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Study on the burn-up characteristics of a thermal neutron filter containing B_4C particles for NTD-Si irradiation

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ABSTRACT

We investigated an alternative irradiation method for silicon doping using a thermal neutron filter to increase the irradiation efficiency in the Japan Research Reactor No. 3 (JRR-3). Aluminum mixed B_4C particles were adopted as the filter material. The burn-up characteristics of a thermal neutron filter containing a large number of B_4C particles are described. We have developed a method for evaluating neutron transmissivity as Boron burning in the neutron filter and clarified the applicable scope of the method via experiments using thermal neutron filters. We derived a method by which we could calculate the suitable diameter and density of B_4C particles.

1. Introduction

Neutron Transmutation Doping (NTD) (Lark-Horovitz, 1951; Tanenbaum and Mills, 1961) is one of the doping methods for silicon, and the dopant exists more uniformly in silicon in this type of doping than in other doping methods (Blowfield, 2007). The NTD silicon is mainly used in the Insulated Gate Bipolar Transistor (IGBT), which needs to operate at high voltages and currents. There is growing demand for the IGBT because it is effective for energy saving. The JRR-3 is producing the NTD silicon, and the present irradiation method of JRR-3 is the so-called reverse method, which is simple and easy. However, this method requires two irradiation sessions for a silicon ingot in order to have uniform irradiation along the long axis direction. A long cooling time of about 2 days is necessary in the interval between the first and second irradiations. The low irradiation efficiency is additionally caused by the long irradiation time in this method, and the axial flux distribution is not used efficiently. Here, JRR-3 has investigated an alternative irradiation method using a thermal neutron filter, called the filter method or screening method, to obtain the increasing of the irradiation efficiency. The screening method was utilized in some research reactors; nickel filters in FRM II (Li et al., 2009) of Germany and OSIRIS (Breant et al., 1980) of France, and stainless (SUS) filters in HANARO (Kim et al., 2006) of Korea. JRR-3 cannot use these types of filter because of a potential radiation problem. Although in other reactors irradiated silicon is usually cooled in the pool for a long time, this is not possible at JRR-3, because there is thick shielding on the pool and silicon cannot be moved in the pool. In JRR-3, irradiated silicon is cooled in a cask. Since silicon is handled in the cask, there is little distance between the human operators and the silicon holder.

Although the filter material is attached to an irradiation pipe in other reactors, the filter material is requisite to be attached to a silicon holder in JRR-3 due to limitations in the mechanical design. Moreover, activation materials, including nickel and SUS, cannot be applied for the filter method in JRR-3. Because of the restrictions caused by engineering design and activity, aluminum mixed B_4C particles were adopted as a screening material in JRR-3.

A thermal neutron filter contains a large number of B_4C particles. In the Monte Carlo calculation that is part of filter analyses, it is impossible to describe the position information of all of those large number of particles. To solve the issue, we used the STGM (Statistical Geometry Model; Nagaya and Mori, 2005) function of the Monte Carlo code MVP (Nagaya et al., 2005) developed by the Japan Atomic Energy Agency (JAEA). The STGM was developed for the analysis of coated fuel particles. The transmissive neutrons are changing as B_4C is burning up. The estimation of the transmissive neutrons during burn-up is very important, since it concerns evaluating the life time of a silicon holder with the thermal neutron filter. The burn-up degree of the B_4C depends on the particle diameter, and the neutron shielding ability depends on the B_4C

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density and the particle diameter. The estimation of changing the transmissive neutrons based on the parameters of the B_4C density and the particle diameter is important. However, it is difficult to do the estimation by STGM calculations alone, since there are an enormous number of calculations that would need to be completed.

The purpose of this study was to clarify the grain effect of B_4C particles in a thermal neutron filter and develop an estimation method of the filter's ability. We derive the relational expression of the changing of the transmissive neutrons with parameters of the B_4C density and the particle diameter using Monte Carlo simulations, and by using that expression we are able to analyze the thermal neutron filter in detail.

2. Estimation method of transmissive neutrons

We investigated the change in the neutron capture cross section to clarify the decreasing filter ability of neutron capture during B_4C burn-up. The burn-up of B_4C (B_p) can be expressed as follows:

$$B_p(t) = \frac{\int_0^t \sum_a(t)\varphi dt}{N_0}, \tag{1}$$

where N_0 is the initial atomic density of ^{10}B, φ is the neutron flux and Σ_a is the macro capture cross section of the thermal neutron filter.

In this work, we assumed that Σ_a is proportional to B_p, and the ambit of the supposition is described in the next section. Here, Σ_a can be given as Eq. (2a) when α is the proportionality factor between the macro capture cross section (Σ_a) and the burn-up of B_4C (B_p). As we will describe later, α declines as the diameter of a B_4C particle increases. Σ_0 is defined as the initial macro capture cross section of the thermal neutron filter. Eq. (2a) is assumed to become Eq. (2b), as follows:

$$\Sigma_a(B_p) = \Sigma_0 \times (1 - \alpha B_p), \tag{2a}$$

$$\Sigma_a(t) = \Sigma_0 \times (1 - \alpha B_p(t)). \tag{2b}$$

when Eq. (2b) is substituted into Eq. (1) and it is differentiated with respect to the time evolution t, Eq. (3) is derived, as follows:

