma_R \phi_f(0)}{D_s(k^2-\Sigma_R^2)}\right] e^{-kx} + \frac{\Sigma_R \phi_f(0)}{D_s(k^2-\Sigma_R^2)}\ e^{-\Sigma_R x}.$$

With this expression for the thermal neutron flux for one non-multiplying material outside the core, we can now calculate the gamma source and dose rate at the outside surface of the material.

To calculate the gamma energy flux at a surface a distance X in the material due to a volume source, we consider an infinite slab of material of thickness dx. If S_v is the volume source (Mev/cm^{3}sec) distributed according to the thermal neutron flux $\phi_s(x)$, the energy flux (Mev/cm^2-sec) of gamma rays of energy E_γ (Mev) per unit area, S_a, from the surface of the element dx is given by

$$S_a = S_v dx = \phi_s(x)\, \Sigma_c E_\gamma\, dx.$$

The flux at x is determined by the attenuation for an infinite plane source of gamma rays and is given by (19)

$$\phi_\gamma(x) = \frac{1}{2} S_a E_1(\mu x) = E_\gamma \frac{1}{2} \Sigma_c \phi_s(x)\, E_1(\mu x) dx.$$

To obtain the total gamma ray energy flux at a surface a distance X in the material, we integrate the expression for $\phi_\gamma(x)$ over the thickness of the non-multiplying material. Then we have

$$\phi_\gamma(X) = \frac{\Sigma_c}{2} \int_0^X \phi_s(x) E_1(\mu x) dx.$$

When the previously derived expression for $\phi_s(x)$ is substituted into the equation for $\phi_\gamma(x)$ and integrated the result is

$$\phi_\gamma(x) = \left[\phi_s(x) - \frac{\Sigma_R \phi_f(0)}{D_s(k^2 - \Sigma_R^2)}\right] \frac{\Sigma_c E_\gamma e^{-kx}}{2k} \left\{ e^{kx} E_1(\mu x) + Ei\left[\mu x(\nu_1 - 1)\right] + \ln\left(\frac{\nu_1 + 1}{\nu_1 - 1}\right)\right\}$$

$$+ \left[\frac{\Sigma_R \phi_f(0)}{D_s(k^2 - \Sigma_R^2)}\right] \frac{\Sigma_c E_\gamma e^{-\Sigma_R x}}{2\Sigma_R} \left\{ e^{\Sigma_R x} E_1(\mu x) + Ei\left[\mu x(\nu_2 - 1)\right] + \ln\left(\frac{\nu_2 + 1}{\nu_2 - 1}\right)\right\}$$

where

Ei(y) is the exponential integral of the argument y, and

$$Ei(y) = -E_1(-y)$$

also

$$\nu_1 = \frac{k}{\mu} \quad \text{and} \quad \nu_2 = \frac{\Sigma_R}{\mu} .$$

The evaluation of the integrals given are for ν_1 and ν_2 greater than one. The corresponding expressions for ν less that one and equal to zero can be found in Etherington (25). Graphs of the exponential integrals are to be found in Rockwell along with a synopsis of the properties of these functions (28,29).

The above equations were derived assuming a build-up factor of unity. The actual build-up factors are uncertain, because of the variation of the actual numbers of relaxation lengths traveled by the photons. As was mentioned previously the calculated results of the total dose rate from all sources agree well with the measured values of the gamma dose rates. Thus one concludes that the above assumptions do account for the gamma dose rate around the reactor.

For multi-layer shields the above calculation can be made for each material. The fast and thermal fluxes at the interface are assumed to be the initial fluxes $\phi_s(0)$ and $\phi_f(0)$ for the next layer. This assumption departs from the usual diffusion theory approach that requires both neutron flux and net current to be continuous at the interface. The assumption has been made for ease of calculation and will be valid for materials that have similar neutron diffusion properties.

The gamma ray energy flux that is calculated at the interface is then assumed to approximate an isotropic plane source and the attenuation equation for this case is used to find the flux at the outside face of the shield.

