# Measurement of $^{233}$U-HU Substitution Reactivity Worth in KUCA for Integral Validation of $^{233}$U Nuclear Data

*Tadafumi* Sano[1*], *Jun-ichi* Hori[2], *Yoshiyuki* Takahashi[2], *Kazushi* Terada[2], *Hiroshi* Yashima[2], *Takashi* Kanda[1] and *Hironobu* Unesaki[2]

[1]Atomic Energy Research Institute, Kindai University, 3-4-1 Kowakae, Higashiosaka-shi, Osaka, Japan
[2]Institute for Integrated Radiation and Nuclear Science, Kyoto University, 1010 Asashiro, Kumatori-cho, Osaka, Japan

**Abstract.** In order to perform an integral evaluation of $^{233}$U nuclear data, measurements of substitution reactivity worth between $^{233}$U sample and high enriched uranium (HU) sample in the Kyoto University Critical Assembly (KUCA) were carried out. In this study, the experimental core was consisted of 24 fuel elements and 1 special fuel element with the $^{233}$U or HU sample. The fuel element was consisted of 31 unit cells and sandwiched by a upper and a lower polyethylene reflector. The unit has one enriched uranium plate of 1/16" thickness, three polyethylene plates of 1/8" thickness. The special fuel element was loaded at the center of the core. The substitution reactivity worth was defined as difference between the excess reactivity of the core in which the $^{233}$U sample or the HU sample inserted the core. The excess reactivities ware measured by the positive period method. As the result, the substation reactivity was obtained 0.0146±0.0006 %Δk/k. A numerical result by MVP3 with JENDL-4.0 was 0.0141±0.0002 %Δk/k and a C/E was 0.964 ± 0.041. On the other hand, the calculated substitution reactivity worth with JENDL-5 was 0.0168±0.0002 %Δk/k and a C/E was 1.151±0.047.

## 1 Introduction

In engineering discussions of the feasibility of new reactor systems, it is necessary to evaluate the impact of the fuels and materials for the neutronics characteristics such as criticality, conversion rate, and fuel balance. In order to develop of Thorium (Th) nuclear system, critical experiments on Th loaded cores using the Kyoto University Critical Assembly (KUCA) with solid moderator core have been systematically carried out to perform neutronics characteristics measurements of Th loaded thermal neutron systems and integral evaluation of Th cross sections. In those studies, integral validations of $^{232}$Th capture cross sections at KUCA [1] and differential validations of $^{232}$Th capture and total cross section using TOF method at KURNS-LINAC [2] have been performed. In this study, we focused on $^{233}$U cross section. In order to perform a integral validation of $^{233}$U fission cross section, new measurements of $^{233}$U-high enriched uraniumu (HU) substitution reactivity worth at KUCA were carried out, and the validation of the $^{233}$U nuclear data stored in the evaluated nuclear data library was performed by comparing them with numerical calculations.

## 2 Measurement of $^{233}$U-HU Substitution Reactivity Worth in KUCA

### 2.1 KUCA Core Configuration

KUCA solid moderated core is able to consist of enriched uranium fuel plate and various moderator plates (i.e. polyethylene and graphite). In the core, fuel and moderator plates in fuel element were set in a 1.5mm thickness aluminium sheath and all material plates have nominal cross section of 50.8 mm (2 in.) square. In this experiment, two types of fuel element were loaded into the KUCA solid moderated core shown in Figure 1 [1].

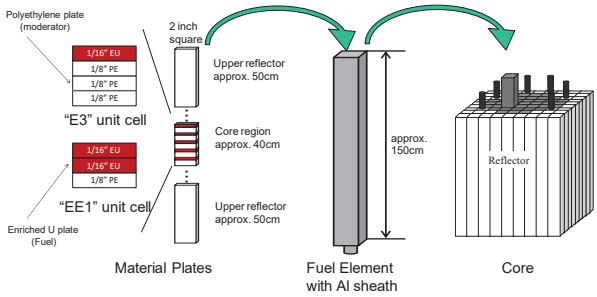


**Fig. 1.** Schematic view of the KUCA solid moderator core

**Figure 2** shows composition of a fuel element "F" and a special fuel element "S". A smallest component unit of the fuel element in KUCA is a composition of a fuel plate and a moderator plate called "unit cell". The fuel element "F" was consisted of 31-unit cells and sandwiched by an upper and a lower polyethylene

