Comparison of Absorption Cross Sections for Neutron Absorbing Materials

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1. Introduction

The use of neutron absorbing materials as an additive to packages that contain fissile material has been identified as an opportunity for the safe transport and storage packaging of radioactive material. Their use for criticality safety is expected to result in a reduction in neutron flux within a system to a level at which neutron-induced chain reactions are not sustainable. The International Atomic Energy Agency (IAEA) has provided the international regulatory basis for the safe transport of fissile material [1]. The primary goal of the regulations is to protect the public, workers, property and the environments from the effects of radiation during transit. The transport packages that contain fissile materials need to ensure that contents will remain sufficiently subcritical under the normal and accident conditions [2]. Besides, similar safety requirements are applied to a spent nuclear fuel storage and its management. The safety of a spent fuel storage is ensured by appropriate containment, criticality safety, heat removal and radiation shielding. In the IAEA standards, various means of maintaining subcriticality are described, which include geometrical configurations of spent fuel, fixed neutron absorbers and the use of burnup credit [3].

Spent fuel is generated constantly by the operation of nuclear reactors worldwide. It is stored in the reactor fuel storage pool for a period of time and then may be transferred to a wet or dry storage facility. Neutron absorbing materials are required for interim storage of spent fuel, as well as long-term disposition. In order to keep the spent fuel facility safely for a long time, effective neutron absorbers are to be examined as a preferred option. In this study, we investigated the effective absorbers for neutrons from thermal through epithermal ranges by comparing the absorption cross sections for various elements. In evaluating the group-averaged neutron cross sections, we used the latest version of the ENDF libraries and performed the numerical simulation. Finally, the absorption cross sections of a model alloy for fixed neutron absorber were generated for a quantitative comparison.

2. Methods

In this section, the procedure of calculating neutron absorption cross sections for specified elements is described, which are obtained from the ENDF/B-VIII library provided by the BNL NNDC site [4].

2.1 Neutron Absorption Elements

There are well-known elements with high neutron absorption cross sections such as B, Gd, Cd, Hf, Ir and the rare earth elements. Among these, boron is used prevalently for the absorbing materials because of its availability and cost. However, the effectiveness of neutron absorption in the whole energy range (thermal to fast) is of our interest. In this work, the addition of various high neutron absorbing elements to Ti-matrix is taken into account. Though the specific amount of the element for a model alloy is not determined yet, chemical composition of several neutron absorbing materials, including two commercial ones of Boral® and borated-stainless steel (B-SS), is listed in Table I.

Table I: Chemical compositions of neutron absorbing materials

Materials	Elements (w/o)						
Boral®	Al (60.7)		B (30.8)		C (8.5)		
B-SS	Fe (66.8)	Cr (18)	Ni (12)	Mn (1)	Si (0.2)	C (0.02)	B (1.7)
Model alloy	Ti (80)		Mo (12.5)		Nb (3.1)		Gd* (4.39)
*other absorbing elements considered : Cd, Dy, Hf, Sm, In							

The cross sections for target elements were obtained from the recent version of the ENDF/B library. Because the library does not provide the absorption cross section in one file, we obtained both total and elastic scattering cross sections and then processed to calculate the group averaged-value.

2.2 Calculation of Group-Averaged Cross Sections

Normally, the neutrons in a reactor have energies spanning the rage from 10 MeV down to below 0.01 eV – nine orders of magnitude. We break the neutron energy range into 20 energy groups at random between 0.001 eV and 1×10^7 eV, which visualize the intensity of neutron absorbing capacity for given energy range. Since there is no neutron flux spectrum for the spent fuel storage, collapsing the cross section data over the spectrum was not possible. For convenience sake, the FORTRAN subroutine for generating a group-averaged value was used in the calculation, which was included in the SPECTER code [5].

Total and elastic scattering cross sections for Gd-157 are shown in Fig. 1. where the solid curves are from the ENDF/B library and the dotted lines represent the group-averaged values. We have calculated the group-averaged cross sections for all elements listed in Table I, including natural isotopes.

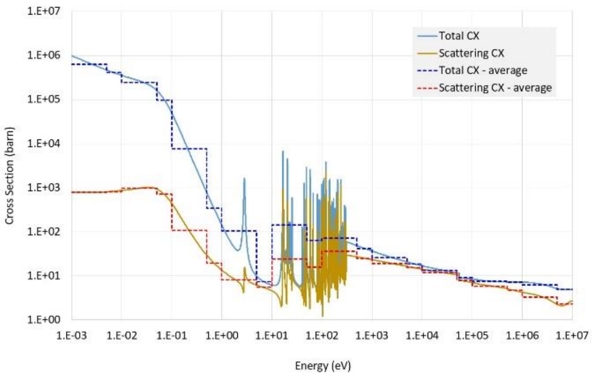

Fig. 1. Total and elastic scattering cross sections for Gd-157 (solid: ENDF/B-VIII, dotted: group-average)

3. Results

With the group-averaged cross sections calculated previously, the absorption cross sections for neutron absorbing materials are generated in this section.

