Analytical and statistical calculation of gamma dose rate for the accident of losing the shield for Tehran Research Reactor

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Abstract In this paper we study the analytical and statistical results of estimating the gamma dose rate at pool access floor in TRR when the core shield accidentally decreases to some non-permitted levels. Due to the risk of experimental techniques, we use the analytical and statistical methods. In normal conditions (no risk), the discrepancies between experiment and two methods are justified and it is found that for such problems we have to normalize these methods to experimental results as follows: the analytical method by factor 0.13 and MCNP by 1.7.

Key words TRR, gamma-ray, dose rate, MCNP4C

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1 Introduction

In today's world, nuclear facilities play important roles in developing science and technology, and obviously special attention should be paid to ensure that nuclear reactors remain safe to protect the stuff and people from radiation. Therefore predicting the accident effects is the most important problem that has to be considered. In this paper, the accident of non-sufficient water shield for the Tehran Research Reactor (TRR) is studied. The Tehran Research Reactor is an open pool MTR type light water reactor operated with a thermal power of 5 MW. The TRR was originally loaded with high enriched uranium (HEU) and now has been converted to low enriched uranium fuel (LEU) which is U_3O_8-Al with 20% enriched uranium. The TRR core configuration has been shown in Fig. 1. The water level at normal operation is 730 cm over the core surface, which is sufficient to shield the radiation generated by the core. In the normal condition the equivalent dose rate is almost 15 mrem/h[1, 2]. Due to the risk of experimental test of losing the biological shield (water), one can see the importance of such calculation for these accident outcomes. On the other hand as we will see it needs to normalize the MCNP results for this kind of problems. Therefore in this paper we study the accident in which the biological shield (water) will be decreased to the level less than TRR safety limit.

To this end, we use both the deterministic and statistic techniques considering the advantages and disadvantages of each technique. The deterministic methods such as analytical solution can give exact result but are not convenient for the problems with complex geometries such as the core and all its components and complexities. Hence when we calculate analytically, due to some simplicity which we have to make, it is possible to increase the uncertainty in the results. The statistical techniques can handle the problems with complex geometries but the results are based on probabilities and therefore the uncertainties[3]. We have to emphasize that, in the Monte Carlo calculations, the meaning of precision of the results is different from the meaning of accuracy. Monte Carlo may calculate a very precise result, which can be far from the physical truth[4]. Regarding these reasons, in this paper due to the problem nature we attempt to find out the normalization factor for both methods. In Sec. 2, we introduce the analytical method. In Sec. 3, the results of our analytical calculations are compared with MCNP4C as the sta-

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tistical technique. In Sec. 4 we discuss and justify the results by presenting the normalization factors in the observed dose rate at pool for normal operation. In Sec. 5 we present the conclusion.

9	IR BOX	graphite reflector	graphite reflector	graphite reflector	IR BOX	graphite reflector
8	SFE	CFE	SFE	CFE	SFE	SFE
7	SFE	SFE	SFE	SFE	SFE	SFE
6	SFE	CFE	SFE	IR BOX	CFE	SFE
5	SFE	SFE	CFE	SFE	SFE	SFE
4	IR BOX	SFE	SFE	SFE	SFE	IR BOX
3	graphite reflector		SFE	SFE	IR BOX	graphite reflector
2	graphite reflector	graphite reflector	graphite reflector	graphite reflector	graphite reflector	graphite reflector
1	graphite reflector		graphite reflector	graphite reflector	graphite reflector	graphite reflector
	A	B	C	D	E	F

(a)

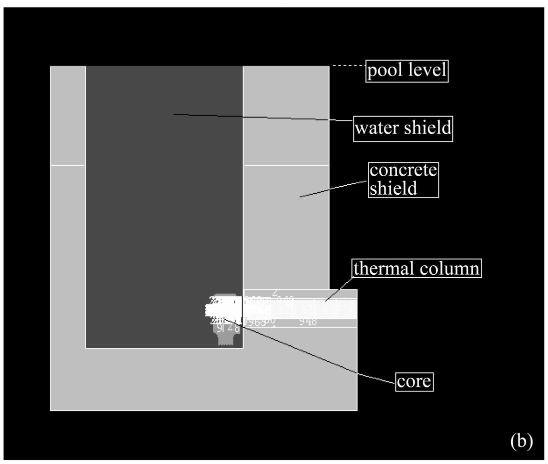

Fig. 1. (a) shows the core configuration of TRR; (b) shows the horizontal cross section of one SFE plotted by MCNP.

