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The Effect of the Neutron Multiplication Factor on a Nuclear Material Measurement in Dry-Processed Spent Fuel Material

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This study addresses the neutron multiplication which could occur in the assay sample in dry-processed spent fuel material. The neutron multiplication may cause measurement error in case of accounting by a coincident neutron counter. In this study, the multiplication effects in the dry processed DUPIC pellets were investigated by two methods : direct measurement and simulation by MCNP code. From the direct measurement, it showed that about 0.5% ~ 1% of coincident neutron was due to the multiplicity by induced fission, and that the results of the measurement method also showed good agreement with those of simulation by the MCNP code. The neutron multiplicity correction curve obtained from this study will be used to increase measurement accuracy of the well-typed neutron accounting system in the future.

KEY WORDS : NDA, neutron accounting, neutron multiplication, MCNP code

I. Introduction

Generally, two analysis methods, destructive and non-destructive analysis, have been used for accounting the nuclear material. The destructive analysis has high accuracy, but it is difficult to use in the large scale facilities because of long analysis time. Accordingly, non-destructive analysis by use of detecting gamma or neutron has been developed for nuclear material accounting in a dry-processing facility like DUPIC (Direct Use of PWR fuel In CANDU) facility.

In this study, the neutron multiplications by induced fission are measured and corrected to reduce the measurement error in coincidence neutron counting. The results were compared with those by Monte Carlo simulation methods. The measurement method and correction method of neutron multiplication effect were described in Chapter II, and the results of the measurement and their comparison with MCNP simulation were described in Chapter III.

II. Neutron Multiplication Correction

In this study, neutron multiplication effects were analyzed with DUPIC sintered pellets. DUPIC is the nuclear fuel cycle technology for reusing spent PWR fuel in CANDU via direct fabrication of spent PWR fuel materials without using the wet reprocessing process. It has been known that the DUPIC fuel, which uses the spent PWR fuel with 35GWD/MTU of burnup and 10 years of cooling time, emits spontaneous fission neutron 97.2% and (α,n) neutron of about 3%. In addition, Cm-244 is a dominant source of spontaneous fission neutrons generated from the spent fuel[1].

KAERI has developed DSNC(DUPIC Safeguards Neutron Counter), which could account the nuclear material in DUPIC process, by introducing the Cm-244 measurement method with Los Alamos National Laboratory.

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The DSNC is a well typed-neutron coincident counter. Neutron signal obtained from DSNC is separated spontaneous neutron from (α,n) neutron through shift register. The shift register can separate coincidence neutron from non-coincidence neutron, however, it can't distinguish spontaneous fission neutron from induced fission neutron. If induced fissions are generated, it can't distinguish spontaneous fission neutron from induced fission neutron. Therefore, it is possible to overestimate mass of nuclear material because the coincidence neutrons from the induced fissions measured by shift register cannot be directly distinguished from the spontaneous fission neutrons[1-2].

In general, coincident neutron measurement by the shifter resister uses the following two parameters (singles and doubles rate) :

$$S_C(M_L) = \varepsilon M_L F_S \overline{v_{S(1)}}(1+\alpha) \tag{1}$$

$$D_C(M_L) = \varepsilon^2 f M_L^2 F_S[\overline{v_{S(1)}} + \frac{M_L - 1}{\overline{v_{I(1)}} - 1}\overline{v_{S(s)} v_{I(2)}}(1+\alpha)] \tag{2}$$

Here S_C and D_C represent singles rate and doubles rate respectively. Other symbols are as follows;

M_L : leakage multiplication,

α : (α,n)neutron/spontaneous fission neutron,

ε : efficiency,

F_S : fission rate,

f : double gate fraction,

$\bar{v}$: factorial moment for fission neutron.

Singles rate means total neutron included (α,n), induced fission and spontaneous fission neutron and doubles rate means coincidence neutron[2].

1. Measurement method

Ensslin[3] has proposed a measurement method to correct neutron multiplication generated in the passive coincidence counter. The measurement method estimates leakage multiplication effect by detecting a single and double rate in well-characterized material and process material, and then calculating single/double ratio. In this study, the method proposed by the Ensslin was used and the procedure is as follows[3)];

1) Let the α = ratio of (α, n) neutron to spontaneous fission neutrons in a sample.
2) Take a standard sample which seems not to be resulted in multiplicity, and measure S_0, D_0 and α_0.
3) Let the dead time corrected single rate and double rate of the sample to be measured be S_C and D_C.
4) Calculate the γ from measurement of parameters as follows;

$$\gamma = \frac{\dfrac{D_C(M_L)}{S_C(M_L)}(1+\alpha)}{\dfrac{D_0}{S_0}(1+\alpha_0)} \qquad (3)$$

5) The leakage multiplication can be derived from relating γ and equations (1) and (2) as follows;

$$M_L = \frac{-B+\sqrt{B^2-4AC}}{2A} \qquad (4)$$

where, $A = k(1+\alpha)$

$B = 1-k(1+\alpha) = 1-A$

$C = -\gamma$

$$k \equiv \frac{\overline{\nu_{S(1)}}}{\overline{\nu_{I(1)}}-1}\frac{\overline{\nu_{I(2)}}}{\overline{\nu_{S(2)}}}$$

In order to use the above procedure, we performed the two kinds of experiment with DUPIC sintered pellets. Experiment 1 is the case that the geometry of samples is unknown, in which pellets for measurement are put in an arbitrary position in a measurement container. On the order hand, experiment 2 is the case when the geometry of samples is known. For this, we used a holed container which can simulate dimension of the DUPIC bundle.

