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Development of NDA Measurement Method to Determine the Fissile Material Contents for DUPIC Fuel

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Abstract

Neutron measurement method by NDA is being developed and simulated for the possible use in determination of fissile contents of DUPIC (Direct Use of Spent PWR Fuel in CANDU) fuel. This method could effectively be applied to DUPIC fuel design, fabrication and its safeguard implementation under the condition of very high radiation environment. The change of neutron count between the induced and non-induced fission by Cm-244 spontaneous fission neutron in spent fuel was analyzed. Results from MCNP calculation model for the two-parameter(singles and doubles) are compared with NDA measurements using PWR spent fuel rod-cuts at KAERI. It shows that the measured neutron count ratio versus quantity of spent fuel material is reasonably well agreed with the calculated values.

1. Introduction

Spent fuel materials contain various heavy metal nuclides such as uranium, plutonium and a hundred of fission products which consists of nuclides produced by actinide fission including their decay and capture products. It is difficult to examine the contents, homogenity and characteristics of spent 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 contents and characteristics of spent fuel contained specially in fission products.

In order to determine the fissile contents and charateristics of spent fuel, the NDA neutron measurement method has been studied by applying a neutron count ratio. The fissile content of spent fuel 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 fission products poisons contained in the spent fuel. The MCNP code[6] was used to design the fissile material measurement model for examination of fissile contents in spent fuel. The effects of neutron count on fissile content, burnup and initial enrichment of fuel are investigated by using the MCNP model.

2. Theoretical Concept of NDA Measurement Method

Spent fuel contains the heavy metal nuclides such as uranium, plutonium and curium called as actinides plus their daughters. The primary emission neutrons originate from the spontaneous fission of curium and plutonium as well as some (α,n) reaction neutrons from the fuel materials. The spontaneous fission and (α,n) neutron source terms are dependent on fuel burnup and initial enrichment. But the dominant source term of neutrons is spontaneous fission from Cm-244. Fig.1 shows the Cm-244 and Cm-242 concentration versus PWR spent fuel burnup. This figure illustrates the major spontaneous fission neutron source of Cm-244.

However, there are an additional neutron source produced from the multiplication process from spent fuel. This multiplication is significantly increased when the spent PWR fuel is measured under moderator material such as water, graphite and polyethylene. The Cm-244 spontaneous fission neutrons will be the dominant 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 spent fuel with Cm-244 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

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

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

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

$$SCR = \frac{S_0}{S_{Cd}}, \quad DCR = \frac{D_0}{D_{Cd}} \tag{3}$$

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, Pu, U-235 and fission product poisons.

3. Fissile Content Measurement Model

The fissile content in spent fuel was not easy to measure because of the high gamma-ray backgrounds. The 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 MCNP model.

For comparison with the Monte Carlo calculations, a series of PWR fuel rod-cut were measured with DSNC(DUPIC Safeguard Neutron Counter) which was developed at KAERI. The fuel material in the cavity are composed of 4 PWR spent fuel rod-cuts with 10 cm in length and with 1.31 cm in diameter. These are made by cutting a single spent PWR fuel rod and then placed into encapsulated by CANDU fuel sheath. The fuel rod has a history of 39 Gwd/Mtu and 9 years cooling at Kori-1 nuclear power plant. The polyethylene reflector is placed between 4 fuel samples and inner stainless steel shell. The neutron multiplication in the fuel rod-cuts is caused by thermal neutron which the fast neutrons due to primary spontaneous fission and (α, n) emmision are moderated in poly reflector layer. When one rod-cuts is inserted, the cavity space is filled with three dummy stainless steel rod-cuts equivalent to thermal absorption of fuel rod.

The Cd shutter between the fuel rod-cuts and poly is placed and removed for measuring Cd ratios. The thick tungsten layer gives gamma-ray shielding of the detector tubes. Air gap is outside tunsgten(W) shield. The poly encased with stainless steel shell has 18 holes for He-3 detector tubes which can detect neutrons by (n, p) reaction. The neutron detector tubes have approximately 75 cm long enough to get the constant response for all fuel rod-cuts. Nickel reflector is botton of the fuel rod-cuts.

4. Results and Discussions

The fissile material content for PWR spent fuel rod-cuts 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 spent fuel sample. The change of Cd ratio due to induced fission of fissile material was suggested to determine the fissile material content of spent fuel.