2 The calculation method

The calculation for more simplicity is based on four groups of gamma energy. We take the core as a homogenous spherical geometry in radius of R, which can be found from $\left(R=\sqrt[3]{\dfrac{3V_{\text{core}}}{4\pi}}\right)$ in which V_{core} is the volume of TRR core. Now we introduce the calculation method as follows.

Regarding the Point Kernel method, we calculate the gamma flux, $\varphi_R(E)$ at the core surface and then take the core as an isotropic point source. Including the Taylor's Buildup factor expression, the flux at distance "t" from the core surface surrounded by the shield of thickness "t'" can be derived as below:

$$\varphi_t(E) = \frac{Sj(E)R^2}{2\mu_{\text{c}}(E)(R+t)^2}\left(1-\mathrm{e}^{-2\mu_{\text{c}}(E)R}\right)\times \left[A\mathrm{e}^{-(\alpha_1+1)\mu(E)t'}+(1-A)\mathrm{e}^{-(\alpha_2+1)\mu(E)t'}\right], \quad (1)$$

in which, "α_1", "α_2" and "A" are the Taylor's Buildup factor parameters[5], "$\mu(E)$" is the shield attenuation coefficient, "$\mu_{\text{c}}(E)$" is the total attenuation coefficient of core, "S" is the fission rate density and "$j(E)$" represents the contribution of both the prompt and the decay γ-rays of each group of energy per fission.

Finally, the Flux to Dose Rate conversion factors will be used to give the gamma dose rate[6].

Now we include the main gamma sources in our calculation. When adding to the prompt and decay γ-ray, we have to take into account the inelastic neutron scattering and radiative captures of thermal neutrons too. Because of the importance of gamma generated from radiative capture in the core materials, we ignore the gamma generated by neutron scattering[7]. We can find the thermal neutron flux, φ_{th} as $\varphi_{\text{th}} =$ (fission rate density)/$(N\sigma_{\text{f}})$ and so the number of gamma produced by neutron capture in the i-th material which is expressed as below:

$$S^i_{\text{cap.}}(E)=\frac{\sqrt{\pi}}{2}\varphi_{\text{th}}N^i\sigma^i_{\text{cap.}}f^i(E)\ , \quad (2)$$

where, $f^i(E)$ is the number of gammas produced per one capture in the i-th material[7] and $N^i\sigma^i_{\text{cap.}}$ represents the macroscopic capture cross-section for the i-th material. To modify the source in Eq. (1), we replace $Sj(E)$ with $Sj(E)+\sum\limits_1^i S^i_{\text{cap.}}(E)$ to include the γ-rays produced by neutron capture. Here we consider hydrogen, uranium and aluminum as the materials in which thermal neutron radiative capture will occur. As we will see in Table 2, $S_{\text{cap.}}$ has effective contribution especially for energies more than 5 MeV.

3 The results

Since the reactor power is 5 MW and the core volume, V = 107525.28 cm^3 so, R = 29.49 cm, the fission density will be equal to $1.47\times10^{12}\ \dfrac{\text{fission}}{\text{cm}^3}$ and the thermal neutron flux will be $4.6\times10^{13}\ \dfrac{\text{neutron}}{\text{s}\cdot\text{cm}^2}$, which acceptably are near to the experimental measurements. To calculate the gamma attenuation in the core and shield we use the Table 1 in which the data are calculated according to the weigh fraction and mass attenuation, $\dfrac{\mu}{\rho}$ of the materials for each group of energies[5].

Table 1. Linear attenuation coefficient of the main materials in core and the total attenuation coefficient of the core contributing for 4 groups of energy.

energy/ MeV	μ_{water}/ cm^{-1}	$\mu_{aluminum}$/ cm^{-1}	$\mu_{aluminum}$ (in fuel)/cm^{-1}	$\mu_{uranium}$/ cm^{-1}	$\mu_{graphite}$/ cm^{-1}	μ_{oxygen}/ cm^{-1}	total μ_C/ cm^{-1}
weight fraction	0.3018	0.1866	0.0538	0.1259	0.3091	0.0228	
1	0.0706	0.1655	0.076	0.2187	0.0953	0.0334	0.114
3	0.0396	0.0953	0.0438	0.1262	0.0534	0.0188	0.0649
5	0.0303	0.0767	0.0352	0.1285	0.0407	0.0146	0.0544
7	0.025	0.06890	0.03164	0.1334	0.0232	0.0126	0.0464

Table 2. Number of gamma generated per (s·cm^{-3}).

energy/ MeV	j	prompt & delayed ($S(E)\times j$)	photos from neutron capture in H, U and Al	total	delayed gamma fraction
0—1	8.4	1.2×10^{13}	6.19×10^{12}	1.8×10^{13}	1.3
1—3	3.3	4.85×10^{12}	7.58×10^{12}	1.24×10^{13}	1.2
3—5	0.4	5.9×10^{11}	1.16×10^{12}	1.75×10^{12}	1.2
5—7	0.046	6.8×10^{10}	2.68×10^{11}	3.36×10^{11}	1.1

In Table 2, the strength source of gamma generated from the core is divided to four groups of energy for the prompt, delayed and those from neutron capture.