2. MCNP code method

The Monte Carlo method is suited for computing coincidence response because it can yield information about events involving integral number of neutrons. The MCNP(Monte Carlo N-Particle transport) code calculates a sufficient number of neutron histories to yield a multiplication factor k_P that is then used to calculate the multiplication M from following equation[3-5)];

$$M = \frac{1}{1-k_P} \qquad (5)$$

$$M_L = M \cdot p_L \qquad (6)$$

III. Result and discussion

1. Experiment 1

Figure 1 shows the result of multiplications obtained from measurement in the case that the geometry of samples is unknown. It was indicated that it is difficult to induce the value of multiplications.

Multiplications of pellets calculated by MCNP code are illustrated in **Fig.2**. The geometry of the pellets was assumed to be the form of sphere. It shows that the multiplications are increased according to increase of the mass of pellets as expected.

The results of the experimental value in comparison with MCNP code calculation are shown in **Fig.3**. We can see that the measurement data differ largely with the simulation data by the MCNP code. The calculation data of MCNP code in **Fig.3** come from the assumption that the geometry of pellets was piled up and then formed sphere when the pellets were put into the container. Therefore, model of MCNP code is modified to simulate the size of DUPIC bundle as shown in **Fig.4**. **Figure 5** shows the multiplications obtained by MCNP code using the model of **Fig.4**. Distribution of multiplication is divided into two parts and depends on the geometry of pellets. It was indicated from comparison of **Fig 5** with **Fig. 4** that more induced fissions can generate if the pellets are lumped, compared with the form of element. Furthermore, measurement values a little differ with MCNP values because the position and geometry of samples could be various. In order to use MCNP curve for correction, we found out that more accurate calculation and experiment (Experiment 2) are needed.

