NDA Measurement to Determine the Fissile Material for Fuel Powder

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1. Introduction

Active NDA neutron counting method have been developed for the assay of nuclear fuel material[1-5]. 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 by using neuron source.

In order to determine the fissile contents of a fuel sample powder, 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[6] was used to calculate the neutron multiplicity count model for the examination of fissile contents in a fuel material sample. The calculations by MCNP code are compared with the measurement of neutron counts using the cylinder-type neutron detector.

2. Active Neutron Count Method

The neutron sources are more important in active nondestructive assay measurements. The dominant source term of neutrons is spontaneous fission from Cf-252 for active NDA. The 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 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

$$S = \varepsilon M_L F_S \nu_{S1} (1 + \alpha)$$

$$D = \varepsilon^2 M_L^2 f F_s [\nu_{S2} + \frac{M_L - 1}{\nu_{I2} - 1} \nu_{s1} \nu_{i2} (1 + \alpha)]$$

3. Measurement Test Model

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 shown in Fig.1. 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 detector tubes. The singles and doubles rate neutrons were measured by using 32 He-3 tubes. The MCNP calculations were compared with neutron count measurements of neutron counter.

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. The fissile content for fuel material has been studied by the comparison of the experiment measurement between the MCNP calculations. Fig. 2 shows a comparison of the measured and calculated count rate versus UO_2 powder enrichment by using Cf-252 neutron source with.slightly 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.

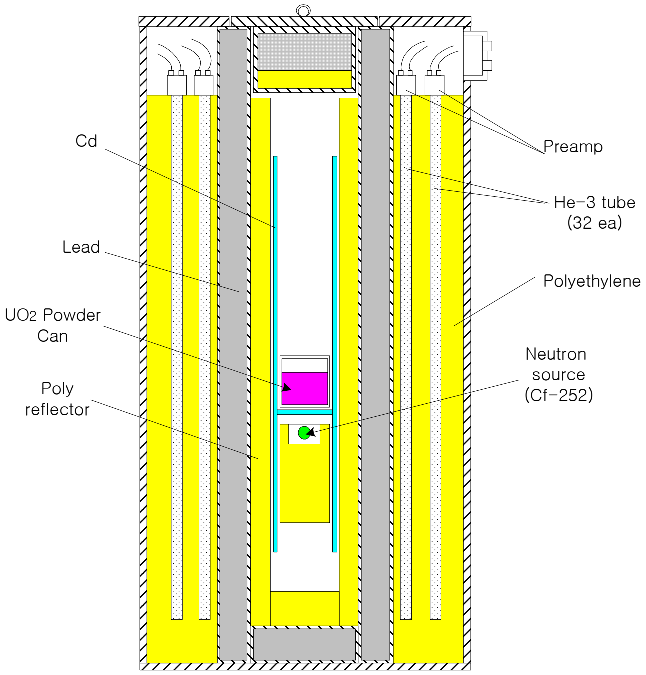

Figure 1 Neutron counter for UO_2 Powder measurement

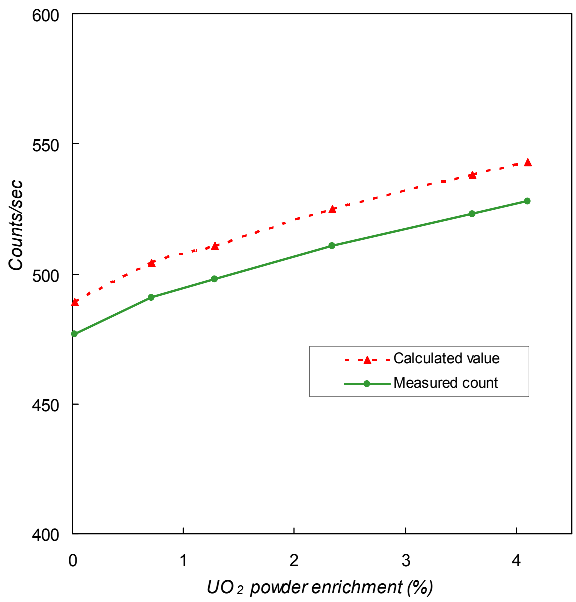

Figure 2 Doubles rate versus UO_2 powder enrichment

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