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Method for calculating delayed gamma-ray transport from aluminum in HANARO

Byung-Gun Park*, G. M. Sun and Myong-Seop Kim
Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon 34057, Korea
**Corresponding author: bgpark@kaeri.re.kr*

1. Introduction

HANARO is a research reactor that uses neutrons to serve various purposes including material testing, neutron activation analysis, NTD, and RI production. Various types of radiation are generated, and temperature of the reactor and samples is increased by nuclear heating. In order to accurately predict the temperature of the core (nuclear fuel and structural materials), sample, and test device, the heating value must be evaluated.

Among the factors contributing to nuclear heating is decay radiation from the radioactivity of structural materials [1]. There are many aluminum (Al) structures in the HANARO core. The cladding material of the nuclear fuel rods is pure aluminum with high ductility, and the chimney located on the top of the core and heavy water reflector tank are made of aluminum alloy Al-6061-T6. In addition, the rigs and capsules used for irradiation tests and isotope production are mostly made of Al or Al alloys. In this study, a method for evaluating the nuclear heating of Al-containing irradiated materials based on the Monte Carlo method is presented, and a method for performing the transport calculation of delayed gamma-ray is presented.

2. Methods and Results

When nuclear fission occurs in a reactor, various forms of energy are released. Main components that contribute to the nuclear heating can be divided into (1) fission products, (2) fission neutrons, (3) fission gamma rays, (4) gamma rays captured by neutron absorption, (5) beta and gamma rays from the decay of fission products, and (6) beta and gamma rays from the decay of radioactive products in structure materials. In addition to the mentioned components, an additional consideration is (7) beta and gamma radiation from the decay of radionuclides in the irradiated material. This is often included in (6), but it can be evaluated separately from (6) because it is not a fixed structure in the core.

When the test rigs, capsules containing Al are irradiated by neutrons, Al-27 captures neutrons and becomes Al-28, and Al-28 beta decays ($T_{1/2}$=2.245 m) to Si-28. The beta ray emitted by Al-28 has a maximum energy of 2863.27 keV and an average energy of 1241.8 keV, and it also emits a gamma-ray of 1778.987 keV [2]. The prompt gamma-ray spectrum of Al is shown in Figure 1. It includes the 1778.92 keV gamma-ray, which is the decay gamma-ray of Al-28, which is treated as the prompt gamma-ray due to the short half-life of Al-28. Compared to the lines of the prompt gamma-ray, the 1778.92 keV delayed gamma-ray has a relatively high emission rate.

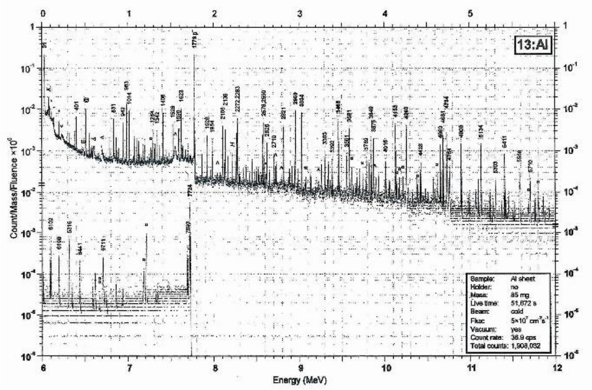

Fig. 1. Prompt gamma-ray spectrum of Al [2]

In order to evaluate the nuclear heating of Al-containing materials, transport calculations for these beta and gamma rays are necessary. Beta rays lose all of their energy within a few millimeters, while gamma rays transfer some of their energy into the workpiece and some escape. In the case of delayed gamma-ray, the (n,γ) reaction rate is calculated by region, and the gamma-ray source distribution is used to estimate the heating value accumulated by the gamma-ray. This evaluation method is called the two-step method, and in this case, Monte Carlo particle transport calculations must be performed twice. In the Monte Carlo calculation, especially MCNP codes, transport calculations of delayed gamma rays can be performed in four main ways.

1) Two step method

The two-step method involves two stages of calculation: the production rate of radioactive isotopes and the source term of delayed gamma rays. This method involves dividing the sample into smaller pieces and using mesh tally to divide it into multiple cells due to variations in neutron flux gradients.

2) Activation option

The MCNP code can perform radiation transport calculations using the ACT option, which includes gamma-ray data from the CINDER90 database and emission rate data for many isotopes. However, the ACT option has limitations, such as its inability to perform kcode calculations and multi-thread calculations, and its slow calculation time. As a result, it may not be suitable for complex problems but can be used for simple geometric structures.

3) Modification of the nuclear library [4,5]

The continuous-energy and discrete neutron data library in the MCNP code can be modified. This library contains Gamma Production Data (GPD), and it contains nuclear data for prompt gamma rays. However, this method is very cumbersome if the nuclear data library is large or if there are many types of gamma rays, because it is very difficult to match the formatting and indexing it.

