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Fissile Measurement of Fuel Material by Active Neutron Source

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Abstract

A fissile content measurement has been developed to determine the fissile content in fuel material by active neutron source. Active neutron multiplicity measurements to assay enrichment of oxide 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 with equivalent results obtained from MCNP calculation. It shows that the measured neutron counts versus quantity of enrichment agreed reasonably well with the calculated values.

1. Introduction

Active neutron counting method have been developed to assay the nuclear material content in fuel cycle[1-6]. The 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 whose 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 of a fuel sample, the 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 a fuel material sample. Because of neutron absorption and multiplication in oxide uranium powder samples, the neutron count are effected on the geometry, powder density and enrichment. The Monte Carlo calculations by the MCNP code are compared with the measurement of neutron counts using the well type neutron counter.

2. Measurement Method

The neutron sources are more important in active nondestructive assay measurements. The neutron sources originate from 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 the 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 $\nu=1$. This characteristic 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. The energy spectrum of the neutrons emitted the AmLi source and the mean energy is 0.3 MeV.

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

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

The concept for fissile measurement in fuel samples is to use a neutron counting rate in terms of the counting rate to separate the primary emission neutrons from secondary fission neutrons induced in the fissile material. When the Cd tube inside the 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]. The change of counting rate due to induced fission dependent on the contents of various fuel materials was proposed to determine the fissile content of fuel material.

Some simplifications of the geometry in the Monte Carlo model were used for neutron counter measurement using the MCNP code. Fig. 1 shows a horizontal and vertical view of the neutron counter model for comparison with experimental measurements.

3. Experimental Measurement by Neutron Counter

Total neutron counting accepts all pulses arising from neutron reactions in the sensitive volume of a neutron detector. A simple total neutron counting system consists of the components such as detector, preamplifier, amplifier, integral discriminator and scaler. The fuel material in the cavity is composed of UO_2 powder cans with 13 cm in length and with 3.8 cm in diameter. These are made by selecting a series of enrichments from 0.71 to 4.1 % and then placed into encapsulated stainless steel can shown by Fig. 2. The polyethylene reflector is placed between the powder can and the 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) emission are moderated in poly reflector.

The Cd shutter between the UO_2 can and poly is placed for measuring no fission ratio. The thick lead layer gives gamma-ray shielding of the He-3 tubes for protection from gamma emission of AmLi source. The air gap is outside the 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 32 detector tubes. The neutron detector tubes are 50 cm long, enough to get the constant response for all long fuel material sample.

Poly reflector is also placed at the bottom of the neutron source.

The experimental measurements are carried out by a Fissile Neutron Counter with 32 He-tubes using an AmLi neutron source. Total neutrons were measured by using 32 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 rate based on multiplicity of induced fission. And this method would be utilized in determining the total fissile content in a given sample. The neutron counting rate due to the variation of induced fission was suggested to determine the fissile content of the fuel material sample.

The measured singles rates(total neutron count rate) in neutron counter is slightly increased as UO_2 powder enrichment in Fig. 3. Fig. 4 shows a comparison of the measured and calculated values versus UO_2 powder enrichment by using AmLi neutron source. The set of two curves are representative with the calculated and experimental count rate. These Cd ratios 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 3 % for the experimental value. The fissile content measurement by using neutron counter could be available to assay powder enrichment.

5. Conclusion

The fissile content measurement were carried out to study the enrichment measurement by using neutron counter. A MCNP calculation and experimental measurement was successfully accomplished with the fissile neutron counter. 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 neutron counter 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 neutron count method will be continually developed by further study.

