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SHIELDING DESIGN CALCULATION OF A 50 MW RESEARCH REACTOR

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ABSTRACT

The computer code ANISN/PC has been applied to calculate the group flux distribution across different shield layers of a 50 MW light water research reactor. The code has been run in P3 approximation and S8 discrete ordinates. The calculated group fluxes multiplied by appropriate flux-to-dose rate conversion factors have been used to give the dose distribution across the shield layers.

The thickness of the concrete shield has been determined to give the dose rate at the outer surface of the shield as 0.5 nSv/sec. The same calculation have been also performed in axial direction to determine the thickness of water needed above the core to reduce the dose level to 2.5 nSv/sec.

The result of calculation shows that the contribution of capture gamma rays to the total dose at the outer surface of the shield is more than 50 percent. This simplifies the calculations to determine the shield layer thickness, especially in preliminary stages of the shield design.

KEYWORDS

ANISN, Boltzman transport equation, discrete ordinates method, dose, flux to dose ratio, gamma rays spectrum, IRAN.LIB cross-section library, neutron spectrum, gamma ray spectrum.

INTRODUCTION

In this work we have performed shielding calculation for a light water reactor with Beryllium reflector around the core. The fuel of the reactor is 19.5% enriched Uranium with 10 kg of U-235 isotope. The thickness of Be reflector is 23 cm and the thickness of water layer between the reflector and the reactor tank is 107 cm and the thickness of water around the tank is 146 cm [1,2,3]. The thickness of ordinary concrete biological shield beyond water, have been calculated to reduce the equivalent dose rate to 0.5 nSv/sec. Furthermore, the height of water above the core has been also calculated to reduce the equivalent dose rate to 2.5 nSv/sec.

The horizontal cross section dimensions of the parallel-piped shaped core is

39x54 cm and its height is 75 cm. A simplified vertical cross section of the reactor is shown in fig. 1.

APPLICATION OF COMPUTER CODE ANISN/PC

The computer code ANISN/PC solves the Boltzman transport equation in one dimension (slab, cylindrical or spherical geometry) using discrete ordinates (S_N) method [4]. The energy dependence is treated using multigroup method. The code takes into account the scattering anisotropies by expanding the scattering cross sections in Legendre's polynomials, so called anisotropic S_N method.

The Boltzmen transport equation integrated over energy group g is given below [5]:

$$\Omega.\nabla\varphi_g(r,\Omega)+\Sigma_{tg}\,\varphi_g(r,\Omega)=\sum_{l=0}^{L}\sum_{g'}^{G}\int_{4\pi}d\Omega'\left(\frac{2l+1}{2}\right)\Sigma^{(l)}_{g'\rightarrow g}(r)P_l(\mu)\,\varphi_{g'}(r,\Omega')$$
$$+\frac{\chi_g}{k_{eff}}\sum_{g'}^{G}\left(\frac{\nu\Sigma_f}{4\pi}\right)_{g'}\varphi_{g'}(r)+S^{g}_{ext}(r) \qquad ,\ g=1,2,3,\dots,G \tag{1}$$

where $\varphi_g(r,\Omega)$ is the group angular flux for group g defined as:

$$\varphi_g(r,\Omega)=\int_{E_g}^{E_{g+1}}\varphi(r,E,\Omega)\,dE \tag{2}$$

and $\varphi_g=\int_\Omega\varphi_g(r,\Omega)d\Omega$ is the group flux in group g. The quantities $(\nu\Sigma_f)_g,\Sigma_{tg},\Sigma^{(l)}{}_{g'\rightarrow g}$ and χ_g are group constants averaged over group g. For example:

$$\Sigma_g(r)=\frac{\int_g\Sigma(r,E)\varphi(r,E)dE}{\varphi_g(r)} \tag{3}$$

and

$$\Sigma^{(l)}_{g'\rightarrow g}(r)=\frac{\int_g dE\int_{g'}\Sigma^{(l)}(r,E'\rightarrow E)\int_{4\pi}\varphi(r,E',\Omega')\,P_l(\mu)\,d\Omega'dE'}{\int_{g'}\int_{4\pi}\varphi(r,E',\Omega')\,P_l(\mu)\,d\Omega'dE'} \tag{4}$$

In solving shield problems in which capture gamma rays (n,γ) play an important role in the total dose at the shield outer surface, the coupled neutron-gamma cross sections should be included in the cross section libraries. The library "IRAN.LIB" includes these coupled cross sections [6]. In our calculations, we have used IRAN3.LIB, which includes 7 groups of neutrons (1-7) and 18 group of gamma rays (8-25) in ISOTXS format.