Fig. 3 shows a comparison of the measured and calculated Cd ratio versus number of fuel rod-cut. The Cd ratio for singles and doubles rate are given according to fuel rod-cut number. This figure is illustrated that the Cd ratio is increased with the number of fuel rod-cuts contained with Pu-239, U-235 fissile materials. Here the shapes are in good agreement except doubles rate. The Cd ratio of spent fuel is also slightly changed by the thermal neutron absorber due to fission products. 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 in DSNC at KAERI. A MCNP modelling conversion of fuel specifications to MCNP model was successfully accomplished with DSNC. The problem of data difference between the measured and the calculated values in doubles rate could be resolved by increasing the measured time. To determine the fissile contents in spent fuel material, the Cd ratio by NDA neutron count is considered to be an appropriate method. The Cd ratio depends on the fuel size, mass, Pu-239, U-235 and fission products. 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

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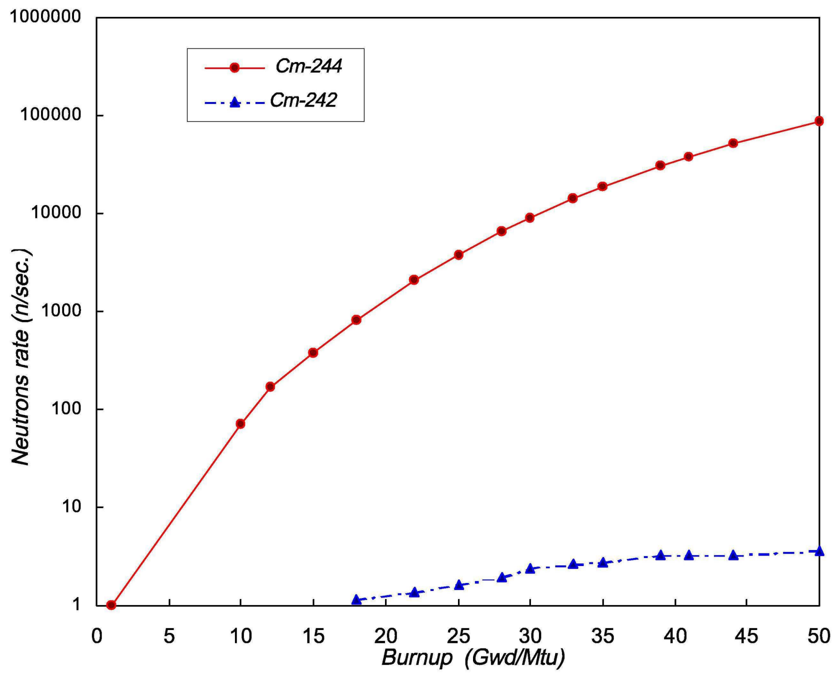

Fig. 1 Cm-244 and Cm-242 spontaneous fission neutron rate versus fuel burnup (based on 1 rod-cut)