VI. DOSE RATE CALCULATIONS

The calculated dose rates were based on a power level of 100 kw. The prompt and delayed fission gamma ray source strengths were based on 3.1×10^{10} fissions/watt-second of reactor power. The capture gamma source in the core was based on an uniform thermal flux of 1×10^{12} neutrons/cm^2-sec. This number is approximately 10^3 times the flux level as measured at 0.1 kw (4). The fast and thermal neutron fluxes at the reflector core interface were taken as 9.3×10^9 neutrons/cm^2-sec and 6.2×10^9 neutrons/cm^2-sec respectively and were assumed to be uniform over the entire core. These values of fluxes represent a factor of 10 increase over the values assumed in the original shield design (30). The main justification of these choices of source strengths is that they predict approximately the same dose rates at 10 kw as have been measured. The only departure from the above set of data was that the thermal flux at the thermal column shield interface was taken as 5.5×10^8 thermal neutrons/cm^2-sec, so as to agree with the data given by Stam (5). This was necessary as the value given by the assumed treatment does not take into account the flux depression due to the interface between the dissimilar

mediums graphite and heavy concrete.

With the above sources and using the nuclear data as given in Appendix II, the results given in Table 1 were obtained.

The points at which the dose rates were calculated can be correlated with the numbers given on Figure 2.

A sample procedure showing the intermediate results can be found in Appendix III.

TABLE I

Calculated Dose Rates

Position No.	Dose Rate due to prompt and delayed fission and capture gammas in Water	Dose rate due to capture gammas in Graphite	Dose rate due to capture gammas in material outside Core	Total dose Rate	Measured dose rate (5) X 10
	mr/hr	mr/hr	mr/hr	mr/hr	mr/hr
1	4.27	0.716	33.8	38.8	31.6
2	17.1	1.42	137.5	156	53.5*
3	0.653	0.088	106	107	94.9
4	7.76	1.17	11.65	20.6	25.6
5	0.501	0.00225	7.70	8.20	16.2**
6	0.511	0.128	12.44	13.1	31.8**
7	0.0278	0.00237	7.70	7.70	39.2**

* Includes steel plate not considered in calculations.

