Neutron Count Rate Measurement of UO_2 powder by Neutron Source

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Abstract

Neutron count rate measurements to assay fissile content of uranium powder have been carried out in a neutron counter. The induced fission neutrons by Cf-252 neutron source are counted as the variation of fissile material in fuel material. The measured counts are compared with equivalent results obtained from calculation. It shows that the measured neutron counts versus quantity of UO_2 powder enrichment agreed reasonably well with the calculated values.

Keyword : Neutron count rate, Uranium oxide, Enrichment

1. Introduction

Neutron detectors do not usually preserve information about energy of the detected neutrons. Consequently, neutron assay consists of counting the number of emitted neutrons without knowing their specific energy. It can be obtained a neutron signal that is proportional to the quantity of the isotope to be measured. Active NDA neutron counting method were developed for the assay of nuclear fuel material[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 and coincidence count rates from a sample are used to solve the neutron multiplication from the spontaneous fission neutron yield.

In order to determine the fissile contents of a fuel sample, the NDA neutron measurement method has been applied by the variation of neutron counts. The fissile content of fuel material is measured by neutron counts 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 the examination of fissile contents in a 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 calculations by MCNP code are compared with the measurement of neutron counts using the neutron detector.

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

The concept theory for fissile content measurement is to use a neutron counting ratio in terms of the multiplicity 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 multiplicity ratio due to induced fission dependent on the contents of various fuel materials was proposed to determine the fissile content of fuel material. The multiplicity ratios means to measure neutron count for fuel material with removable Cd shutter between the fuel rods and moderator, and then to measure neutrons without Cd shutter. The effects of multiplicity ratios varied with fuel fissile material.

Table 1. Characteristics of Cf-252 neutron source

Item	Description
Total half-life	2.646 yr
Neutron yield	2.34×10^{12} n/s-g
Average neutron energy	2.14 MeV
Neutron activity	4.4×10^{9} n/s-Ci
Neutron dose rate	2300 rem/h-g at 1m
Average spontaneous fission neutron multiplicity	3.757

3. Measurement Test Model

The fissile measurement model was to develop the MCNP code simulation capable of measuring the neutron counting ratio due to the induced fissions. Some simplifications 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 detector for neutron count rate measurement.

For comparison with the MCNP neutron calculations, a series of UO_2 powder can were measured with the Fissile Neutron Counter which was developed at KAERI. 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. The polyethylene reflector is placed between the powder can and the inner stainless steel shell. The neutron multiplication in UO_2 powder is caused by a thermal neutron, which the fast neutrons due to Cf-252 emission are moderated in polyethylene reflector.

The Cd shutter between the UO_2 can and poly is placed and removed for measuring multiplicity ratios. The thick lead layer gives gamma-ray shielding of the He-3 tubes for protection from gamma emission. The polyethylene encased with stainless steel shell has 32 holes for He-3 detector tubes which can detect neutrons by (n, p) reaction. The neutron counts from Cf-252 neutron source has been measured in 32 He-3 detector tubes. The singles and doubles rate 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

A series of uranium oxide powder were measured in a fissile neutron counter. The fissile content has been studied by using neutron count rate based on multiplicity of induced fission. The fissile measurement by using an active neutron count method could be available to assay powder enrichment. And this method would be utilized in determining the total fissile content in a given sample. The fissile content for fuel material has been studied by the comparison of the experiment measurement between the MCNP calculations. Fig. 3 shows a comparison of the measured and calculated count rate versus UO_2 powder enrichment by using Cf-252 neutron source. The plot is in good agreement within 4 % difference.

5. Conclusion

This experiment was 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 at KAERI. To determine the enrichment in fuel material, the neutron count rate by active neutron source 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.

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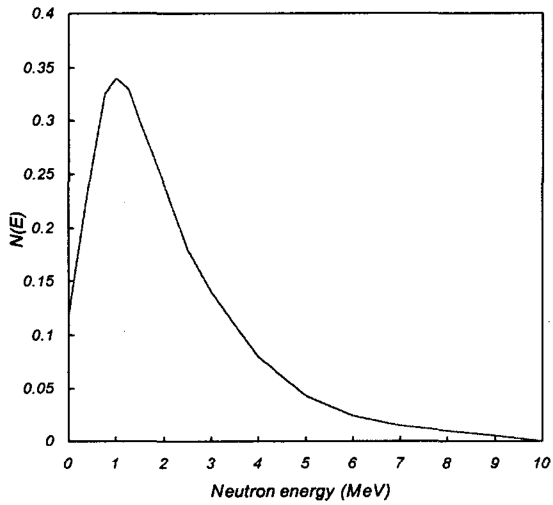

Figure 1 Neutron spectrum from spontaneous fission of Cf-252 source

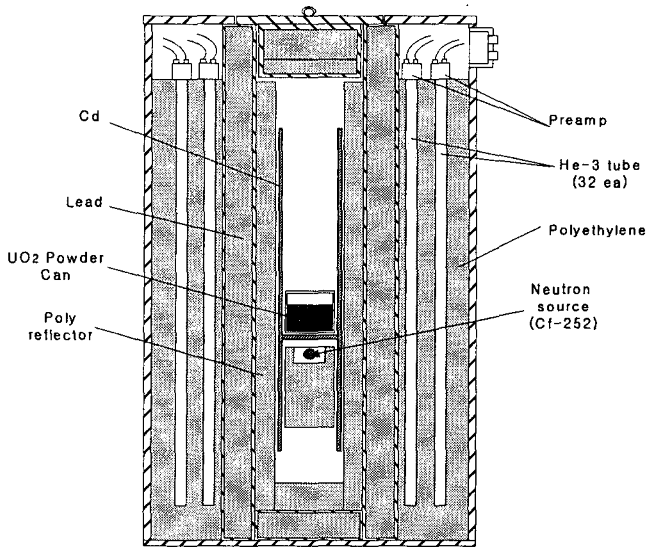

Figure 2 Neutron counter for neutron count rate measurement of UO_2 Powder

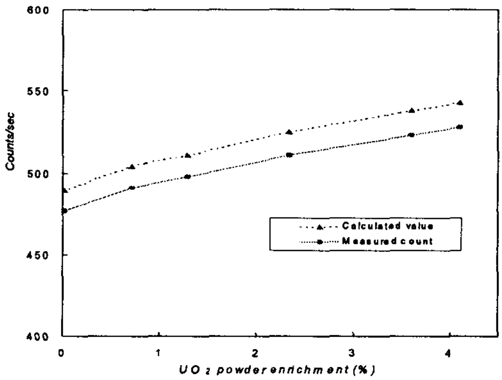

Figure 3 Doubles rate versus UO_2 powder enrichment used by Cf-252 source