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Comparison of Fuel Temperature Coefficients of MTR U_3Si_2 Fuel Depended on Nuclear Data Library (ENDF/B-VI.0 and VII.1)

Kyung-O KIM*, Gyuhong ROH, and Byungchul LEE
Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-gu, Daejeon, Korea
**Corresponding Author: k5kim@kaeri.re.kr*

1. Introduction

The interaction probability of incident neutron with nuclear fuel depends on the relative velocity between the neutron and the target nuclei. However, since the nucleus are in continual thermal motion, this relative velocity can be changed to be faster or slower than the neutron speed, and their thermal motion is mainly influenced by the change of fuel temperature. Especially, the resonance in neutron absorption cross-section is broadened with increasing fuel temperature, while its peak magnitude is decreased. This phenomenon is called as Doppler effect or Doppler broadening of cross-section, which leads the change of the neutron multiplication factor (k_{eff}) in nuclear core.

The Doppler defect ($\Delta\rho_{DD}$) can be simply evaluated from the reactivity difference between the Hot Full Power (HFP) and Hot Zero Power (HZP) conditions, which is mathematically presented as follows;

$$\Delta\rho_{DD} = \frac{k_{eff}^{HFP} - k_{eff}^{HZP}}{k_{eff}^{HFP} \times k_{eff}^{HZP}} \quad (1)$$

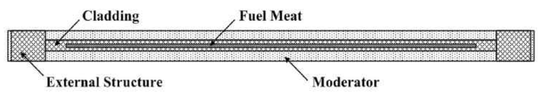
The Doppler coefficient (D_c or FTC) is defined as the change of Doppler defect with respect to the change in fuel temperature (ΔT_{Fuel}) without any other change such as moderator temperature, moderator density, etc., and it can be denoted as follows;

$$D_c = \frac{\Delta\rho_{DD}}{\Delta T_{Fuel}} \quad (2)$$

In this study, the FTCs for U_3Si_2 fuel were evaluated by using MCNP6.1code [1] based on a Monte Carlo method. In addition, the latest neutron cross-sections (ENDF/B-VI.0 and VII.1) were applied to analyze the effect of these data on the evaluation of FTC, and nuclear data used in MCNP calculations were generated from the makxsf code [2].

2. Methods and Materials

The world's research reactors have been used for research and training, material testing, or the production of radioisotopes, and the most common type of those is called as MTR (Material Test Reactor). The MTRs have generally used plate-type nuclear fuel for their operation, and these plates are arranged in parallel bundles. Also, the fuel material is surrounded by aluminum cladding for isolating the fission products, and the pure water as coolant/ moderator is streamed through the space between the plates. The uranium silicide-aluminum (U_3Si_2-Al) was considered as fuel material because it is widely used for the standard LEU fuel. Hence, a unit cell of MTR U_3Si_2-Al fuel can be assumed to be a long rectangular form with a fuel plate in a center and external structures at both sides, and its specification applied in this study is presented in **Figure 1**.



Parameter	Value
Unit cell [cm]	
Thickness × Width × Length	0.42 × 7.7 × 64
Fuel meat [cm]	
Thickness × Width × Length	0.05 × 6.0 × 60
Cladding [cm]	
Thickness × Width × Length	0.15 × 6.6 × 64
Moderator [cm]	
Thickness × Width × Length	0.42 × 6.6 × 64
External structure [cm]	
Thickness × Width × Length	0.42 × 0.5 × 64
Fuel material	U_3Si_2-Al
Fuel density [g/cc]	12.0
Cladding material	Al
Cladding density [g/cc]	2.67
Moderator material	H_2O
Moderator density [g/cc]	0.9975

Figure 1. Unit Cell Model of MTR U_3Si_2 Fuel

The unit cell model was composed of fuel meat, cladding, side plates, and moderator region and was designed with reference to the fuel assemblies used in the Argentina RA-6 [3] and Japan MTR [4]. The materials of fuel, cladding and side plate, and moderator are U_3Si_2-Al (4.8 gU/cm^3), aluminum, and pure water respectively. The Doppler coefficients were evaluated by changing fuel temperature alone from 296K to 423K, and unlike the evaluation of PWR UO_2 fuel, it was assumed to be no difference in fuel meat dimension and density at those temperatures. The uranium enrichments were considered from 19.89% up to 90.0% in the evaluations of the neutron multiplication factor and Doppler coefficient. The unit cell model shown in **Figure 1** was modeled in 3-D geometry, and the reflective boundary condition was applied in the axial and other four sides of this model. Each evaluation was performed with 10,000 neutrons per cycle and an initial guess for k_{eff} of 1.0. The first 500 cycles were skipped before k_{eff} data accumulation, and a total of 3,500 cycles were run.