Figures 2—5 show the variations of γ-ray dose rate at pool surface in terms of water height over the core surface for energies 0—1 MeV, 1—3 MeV, 3—5 MeV and 5—7 MeV respectively.

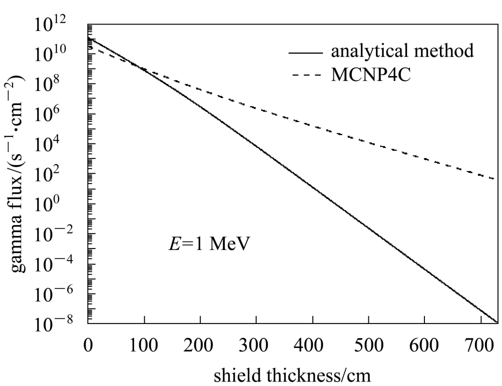

Fig. 2. Flux of γ-ray with energy 0—1 MeV at pool level in terms of water shield over the core.

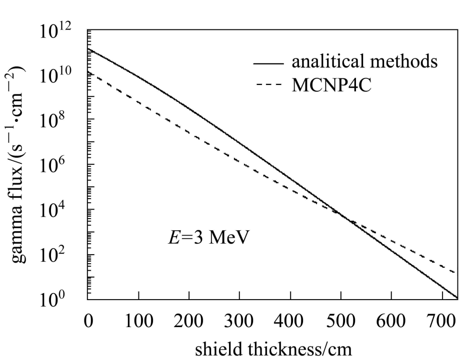

Fig. 3. Flux of γ-ray with energy 1—3 MeV at pool level in terms of water shield over the core.

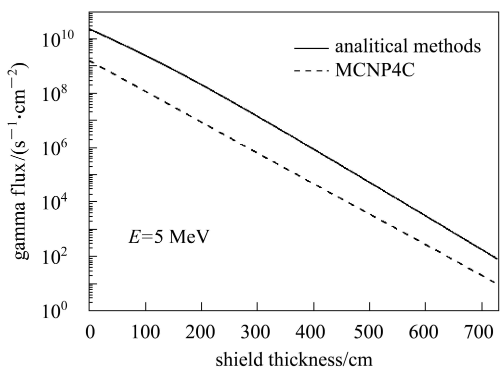

Fig. 4. Flux of γ-ray with energy 3—5 MeV at pool level in terms of water shield over the core.

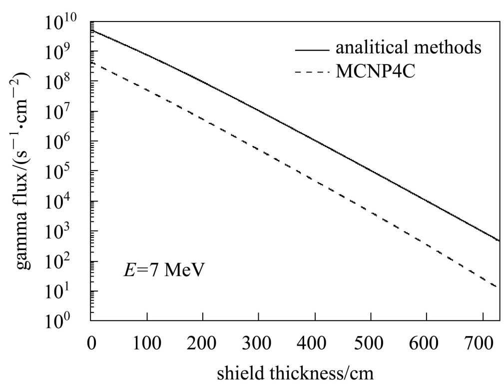

Fig. 5. Flux of γ-ray with energy 5—7 MeV at pool level in terms of water shield over the core.

The solid line represents the results of the analytical calculation and the dashed line shows the MCNP out-put according to the core configuration with 24 SFE and 5 CFE and the reactor structure shown in Fig. 1. Here we emphasize that MCNP4C does not generate delayed γ-rays from fission products[4]. To reprieve this problem in MCNP we have multiplied the flux of each group by the factor listed in Table 2. The MCNP results are in one σ level of confidence and the errors are less than 0.06[4, 8].

Figure 6 shows the total gamma dose rate at pool access floor in terms of water height over the core surface. The solid line represents the results of the analytical calculation and the dashed line corresponds to the MCNP. Despite of the dissimilarity of flux curvatures in Figs. 2—5, in this figure both lines have the same curvature so one can find out the normalization factor for both techniques easily. From this figure one can see the prediction of MCNP for dose rate at normal operation (730 cm water level) is 0.29 mrem/h and that the other technique is 3.85 mrem/h. To determine this normalization factor we have to compare the dose rate at normal condition which can be found in Fig. 6 for the shield thickness at 730 cm with the experimental measurement presented in the next section.