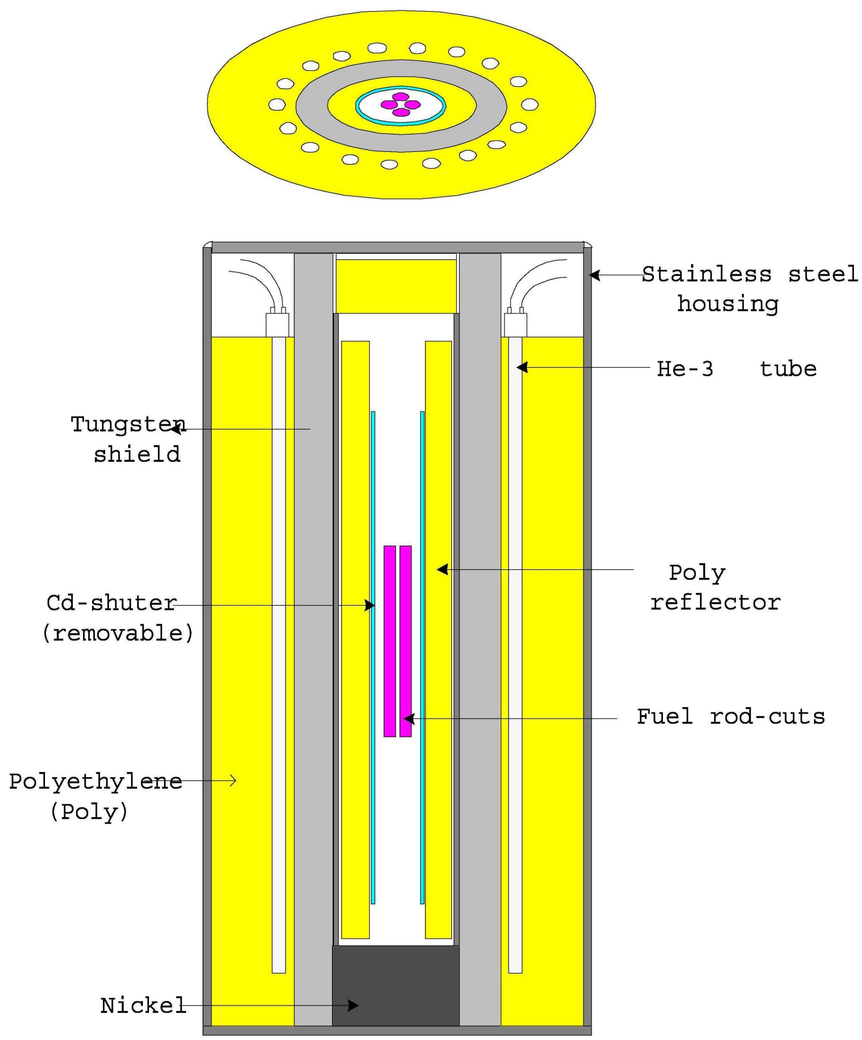

Fig. 2 Test model for Fissile content measurement for PWR spent fuel rod-cuts

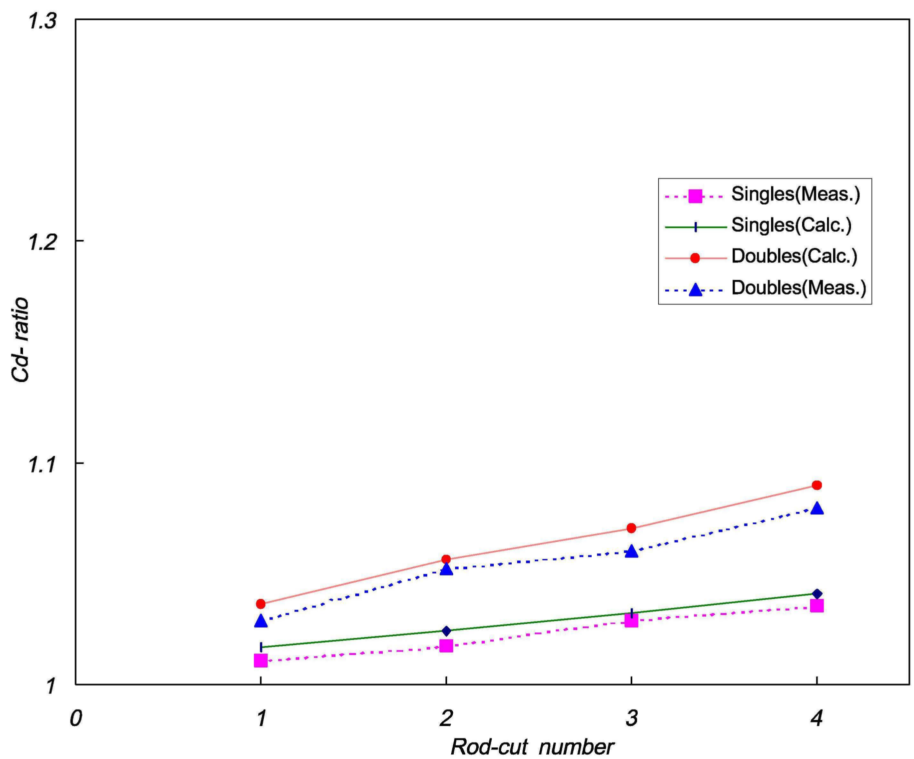

Fig. 3 Cd ratio(Neutron counts ratio) versus number of PWR spent fuel rod-cuts