The number densities of different isotopes in the reactor core (n_i) have been calculated using $n_i=\frac{N_0\,\rho_i\,V_i}{A_iV}$, where N_0 is the Avogadro's number, V is the volume of the core and ρ_i,V_i and A_i are density, volume and atomic mass number of the ith isotope, respectively. The number densities of different isotopes in the reactor shield materials have been calculated using $n_i=\frac{N_0 W_i \rho}{A_i}$, where W_i is the weight fraction of the ith isotope of the shield material, ρ is the density of the shield material and N_0 and A_i have the same definitions as above. The number densities given to the code should be in 1/cm-barn. The number densities calculated for different isotopes in the reactor core and shield are given in tables 1 and 2 [7].

We have applied ANISN/PC computer code in P3 approximation and S8 discrete ordinates to calculate the flux distribution in shield layers in each energy group. We have used the distributed fixed source option of the code. In order to distinguish between capture gamma rays and core gamma rays contribution to the total dose, we have run the program with different input files. In one case we have considered only the neutron sources that during slowing down and thermalization produce capture gamma rays. In in the second case, we have considered only the core gamma rays including prompt and fission product decay gamma rays.

The source density of neutrons has been calculated from Watt spectrum given below:

$$S_n^g=3.13\times10^{10}\, \nu\, \chi_g\, \rho \qquad \text{n/cm}^3\text{-sec.} \tag{5}$$

Where ν is 2.45 neutrons per fission and ρ is the power density in W/cm^3 and χ_g is calculated by $\chi_g= \int_g \chi(E)dE$ where

$$\chi(E)= 0.484 \sinh \sqrt{2E} \exp(-E) \tag{6}$$

Where E is the neutron energy in MeV.

The values of χ_g calculated for 7 energy groups of neutrons are given in table 3.

The source density of prompt core gamma rays has been calculated as follows:

$$S_{\gamma F}^g= 3.13\times10^{10}\, \Gamma_F^g\, \rho \qquad \gamma/\text{cm}^3\text{-sec} \tag{7}$$

$$\Gamma_F^g= \int_g \Gamma_F(E)dE \tag{8}$$

$$\Gamma_F(E)= 6.94\ ; \quad 0.1<E<0.6 \text{ MeV} \quad \gamma/\text{fiss.} -\text{MeV} \tag{9}$$
$$\Gamma_F(E)= 20.2 \exp(-1.78E); \quad 0.6<E<1.5 \text{ MeV} \quad \gamma/\text{fiss.} -\text{MeV} \tag{10}$$
$$\Gamma_F(E)= 7.2 \exp(-1.09E); \quad 1.5<E<10.5 \text{ MeV} \quad \gamma/\text{fiss.} -\text{MeV} \tag{11}$$

The source density of fission product decay gamma rays has been calculated as follows:

$$S_{\gamma FP}^g= 3.13\times10^{10}\, \Gamma_{FP}^g\, \rho\, \gamma/\text{cm}^3\text{-sec} \tag{12}$$

$$\Gamma_{FP}^g= \int_g \Gamma_{FP}(E)dE \tag{13}$$

$$\Gamma_{FP}(E) = 7.4\ \exp(-1.1E) \quad \gamma/\text{fiss.}-\text{MeV} \tag{14}$$

The values of $\Gamma_F{}^g$ and $\Gamma_{Fp}{}^g$ in 13 groups out of 18 gamma energy groups (group # 8 to group # 20) and the $\Gamma_C{}^g = \Gamma_F{}^g + \Gamma_{FP}{}^g$ are given in table 4.

The spatial distribution of fission sources inside the reactor core is considered as $\cos(\beta x)$ where x is the distance from the center of the core. The value of β is calculated to give

$$\phi_{max}(x)/\bar{\phi}(x) \approx (1.9)^{1/3} = 1.24 \tag{15}$$

where 1.9 is the $\phi_{max}/\bar{\phi}$ over the whole reactor core in 3 dimensions.

RESULT OF CALCULATION

Using the data mentioned in the previous section, the computer code ANISN/PC has been applied for different concrete shield thicknesses in radial dimension and for different water heights above the reactor core in axial dimension to give the neutron and gamma fluxes at the outer surface of the concretes shield and at the surface of water.