3.1 Elemental Absorption Cross Sections

The absorption cross section (σ_{abs}) for each isotope is calculated from the difference between total (σ_t) and elastic scattering cross section (σ_{el}), which is given by:

$$\sigma_{abs} = \sigma_t - \sigma_{el} \qquad (1)$$

Considering the natural abundance of the isotopes, we obtain the absorption cross sections for an element. Fig. 2. shows the absorption cross sections for Ti (dotted), as well as its five isotopes (solid). Besides, for five elements which are known as strong neutron absorbers, their absorption cross sections are plotted in Fig. 3. Depending on the range of neutron energy, the absorption cross sections change significantly.

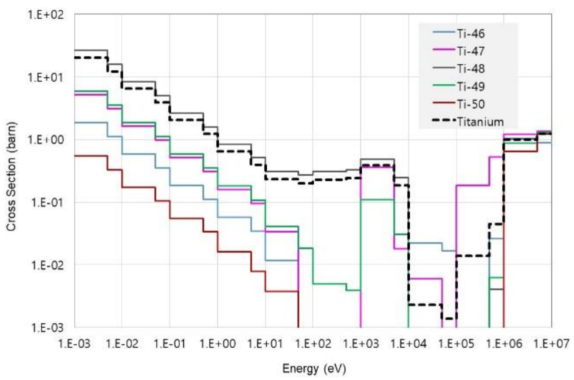

Fig. 2. Absorption cross sections for Ti (dotted) and its isotopes (solid)

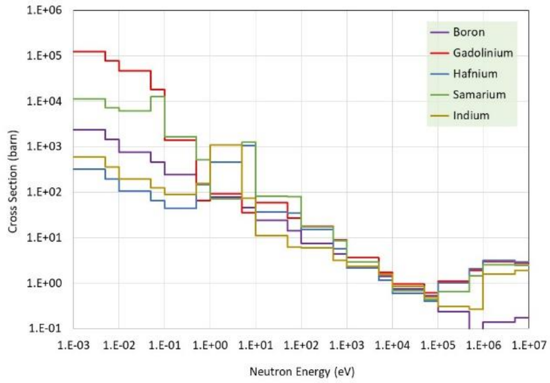

Fig. 3. Absorption cross sections for five neutron absorbing elements including B, Gd, Hf, Sm and In

3.2 Neutron Absorption of Neutron Absorbing Material

The absorption cross sections for three absorbing materials in Table I are plotted in Fig. 4. Though the chemical components for the Ti-model alloy is not determined, the materials containing Gd demonstrates higher neutron absorbing capacity in the thermal energy range. On the other hand, boron exhibits higher neutron absorption in the epithermal and fast energy region.

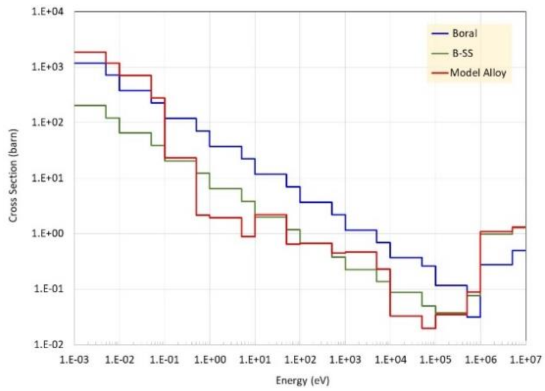

Fig. 4. Absorption cross sections for neutron absorbing materials of Boral, B-SS and Ti-based model alloy

4. Discussion

The use of neutron absorbing material is important to spent fuel storage and transport devices such as fuel rack and dry storage casks. Boron has been regarded as an effective neutron absorbing element and is being used prevalently as a major component. Due to low solubility of boron in an alloy, its addition is limited, though. Considering two factors of neutron absorbing capacity and structural integrity, it is meaningful to look into other choices of neutron absorbing elements. In this study, we investigate the neutron absorption cross sections for elements of interest, which will be useful in determining the effective neutron absorbing ones. Currently, we selected Gd as an absorbing element in the Ti-based model alloy, which gives strong absorption capacity in the thermal energy range of neutrons. However, other elements such as Hf, In, *etc.* would be an alternative when considering the epithermal and fast neutrons.

REFERENCES

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