Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22, 2009

The Quantity Determination for ^{235}U and ^{239}Pu in Nuclear Fuel using a Monte Carlo Simulation

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1. Introduction

During the recent decades, the measurement system for fissile isotope contents in the spent fuel from LWRs has been developed. The objective of measurement system introduced in the paper is for accurate and quantitative measurements of fissile contents in the spent fuel or the reproduced fuel such as a DUPIC and a pyro-processed fuel using a Monte Carlo method. Based on the fact that the fissile materials have different neutron fission cross sections each other, the amount of fissile materials was accurately measured depending on its isotopes from the fission detection rate in fission chamber mainly made of ^{238}U. In this study, the neutron source emits mono-energetic neutrons so that we can easily get the fission rate depending on the given neutron source. Because the method has a limitation that the neutron source intensity should be larger then the back ground neutron intensity from Curium in a spent fuel, back ground neutron intensity is neglected. The method is mainly calculated by the Monte Carlo simulation and if neutron sources generating mono-energy with enough large intensity are available, the method will be applied in various fields of nuclear engineering or the fields associated with fuel managements.

The previous method [1] only treated with two contents, ^{235}U and ^{239}Pu, using a slowing-down neutron energy in the lead medium. In the study, the analytic formulation is generalized and extended so that it can be used to measure the quantities of fissile contents depending on the number of fission neutrons with given neutron energies.

The fission chamber consists of ^{238}U, which has threshold energy (0.1MeV) to count the fission neutrons. Therefore, the chamber detects fast fission neutrons. To demonstrate the method, a Monte Carlo code, MCNPX, is used. The simulation is conducted by the following procedure: i) neutron source with 0.025eV is emitted, then fission rates are counted, ii) neutron source with 0.3eV is emitted, then the fission rates are counted, iii) it is conducted to reference calculation with known fuel contents to know normalizing factors. From these calculations, the quantities of fuel contents such as ^{235}U and ^{239}Pu are obtained.

When the conceptual method is extended to N types of contents in the fuel, the accurate quantities for unknown contents can be acquired. In addition to that, it gives accurate quantities of unknown fissile contents in the use of LSDTS (Lead Slowing Down Time Spectrometer).

2. Methodology

The fission neutron counting in the detector is characterized as the following formulations depending on the incoming neutrons (source neutrons) as shown in Eq. (1).

$$\begin{bmatrix} D_1 \\ D_2 \\ \vdots \\ D_{j-1} \\ D_j \end{bmatrix} = \begin{bmatrix} B_1a_{1,1} & B_1a_{2,1} & \cdots & B_1a_{i-1,1} & B_1a_{i,1} \\ B_2a_{1,2} & B_2a_{2,2} & \cdots & B_2a_{i-1,2} & B_2a_{i,2} \\ \vdots & \vdots & \ddots & \vdots & \vdots \\ B_{j-1}a_{1,j-1} & B_{j-1}a_{2,j-1} & \cdots & B_{j-1}a_{i-1,j-1} & B_{j-1}a_{i,j-1} \\ B_ja_{1,j} & B_ja_{2,j} & \cdots & B_ja_{i-1,j} & B_ja_{i,j} \end{bmatrix} \begin{bmatrix} N_1 \\ N_2 \\ \vdots \\ N_{i-1} \\ N_i \end{bmatrix} \quad (1)$$

where a_{ij} is defined as

$$a_{ij} \equiv \int \sigma_{fi}(E) \cdot \phi_j(E)dE, \quad (i, j = 1, \cdots, N) \qquad (2)$$

and $\sigma_{fi}(E)$ is a fission cross section for i-th fuel content with a given incoming source neutron and $\phi_j(E)$ is a neutron flux in a j-th neutron energy in the unknown fuel. D_j and N_i mean a fission neutron counting at the fission chamber using j-th incoming neutron source and an atomic density of i-th contents in the unknown fuel, respectively. B_j is a normalizing factor for the neutron intensities.