* Corresponding author: t-sano@kindai.ac.jp

reflector. The unit cell has one enriched uranium plates of 0.159 mm (1/16 in.) thickness, three polyethylene plates of 0.318 mm (1/8 in.) thickness. The fuel element "S" was consisted of 30-unit cells and one sample case made of aluminium. The geometry of sample case has 50.8 mm × 50.8 mm × 11.11 mm (7/16 in.).

**Figure 3** shows a core configuration of this experiment. The experimental core has 24 fuel elements "F" and one special fuel element "S". The "p" are polyethylene reflectors. The C1 – C3 are control rods and the S4 – S6 are safety rods.

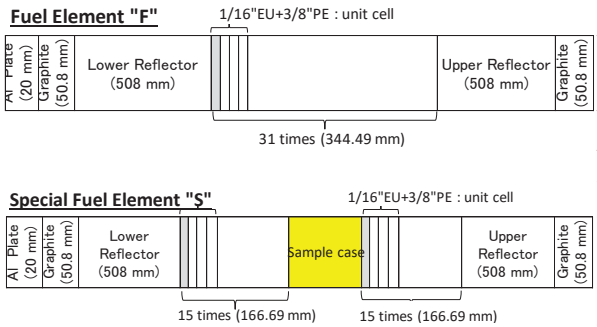


**Fig. 2.** Composition of a fuel element "F" and a special fuel element "S".

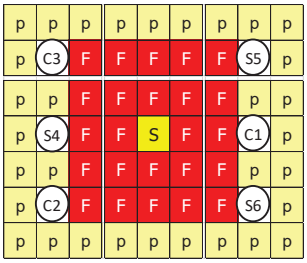


**Fig. 3.** Core configuration in this study.

## 2.2 Sample Information

A sample information of $^{233}$U plate is shown in **Table 1**. The $^{233}$U enrichment is 99.4 wt%. The sample plate was made of $U_3O_8$-Al and the amount of $^{233}$U in a plate was about 0.09 g. 9 plates of $^{233}$U sample were set into the sample case. **Figure 4** shows the sample geometry (left), the samples in the sample case (center) and the case installed into the fuel element "S" (right). The geometry of a plate was 12.66 mm × 12.68 mm × 1.08 mm and has a round chamfering (R = 3 mm). In the numerical analysis, a shape of the sample plate was assumed to be rectangular to keep the weight of $U_3O_8$.

A HU plate has almost same geometry as the $^{233}$U plate. A sample information and a geometry of HU plate is shown in **Table 2** and **Figure 5**.

**Table 1**. Sample information of $^{233}$U plate.

| Nuclide | Density (n/cm$^3$) * | Mass (g) |
|---|---|---|
| $^{233}$U | $1.342\times10^{21}$ | $9.034\times10^{-2}$ |
| $^{234}$U | $7.133\times10^{18}$ | $4.823\times10^{-4}$ |
| O | $3.598\times10^{21}$ | $1.689\times10^{-2}$ |
| Al | $4.491\times10^{22}$ | $3.502\times10^{-1}$ |

*The sample geometry is assumed to rectangular


**Fig. 4**. The sample plate geometry of $^{233}$U (left), the sample plates in the Al case (center), the case installed into the special fuel element "S" (right).

