MEASURING THE AXIAL PROFILE OF SPENT NUCLEAR FUEL BURN-UP BY SPONTANEOUS-FISSION-NEUTRONS

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ABSTRACT

Burning-up of nuclear fuel depends on neutron flux density and energy spectrum distribution during fuel operation in nuclear reactor. Various methods are used to determine burn-up of spent nuclear fuel in safeguard systems or fuel reprocessing. A method of measurement of burn-up spatial distribution of spent nuclear fuel element has been developed in National Centre for Nuclear Research (NCBJ) in Świerk, Poland. The method based on recording of the neutron emission from investigated fuel element. The results of measurements made over a period of several years by means of described method are presented in the paper.

1. Introduction

Burning-up of nuclear fuel consists in decreasing of 235U content as an effect of neutron induced reactions in the reactor core. The burn-up quantity is often defined as an amount of energy generated during fission reactions in a mass unit of a fuel or in whole fuel elements. Besides of nuclear fission reactions, the neutron radiative captures occurs in nuclear fuel during its operation as well. Both fission and radiative capture reactions lead to many nuclides, usually radioactive. There-fore, the burn-up value is directly connected with activity of nuclides accumulated in the fuel. That has significant impact on the heat generation in spent fuel elements and accordingly their storage security. The spatial distribution of 235U concentration in spent fuel elements need to be determined due to storage criticality safety as well.

The spatial distribution of neutron emission rate from spontaneous fission of transuranic nuclides in spent nuclear fuel is strongly correlated with local neutron fluence during fuel operation and, therefore, with fuel burn-up value. This relation is not trivial due to several factors influence on neutron emission, e.g. operating conditions in reactor, cooling time, initial uranium enrichment and initial uranium mass. Fluctuations of vertical burn-up distribution in fuel element are results of heterogeneous distribution of neutron flux density in reactor core. It is influenced by surrounding moderator, fuel elements, absorbing rods, irradiation targets etc.

2. Measurements

The subjects of investigation were HEU fuel elements used in MARIA research reactor, operated by NCBJ in Świerk. The spent HEU fuel assemblies had been stored in the MARIA reactor storage pool.
In described method the measurement of burn-up distribution consist in recording of neutron emission from separate sections of a fuel element. The neutron emission is scaled down to the burn-up value of particular section of spent fuel element.

A series of calculations have been performed in order to determine the inventory of neutron emitters in spent nuclear elements. The relative concentration of neutron emitters depends on the fuel burn-up and cooling time.

The calculations based on the cross-sections of particular nuclear reactions and decay constants of their products, allow identifying the main spontaneous fission neutron sources. The relation of neutron emission rate from spontaneous fission of transuranic nuclides in spent nuclear fuel and fuel burn-up value has been calculated.
Based on performed analyses and calculation, suitable measuring set-up has been designed and constructed. The measuring set-up allows counting the number of neutrons emitted in time unit from particular section of the fuel assembly. The recorded data were used to determine the absolute value of the fuel burn-up along the fuel assembly.

3. Results

Measuring of vertical burn-up distribution of fuel assembly consist in counting the number of neutrons emitted in time unit from particular section of the fuel assembly. Construction of measuring stand allows to count the neutrons emitted only from specified parts of fuel element separately. Measurements have been done in nine points along each fuel assembly. The investigations have been performed on 110 pcs of MR-6/80 fuel assemblies (80% enriched, used in MARIA reactor until 1999) with total burn-up values varying from 40 MWd up to 127 MWd and cooling time from 5 to 27 years. The results are presented on fig. 1.

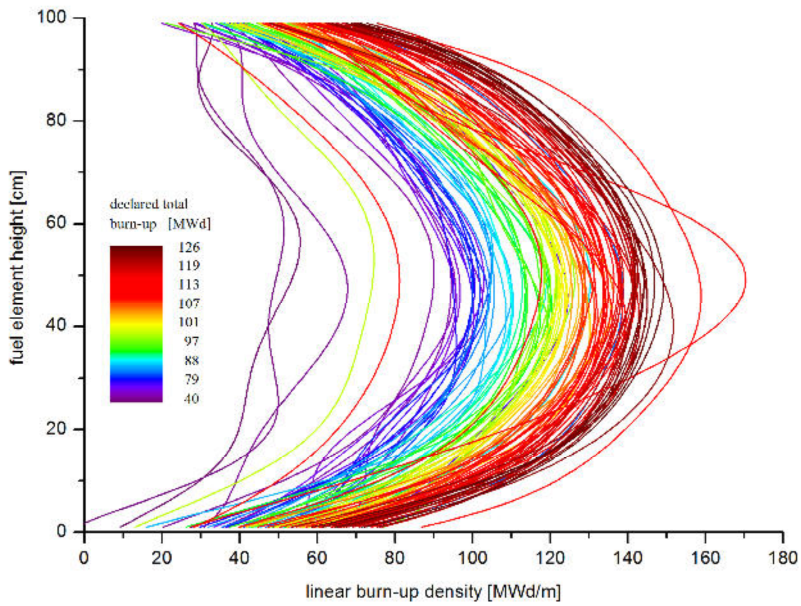

Fig. 1. Measured space distribution of burn-up of spent fuel assemblies; the warmer color, the higher declared burn-up value [1].

The relative burn-up distribution of measured fuel elements has been determined as well. The collected measuring data have been then used to establish one universal relation. The relation describes a shape of relative linear burn-up distribution as a function of fuel height for a given total (or average) burn-up value. This relation is characteristic for a given reactor core – it depends on the neutron flux density and energy spectrum distribution and does not depend on the fuel type. Therefore, it can be used in further analyses of operation and storage of other types of fuel elements used in MARIA reactor.

5. Reference

R. Prokopowicz, K. Pytel, Determination of nuclear fuel burn-up axial profile by neutron emission measurement, Nucl. Instrum. Methods Phys. Res. Sect. A 838 (2016) 18–23.