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Development of a simple engineering design tool for neutron flux calculation in radiation tube

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1. Abstract

A simple and a user-friendly engineering tool for evaluating the neutron beam flux is presented. The flux of neutrons, emitted from semi-infinite plane source, is calculated by means of the diffusion equation assuming ballistic motion. The Engineering tool enable to examine and to understand the influence of the neutron beam geometry and its neutron absorbing layers on the overall neutron flux.

Keywords: Radiation tube, Neutron beam, Neutron flux, Scientific Instruments

2. Introduction

Execution of scientific research using neutron flux is a major objective in research reactors. Essentially, transport of neutron flux from the core towards experimental facility in the reactor hall is carried out by radiation tubes.

The general structure of radiation tubes in pool type reactors consist of several basic components: a thimble, a main structural body and a shutter. The thimble is the wetted part of the radiation tube and is immersed in the reactor pool. Its main function is to transfers neutron flux from the core to the main structural body of the tube, hence it is designed for high neutron transport efficiency. The radiation tube main structural body is fixed within the pool biological shield and provides the mechanical strength to the tube. This part consists of neutron optical components, e.g., collimators and filters, for neutron beam manipulation and held at low absorbing atmosphere for high neutron transfer efficiency. The shutter function is to enable or to block the transport of flux from the radiation tube to the experimental facility outside the tube and hence it has both operational and safety functions. The shutter consists of materials with high absorption cross-section for thermal neutrons for neutron flux blocking and also has a specific and defined area for neutron transport. Therefore, in its general scheme, a beam tube consists of several layers of components, materials and atmospheres which their presence may decrease the required neutron flux for the scientific instruments, since they behave as neutron absorbers. Hence, it is required to understand their influence of each layer on the neutron flux.

Usually, neutron fluxes are being calculated by expert numerical software based on Monte-Carlo methods. Although those software calculations results are quite reliable, the demand expertise and computational resources. In order to shorten the examination time and to prevent the usage of an expert software, a simplified physical model was written, as an engineering design tool. The model can examine the changes of the neutron flux in the radiation tube according to design parameters. The model gets the initial neutron flux value and due to its absorbing materials and their absorption cross-section for thermal neutron, it calculates the flux value along the radiation tube. Parameters as structural materials, atmosphere flow materials, geometries and absorbing materials can be examined.

3. Model description

As a rule, the intensity of neutron source can be measured due to its neutron emission rate. A spatial source intensity defines as neutron emission rate to material quantity, and has units of ($n/s/cm^2$). In the developed model, the neutron source has a semi-infinite plane geometry which located in the entry of a cylindrical pipe, hence it defined as a disc geometry. The source is assembled from infinitesimal areal cells (dA), and every cell functions as a specific neutron source with intensity:

(1) $$I_0 = I_A dA$$

when I_A is specific intensity of the areal source under infinite isotropic assumption. The contribution of each specific source to the flux density can be calculated by using diffusion equation (2):

(2) $$\phi = \frac{I_0}{\pi 4 l^2} e^{-\Sigma x_i \mu_i}$$

When l is the distance between the source and the detector (the detector indicates the spatial location in the radiation tube volume which the overall flux is being calculated for), X_i is the total thickness of absorber i (if couple of absorber layers from the same material exist, the total thickness is the algebraic sum of each thickness layer) and μ_i is the reduction factor of the material which can be calculated from the macroscopic absorbing properties:

(3) $$\mu_i = \sigma_i \eta_i = \sigma_i \frac{\rho_i N_A}{\mu_i}$$

When σ_i is the total absorption cross section for neutron removal and ρ_i is the density of each i material.

Hence, the overall neutron flux in the detector, which is located in spatial coordinate (z,d) in radiation tube volume (d is the distance from the symmetry axis z) will be the sum of every flux infinitesimal contribution from all the sources in the disc shape source, where every source has area of dA =drdθ, as can be shown in Figure 1.

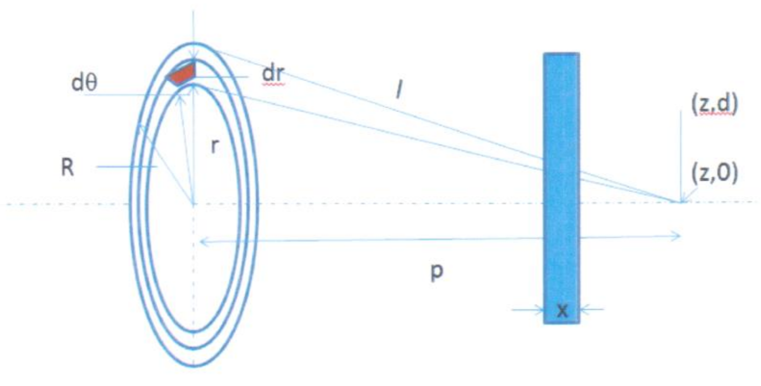

Figure 1: Plain (z,d) which crosses the radiation tube

According to eq. (2) and (3), the neutron flux at any location in the radiation tube will be:

(4) $$\phi_{total}(z,d) = \iint \frac{I_0}{\pi 4 l^2} e^{-\Sigma x_i \mu_i} r d\theta dr$$

A parametric numerical examination of eq. (4) (geometrical dimensions, different absorbers, source intensities) was done with specially developed OCTAV [2] open source code. In order to examine parameters in easy way, a user-friendly interface was also added. The results of the calculation describe the total radiation tube flux in plain (z,d) which divides the radiation tube, so its symmetry axis is in the plain, and its radius in orthogonal to the plain, as can be seen in Figure 2.

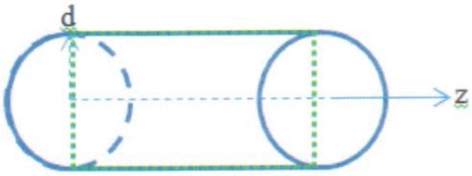

Figure 2: description of (z,d) plain

According to Figure 2, the boundary condition on d axis is the neutron flux entry rate, and the model describes the neutron flux distribution along the radiation tube, as function of the distance z.

4. Model results

An example of the model results is shown in Figure 3. In this example, a cylindrical radiation tube based on 20 cm diameter and length of 500 cm was examined. The configuration consists of different layers such Aluminum, Stainless Steel, Helium and water, which function as neutron absorbers. The initial flux value in the radiation tube entry was 10^{11} (n/cm^3/s). The legend of Figure 3 had logarithmic scales which describes the neutron flux.

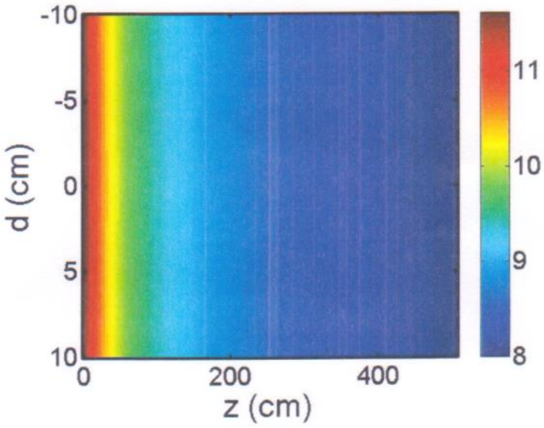

Figure 3: Neutron flux contribution on plain (z,d)

The entry flux value can be seen in Figure 3 within the red shades colors, and the calculation show that the neutron flux is 10^8 (as shown in deep blue colors) on the tube outlet, e.g., losses of 3 orders of neutron fluxes in transportation in the radiation tube. In this manner, tube geometrical parameters as diameter and length can be examines, as well as different absorbing materials and thicknesses, and their influences on the outlet neutron flux can be calculated.
Another calculation examines the flux distribution along the radiation tube, as shown in Figure 4. It can be seen that a flux distribution in the radial direction (d axis) in the entry of the radiation tube has is higher by factor 2 from highest value to the lowest value. The flux

distribution is becoming along the radiation tube (z axis), until it has almost equal distribution on the tube outlet.

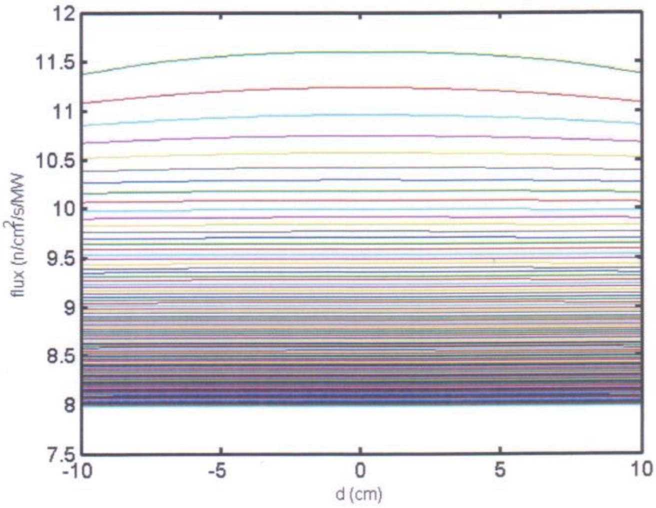

Figure 4: Flux distribution along d axis

5. Summery

This paper presents a simple and user-friendly engineering tool for parametric examination of neutron flux transport in radiation tube. The engineering tool based on numerical model which calculates the diffusion equation for neutrons which are emitted from areal source and progress in straight lines, without reflections from the tube structure. The model results describe the neutron flux behavior along the radiation tube and enable to understand the influence of each tube geometric parameters or each absorbing layer on the overall neutron flux.

6. References

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