**Table 2**. Sample information of HU plate.

| Nuclide | Density (n/cm$^3$)* | Mass (g) |
|---|---|---|
| $^{235}$U | $1.308\times10^{21}$ | $8.421\times10^{-2}$ |
| $^{238}$U | $9.510\times10^{19}$ | $6.202\times10^{-4}$ |
| O | $3.741\times10^{21}$ | $1.756\times10^{-2}$ |
| Al | $4.745\times10^{22}$ | $3.700\times10^{-1}$ |

*The sample geometry is assumed to rectangular

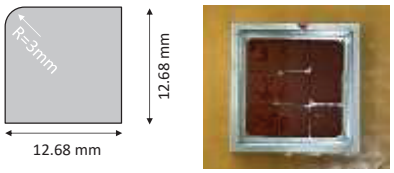


**Fig. 5**. Sample plate geometry of HU (left) and the sample plates in the Al case (right).

## 2.3 Measurement of substitution reactivity worth

The substitution reactivity worth was defined as the difference of excess reactivity between the core with the aluminium case installed the $^{233}$U plates and the core with the aluminium case installed the HU plates. The excess reactivity measurements were carried out by the positive period method. In each measurement, the C2 and C3 control rods were set to the upper limit (U.L.) and the C1 control rod was adjusted to the critical position. And the C1 control rod was drawn out to the U.L. from the critical position and the reactivity was inserted into the core.

The results of measured excess reactivity and the control rod position are shown in **Table 3** and **Table 4**. Here, the kinetics parameters using the reactivity measurements were calculated by the continuous energy Monte-Carlo code MVP3.0 [3] with JENDL-4.0 [4]. The parameters are shown in **Table 5**. As the result, the excess reactivity with $^{233}$U sample was 0.1003±0.0037 %Δk/k and with HU sample was 0.0857±0.0019 %Δk/k. Therefore, the sample worth was obtained as 0.0146±0.0043 %Δk/k. In this study, the experimental uncertainty is only considered measurement error.

**Table 3.** Excess reactivity with $^{233}$U samples

| Run No. | C1 (mm) | Core Temp. (℃) | Period (sec) | Reactivity (%Δk/k) |
|---|---|---|---|---|
| 9955 | 725.27 | 24.1 | 71.43 | 0.1005 |
| 9958 | 727.62 | 24.0 | 77.52 | 0.0948 |
| 9961 | 724.56 | 24.0 | 72.64 | 0.0995 |
| 9966 | 721.35 | 24.0 | 67.35 | 0.1051 |
| 9968 | 722.81 | 24.0 | 70.52 | 0.1016 |

**Table 4.** Excess reactivity with HU samples

| Run No. | C1 (mm) | Core Temp. (℃) | Period (sec) | Reactivity (%Δk/k) |
|---|---|---|---|---|
| 9959 | 734.95 | 24.0 | 92.22 | 0.0831 |
| 9962 | 734.38 | 23.9 | 90.07 | 0.0847 |
| 9965 | 731.49 | 23.9 | 85.83 | 0.0878 |
| 9967 | 732.41 | 23.9 | 86.78 | 0.0871 |
| 9969 | 733.26 | 23.9 | 88.31 | 0.0859 |

**Table 5.** Kinetic parameters

| | |
|---|---|
| Effective delayed neutron fraction | $7.895\times10^{-3}$ |
| Neutron generation time (sec) | $3.463\times10^{-5}$ |

## 3 Numerical Analysis

In numerical analysis, the substitution reactivity worth ($\rho_{calc}$) was defined as difference between the reciprocal of effective multiplication factor ($k_{eff}$) in the core with the $^{233}$U samples and with HU samples.