**Does not include 2" lead slab between reflector and shield.

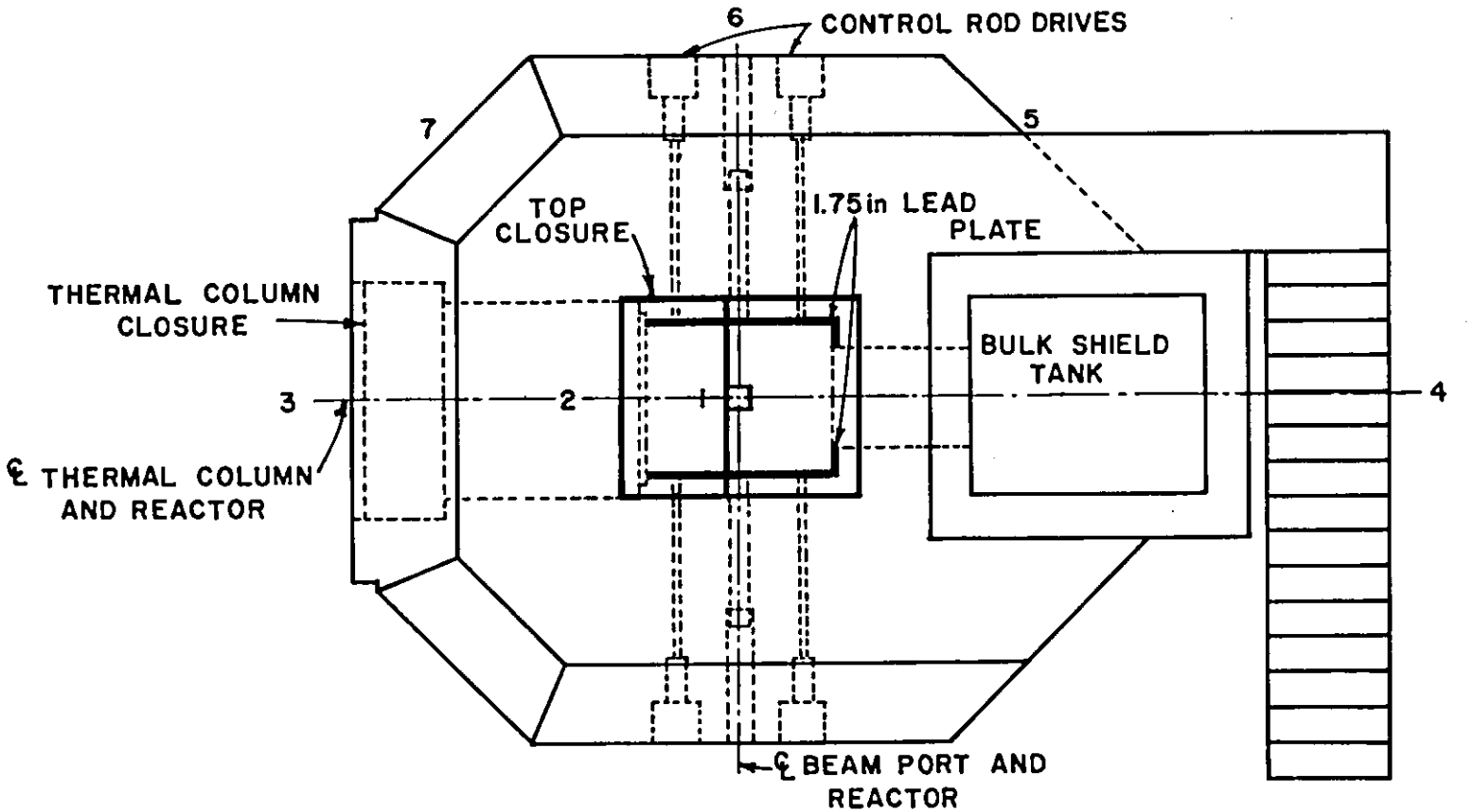

FIG. 2. DOSE RATE CALCULATION POINTS

VII. CONCLUSIONS

The results show that the major contribution to the gamma ray dose rate is due to the capture gamma sources in the concrete shielding materials. Thus to lower the gamma ray dose rate appreciably, the shielding material will have to be placed on the outside face of the reactor system.

Detailed calculations of the neutron dose rates have not been made because the increase in power will increase the neutron flux by an approximately factor of 10 and should raise the measured neutron dose rate accordingly. Detailed gamma ray calculations are necessary because the gamma sources are distributed throughout the reactor system whereas neutron production is confined to the core.

The main distributed gamma source is due to the fast leakage neutrons, if the fast neutron leakage flux could be reduced, a reduction in the gamma source would also be effected.

The only serious high dose rate regions are on the top of the reactor and at the thermal column door. Additional shielding should be placed at these positions before an increase in reactor power is undertaken.

VIII. ACKNOWLEDGMENTS

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XI. APPENDIX I

UNITS OF RADIATION DOSE (26)

ROENTGEN: The roentgen (r) is that quantity of X or δ radiation such that the associated corpuscular emission per 0.001293 gm of air ($1cm^3$ of dry air at 0^oC and 760 mm Hg) produces in air ions carrying one electrostatic unit of quantity of electricity of either sign.

REP: The roentgen equivalent physical (rep) is a unit of absorbed dose corresponding to .93 ergs/gm of soft tissue. (obsolescent)

RAD: The radiation absorbed dose (rad) corresponds to an energy absorption of 100 ergs/gm of any medium.

REM: The roentgen equivalent man (rem) is an Relative Biological effectiveness (RBE) (22) dose unit that is defined by the relation number of rem = RBE times (number of rad).

$$\text{RBE} = \frac{\text{Physical dose of 200-kev x radiation to produce effect of interest.}}{\text{Physical dose of comparison radiation to produce same effect.}}$$

Absorbed Dose = Energy imparted to matter by ionizing particles.