3. Results and Discussions

For eight different uranium enrichments (19.89 wt% – 90.0 wt% ^{235}U), k_{eff} values were calculated by using MCNP6.1 code (see **Table 1**). As shown in the table, k_{eff} values were linearly increased with the increase of uranium enrichment, and the difference within about 400 pcm is found between each value at each enrichment due to the difference of the used nuclear data. To review those libraries, they had some differences in the neutron absorption cross-section of ^{27}Al isotope, and especially the ENDF/B-VI.0 library showed higher neutron absorption probability at energy regions around resonance peak (5.9 keV and 35.94 keV) than the other data (see **Figure 2**). Because of this, it was estimated that k_{eff} values evaluated by the ENDF/B-VI.0 cross-section were generally lower than those from the ENDF/B-VII.1 library.

Table 1. Calculation Results for the Unit Cell Model of MTR U_3Si_2 Fuel

^{235}U wt%	k_{eff} at HFP (Standard Deviation)		k_{eff} at HZP (Standard Deviation)	
	ENDF/B-VI	ENDF/B-VII	ENDF/B-VI	ENDF/B-VII
19.89	1.64765 (0.00010)	1.65098 (0.00010)	1.65432 (0.00010)	1.65742 (0.00010)
30.0	1.68184 (0.00010)	1.68473 (0.00010)	1.68847 (0.00010)	1.69166 (0.00010)
40.0	1.6995 (0.00010)	1.70290 (0.00010)	1.70618 (0.00010)	1.70935 (0.00010)
50.0	1.71219 (0.00010)	1.71532 (0.00010)	1.71862 (0.00010)	1.72196 (0.00010)
60.0	1.72294 (0.00010)	1.72628 (0.00010)	1.72934 (0.00010)	1.73274 (0.00010)
70.0	1.73384 (0.00010)	1.73710 (0.00010)	1.73978 (0.00010)	1.74353 (0.00010)
80.0	1.74584 (0.00010)	1.74911 (0.00010)	1.75149 (0.00010)	1.75508 (0.00010)
90.0	1.76218 (0.00010)	1.76562 (0.00010)	1.76679 (0.00010)	1.7706 (0.00010)

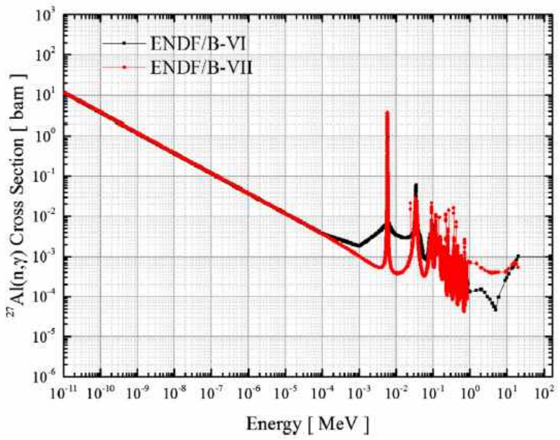


Figure 2. $^{27}Al(n,\gamma)$ Cross-section of ENDF/B-VI and VII (at 300K)

The Doppler coefficients were obtained from k_{eff} values, which was fitted by a 4th order polynomial equation (see **Figure 3**), and the deviations of them were derived from standard deviations of k_{eff} values. As a result, unlike the case of PWR UO_2 fuel, the Doppler coefficient of MTR U_3Si_2-Al fuel was exponentially changed to be less negative with increasing uranium enrichment. Particularly, there was a difference more than two standard deviation (2σ) between the Doppler coefficients at the range above enrichment of 60%. Also, the coefficients evaluated from ENDF/B-VII.1 were generally lower than those from ENDF/B-VI.0, except for the value at 19.89% uranium enrichment.

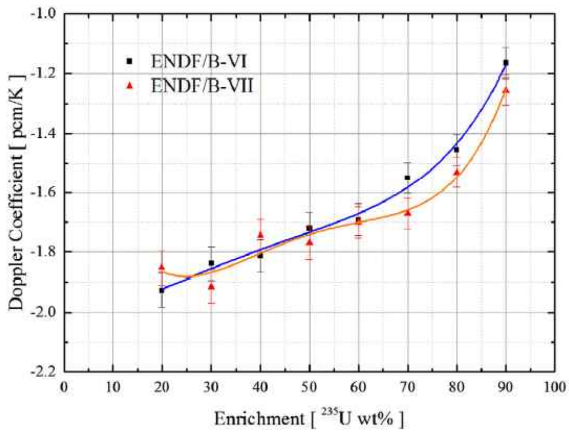


Figure 3 Doppler coefficient for MTR U_3Si_2-Al fuel

4. Conclusions

An evaluation of the Doppler effect and FTC for U_3Si_2-Al fuel widely used in MTR was conducted using MCNP6.1. The ENDF/B-VI.0 and VII.1 were also applied to analyze what effect these data has on those evaluations. All cross-sections needed for MCNP calculation were produced using makxsf code. The calculation models used in the evaluations were based on the typical MTR U_3Si_2 lattice. As a result, there was a difference within about 300–400 pcm between k_{eff} values at each enrichment due to the nuclear data used in the evaluations. The FTC was changed to be less negative with the increase of uranium enrichment, and it was dramatically increased. However, it is necessary to perform additional study for investigating what factor causes the differences more than two standard deviation (2σ) among the FTCs at partial enrichment region.

REFERENCES

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