$$\rho_{calc} = 1/k_{eff,HU} - 1/k_{eff,233U} \quad (1)$$

The $k_{eff}$ was calculated by the MVP3.0 with JENDL-4.0, JENDL-3.3[5], ENDF/B-VII.1[6] and JENDL-5 [7]. Here only the $^{233}$U cross sections were taken from JENDL-4.0, JENDL-3.3, ENDF/B-VII.1 or JENDL-5 but all other nuclides were done from JENDL-4.0, were carried out to evaluate validations of $^{233}$U cross section among these nuclear libraries to the sample reactivity worth. In the MVP calculations, $5 \times 10^{10}$ neutron histories (1000000 histories × 50000 cycles) are generated to suppress the statistical error of $k_{eff}$ to less than 0.0005 % (1 σ ). The numerical results are shown in **Table 6**. The calculational result by JENDL-4.0 is agreement with the experimental results within the C/E error margin. On the other hand, the calculated substitution reactivity worth by JENDL-3.3, ENDF/B-VII.1, and JENDL-5 overestimates the experiment value by about 7 % - 15 %. Those results are able to be explained by difference of $^{233}$U cross sections in each library. Figure 6 shows the difference of $^{233}$U fission cross section between JENDL-4.0 and JENDL-5. The cross section of JENDL-5 is larger than that of JENDL-4.0 at the resonance energy region and the thermal energy region.
Figure 7 shows the numerical result of neutron spectrum and $^{233}$U fission rate. Those results were calculated by MVP3 and JENDL-4.0. That figure indicates that the difference of the calculated substitution reactivity worth between JENDL-4.0 and JENDL-5 is due to the energy range of thermal region.

**Table 6.** Numerical results of substitution reactivity worth

| Library | Numerical result (%Δk/k) | C/E |
|---|---|---|
| JENDL-4.0 | 0.0141±0.0002 | 0.964±0.041 |
| JENDL-3.3 | 0.0157±0.0002 | 1.072±0.044 |
| ENDF/B-VII.1 | 0.0165±0.0002 | 1.127±0.046 |
| JENDL-5 | 0.0168±0.0002 | 1.151±0.047 |

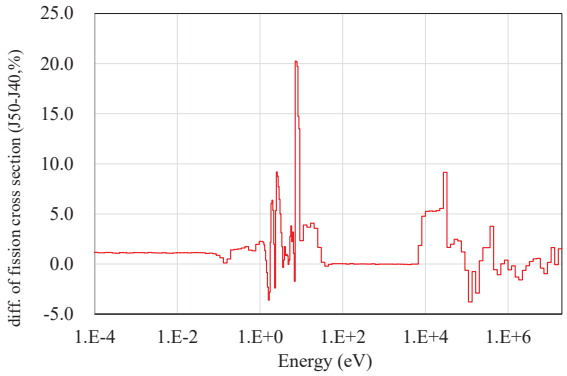


**Fig. 6.** Difference of $^{233}$U fission cross section between JENDL-4.0 and JENDL-5

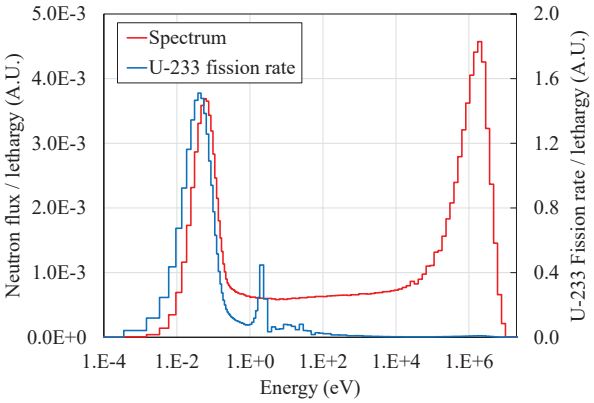


**Fig. 7.** Neutron spectrum and $^{233}$U fission rate at the sample region

In addition, an evaluation for the energy width contribution of cross section changes (JENDL-4.0 to JENDL-5) to the calculated substitution reactivity worth were performed by using a sensitivity analysis. The sensitivity coefficient is defined as following equation.

$$S = \frac{d\rho/\rho}{d\sigma/\sigma} \quad (2)$$

Where, ρ is the substitution reactivity worth and σ is the cross section respect for nuclide of interest. By multiplying the obtained sensitivity coefficient by the cross-section difference, an energy depend contribution of the reactivity difference due to the cross section change is obtained. The sensitivity coefficients were calculated by the sensitivity calculation code SAGEP [8]. **Table 7** shows the energy integrated components of the reactivity difference and **Figure 8** shows the energy wise contribution of the reactivity difference. Those results indicate that the difference of fission cross section in the thermal region has large impact for the difference of the calculated substitution reactivity worth.