Exposure Dose = Ionization produced by radiation.

XII. APPENDIX II

NUCLEAR DATA

Gamma Ray Spectra

Energy (Mev)	Prompt Fission (Mev/fiss)	Delayed Fission (Mev/fiss)	Capture Gamma Spectra (Mev/Neutron Capture) in				
			Barytes Concrete	Ordinary Concrete	Graphite	Water	Lead
1	3.45	5.16	0.279	0.546			
2	2.36	1.737	0.548	1.066		2.23	
3	1.175	0.322	1.278	2.217			
4	0.477		1.932				
5	0.203				5.0		
6	0.136		1.224	0.856			4.2
7	0.026						
8			0.736	0.4336			6.86
Total	7.827	7.219	5.997	5.1186	5.0	2.23	11.06

Total Gamma Ray Attenuation Coefficients (cm^{-1})

Energy (Mev)	Water $\rho = 1$ (gm/cm^3)	Ordinary Concrete $\rho = 2.3$ (gm/cm^3)	Barytes Concrete $\rho = 3.5$ (gm/cm^3)	Graphite $\rho = 1.67$ (gm/cm^3)	Lead $\rho = 11.4$ (gm/cm^3)
1	.0706	.146	.222	.1062	.780
2	.0493	.1025	.156	.0741	.521
3	.0396	.0835	.127	.0595	.480
4	.0339	.0729	.111	.0508	.479
5	.0301	.066	.1005	.0451	.486
6	.0275	.0616	.0937	.0409	.497
7	.0258	.0589	.0896	.0383	.511
8	.0240	.0559	.085	.0356	.524

Flux to Dose Rate Conversion Factors

Energy (Mev)	1	2	3	4	5	6	7	8
$\frac{mr/hr}{Mev/cm^2\text{-}sec} \times 10^3$	1.93	1.64	1.45	1.33	1.24	1.18	1.14	1.04

Neutron Diffusion Properties of Materials

Material	Density (gm/cm^3)	Σ_R (cm^{-1})	D_s (cm)	k (cm^{-1})	Σ_c (cm^{-1})
Ordinary Concrete	2.3	0.085	0.484	0.139	0.0094
Barytes Concrete	3.5	0.125	0.44	0.212	0.197
Water*	1.0	0.1	0.159	0.351	0.0195
Graphite*	1.67	0.085	0.879	0.0202	2.86×10^{-4}
Lead*	11.35	0.117	0.918	0.0784	5.63×10^{-3}

*Based on data from Etherington (23).

Coefficients for an Analytical Fit to Build-up Factor Equation

		Gamma Ray Energy (Mev)					
		1	2	3	4	6	8
Concrete	A	10.0	6.3	4.7	3.9	3.1	2.8
	$-\alpha$	0.088	0.069	0.069	0.059	0.059	0.056
	β	0.029	0.058	0.073	0.079	0.083	0.086

Except where noted all of the above data is taken from "Reactor Physics Constants" (32).

XIII. APPENDIX III

Source Strengths (based on 100 kw)

Point Sources in Fuel Regions (Mev/sec).

Energy (Mev)	Total Prompt Fission Source	Total Delayed Fission Source	Total Capture Gammas in Water	Individual Point Source Strength
1	1.07×10^{16}	1.60×10^{16}		1.34×10^{16}
2	7.31×10^{15}	5.38×10^{15}	1.88×10^{15}	7.27×10^{15}
3	3.64×10^{15}	1.00×10^{15}		2.32×10^{15}
4	1.48×10^{15}			7.40×10^{14}
5	6.29×10^{14}			3.15×10^{14}
6	4.22×10^{14}			2.11×10^{14}
7	8.06×10^{13}			4.03×10^{13}

Capture Gamma Source in Graphite Reflector (Mev/sec)

Energy (Mev)	Total Source
5	7.85×10^{15}

Sample Calculation Procedure:

This data will be taken from calculations made for the dose at the top center of the Reactor, Point I.

