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Pulsed neutron imaging for non-destructive testing using simulated nuclear fuel samples

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Abstract

An integrated assessment method for a nuclear fuel with high decay heat and high radioactivity is required to establish fast reactor system with Trans-Uranium (TRU) fuel containing minor actinides. In addition, a Pu quantitation method with rapidity and accuracy is also necessary in a viewpoint of nuclear security. For these demands, a quantitative evaluation technique for nuclei concentration, thermal property and physical information of such fuel has to be developed. The present study focuses on the non-destructive imaging using pulsed neutrons. Experiments are carried out at Hokkaido University Neutron Source (HUNS) and a gas electron multiplier (GEM) is applied to obtain 2-D information of time-of-flight (TOF). To simulate a nuclear fuel pellet, a sample with equivalent thermal neutron cross-section to the enriched uranium fuel is prepared and the transmitted images of the simulated sample are acquired. Furthermore, a small piece of In, which simulates the Pu spot in the actual fuel, is inserted into the sample and the detectability of the small spot is discussed.

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Keywords: nuclear fuel; non-destructive testing; simulated fuel pellet; GEM

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1. Introduction

Nuclear power generation is one of the options for long-term stable energy supply. However, for reduction of the environmental load due to the high-level radioactive waste from the nuclear power plants, a fast reactor system using TRU fuel with minor actinides (MA) should be developed (Salvatores et al., 2009). An integrated assessment method for a nuclear fuel with high decay heat and high radioactivity is required to establish the fast reactor system. In addition, a Pu quantitation method with rapidity and accuracy is also necessary in a viewpoint of nuclear security. Especially, Pu spots attributed to the insufficient mixing in fuel production process is very important for safety operation of the reactor and its size and local distribution has to be quantified non-destructively. For these demands, a quantitative evaluation technique for nuclei concentration, thermal property and physical information of such fuel has to be developed.

Non-destructive testing method using neutrons is very effective tools to study materials including heavy metals such as nuclear fuel. Previously, some studies applied conventional neutron radiography to visualized nuclear fuels (Yasuda et al., 2005; Grosse, et al., 2011; Craft et al. 2015). The shape and the structure inside the fuel could be understood from the neutron radiographs. On the other hand, a pulsed neutron resonance absorption spectroscopy with computed tomography has been developed, and it has been applied to quantify the temperature distribution (Kamiyama et al., 2005) and a nuclide density distribution in the sample (Kamiyama et al., 2009). This technique would be applicable to non-destructive testing of the nuclear fuel and also to detecting the Pu spot in the fuel pellet.

In the present study, the pulsed neutron resonance absorption spectroscopy with a position sensitive detector was applied to study the feasibility of the Pu spot detection in the nuclear fuel pellet. For this purpose, a simulated fuel sample with similar neutron transmission characteristics to the actual UO_2 fuel was prepared and the small Indium pieces, which have relatively similar resonance absorption properties to ^{240}Pu, were inserted into the sample. Then, the possibility of the small piece detection was investigated from the obtained radiographs and calculated distributions.

2. Experimental details

To prepare a simulated nuclear fuel sample with a thermal neutron absorption cross-section equivalent to that of 3.4% enriched $^{235}UO_2$ fuel pellet, bismuth oxide (Bi_2O_3) and neodymium oxide (Nd_2O_3) were mixed. Monte-Carlo simulation code (MVP2.0 + JENDL-4.0) was applied to calculate the neutron capture cross-section. The concentration of the mixing ratio between Bi_2O_3 and Nd_2O_3 was adjusted and the thermal neutron capture cross-

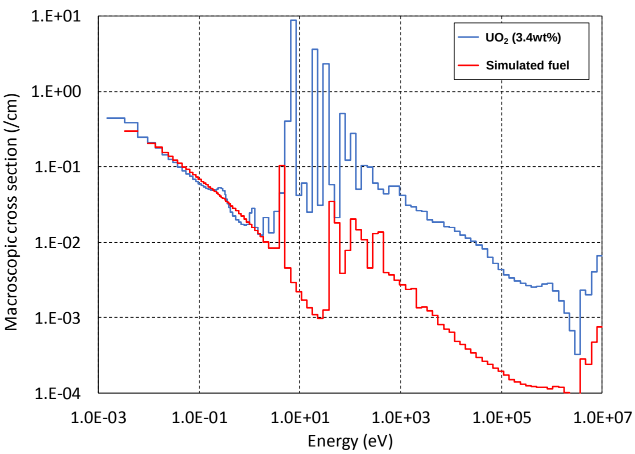

Fig. 1. Calculated macroscopic capture cross-section of UO2 and simulated fuel.