4) Use a nuclear data library for delayed gamma rays

Certain nuclear data libraries contain delayed gamma-emission data for Al-28. In the MCNP code, ACTI library based on ENDF-B/VIII.0 published in 2002, contains emission data for the Al-27(n,γ)Al-28 reaction. In order to compare the gamma-ray flux and the gamma heating value using different methods for delayed gamma rays of Al, a simple model was designed and MCNP calculations were performed, as shown in Figure 2. The model is a cylinder with a diameter of 10 cm with heavy water at the bottom of the cylinder at a height of 1 cm, Al at a height of 3 cm in the middle, and heavy water again at a height of 1 cm above the Al. In the calculation using two-step method, the Al cylinder subdivided into 10 pieces. Neutrons are emitted from the bottom of the cylinder toward the Al with an energy of 0.025 eV.

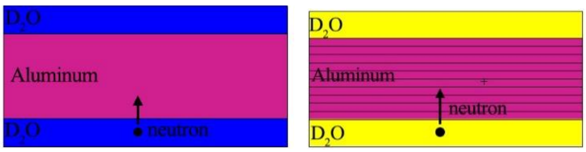

Fig. 2. MCNP models for gamma-ray transport in Al (simple model and model in the two-step method)

Using MCNP 6.2 code, the photon flux and gamma heating value at the Al position were calculated for each of the following cases: without considering delayed gamma rays, using the ACT option (multi group and line mode), using the delayed gamma-ray data, and using the two-step method. Figures 3 and 4 show the photon flux and gamma heating distribution as a function of energy. In both figures, the y-axis refers to the value for one neutron. In the figures, we can see that the photon flux and gamma heating value between 1 and 2 MeV increase significantly when applying the ACT option or using the delayed gamma ray library. The photon flux between 50 keV and 1 MeV, where the 1778.987 keV gamma-ray reacts with the material and loses some of its energy, also increases.

In the case of two-step method, the neutron capture reaction rate and delayed gamma-ray emission rate of each piece is listed in Table 1, where the lowest piece of aluminum is called Al1, and the higher pieces are called Al2, Al3 ... , Al10 as shown in Figure 2. Using the emission rate of delayed gamma-ray in Table 1 as the source term and calculating the gamma-ray transport of 1778.987 keV, the gamma heating of delayed gamma-ray is 2.229×10^{-5} MeV/g/neutrons.

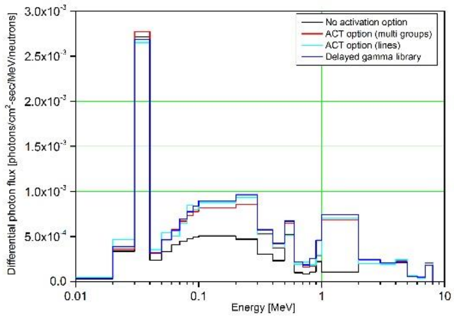

Fig. 3. Distribution of photon flux using different estimation methods for delayed gamma rays from Al

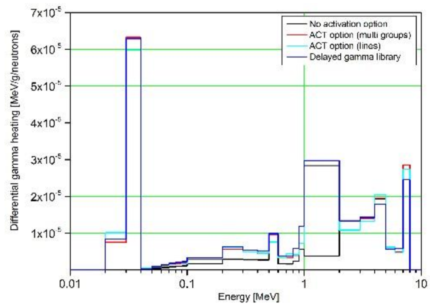

Fig. 4. Distribution of gamma heating using different estimation methods for delayed gamma rays from Al

Table I: Neutron capture reaction rate and delayed gamma-ray emission rate in the two-step method

Position	Neutron capture reaction rate [reactions/cm^3/n]	Delayed gamma-ray emission rate [photons/n]
Al1	2.611×10^{-4}	6.153×10^{-3}
Al2	2.510×10^{-4}	5.915×10^{-3}
Al3	2.430×10^{-4}	5.726×10^{-3}
Al4	2.359×10^{-4}	5.559×10^{-3}
Al5	2.297×10^{-4}	5.413×10^{-3}
Al6	2.247×10^{-4}	5.295×10^{-3}
Al7	2.204×10^{-4}	5.192×10^{-3}
Al8	2.171×10^{-4}	5.115×10^{-3}
Al9	2.150×10^{-4}	5.065×10^{-3}
Al10	2.148×10^{-4}	5.061×10^{-3}

3. Conclusions

Neutron and gamma-ray transport calculations were performed, and photon fluxes and gamma heating values were evaluated to compare methods for calculating delayed gamma-ray transport in Al. The total gamma heating values calculated by the two-step method, the ACT option, and the delayed gamma-ray library were similar, therefore, any of the methods can be used to perform delayed gamma-ray transport for Al. However, when performing core calculations, the ACT option of the MCNP code cannot be directly applied, and the two-step method requires manual cell splitting and multiple calculations. Therefore, the most convenient method is currently considered to be to use the ACTI library, which contains the delayed gamma-ray data of Al-28. When the delayed gamma rays of Al-28 were considered, the photon flux increased by about 1.54 times compared to the prompt gamma rays only, and the gamma heating value increased by about 1.28 times. Therefore, when conducting neutron irradiation experiments on materials containing Al, it is essential to consider delayed gamma-ray transport.

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