To convert the calculaed fluxes to equivalent dose rate the computer program DOSEFLUX has been developed in FORTRAN 77 language. The flux output of ANISN called ANISNC4.PUN has been used as input file for DOSEFLUX program. Flux-to-dose rate conversion factors have been used from references 8,9 and 10.

The variation of equivalant dose rate at the outer surface of the concrete shield versus concrete thickness is shown in fig. 2. This curve shows that for the thickness of 270 cm of concrete, the equivalant dose at the outer surface will be 0.5 nSv/sec. The same curve is drawn versus water height above the core in fig. 5. This figure showes that 1000 cm of water is needed above the core to reduce the equivalent dose rate to 2.5 nSv/sec. The variation of neutron and gamma ray fluxes accross the shield in radial direction is shown in fig. 4.

Using the calculated flux distribution as input for DOSEFLUX program, depth dose distribution accross the shield in radial and axial directions are calculated and shown in fig. 3. and fig. 6. The neutron dose, capture gamma dose, core gamma dose and the total dose are drawn in these figures simultaneously.

We found that the contribution of core gamma rays to the total dose at the surface of the shield is less than 50%. This result concludes that in preliminary design calculation for determining the thickness of the reactor shield, ignoring core gamma rays does not affect the results so much.

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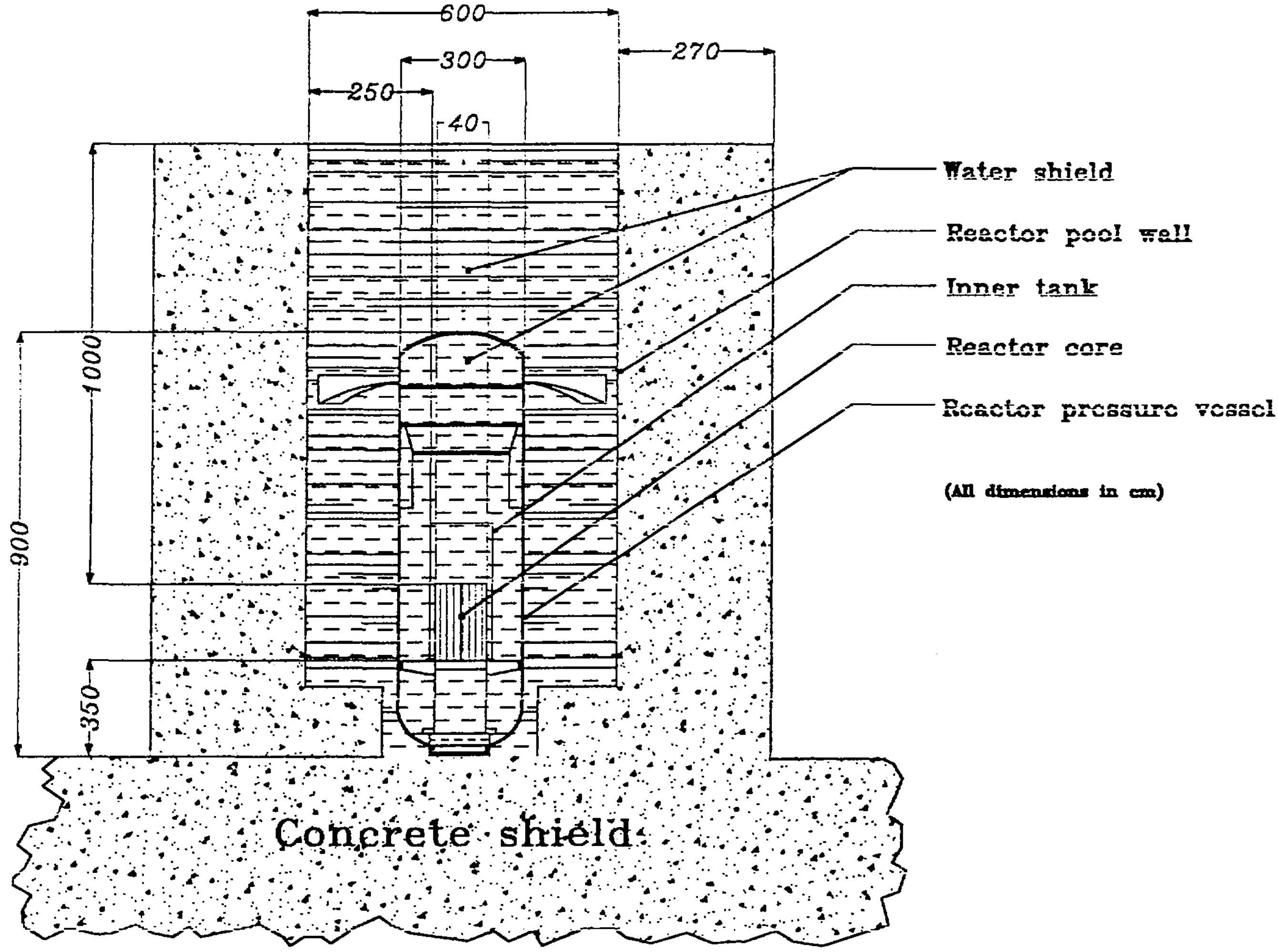


