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Enrichment Measurement Development of Fuel Material by NDA Neutron Counter

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Abstract

A neutron count method has been developed to determine the fissile content by NDA neutron counter. Active neutron multiplicity measurements to assay enrichment of uranium powder have been carried out. The total neutron counts between the induced and non-induced fission by neutron source in sample were measured. These results are compared to equivalent results obtained from MCNP calculation. It shows that the measured neutron count ratio versus quantity of UO_2 enrichment is reasonably well agreed with the calculated values.

1. Introduction

Passive and active NDA neutron counting method were developed for the assay of nuclear fuel material[1-6]. Active neutron measurement technique would be useful to determine the amount of fissile content in the fuel material. This is sensitive to the enrichment, density and material composition of samples. Active neutron multiplicity counting has become a nondestructive analysis technique for the assay of UO_2 powder samples which characteristics are well known. The measured total count rates from a sample are used to solve for neutron multiplication from (alpha, n) neutron yield.

In order to determine the enrichment and fissile contents of fuel sample, the NDA neutron measurement method has been applied by 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 fissile materials. The MCNP code[7] was used to calculate the neutron multiplicity count model for examination of fissile contents in fuel material sample. Because of neutron absorption and multiplication in uranium powder samples, the neutron count are effected on the geometry, powder density and enrichment. The Monte Carlo calculations by MCNP code are compared with the measurement of neutron counts using the NDA neutron counter.

2. Active NDA Measurement Method

The neutron sources are more important in active nondestructive assay 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. This characteristics is exploited by neutron coincidence counters which can distinguish between spontaneous fission neutrons and neutrons from (α,n) reactions. The AmLi source is a typical (α,n) reaction neutron source for active NDA measurements.

The active source term of neutrons is the (α,n) reactions for active NDA measurements. Fig. 1 shows an energy spectrum of the neutrons emitted the AmLi source. The mean energy is 0.3 MeV.

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 will be used as active neutron driving term. The U-235 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 in fuel materials with AmLi (α,n) neutron source.

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

$$S = \varepsilon M_L F_a \nu_{s1} \qquad (1)$$

where,

$S =$ *Singles count rate*

$\varepsilon =$ *detector efficiency*

$F_a =$ *AmLi neutron source yield(neutrons/sec)*

$M_L =$ *leakage multiplication of fuel material*

$\nu_{s1} =$ 1*st reduced moments of the first−fission neutron distribution(n/ fission)*

The concept for fissile measurement in fuel sample is to use a neutron counting ratio in terms of the Cd ratios to separate the primary emission neutrons from secondary fission neutrons induced in the fissile material. When the Cd plate inside poly reflector is used in neutron counter, the induced fission is not generated. The fissile 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,

$$CR = \frac{S_0}{S_{Cd}} \qquad (2)$$

where, CR is the Cd ratio for singles neutron count rate measured in all detectors. The Cd ratio(CR) depends on the size, mass, density and enrichment.

3. Neutron Counter Measurement Model

The neutron counter model 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 neutron counter measurement using the MCNP code. Fig. 2 shows a horizontal and vertical view of the neutron counter model.

For comparison with the MCNP neutron calculations, a series of UO_2 powder can were measured with Fissile Neutron 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 (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 He-3 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 measurement model 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 Fissile Neutron Counter with 16 He-tubes using AmLi neutron source. Total neutrons were measured by using 16 He-3 tubes. The MCNP calculations were compared with experimental measurements. A series of measurements were done from the empty can to 4.1% enriched powder can.

4. Results and Discussions

For comparison with the neutron counter model calculations, a series of uranium oxide powder were measured in fissile neutron counter. The fissile content has been studied by using neutron count ratio based on multiplicity of induced fission. And this method would be utilized in determining the total fissile content in a given sample. The Cadnium ratio due to induced fission of fissile material was suggested to determine the fissile content of fuel material sample.

The Cd ratios for singles rate is slightly increased as powder enrichment in Fig. 3 It shows a comparison of the measured and calculated Cd ratio versus UO_2 powder enrichment by using AmLi neutron source. The set of two curves are representative with the calculated and experimental Cd ratios. These Cd ratios are varied within 3 % difference. The 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. The enrichment measurement by using neuron count could be available to assay powder enrichment.