References

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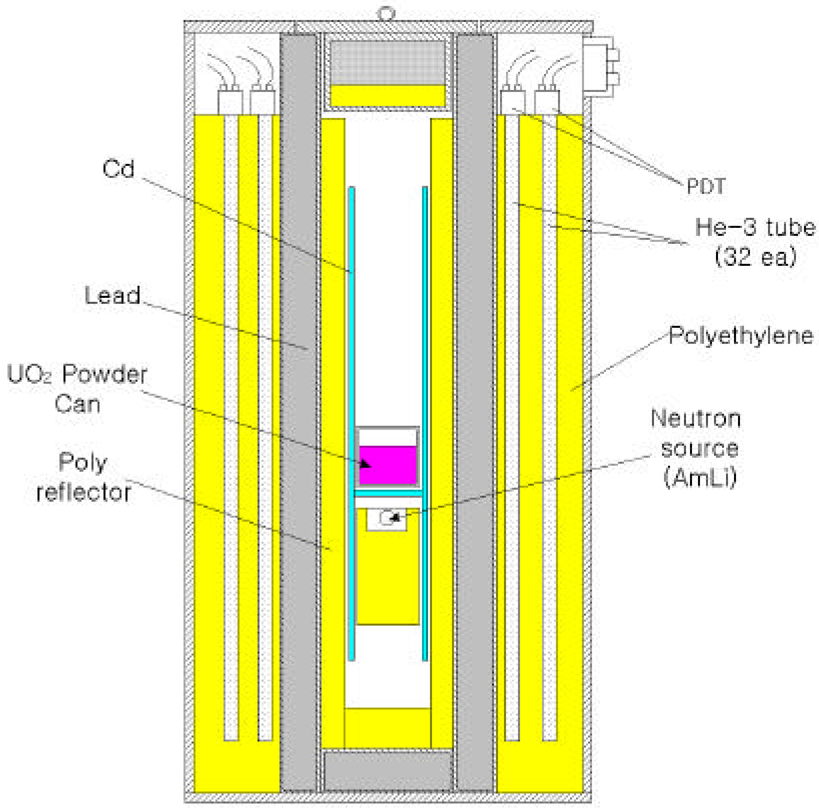