$$\frac{d\sum_a(t)}{dt} = -\frac{\alpha\Sigma_0\varphi}{N_0}\sum_a(t). \tag{3}$$

Eq. (4) is obtained by solving Eq. (3), as follows:

$$\sum_a(t) = \Sigma_0 \times e^{-\frac{\alpha\Sigma_0\varphi}{N_0}t}. \tag{4}$$

Eq. (4) indicates a change of neutron macro capture cross section as burn-up, and Eq. (5) gives the relation between time and transmissive neutrons, as follows:

$$\frac{e^{-\sum(T)\times d}}{e^{-\sum(0)\times d}} = \frac{X+100}{100}, \tag{5}$$

where T is defined as the time when transmissive neutrons passing through the filter are increased by X%, and d is the filter thickness (cm). Eq. (6) used to estimate the filter's performance degradation. It is derived from Eqs. (4) and (5), as follows:

$$T = \frac{N_0}{\alpha\phi}\ln\left(\frac{-\ln\left(\frac{X+100}{100}\right)}{d\Sigma_0} + 1\right). \tag{6}$$

3. Parameter calculation with calculation code

The values of α and Σ_0 in Eqs. (4) and (6), respectively, were calculated on the basis of the Monte Carlo neutron transport code MVP. Filter pieces containing 1.0 wt.% B_4C density of 2 mm thickness are irradiated by thermal neutrons, and the macro capture cross sections are calculated.

The neutron macro capture cross sections can be given as follows:

$$\Sigma_a = \frac{\int R(\vec{r})dV}{\int \varphi_{mat}(\vec{r})dV + \int \varphi_{B_4C}(\vec{r})dV}, \tag{7}$$

where the $^{10}B(n,\alpha)^7Li$ reaction rate in the filter is R (1/cm^2/s), the neutron flux of matrix (aluminum) is φ_{mat} and the neutron flux in B_4C is φ_{B_4C}. The neutron absorption in aluminum is neglected, and Σ_a can be given by calculating Eq. (7), and the Σ_a value determined by Eq. (7) is only the initial number. Here, the macro cross sections as burn-up are calculated, and then burn-up calculations reduce the density of ^{10}B atoms. The calculation results are shown in Fig. 1, and they show that there is a good proportional relationship between 0% and 30%. However, the proportional relationship diminishes due to self-shielding effect over 30%. A homogeneous model that does not use STGM has a good proportional relationship because it does not have the self-shielding effect. As shown in Fig. 1 and Table 1, the y-intercept and gradient are obtained by the values of Σ_0 and α, and the value of Σ_0 declines with increasing diameter due to self-shielding, although the value of 2 μm is larger than in the homogeneous model. This tendency was considered to be caused by an error of STGM calculation. In Table 1, the value of α also declines with increasing diameter, which means that larger particles have smaller burn-up. The α value of the homogeneous model has a much larger value than others. Especially the value of 2 μm is much less than in the homogeneous model, although the value Σ_0 of 2 μm is almost the same or barely larger in the homogeneous model. This indicates that the influence of self-shielding at burn-up is strong.

Using the results shown in Table 1, the transmissive neutrons through a filter during irradiation by neutron flux 2×10^{13} (1/cm^2/s) is shown in Fig. 2. The transmissive neutrons increase by 10% to 2 μm by the 149th day, to 20 μm by 180th day, to 40 μm by the 223rd day, to 100 μm by the 435th, to 200 μm by the 1504th day, and 121st day in the homogeneous model. They have the same B_4C density. The 100 μm lifetime is 2.4 times as long as 20 μm. The filter lifetime is obviously longer when the particles have a large diameter. The ability of neutron capture reactions is decreased in the case of the large particles due to self-shielding, and it is necessary to increase the number of B_4C particles: to raise the B_4C density to increase the ability of neutron capture reactions. Next, Fig. 3 shows the calculation results of the relation between B_4C density (wt.%) and the neutron capture cross section using STGM of MVP. As shown in Fig. 3, the B_4C density and Σ_a are in good proportional relationship. Approximate equations at each density are as follows, where x is defined as B_4C density (wt.%):

$$\left.\begin{aligned}
&2\ \mu\text{m} : \Sigma_a(x) = 0.8871x + 0.0031\\
&10\ \mu\text{m} : \Sigma_a(x) = 0.8655x + 0.0027\\
&40\ \mu\text{m} : \Sigma_a(x) = 0.7945x - 0.0003\\
&60\ \mu\text{m} : \Sigma_a(x) = 0.7528x - 0.0029\\
&100\ \mu\text{m} : \Sigma_a(x) = 0.6773x - 0.0050\\
&\text{homogeneous model} : \Sigma_a(x) = 0.8654x + 0.0003
\end{aligned}\right\}, \tag{8}$$

The value of Σ_a in Eq. (4) corresponds to that of Σ_0 in Eq. (6). The design of the neutron filter requires the ability of neutron capture reactions, and to satisfy Σ_0 obtained by Eq. (8). By using α obtained in Table 1, changing of the transmissive neutrons as a result of burn-up can be indicated. We can then discuss the necessary diameter of B_4C particles by referring to the necessary lifetime of the particles. When the values of Σ_0 and the diameter are determined, the B_4C density can be calculated by Eq. (8).