Fig. 1. Simplified Vertical Section of the Reactor.

Table 1. Isotopical Composition of Homogenized Reactor Core.

Isotope	Number Density (1/cm - barn)
H^{1}	0.04128
O^{16}	0.02064
Al^{27}	0.01792
U^{235}	0.0001702
U^{238}	0.0007128

Table 2. Elemental Composition of Ordinary (Portland) Concrete with Density of 2.35 g/cm^3.*

Element	Atomic Weight**	wt%	Number Density (1/cm-barn)
H	1.00797	1.0	0.014035
O	15.9994	52.9	0.046774
Si	28.086	33.7	0.016972
Al	26.9815	3.4	0.001783
Fe	55.847	1.4	0.000355
Ca	0.08	4.4	0.001553
Mg	24.312	0.2	0.000116
C	12.01115	0.1	0.000118
Na	22.9898	1.6	0.000985
K	39.102	1.3	0.000470

* From Engineering Compendium on Radiation Shielding, Vol. II, Editor JAEGER, P. 358 (1968).
** Based on C^{12} = 12.00000 a.m.u

Table 3. Average Number of Prompt Neutron Emitted per Fission in each Energy Group ($\bar{\nu}$=2.45).

Group No.	Energy Band	No. of Prompt Neutron per Fission
1	5.221 - 17.33 MeV	0.122
2	1.003 - 5.221 MeV	1.548
3	0.498 - 1.003 MeV	0.442
4	0.0980 - 0.498 MeV	0.302
5	0.00912 - 0.0980 MeV	0.0341
6	0.532 - 9120 eV	0.00104
7	0 - 0.532 eV	4.63E-10

Table 4. Average Number of Gamma - ray Photons, Emitted per Fission in each Energy Group.

Group No.	Energy Band (MeV)	Prompt Fission Gamma (No. per Fission)	Decay Gamma (No. per Fission)	Total (No. per Fission)
8	8 - 14	1.008E-3	9.494E-4	0.00196
9	6 - 8	8.465E-3	8.140E-3	0.0166
10	4 - 6	7.489E-2	7.346E-2	0.1483
11	3 - 4	1.667E-1	1.656E-1	0.3323
12	2.5 - 3	1.821E-1	1.821E-1	0.3642
13	2 - 2.5	3.140E-1	3.156E-1	0.6296
14	1.5 - 2	5.415E-1	5.470E-1	1.0885
15	1 - 1.5	1.129E0	9.481E-1	2.771
16	0.7 - 1	1.352E0	8.762E-1	2.2282
17	0.45 - 0.7	1.679E0	9.870E-1	2.666
18	0.3 - 0.45	1.042E0	7.363E-1	1.7783
19	0.15 - 0.3	1.042E0	8.648E-1	1.9104
20	0.1 - 0.15	3.473E-1	3.229E-1	0.6702
21 - 25	0 - 0.1	Neglected	Neglected	Neglected