Attenuation of Point Sources in Fuel Region

Energy (Mev)	$\sum_{i=1}^{I}\mu_i x_i$	$\frac{1}{4\pi R^2} e^{-\sum_{i=1}^{I}\mu_i x_i}$ (cm^{-2})	$B_I(\sum_{i=1}^{I}\mu_i x_i)$	Dose Rate (mr/hr)
1	35.42	5.95×10^{-22}	220	3.39×10^{-6}
2	24.98	2.03×10^{-17}	33.4	8.56×10^{-3}
3	20.29	2.87×10^{-15}	16.3	1.58×10^{-1}
4	17.72	2.88×10^{-14}	10.05	2.96×10^{-1}
5	15.98	1.66×10^{-13}	8.2	5.33×10^{-1}
6	14.87	5.04×10^{-13}	6.9	8.67×10^{-1}
7	14.21	9.65×10^{-13}	6.2	2.75×10^{-1}
Contribution to Total Dose Rate = 2.14 mr/hr				

Since the two fuel regions are symmetric with respect to the top center of the reactor, the dose due to the point sources in the fuel regions is twice that of each individual source alone. Therefore the total prompt and delayed fission gamma rays and the capture gamma rays from water is 4.28 mr/hr.

Attenuation of Capture Gamma Source in Graphite Reflector

Energy (Mev)	$\sum_{i=1}^{I}\mu_i x_i$	$\frac{1}{4\pi R^2} e^{-\sum_i \mu_i x_i}$ (cm^{-2})	$B_I(\sum_{i=1}^{I}\mu_i x_i)$	Dose Rate (mr/hr)
5	16.65	8.75×10^{-14}	8.5	7.1×10^{-1}

Total dose from all point sources = 4.99 mr/hr.

Dose Rate Due to Distributed Source in Concrete

Energy (Mev)	H_1(Mev/cm^2-sec)	H_2(Mev/cm^2-sec)	Dose Rate (mr/hr)
1	-8.02×10^{-4}	7.64×10^{1}	1.47×10^{-1}
2	-1.86	2.56×10^{2}	4.15×10^{-1}
3	-1.78×10^{1}	6.46×10^{3}	9.33
4	-2.28×10^{1}	4.80×10^{3}	6.35
6	-1.54×10^{3}	7.01×10^{3}	6.45
8	-4.65×10^{3}	1.54×10^{4}	1.11×10^{1}
Contribution to Total Dose Rate = 3.38×10^{1} mr/hr			

Where

$$H_1 = \left[\phi_s(x) - \frac{\Sigma_R \phi_f(0)}{D_s(k^2-\Sigma_R^2)}\right] \frac{\Sigma_c E_\gamma e^{-kx}}{2k} \left\{e^{kx} E_1(\mu x) + Ei[\mu x(\nu_1-1)] + \ln\left(\frac{\nu_1+1}{\nu_1-1}\right)\right\}$$

$$H_2 = \left[\frac{\Sigma_R \phi_f(0)}{D_s(k^2-\Sigma_R^2)}\right] \frac{\Sigma_c E_\gamma e^{-\Sigma_R x}}{2\Sigma_R} \left\{e^{\Sigma_R x} E_1(\mu x) + Ei[\mu x(\nu_2-1)] + \ln\left(\frac{\nu_2+1}{\nu_2-1}\right)\right\}.$$

Total Dose Rate at Point I = 38.8 mr/hr.

ABSTRACT

A re-evaluation of the radiation shield of the V. P. I. UTR-10 nuclear reactor is necessary before an increase in the licensed reactor power from 10 kw to 100 kw can be accomplished. Experimental measurements at 10 kw indicated satisfactory performance of the shield for neutrons but certain regions have a higher gamma dose rate than had been predicted by the original shield design.

The results of the experimental measurements were used to establish a gamma ray source distribution. This distribution was then used to calculate the gamma dose rates at various positions around the reactor after an addition of a lead gamma shield on two sides of the reactor core.

The results indicated that additional shielding may be required on the top of the reactor and on the thermal column closure but the radiation level should not prohibit the increase in the reactor power level.