In order to find atomic densities in the unknown fuel, it is necessary to obtain normalizing factors using the Monte Carlo simulations for non-irradiated fuel. Eq. (3) is applied to get the normalizing factors, B_j .

$$\begin{bmatrix} B_1 \\ B_2 \\ \vdots \\ B_{j-1} \\ B_j \end{bmatrix} = \frac{1}{\tilde{N}} \begin{bmatrix} D_{10}/a_1 \\ D_{20}/a_2 \\ \vdots \\ D_{j'-1}/a_{j-1} \\ D_{j'}/a_j \end{bmatrix} \qquad (3)$$

where $D_{j'}$ is a fission neutron counting in the fission chamber, $\tilde{N}$ is an atomic density, and $a_j \equiv \int \sigma(E) \cdot \phi_j(E)dE$ which is caused by known (non irradiated) fuel. Since the method uses mono-energetic neutron source, it is easy to know fission rates and neutron fluxes in the fuel in the MCNPX simulation.

The normalizing factor, B_j, is substituted into Eq. (1) and then N_i is easily obtained by matrix inverse.

3. Simple Problem Description

A simple problem geometry is shown in Fig. 1. The unknown fuel represented as a blue one is located at the central part of the problem domain. The detectors represented as red boxes occupy surrounding the fuel. Neutrons as a point source are emitted with mono-direction and mono energy to the fuel at the middle point between two detectors. Except for the unknown fuel and detectors, the rest area is vacant.

The calibration samples are defined as shown in Table I. The nuclear fuel is mainly consists of ^{238}U over 96 percent. The weight percent ^{235}U and ^{239}Pu is changed from 1 to 2.5. Samples 1, 2, and 3 as non irradiated fuel are used to obtain the normalizing factors. In the simple problem, fissile contents such as ^{235}U and ^{239}Pu are used as unknown contents.

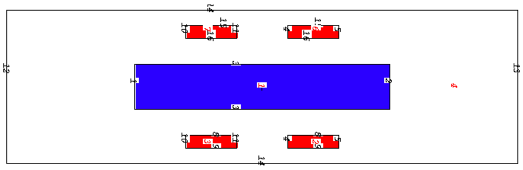

Fig. 1. The problem geometry.

Table I: The Contents Composition for Calibration Samples

No	^{235}U (wt%)	^{239}Pu (wt%)	^{238}U (wt%)	
1	1	0	99	Known fuel
2	1.5	0	98.5	
3	2.5	0	97.5	
4	1	1	98	Unknown fuel
5	1	0.6	98.4	
6	1	0.3	98.7	
7	1.5	1	97.5	
8	1.5	0.6	97.9	
9	1.5	0.3	98.2	
10	2.5	1	96.5	
11	2.5	0.6	96.9	
12	2.5	0.3	97.2	

4. Results

The Table II shows Monte Carlo simulation results using the MCNPX code. The quantity of fissile contents as ^{235}U and ^{239}Pu is provides good results within relative maximum error of 6% in the case of 235U and 14% in the case of 239Pu.

Table II: The Fissile Contents of the Unknown Fuel

No	^{235}U (10^{20}/cm^3)	^{235}U Error (%)	^{239}Pu (10^{20}/cm^3)	^{239}Pu Error (%)	^{235}U Ratio[a]	^{239}Pu Ratio[a]
4	4.89	0.01	4.99	0.00	0.98	0.98
5	5.15	5.25	3.36	12.03	1.63	1.54
6	5.00	2.22	1.70	13.79	3.27	2.94
7	7.77	5.86	5.56	11.42	1.47	1.40
8	7.59	3.43	3.36	12.20	2.45	2.26
9	7.45	1.50	1.70	13.72	4.90	4.38
10	1.26	3.25	5.60	12.11	2.45	2.26
11	1.25	1.80	3.38	12.74	4.09	3.69
12	1.23	0.77	1.71	13.94	8.17	7.23

[a]: (N235/N239 in calibration sample) / (N235/N239 in Monte Carlo result)

5. Conclusions and Future Works

The non-destructive determination for fissile material in unknown fuel is introduced based on Ref. 1. Using the Monte Carlo simulation, the accurate quantity of fissile contents in spent fuel is obtained giving very accurate quantity especially for 235U. As future works, using extended formulations, Eq. (1) and Eq. (3), it will be performed to find the others fissile elements such as ^{241}Pu.

REFERENCES

[1] H. KRINNINGER, S. WIESNER and C. FABER, Nuclear Instruments and Method, 73, 13, North-Holland Publishing, City (1969).