5. Conclusion

We have been carried out to study the enrichment measurement by using neutron counter. A MCNP calculation and experimental measurement was successfully accomplished with fissile neutron counter at KAERI. The difference between the measured and the calculated values in singles rate could be resolved by increasing the measured time. To determine the enrichment in fuel material, the Cd ratio by NDA neutron count is considered to be an appropriate method. To enhance accuracy of the measurement method for predicting the enrichment and fissile content, the passive and active NDA neutron count method is continually developed by further study.

References

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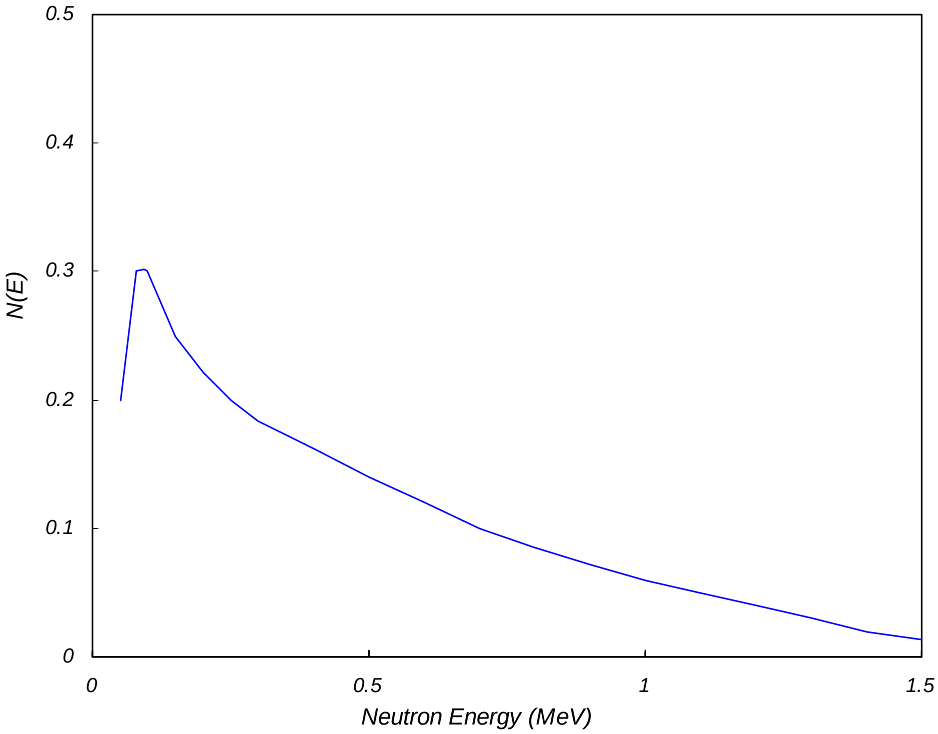

