Applied Radiation and Isotopes 135 (2018) 160–165

Contents lists available at ScienceDirect

Applied Radiation and Isotopes

journal homepage: www.elsevier.com/locate/apradiso

Neutron tomography simulation by MAVRIC/Monaco code

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HIGHLIGHTS

- MAVRIC/Monaco module is a very useful tool for neutron tomography.
- Beam hardening effect on gray level distribution and the reconstructed image was presented.
- MAVRIC/Monaco-simulated projection was adjusted for a correct tomographic reconstruction.
- FBP method was used for 3D image reconstruction.

ARTICLE INFO

Keywords:
Neutron transmission
Beam hardening
SCALE6.1
MAVRIC/Monaco
Mesh tally

ABSTRACT

In this study, MAVRIC/Monaco module integrated with SCALE6.1 package is used to elucidate experimental neutron transmission for various neutron-absorbing materials. Neutron beam hardening and its effect on shielding calculation and tomographic image reconstruction accuracy are presented.

The experimental data for studied materials are used for comparison purpose and to prove the capability of MAVRIC/Monaco code in neutron tomography simulation. The good agreement between experimental and calculated values confirms the capability of SCALE6.1 package in obtaining physical data exactly analogous to an experimental measurement.

1. Introduction

The Monte Carlo simulation of radiation transport within a material system, using appropriate calculation codes, allows a better understanding and interpretation of the neutron transmission process by considering secondary effects such as beam hardening (BH) and background. These calculation codes can give particle counts exactly analogous to experimental measurements (Looman et al., 2009; Hachouf et al., 2012, 2016). During neutron transmission through strong absorbing materials, the BH contribution becomes important and must be considered in theoretical calculations during simulations. This is very important especially for shielding calculations.

BH was originally discovered and studied for the case of X-ray attenuation through matter (Brooks and Di. Chiro, 1976). It can be defined as an increase in high-energy contribution in the beam transmitted through matter in some appropriate conditions and circumstances.

Accordingly, when a polychromatic neutron beam passes through a matter containing strong absorbing material, low-energy neutrons are more likely removed from the beam in the first thickness intervals in the absorber material. Then, the remaining beam becomes rich in high-energy neutrons. As a result, the rest of material cannot attenuate the neutrons as in the first thickness intervals of the sample (Bastürk, 2003).

Therefore, the BH effect is defined as the increase in the transmitted beam average energy with the increase in cross-thickness. This phenomenon is the main cause for the elevation in effective neutron transmission measured through strong absorbing materials (Hachouf et al., 2012, 2016; Zawisky et al., 2004a, 2004b). It leads to a nonlinear evolution of the neutron attenuation, as a function of the object thickness, which results in artifacts appearance in the reconstructed tomographic image. This makes the quantitative interpretation of the obtained images quite difficult and the distinguishing between two different materials impossible (Zawisky et al., 2004a, 2004b; Hachouf et al., 2012).

Strong absorbing materials are also used in the construction of nuclear reactors. In this case, the theoretical shielding calculations will be uncertain if the attenuation coefficient of one of the constituent materials is erroneous. Therefore, the Monte Carlo simulation is necessary for an appropriate estimation of the quantitative contribution of the neutron BH effect to the quality of reconstructed tomographic image, which can be done by the filtered back projection method.

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https://doi.org/10.1016/j.apradiso.2018.01.023
Received 13 August 2017; Received in revised form 13 January 2018; Accepted 14 January 2018

In this work, we study the SCALE6.1 package (Standard Computer Analyses for Licensing Evaluation; MAVRIC/Monaco module) capability in the analysis of experimental neutron transmission data and the determination of neutron attenuation characteristics of different absorbing materials. The effects of neutron BH on transmission data, shielding calculations, reconstruction accuracy, and interpretation of the tomographic images are illustrated. The reconstruction is carried out from the projection data generated using the MAVRIC/Monaco code embedded in the SCALE6.1 system.

The importance of using this code is the possibility of the 3D modeling of the studied material system. It allows making a good estimation of an experimental test in real conditions. Thus, the thermal neutron transmission data of Al, steel, and 304B7 materials are used to validate the simulation results and highlight the capability of the SCALE6.1 code.

2. SCALE6.1 code description

SCALE6.1 is a modular system code developed at Oak Ridge National Laboratory, Oak Ridge, Tennessee (SCALE, 2011). It contains approximately 76 modules, which perform a wide variety of calculations, such as cross-section preparation, eigenvalue calculations, reactor physics analysis, and criticality and safety treatments during radioprotection tests (Matijević et al., 2015a, 2015b; Cédric, 2011).

For Monte Carlo radiation transport analysis of a fixed source in 3D generalized geometry, Monaco/SCALE is an appropriate code. It was developed and integrated within the SCALE6.1 system for shielding calculations. It uses the SCALE6.1 Standard Composition Library and its Generalized Geometry Package (SGGP) that is used by KENO (Kirk, 2010).

In our present work, for neutron transmission characterization and image projection simulation, we used the V7-200n47g cross-section library (ENDF/B-VII 200, neutron group and 47 gamma group libraries) by considering the neutron groups alone.