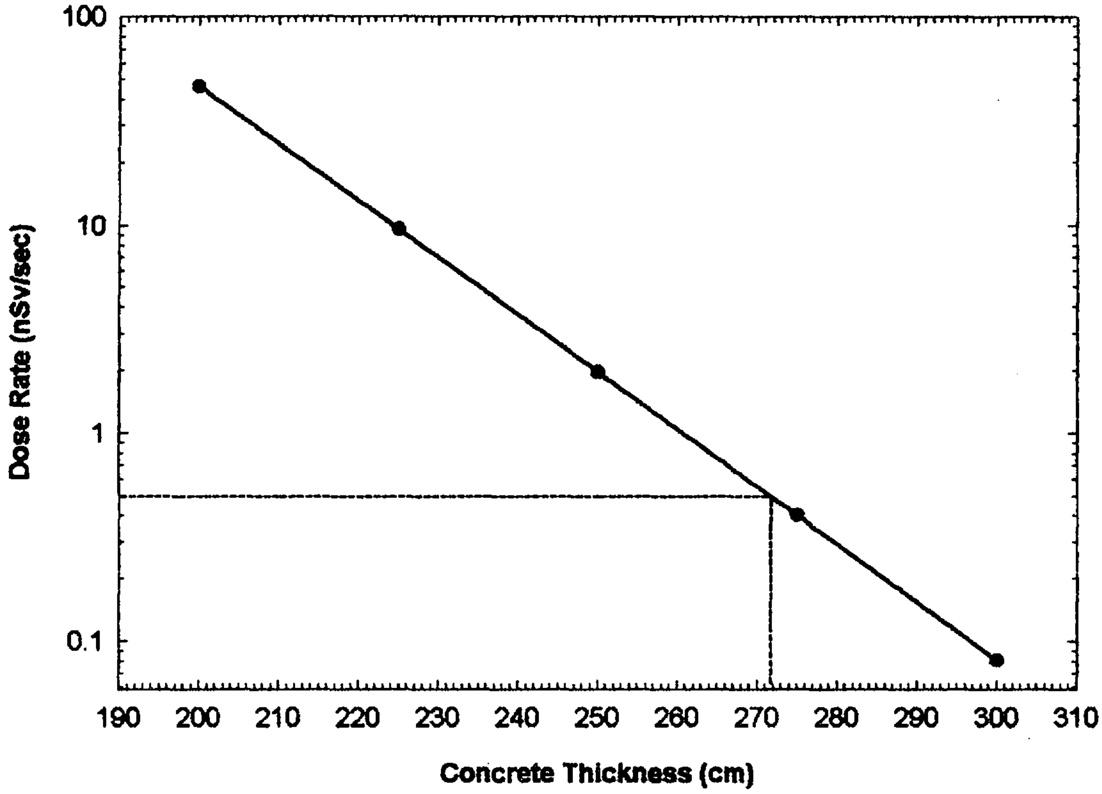


Fig. 2. Equivalent Dose Rate, Behind Different Concrete Shield Thicknesses at 50 MW Power.