Figure 1. Neutron energy spectrum of an AmLi source

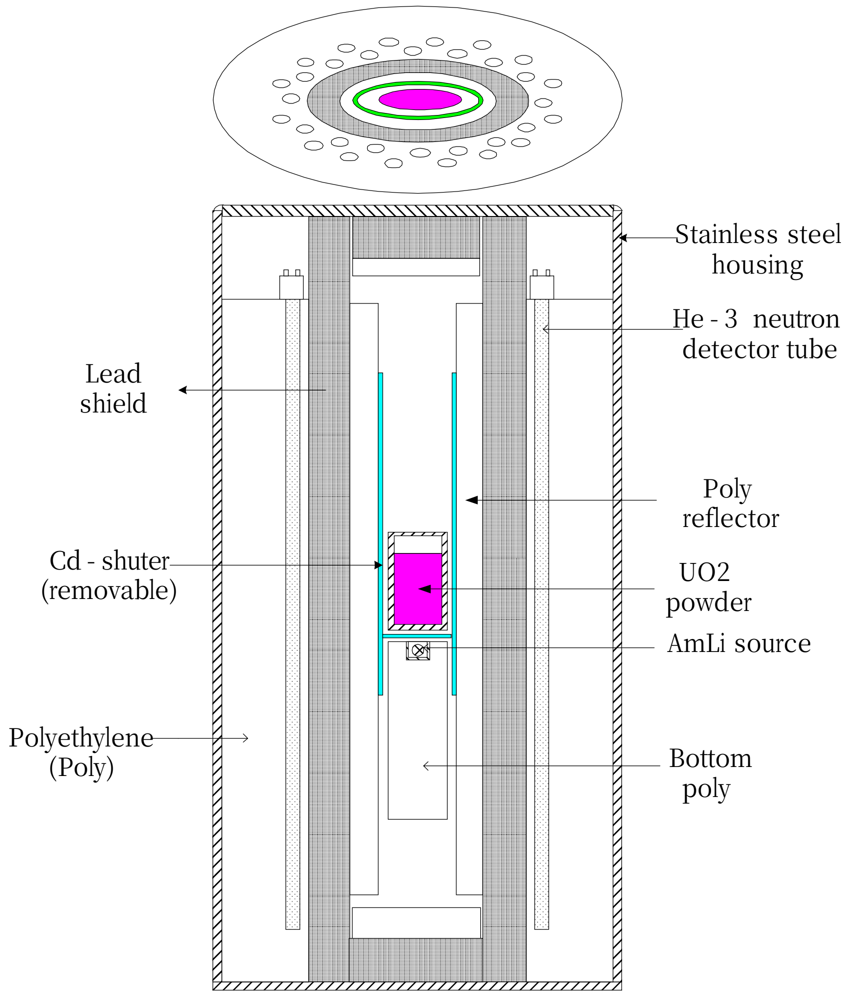

Figure 2 Active fissile neutron counter

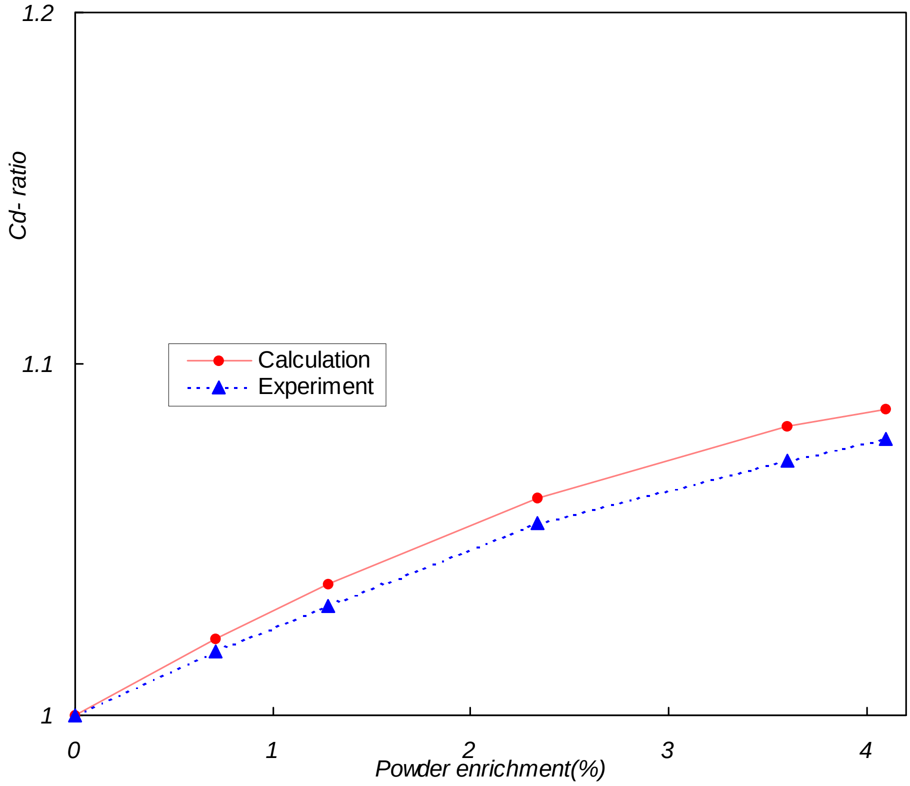

Figure 3 Cd ratios versus uranium powder enrichment