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Active NDA Neutron Measurement Method to Determine Fissile Contents of Fuel Material

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Abstract

Active NDA neutron measurement method has been developed to use in determination of fissile contents for fuel materials. This method can be applied to fuel design, fabrication and its safeguard implementation. The change of neutron count between the induced and non-induced fission by neutron source in fuel material was analyzed. Results from MCNP calculation model for the singles and doubles rates are compared with NDA experimental measurements at KAERI. It shows that the measured neutron count ratio versus quantity of UO_2 enrichment is reasonably well agreed with the calculated values.

1. Introduction

There are several methods to examine the fissile material contents, homogenity and characteristics of fuel materials. The neutron measurement methods have been investigated to measure the contents of special nuclear material of U-235 and Pu-239 under safeguard technology development [1-5]. However, the existing NDA method is not available to determine the unknown fissile contents of fuel material contained specially in mixed U-235 and Pu-239 nuclear material.

In order to determine the fissile contents of fuel material, the NDA neutron measurement method has been studied by applying a neutron count ratio. The fissile content of fuel material is measured by neutron counting ratio due to induced fission dependent on the contents of various fuel materials. The primary difficulty is to determine the fissile material contents in the presence of U-235 and Pu-239 contained in the fuel. The MCNP code[6] was used to design the fissile material measurement model for examination of fissile contents in fuel material. The effects of neutron count ratio versus the fissile contents of fuel material are investigated by using the MCNP model.

2. Theoretical Concept of NDA Measurement Method

The neutron sources are more important in active nondestructive assay(NDA) measurements. The neutron sources originate from the spontaneous fission as well as some (α,n) reaction neutrons for the fuel materials. The spontaneous fission and (α,n) neutron source terms are dependent on kind of isotope and decay time. The energy of the neutron emitted in an (α,n) reaction depends on the energy that the alpha particle has at the time of the reaction and on the Q-value of the reaction in the isotope. An important characteristic of neutrons from (α,n) reactions is that only one neutron is emitted in each reaction. These events constitute a neutron source that is random in time with a multiplicity of ν=1. The AmLi source is a typical (α,n) reaction neutron source for active NDA measurements.

The dominant source term of neutrons is spontaneous fission from Cf-252 for active NDA measurements. Fig. 1 shows an energy spectrum of the neutrons emitted during the spontaneous fission of Cf-252. The mean energy is 2.14 MeV. The spectrum depends on many variables such as fission fragment excitation energy and average total fission energy release, but can be approximated by a Maxwellian distribution N(E),

$$N(E) \approx \sqrt{E}\ \exp(-E/1.43MeV) \qquad (1)$$

This spectrum is proportional to $E^{1/2}$; it then falls exponentially at high energies. Table 1 summarizes some of properties of Cf-252. For active NDA applications it is important to remember that Cf-252 neutrons are emitted with an average multiplicity of ν=3.757. Thus they are strongly correlated in time and will generate coincidence events.

However, there are an additional neutron source produced from the multiplication process from fuel materials. This multiplication is significantly increased when the fuel materials is measured under moderator material such as water, graphite and polyethylene. The AmLi (α,n) reaction and Cf-252 spontaneous fission neutrons will be used as active neutron driving term. The U-235 and Pu-239 fissile contents determine the amount of neutron multiplication. The change of neutron count ratio called as the neutron multiplication is measured as induced fission neutrons of fissile material in fuel materials with AmLi (α,n) neutron source and Cf-252 spontaneous fission source.

The Bohnel point model equations[2] provide a means of predicting an observed neutron count rate from fuel material. The point equations for the real coincidence count rate(doubles rate), and total count rate(singles rate) are summarized below. The singles count rate S and the doubles count rate D are given by

$$S = \varepsilon M_L F_s \nu_{s1}(1+\alpha) \tag{2}$$

$$D = \varepsilon^2 M_L{}^2 f F_s [\nu_{s2} + \frac{M_L - 1}{\nu_{i1} - 1} \nu_{s1} \nu_{i2}(1+\alpha)] \tag{3}$$

where,
S = *Singles count rate*
D = *doubles count rate*
ε = *detector efficiency*
M_L = *leakage multiplication of fuel material*
ν_{s1} = *1st spontaneous fission moment (n/spon. fiss.)*
ν_{i1} = *1st induced fission moment (n/ind. fission)*
α = *ratio of (alpha, n) emission to spontaneous fission*
f = *fraction in the doubles gate*
ν_{s2} = *2nd spontaneous fission moment.(n/s. fiss.)*
ν_{i2} = *2nd induced fission moment (n/ind. fiss.)*

The concept theory for fissile content measurement is to use a neutron counting ratio in terms of the Cadnium(Cd) ratios to separate the primary emission neutrons from secondary fission neutrons induced in the fissile material. Therefore, fissile material content measurement was based on the leakage multiplication theory in the fuel material[2]. One of the initial assumptions in the point model is that all of the neutrons under consideration are born at the same point in time.