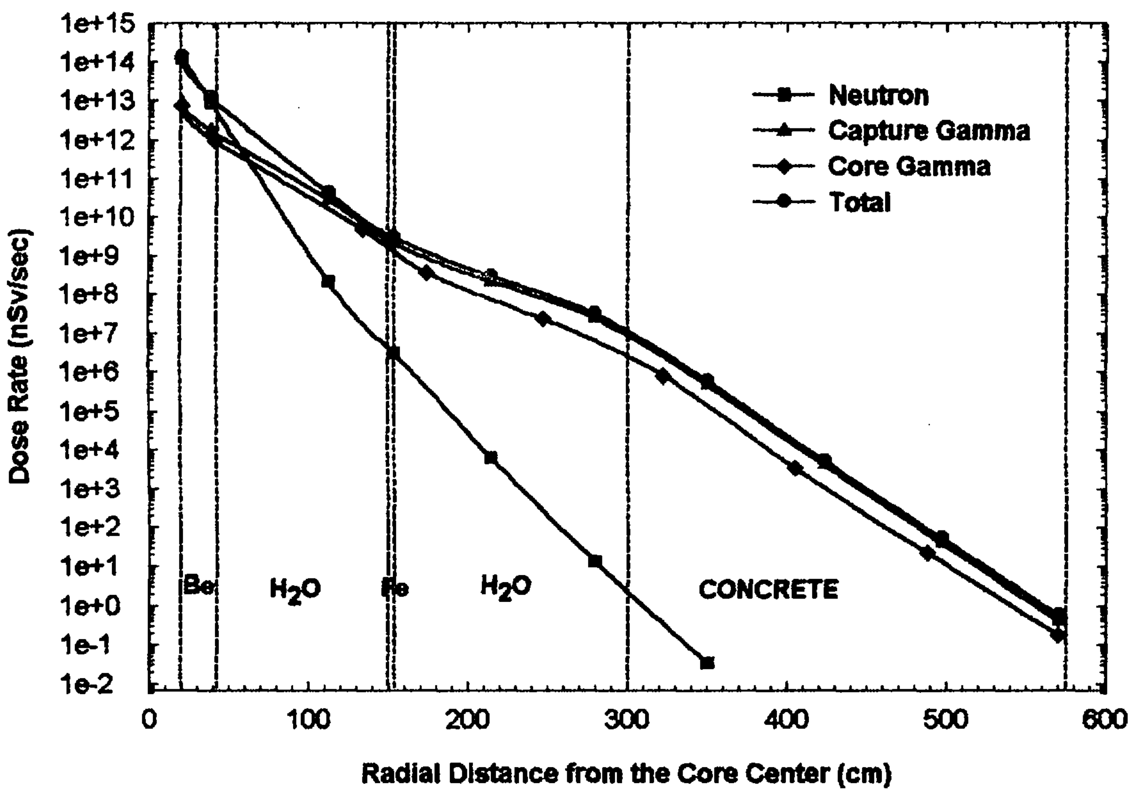
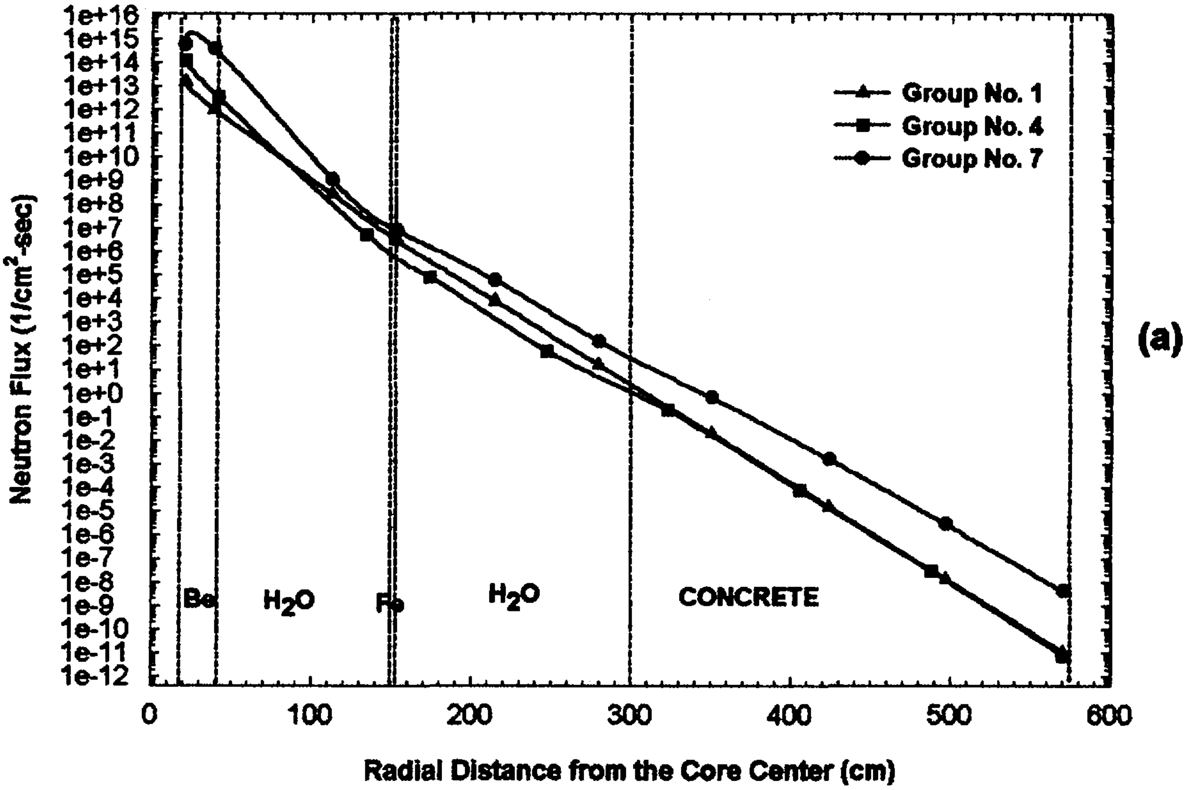


Fig. 3. Radial Dose Distribution Inside the Shield of the Reactor at 50 MW Power.



