Shielding Parameter for 100 MeV Proton Accelerator

Y.S. Min, K.J. Mun, J.S. Cho, J.Y. Kim
Korea Atomic Energy Research Institute, 1045 Daedeokdaero, YuSeong, DaeJeon, Korea
ysmin@kaeri.re.kr

1. Introduction

Shielding of an accelerator requires particular attention, either due to the high intensities required for industrial or research applications, because it is necessary to avoid an unexpected exposure to the public or a worker. The radiation field for the shielding requirement is mainly due to the secondary neutrons produced by the interaction of the proton beam with the target material at a target room or with structures of accelerator.

This study aims at an analytical method for calculating neutron bulk shielding in 100 MeV proton accelerator facility on the basis of the simplified Moyer model [1]. The dose attenuation length is the most important parameter as well as a source term in Moyer model. These parameters are obtained using the MCNPX (version 2.6.0) three dimensional Monte Carlo code [2]. It is extremely useful to have available phenomenological shielding models, such as the Moyer model, which may be used as a rough check on the results of calculations made by more sophisticated techniques. Two shielding materials which are water and concrete are considered in this study.

2. Method (Calculation Models)

The analytical model used to estimate the neutron dose equivalent rate is a simplified Moyer model for 100 MeV proton beam. The model has the following functional form:

$$H(d,\theta) = \frac{1}{r^2} H_\theta e^{-d(\theta)/\lambda} \quad (1)$$

Where H(d, θ) is the neutron dose equivalent rate, Hθ is the source term, r is the distance from the source to the interested point of outside shield, d(θ) is the path length through the shield and λ is the attenuation length for the shielding material.

A mono-energetic and mono-directional proton impinging on rectangular target is produced secondary neutrons which have from thermal energy to proton beam energy.

A dose rate is attenuated according to the thickness of each shielding material. The dose attenuation length (λ) for each shielding material, and the energy & angular distribution of neutrons (the source term H_θ) were calculated by MCNPX.

The target is rectangular type in the code modeling. The area of the targets is 30 by 30 cm, equal to beam size required on user demand and those thickness is water 15 cm, tungsten 2 cm and silicon 10 cm. Calculation models show in fig. 1. Table 1 lists the target parameters used in the calculation.

Table 1 Target Dimensions used in the Calculation

Target	Proton energy	Dimension (cm)	Range (mm)
Water	100 MeV	30 x 30 x 15	75.9
Tungsten	100 MeV	30 x 30 x 2	8.09
Silicon	100 MeV	30 x 30 x 10	39.7

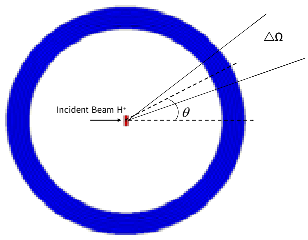

Fig. 1 Calculation Model in MCNPX

3. Neutron Source: energy and angular distributions

The calculation was carried out with MCNPX using a geometry in which one-directional proton beam impinges at the center of the target of 300 cm radius spherical space surrounded by 70 cm thick spherical concrete or water shielding wall.

The energy and angular distributions of neutrons produced by 100 MeV proton impinging on three target materials which are water (1 g/cc), tungsten (19.2 g/cc) or silicon (2.32 g/cc), were calculated by MCNPX and is shown in Fig 2~4. The energy bin (267 groups) of neutron shows in Table 2.

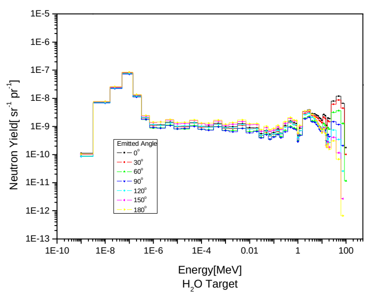

Fig. 2 Neutron Yield per sr for 100 MeV protons impinging on H_2O target

More than 51 % of neutrons produced by 100 MeV protons impinging on the target are emitted between 0 to 60°. The neutron yields below several MeV are almost constant according to their emitted angle. The statistical errors associated to the distribution are below 12 % in the angular interval 0 ~ 90 ° and below 45 % in 90 ~ 180 °.