4 Discussion

It is emphasized that both results are obtained by assuming no impurities or active elements in water, but in fact such materials would be produced through reactions (n,α) or (n,γ) in water, therefore in the real condition there are some factors which strongly contribute to the gamma dose rate. As the most important factors, one can consider the impurities like aluminum which come from the core materials[1] and several other gamma emitters such as ^{24}Na, ^{56}Mn, ^{41}Ar and ^{16}N and so on.

Now we analyze the dose rate measured at the pool level to find out these extra contributions. For instance we estimate the contribution of ^{24}Na, ^{56}Mn and ^{41}Ar as the most important isotopes due to their longer half-life (15 h, 2.57 h and 109 min respectively). According to the quality measurements of TRR pool water[9], when the dose rate at the pool level is approximately 12.2 mrem/h, the maximum activation of ^{24}Na is 115 Bq/cm^3, of ^{56}Mn is 3.3 Bq/cm^3 and that of ^{41}Ar is 2.3 Bq/cm^3 and some small contributions for the rest of the elements, about 2.5 Bq/cm^3. After performing the calculation, the contributions of dose rate approximately will be 11 mrem/h, 0.3 mrem/h, 0.2 mrem/h and 0.2 mrem/h respectively. Due to this fact that increasing the power of reactor causes the increasing of the active elements and their radiation too, we are not able to determine the absolute value of core dose rate because we can never get rid of these elements during the reactor power increasing from 0 MW to 5 MW. Estimation shows that the core dose rate vale is almost 0.5 mrem/h. The other reasons which can raise the uncertainty in absolute core dose rate measurement are the irradiation samples inserted in core for experimental purposes and also the scattered gamma from the reactor hall.

From Fig. 6 and the discussion mentioned before, one can determine that the results of analytical method have to be multiplied by a factor of 0.13 (one order larger than experiment) and the MCNP by a factor of 1.7 to be normalized to the experiment.

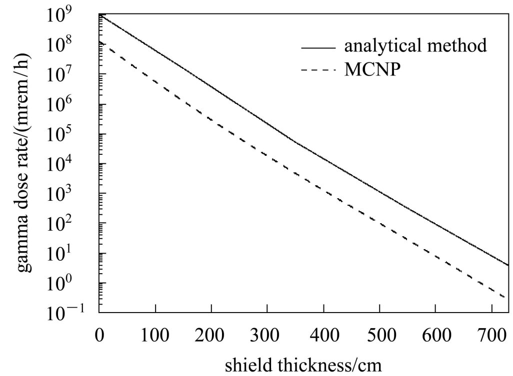

Fig. 6. Gamma dose rate from core, at pool level in term of water height over the core.

In the case of accident, decreasing the water level will increase the core contribution rapidly while the other factors approximately will lose their importance, as shown in Fig. 7. Fig. 7, shows MCNP (solid line), the total dose rate of ^{24}Na, ^{56}Mn and ^{41}Ar (dashed line) and the summation of them (dotted line) in term of water height over the core. It is realized that the effect of impurities for water level less than 550 cm is too small that to be considered but at normal condition, when the water level is 730 cm the importance of the elements mentioned before is quite obvious.

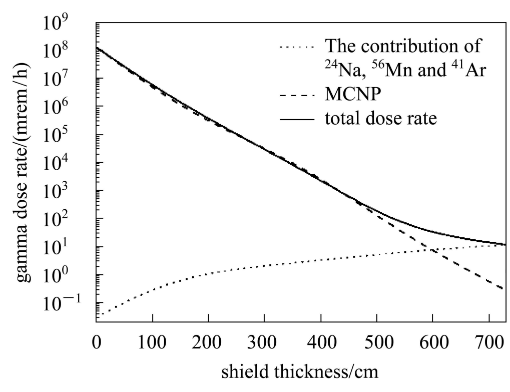

Fig. 7. The summation of core contribution and three main radioisotopes contributions of gamma dose rate at pool surface.

5 Conclusion

As one of the major accidents that may happen in the pool type nuclear reactors is losing the core shield, in this paper we tried to predict the effects of such accident by both analytical and statistical techniques.

We saw that in normal condition with the maximum water level in pool, subtracting the corresponding values of radioisotopes, the results of analytical method have to be multiplied by a factor of 0.13 and the MCNP by a factor of 1.7 to be normalized to the experiment. The normalization factor we have found is comparable with what has been discussed in Ref. [10].

Referring to the previous chapter, the dose rate observed at pool surface mostly (~90%) belongs to the gamma emitted from ^{24}Na and as a smaller contribution to other radioisotopes. It gives an overview of the importance of radioisotopes and impurities to contribute the gamma dose rate in normal condition which use some techniques they have to be reduced.

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