The change of Cd ratio due to induced fission dependent on the contents of various fuel materials was proposed to determine the fissile content of fuel material. The Cd ratios means to measure neutron count for fuel material with removable Cd shutter between the fuel rods and moderator, and then to measure total neutrons without Cd shutter. The effects of Cd ratios varied with fuel fissile material. The relationship between neutron count rate with and without Cd could be expressed as follows,

$$SCR = \frac{S_0}{S_{Cd}}, \quad DCR = \frac{D_0}{D_{Cd}} \tag{4}$$

where, SCR and DCR are the Cd ratio for singles and doubles neutron count rate measured in all detectors. The Cd ratio(SCR and DCR) depends on the size, mass, density, Pu-239 and U-235 enrichment.

Table 1 Characteristics of Cf-252 neutron source

Total half-life	2.646 yr
Alpha half-life	2.731 yr
Spontaneous fission half-life	85.5 yr
Neutron yield	2.34×10^{12} n/s-g
Gamma-ray yield	1.3×10^{13} γ/s-g
Alpha-particle yield	1.9×10^{13} α/s-g
Average neutron energy	2.14 MeV
Average gamma-ray energy	1.0 MeV
Average alpha-particle energy	6.11 MeV
Neutron activity	4.4×10^{9} n/s-Ci
Neutron dose rate	2300 rem/h-g at 1m
Gamma dose rate	140 rem/h-g at 1m
Average spontaneous fission neutron multiplicity	3.757

3. Test Model for Fissile Content

The test modelling was to develop the MCNP code simulation capable to measure the neutron counting ratio due to the induced fissions. Some simplications of the geometry in the Monte Carlo model were used for fissile content measurement using the MCNP code. Fig. 2 shows a horizontal and vertical view of the fissile content test model.

For comparison with the Monte Carlo calculations, a series of UO_2 powder can were measured with DSFC(DUPIC Safeguard Fissile Counter) which was developed at KAERI. The fuel material in the cavity are composed of UO_2 powder can with 13 cm in length and with 3.8 cm in diameter. These are made by selecting a series of enrichment from 0.71 to 4.1 % and then placed into encapsulated by stainless steel can. The polyethylene reflector is placed between powder can and inner stainless steel shell. The neutron multiplication in the fuel UO_2 powder is caused by thermal neutron which the fast neutrons due to primary spontaneous fission and (α, n) emmision are moderated in poly reflector.

The Cd shutter between the UO_2 can and poly is placed and removed for measuring Cd ratios. The thick lead layer gives gamma-ray shielding of the detector tubes for protecting from gamma emission of AmLi source. Air gap is outside lead shield. The poly encased with stainless steel shell has 32 holes for He-3 detector tubes which can detect neutrons by (n, p) reaction. The test model I for AmLi (α, n) neutron source has 16 He-3 detector tubes. The neutron detector tubes have approximately 50 cm long enough to get the constant response for all long fuel. Poly reflector is also placed at the bottom of the neutron source.

The experimental tests are carried out by Test model I with 16 He-tubes using AmLi neutron source. Total neutrons were measured from empty can to 4.1 % powder using 16 He-3 tubes.

4. Results and Discussions

The fissile content for fuel material has been studied by the comparison of the experiment measurement between the MCNP calculations. And this method would be utilized in determining the total fissile content in a given sample. The change of Cd ratio due to induced fission of fissile material was suggested to determine the fissile content of fuel material sample.

Fig. 3 shows a comparison of the measured and calculated Cd ratio versus UO_2 powder enrichment by using AmLi neutron source. Two sets of curves are shown. The upper set of two curves are representative with the calculated and experimental Cd ratios. The Cd ratio is varied from 3 to 5 % constant difference. The lower set of two curves is shown with the normalized Cd ratio. Here the plot is in good agreement within statistical errors (standard deviation) 3 % for the experimental value. Fig. 4 shows plots of both singles and doubles rate versus UO_2 powder enrichment in fissile content test model. The Cd ratio for singles rate is slightly increased as powder enrichment. The plots clearly show the difference in singles and doubles rate. The doubles rate show better sensitivity to fissile content changes.

5. Conclusion

A significant effort was required to prepare the fissile content measurement test model at KAERI. A MCNP modelling for fissile content test was successfully accomplished with DSFC. The difference between the measured and the calculated values in singles rate could be resolved by increasing the measured time. To determine the fissile contents in fuel material, the Cd ratio by NDA neutron count is considered to be an appropriate method. To enhance accuracy of the method for predicting the fissile content, the passive and active measurement method is continually developed by further new model.