Figure 1 Fissile neutron counter model for fissile content measurement

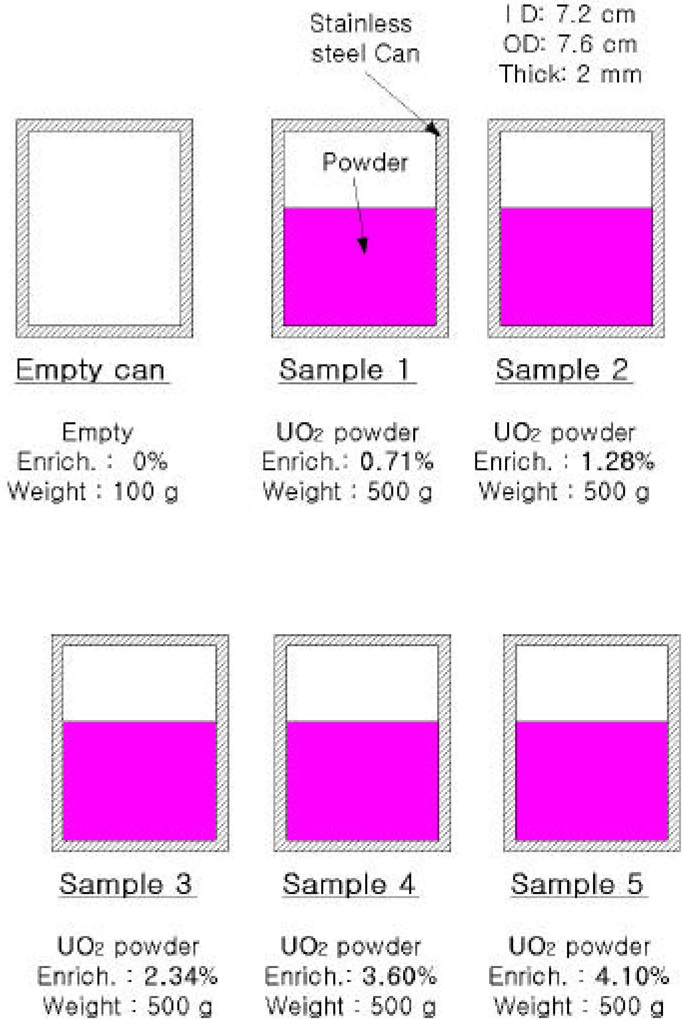

Figure 2 UO2 powder sample can for enrichment measurement

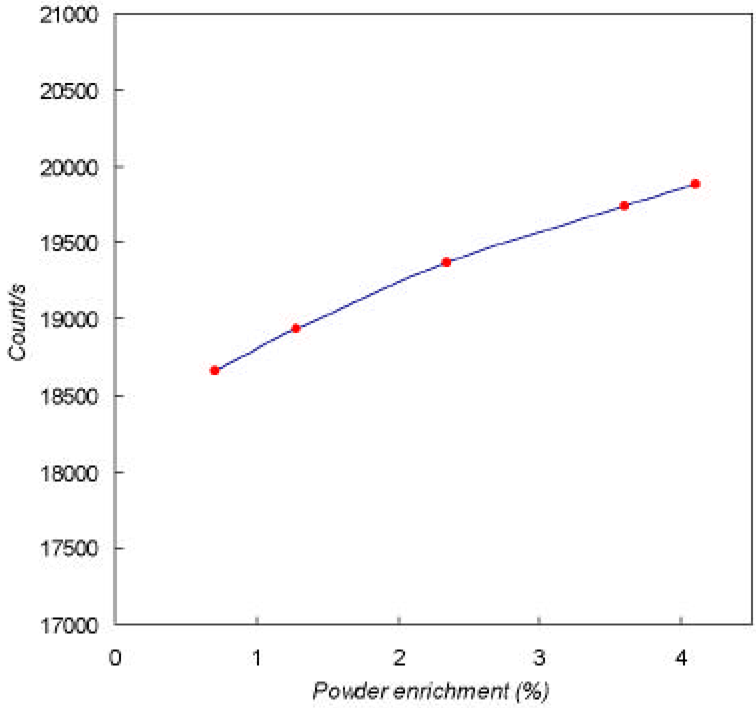

Fig. 3 Total count rate versus UO_2 powder enrichment by AmLi source in fissile neutron counter

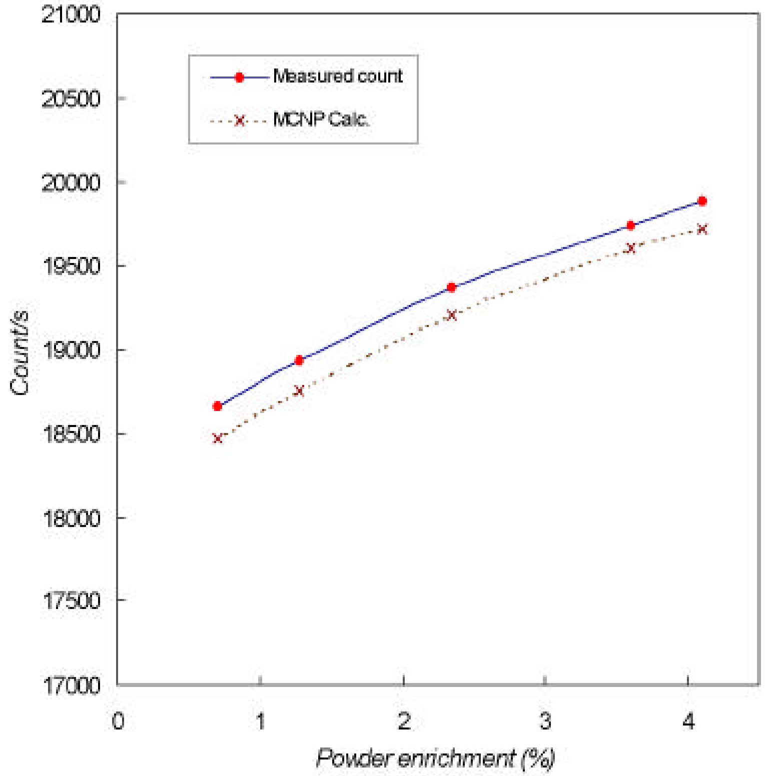

Fig. 4 Comparison of count rate between the measured and calculated value versus UO_2 powder enrichment