For forward angles the neutron build-up region is visible, particularly at the high energies.

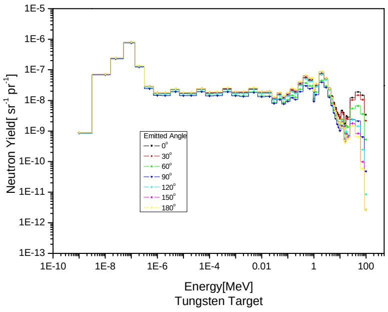

Fig. 3 Neutron Yield per sr for 100 MeV protons impinging on W target

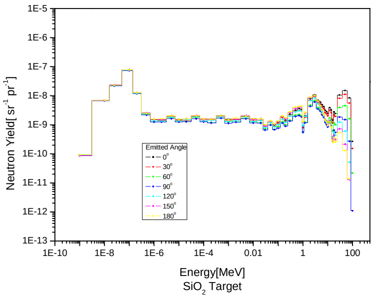

Fig. 4 Neutron Yield per sr for 100 MeV protons impinging on SiO_2 target

Table 2 Energy Structure for MCNPX tally

Upper Energy Limit			
1.0E-09	2.0E-04	3.0E-01	9.0E+00
1.0E-08	5.0E-04	5.0E-01	1.0E+01
2.5E-08	1.0E-03	7.0E-01	1.2E+01
1.0E-07	2.0E-03	9.0E-01	1.4E+01
2.0E-07	5.0E-03	1.0E+00	1.5E+01
5.0E-07	1.0E-02	1.2E+00	1.6E+01
1.0E-06	2.0E-02	2.0E+00	1.8E+01
2.0E-06	3.0E-02	3.0E+00	2.0E+01
5.0E-06	5.0E-02	4.0E+00	3.0E+01
1.0E-05	7.0E-02	5.0E+00	5.0E+01
2.0E-05	1.0E-01	6.0E+00	7.5E+01
5.0E-05	1.5E-01	7.0E+00	1.0E+02
1.0E-04	2.0E-01	8.0E+00	

4. Dose Attenuation Length

The neutron attenuation in water or ordinary concrete was computed with MCNPX for spherical thickness from 0 to 70 cm. Computed particle fluence is scored by means of crossing the boundary every 10 cm in the shield. The neutron dose equivalent is then obtained by multiplying the calculated neutron spectrum with the fluence-to-ambient dose conversion factor as given by ICRP 74[3].

The composition and density of ordinary concrete adopted as a shielding material in this study is shown in Table 3.

Table 3 Typical Compositions of Ordinary Concrete

Element	Density: 2.31g/cm3 Weight Fraction (w/o)
Hydrogen	1
Carbon	0.1
Oxygen	52.9
Sodium	1.6
Magnesium	0.2
Aluminium	3.39
Silicon	33.7
Potassium	1.3
Calcium	4.4
Iron	1.4

Table 4 shows, for three targets, dose attenuation length according to the shield materials. Even though same shield material, the attenuation is slightly different because the energy spectrum of neutron produced by proton is different according to a target.

Table 4 Dose Attenuation Length (g/cm^3)

Target / Shield	Water	Tungsten	Silicon
Water	17.7	13.2	15.4
Concrete	46.8	40.3	42.7

7. Future Work

For different shield materials with a shield depth, we plan to investigate the source terms and attenuation lengths as a function of the energy and emission angle, and the shield thickness. These values can be used to design the shielding of proton accelerators in a realistic geometry.

Acknowledgement

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REFERENCES

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[3] "Conversion Coefficients for use in Radiological Protection against External Radiation," ICRP Publication 74, volume 26 No.2/4 1996.