References

1. M.S. Krick, "Neutron Multiplication Corrections for Passive Thermal Neutron Well Counters," LA-8460-MS, Los Alamos National Laboratory(1980)
2. K. Bohnel, "The Effect of Multiplication on the Quantitative Determination of Spontaneously Fissioning Isotopes by Neutron Correlation Analysis," Nuclear Science and Engineering, 90, 75-82(1985)
3. H.O. Menlove, et al. "CANDUMOX(CMOX) Counter Design and Operation Manual," LA-12192-M, Los Alamos National Laboratory(1991)
4. H.O. Menlove, et al. "Plutonium Scrap Multiplicity Counter Operation Manual," LA-12479-M, Los Alamos National Laboratory(1993)
5. M.S. Krick, et al., "Active Neutron Multiplicity Analysis and Monte Carlo Calculations," LA-UR-94-2440 Los Alamos National Laboratory (1994)
6. J.F. Briesmeister, Ed., "MCNP - A General Purpose Monte Carlo Code for Neutron and Photon Transport," LANL report La-12625-M, Ver. 4A(1993)

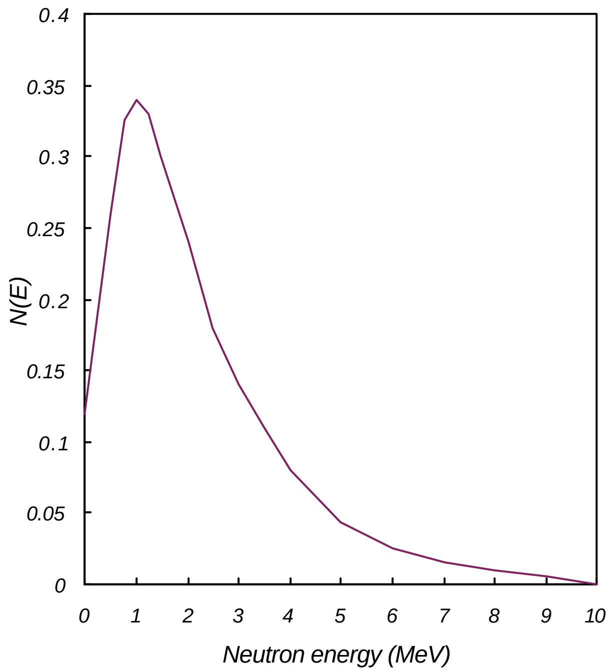

Fig. 1 Prompt neutron spectrum from the spontaneous fission of Cf-252 source

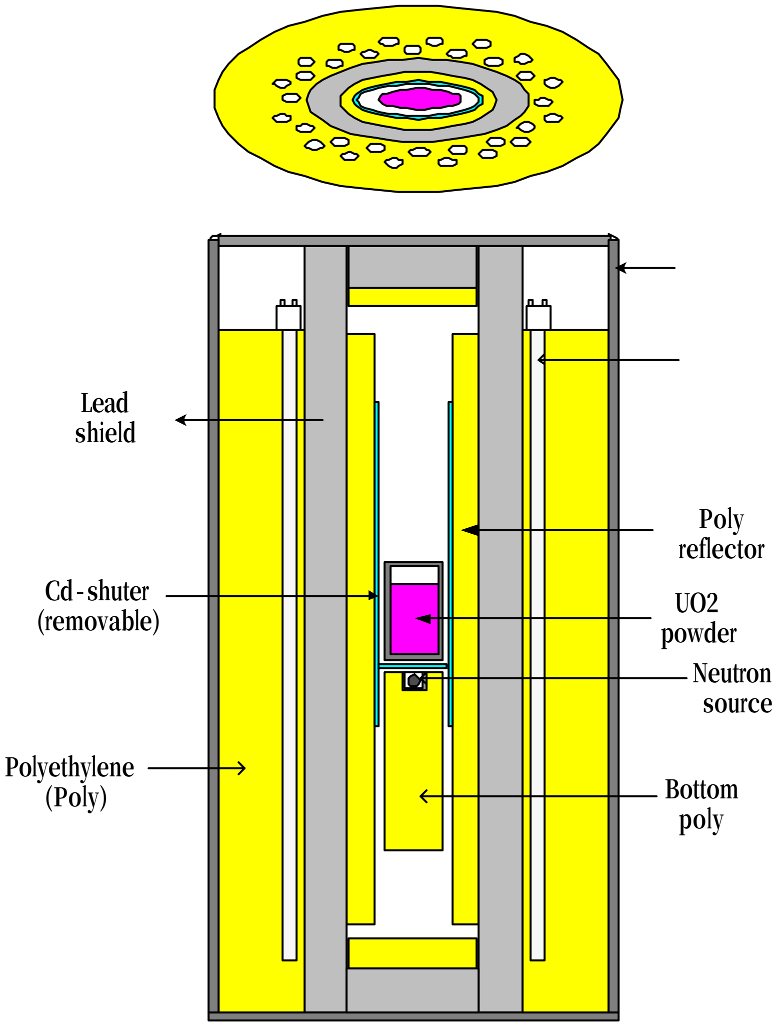

Fig. 2 Fissile content measurement test model
for fuel materials

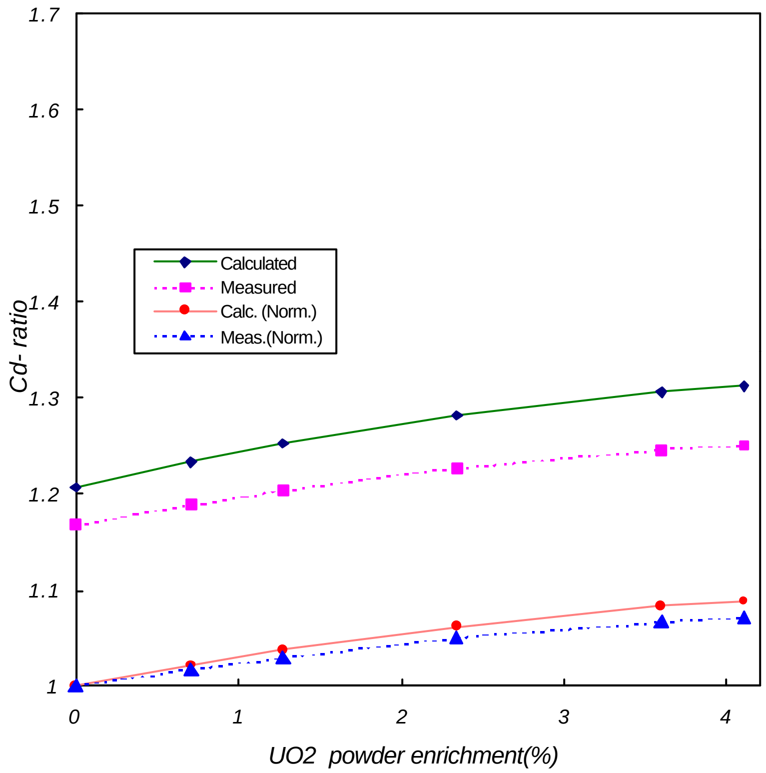

Fig. 3 Cd ratios versus UO2 powder enrichment

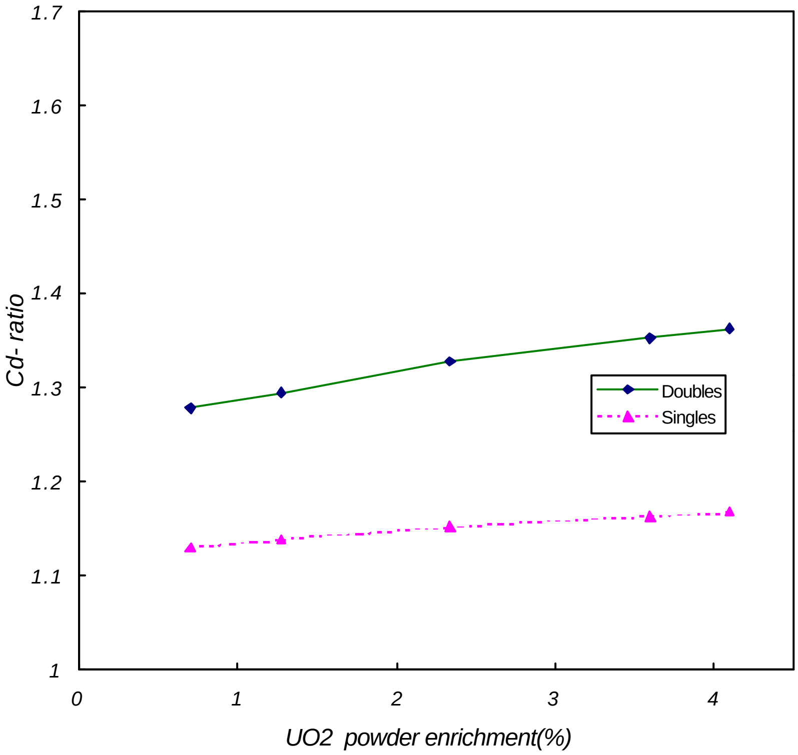

Fig. 4 Cd ratio curve for single and double rate versus UO2 enrich. by Cf-252 source