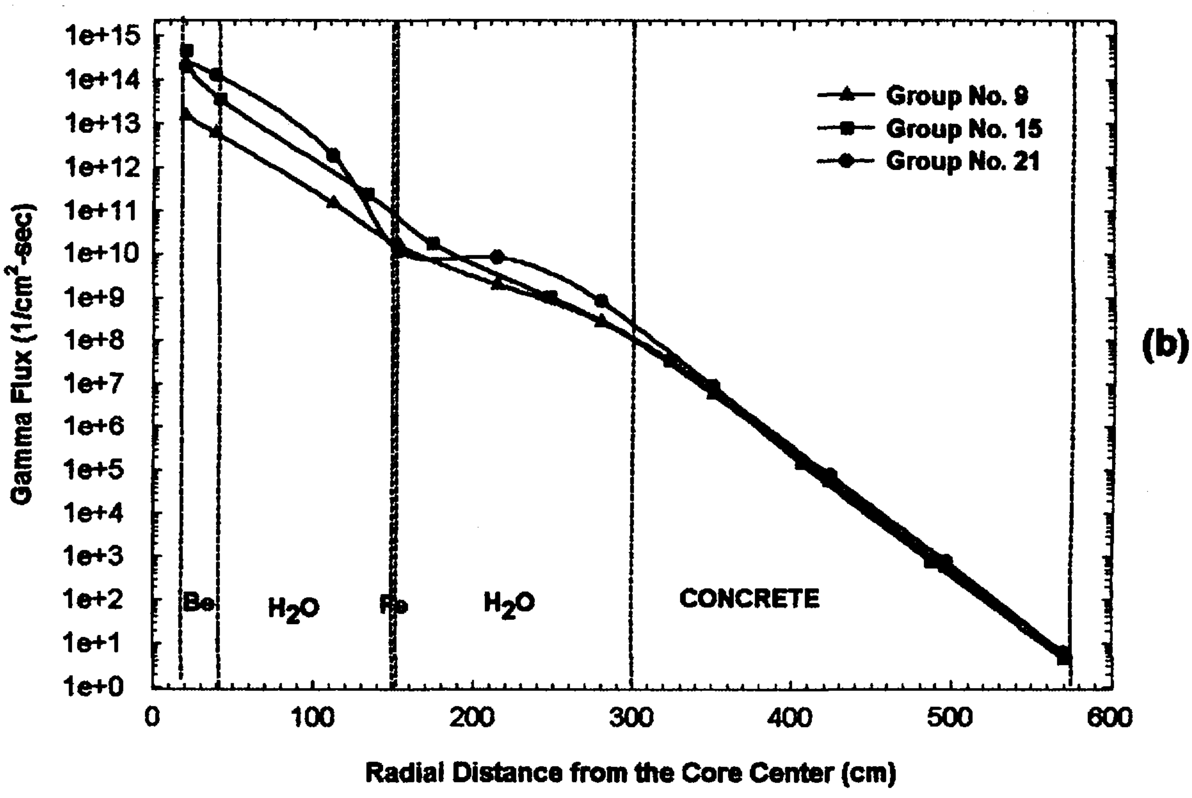


Fig. 4. Group Flux Distribution in the Reactor Shield for 50 MW Thermal Power.
a: Neutron Flux
b: Gamma Flux

