Correlation of Wigner Energy and Neutron Flux Distribution in KRR-2 Graphite

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1. Introduction

As a result of neutron diffusion in the graphite in thermal column of KRR-2 research reactor Wigner energy and various nuclide contents have been shown a kind of very complicated distribution contour maps along with the distance from reactor core[1]. Characteristics of the irradiated graphite caused by neutron collision in the carbon structure are a function of neutron dose. Irradiated graphite dismantled from the thermal column in KRR-2 research reactor, therefore, has been shown to include somewhat degree of Wigner energy, specific radioactivity and the resulting radioactive chemicals/nuclides at different values based on the positions in the study of bulk mode of graphite structure[1].

The differences of residual energy(i.e., Wigner energy) and radioactive chemical property between the individual irradiated graphite in a lump of graphite stack in thermal column might be induced by a self-shielding effect[2,3] and/or by a diffusion ratio[4]. To estimate the neutron energy content affected in graphite moderator system Monte Carlo simulation skill (MCNP code (MCNP4C) and LAHET code) was well used as cited in literatures[2-4].

In this study an analytical thermal neutron diffusion model was used to evaluate the characteristics of the irradiated graphite due to the accumulated neutron energy doses in thermal column graphite. Comparison between an analytical neutron diffusion model and numerical simulation with MCNP code was developed. Variances of Wigner energy and the radioactivity as a function of distance from the reactor core were the major comparison data to correlate with the experimental data.

2. Analytical Diffusion Model

The equation representing the conservation of the neutrons, coming from Fick's law ($J_x = -D\frac{d\phi}{dx}$ in x -coordinate) is shown as following form[5]:

$$D\nabla^2\phi - \sum_a \phi + S = \frac{\partial n}{\partial t} \tag{1}$$

where

$D\nabla^2\phi$ = neutron leakage per unit volume per second

$-\sum_a \phi$ = number of neutrons absorbed in unit volume per second

$-S$ = source strength: the rate of neutron density in time

$-\frac{\partial n}{\partial t}$ = variation rate of the neutron density in unit volume.

When the system is in a steady state condition, the neutron density is constant in time($\frac{\partial n}{\partial t} = 0$). However, when there is no source, $S = 0$. Then, Eq. (1) becomes:

$$D\nabla^2\phi - \sum_a \phi = 0 \tag{2}$$

or takes the form of wave equation:

$$\nabla^2\phi - K^2\phi = 0 \tag{3}$$

where

$$K^2 = \sum_a / D \tag{4}$$

and has a dimension of [cm^{-2}].

2.1 Solution in infinite plane source

Assuming an infinite source in xy-plane, the neutron flux will be independent of x an y at any distance form the source, and the diffusion equation becomes:

$$d^2\phi / dz^2 - K^2\phi = 0 \tag{5}$$

with the boundary conditions:

a) neutron flux is finite except at z=0,

b) a plane source symmetrically emit neutron to both sides.

The general solution of Eq. (5) is as following:

$$\phi(z) = A\exp(-Kz) + B\exp(Kz) \tag{6}$$

where constants A and B are determined from the boundary conditions, as $B = 0$, and $A = \frac{1}{2}KD$. Substituting these values into Eq. (6) yields:

$$\phi(z) = A\exp(-Kz)/2KD \tag{7}$$

The diffusion length L of thermal neutrons:

$$L = 1/K = \sqrt{D/\sum_a} \tag{8}$$

The thermal neutron flux distribution becomes:

$$\phi(z) = \frac{1}{2KD}\exp(-z/L) \tag{9}$$

2.2 Effect in finite dimensions

If the neutrons emitted by an infinite plane source diffuse into a slab of infinite extent, within finite thickness, with boundary conditions available for infinite dimensions, the general solution of Eq. (6) is becomes the followings:

$$\phi_c = A\exp(-Kc) + B\exp(Kc) = 0 \qquad (10)$$

where c = finite dimension of the slab including the extrapolation distance.

$$B = -A\exp(-2Kc)$$

$$\phi(z) = A[\exp(-Kz) - \exp(-K(2c-z))] \qquad (11)$$

$$\phi(z) = \frac{\exp(-Kz) - \exp(-K(2c-z))}{2KD[1+\exp(-2Kc)]} \qquad (12)$$

Eq. (12) will be identical to Eq. (7) for a very high c value.

3. Materials and Results

An analytical and a numerical approach for one dimensional and two dimensional transient and/or steady state neutron flux was estimated for the irradiated graphite in thermal column of KRR-2 research reactor. Some typical values of Wigner energy contents and the radioactivity intensity of the graphite based on the position from the reactor core in thermal column were shown in Fig. 1 and Table 1.

Evaluation results from the model[2,5] derived from Fick's law was focused to the amount of Wigner energy value and the radioactivity intensity, as an imaginary amplitude of total neutron does to the graphite structure, were correlated well with the result from MCNP code.

4. Conclusion

Studies on the estimation for neutron flux distribution and on the comparison of imaginary neutron flux intensity with the amount of Wigner energy and the resultant radioactivity(and/or nuclides) were conducted for the irradiated graphite in thermal column of KRR-2 research reactor. Comparing and applying an analytical model and simulation code(MCNP) for neutron dose to the graphite in terms of Wigner energy and radioactivity was plausibly correlated well with each other.

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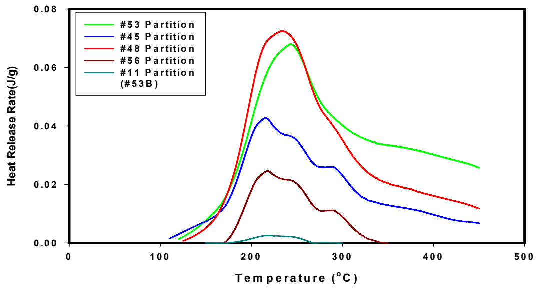

Fig. 1 Typical graphite samples having different values of Wigner energy and the radioactivity intensity(according to the number) in thermal column based on the position from the reactor core.

Table 1 Wigner energy value and Radioactivity for typical graphite samples.

Sample	Position	Dose Rate (mSv/h)	Specific Activity (Bq/g)
11C	1M11	0.17	1049.35
45C	1J01	0.22	4157.38
48C	1J04	0.32	5252.24
53C	1J05	1.95	8679.78
56C	1J12	0.55	5619.96