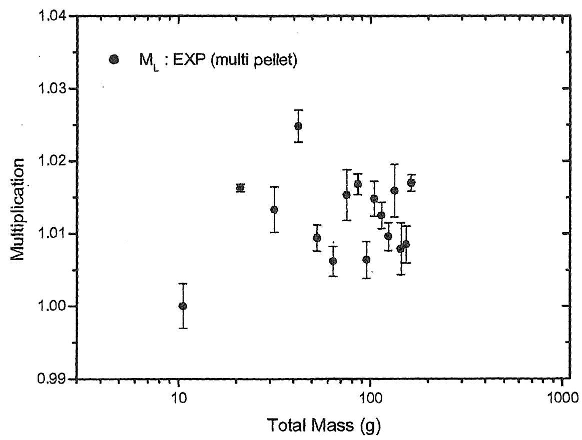

Fig.1 Leakage multiplication obtained from DSNC measurement

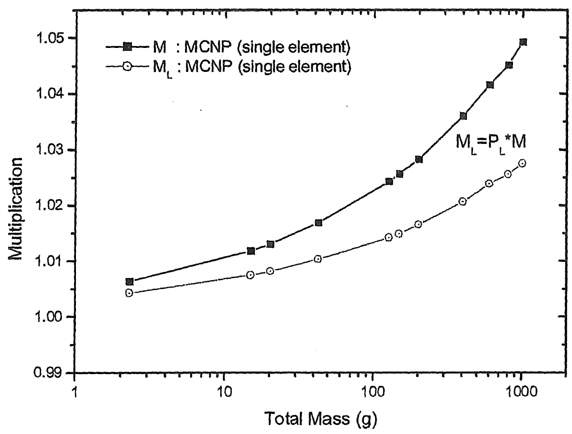

Fig.2 Total and leakage multiplication obtained from MCNP code

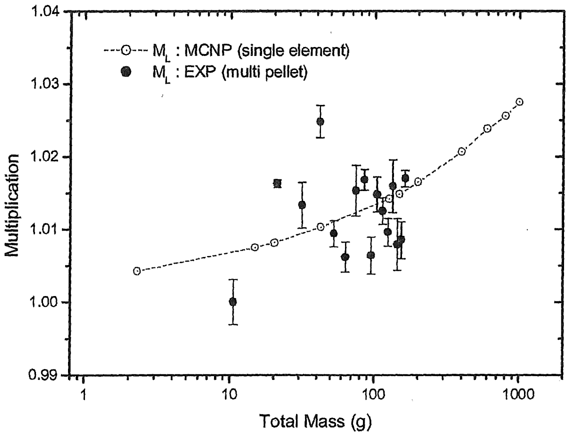

Fig.3 Comparison between DSNC measurement and MCNP calculation

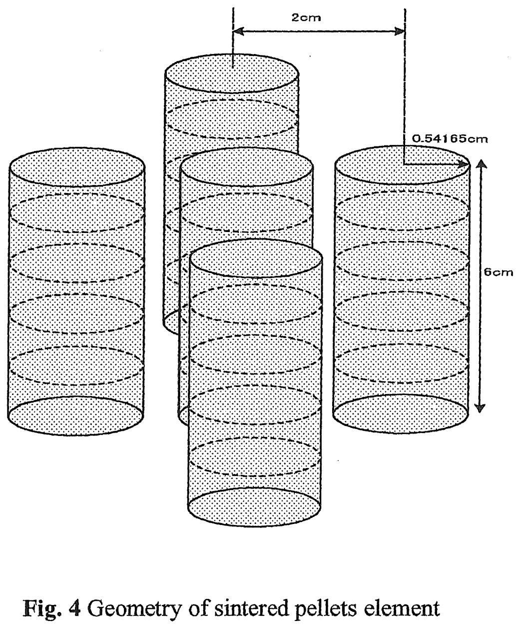

Fig. 4 Geometry of sintered pellets element

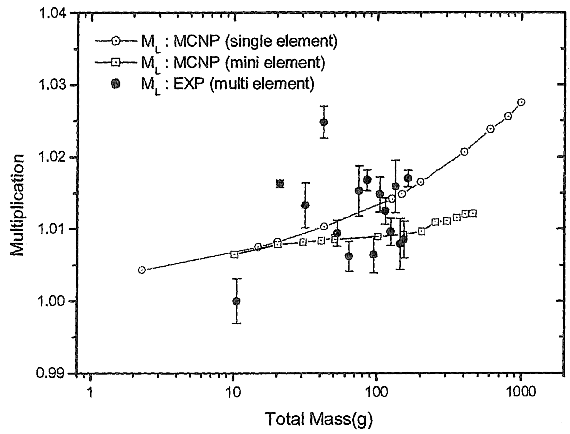

Fig.5 Comparison between DSNC measurement and Modified MCNP calculation

2. Experiment 2

The experiment 2 is the case that the geometry of samples is known. For this, we used a holed container which can simulate dimension of the DUPIC bundle. The container used in this experiment has the same geometry described in **Fig.4**. **Figure 6** shows the result of multiplications obtained by measurement method. In contrast to previous experiment, the multiplications increase as increasing the mass of pellets. However, multiplication effect is small because pellets are far apart.

The left curve of **Fig.7** shows that the comparison of measured values with code calculated values. The right curve of **Fig.7** shows the multiplication of DUPIC bundle obtained from MCNP code. It was found from the figure that the measurement results are well consistent with the MCNP code results and about 0.5% ~ 1% of coincident neutron was due to the multiplicity by induced fission.

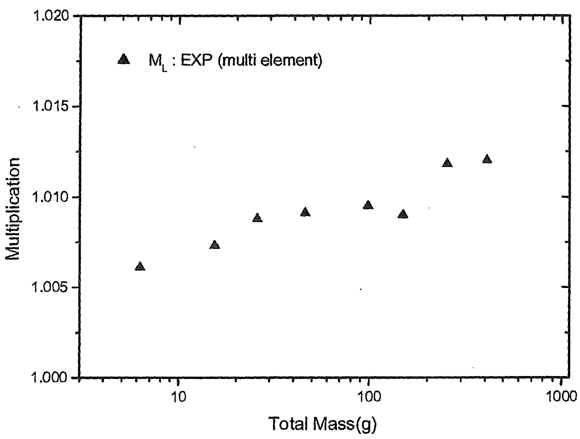

Fig.6 Leakage multiplication obtained from DSNC measurement

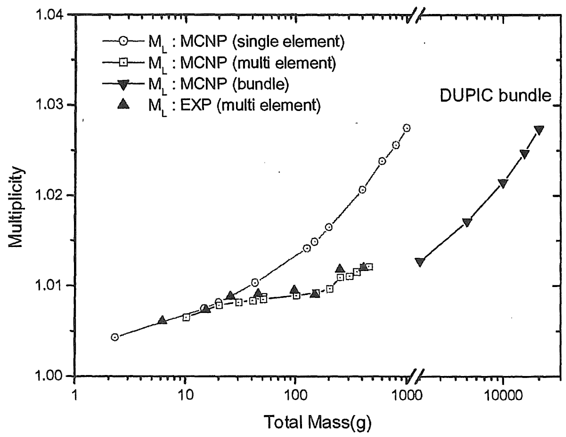

Fig. 7 Comparison between DSNC measurement (Experiment 2) and MCNP code

IV. Conclusions

In this study, the degree of neutron multiplication, which could occur in the assay sample in dry-processed spent fuel material, was investigated by measurement and simulation methods. We found that the results of the measurement method showed to be in a good agreement with those of simulation by the MCNP code. The neutron multiplicity correction curve obtained from this study could be useful for increasing measurement accuracy of the well-typed neutron accounting system.

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