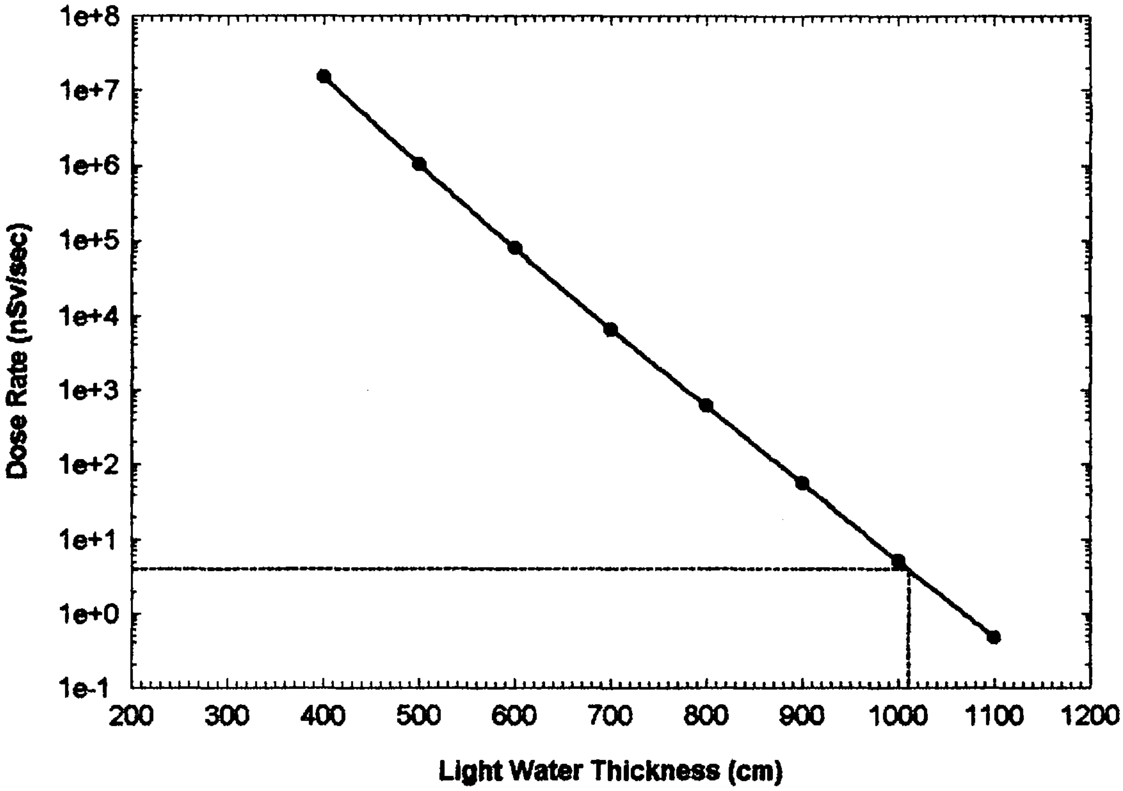


Fig. 5. Equivalent Dose Rate, after Various Thicknesses of Light Water at 50 MW Power.

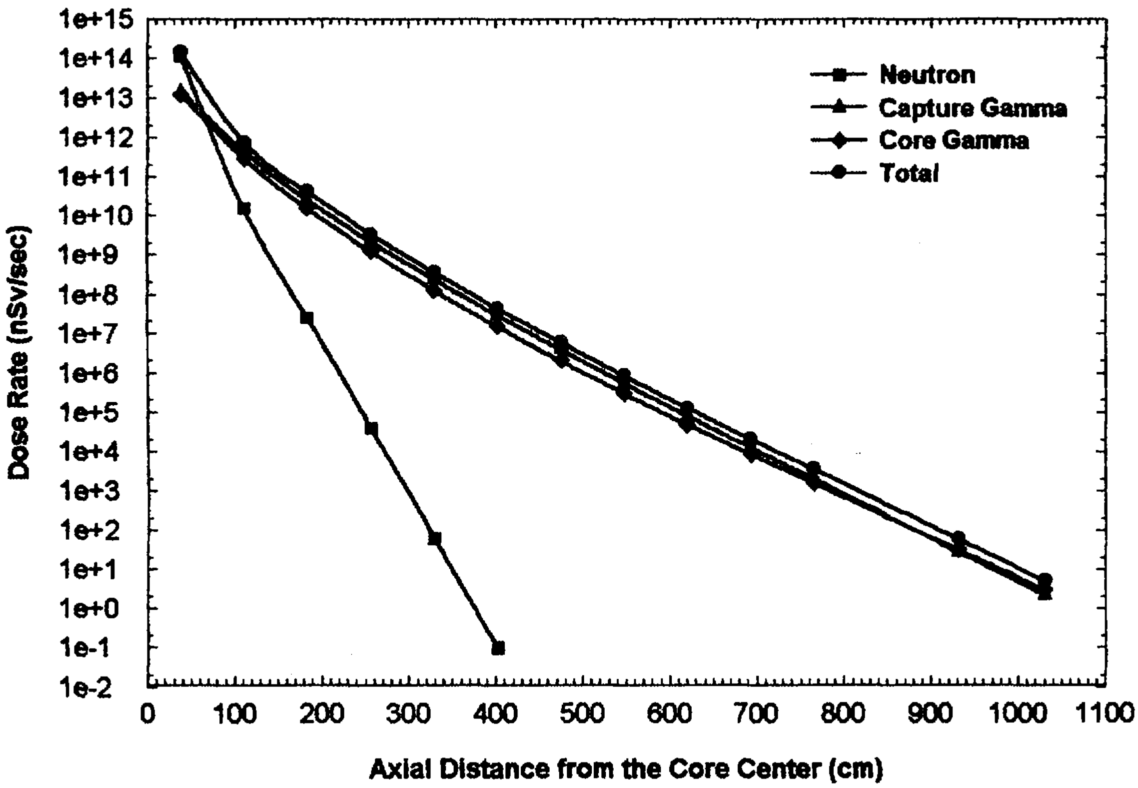


Fig. 6. Axial Dose Distribution Inside the Light Water Shield at 50 MW Power.

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