The MAVRIC/SCALE (Monaco with Automated Variance Reduction using Importance Calculations) sequence was developed, for difficult shielding problems, to provide automated 3D variance reduction by creating a space/energy importance map and biased source distributions for Monaco module (Kirk, 2010). This sequence also generates multigroup cross-sections for working materials. Using these cross-sections, MAVRIC generates problem-dependent cross-sectional data (SCALE, 2011).

The user will also supply the geometry description using the SGGP; source description as functions of position, energy, and direction; tally descriptions (fluxes in each region, point detector, or mesh grid); and response functions (function of energy). The objective of the calculation is to provide flux and dose rates with low uncertainties within a reasonable time by using the response function 9031 (neutron dose), which is the most recent ANSI/ANS 6.1.1-1991 standard flux to-dose conversion factor expressed in (rem/h)/(particle/cm^2/s).

The output consists of tables that describe the region and point detector fluxes (and their responses) and files for mesh tallies (SCALE, 2011). The SCALE6.1 code allows specifying the energy distribution of the emitted particles by the introduction of “abscissa” and “truepdf” values that represent the energy and neutron intensity, respectively.

In our current calculation, the source is defined as a parallel beam of polychromatic thermal neutrons of Maxwellian distribution. The direction perpendicular to the detector plane is considered the propagation direction.

3. Methodology

3.1. Neutron transmission with SCALE6.1

After passing through an absorbing object, the beam intensity decays exponentially with attenuation coefficient and object thickness. It obeys the Beer-Lambert attenuation universal law, which is written in the form of integral as follows:

$$I(x,y) = I_0 \exp\left(-\int_l \mu(x,y)ds\right) \tag{1}$$

where $\mu(x, y)$ is the material linear attenuation coefficient at a point (x, y), which is also denoted by $\Sigma(x, y)$. ds is the differential element in path l. I_0 and I are the incident and the transmitted beam intensities, respectively.

To determine the neutron flux arriving at a point (x, y) in space, a point detector is the most suitable instrument. The neutron transmission quantitative value is the ratio of transmitted to incident flux values:

$$\mathrm{Tr} = \frac{\mathrm{I}}{\mathrm{I_0}} = \exp\left(-\int_l \mu(\mathrm{x},\mathrm{y})\mathrm{ds}\right) \tag{2}$$

The attenuation coefficient or effective macroscopic cross-section (Σ) is calculated from the neutron transmission value obtained from Eq. (2) using the following expression (Oda et al., 1996):

$$\Sigma = \frac{1}{d}\ln Tr \tag{3}$$

For the SCALE6.1 calculations of neutron transmission, the considered object is described by its density and its exact composition. In our work, we used a borated stainless steel 1.88 wt% B-nat with a mass density of 7.74 g/cm^3, assuming that the boron distribution is uniform in 0.173 × 2.5 × 2.5 cm^3 plates. The chemical composition of this material was introduced into weight percents alloys or mixtures card “WTPT”. The compositions of the two other materials studied in this work, steel and aluminum, whose mass densities are 7.94 and 2.702 g/cm^3, respectively, are already defined in the “standard basic composition” of the code. To study the neutron transmission as a function of plate thickness, a “cuboid,” geometry, which represents a plate shape, was modeled and used. The thickness increment (0.173 cm) was carried out by adding one plate each time.

3.2. Projection data with SCALE6.1

For the projection data generation, we considered a cylindrical phantom for our SCALE6.1 modeling. It consists of three incorporated cylinders as shown in Fig. 1. The inner cylinder is made of aluminum with a diameter of 1 cm and height of 2 cm.

The upper outer cylinder of the phantom is made of SS304 stainless steel, and the lower one is made of borated stainless steel containing 1.0 wt% B-nat (7.81 g/cm^3).

These two cylinders have an external diameter of 2 cm and a height of 1 cm. The cylindrical phantom was accurately described using the

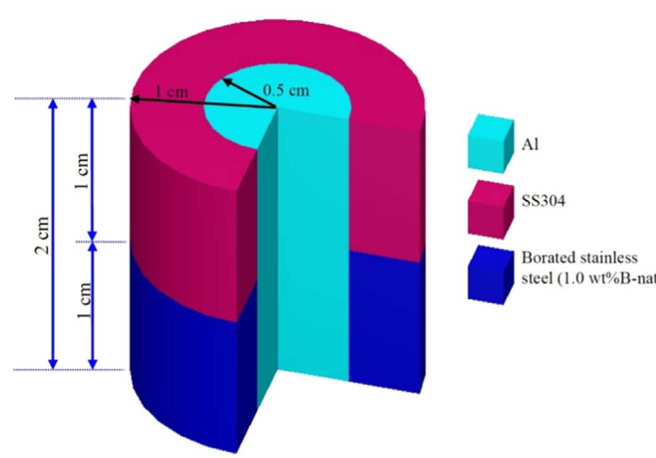

Fig. 1. 3D graphical geometry representation of the studied phantom.