**Table 7.** Energy integrated components of the calculated reactivity difference

| Reaction | Diff. of calculated reactivity worth (%Δk/k) |
|---|---|
| Fission | 0.0198 |
| Capture | -0.0040 |
| Nu | 0.0003 |
| Total | 0.0161 |

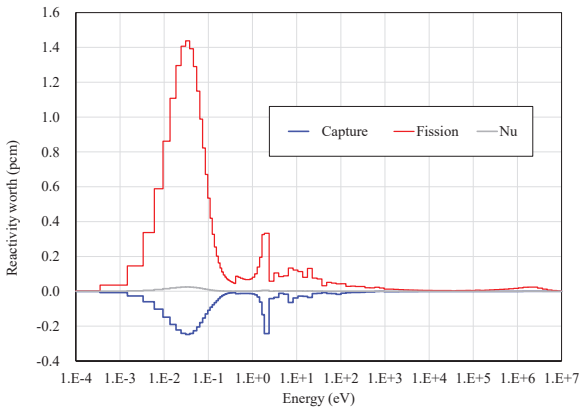


**Fig. 8.** Energy-depend component of the calculated reactivity difference

## 4 Summary

In this study, $^{233}$U-HU substitution reactivity worth was measured at the KUCA in order to perform the integral validation for $^{233}$U fission cross section. The samples were 9 $U_3O_8$-Al plates, and the total weight of $^{233}$U was about 0.81 g. As the experimental results, the substitution reactivity worth of 0.0146±0.0043 %Δk/k was obtained.

In the numerical analysis using the MVP3.0, the respective calculations for the sample reactivity worth with JENDL-4.0, JENDL-3.3, ENDF/B-VII.1 and JENDL-5 lead to 0.0141±0.0002 %Δk/k, 0.0157± 0.0002 %Δk/k, 0.0165±0.0002 %Δk/k and 0.0168± 0.0002 %Δk/k, respectively. The C/E values were 0.964 ±0.041, 1.072±0.044, 1.127 ± 0.046 and 1.151±0.047. In those calculation, only the $^{233}$U cross sections were taken from JENDL-4.0, JENDL-3.3, ENDF/B-VII.1 or JENDL-5 but all other nuclides were done from JENDL-4.0. As the results, the calculation results by the JENDL-4.0 agreed with the experimental data within the error range. In order to quantitatively evaluate the contribution of the cross section difference between JENDL-4.0 and JENDL-5 for the calculated substitution reactivity worth, the sensitivity analysis were carried out. As the results, the difference of $^{233}$U fission cross section in the thermal region has large impact for the difference of the calculated substitution reactivity worth.

## Acknowledgement

A part of this study was supported by CHUBU Electric Power Co., Inc. This work is done in part under the visiting researcher's program of the research reactor institute, Kyoto University, in 2021.

## References

1. T. Sano, J. Hori, J. Lee, Y. Takahashi, H. Unesaki,l. EPJ Web of Conferences, **247**, 09013, (2021).
2. J. Lee, J. Hori, T. Sano, Y. Takahashi, H. Unesaki, K. Nakajima, Energy Procedia, **131**, 306, (2017).
3. Y. Nagaya, K. Okumura, T. Sakurai, T. Mori, JAEA-Data/Code, **2016-018**, (2017).
4. K. Shibata, et. al., J. Nucl. Sci. And Thecnol., **48**, 172, (2011).
5. K. Shibata, et. al., J. Nucl. Sci. And Thecnol., **39**, 1125, (2002).
6. M.B. Chadwick, et. al., Nuclear Data Sheets, **112**, 2887, (2011).
7. O. Iwamoto, N. Iwamoto, K. Shibata, A. Ichihara, S. Kunieda, F. Minato, and S. Nakayama, EPJ Web of Conferences, **239**, 09002, (2020).
8. A. Hara, T. Takeda, Y. Kikuchi, JAERI-M **84-027**, (1984), (in Japanese).