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Neutron spectra at two beam ports of a TRIGA Mark III reactor loaded with HEU fuel

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HIGHLIGHTS

- Neutron spectra of a TRIGA reactor were measured.
- The reactor core is loaded with HEU.
- The spectra were measured at two reactor beam ports.
- Measurements were carried out at 5 and 10 W.

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ABSTRACT

The neutron spectra have been measured in two beam ports, one radial and another tangential, of the TRIGA Mark III nuclear reactor from the National Institute of Nuclear Research in Mexico. Measurements were carried out with the reactor core loaded with high enriched uranium fuel. Two reactor powers, 5 and 10 W, were used during neutron spectra measurements using a Bonner sphere spectrometer with a ^{6}LiI(Eu) scintillator and 2, 3, 5, 8, 10 and 12 in.-diameter high-density polyethylene spheres. The neutron spectra were unfolded using the NSDUAZ unfolding code. For each spectrum total flux, mean energy and ambient dose equivalent were determined. Measured spectra show fission, epithermal and thermal neutrons, being harder in the radial beam port.

1. Introduction

Nuclear reactors are classified in several ways; in terms of their applications reactors are classified as reactors for teaching, research, isotopes production, and for power production. TRIGA is the acronym of Training, Research, Isotopes, General Atomics; there are four different versions of these reactors: Mark I, II, III and F; the main differences are the transient or stationary power level, the location of reactor pool, the amount and type of irradiation facilities and the mobility of reactor core (IAEA, 2005).

Under the leadership of F. Dyson a team did the original design of TRIGA reactor with the aim of this being installed and operated in scientific institutions and universities (GAES, 2011). In 1956 the General Atomic Division of General Dynamics did create a group of scientists, under the direction of E. Teller, with the task to design a nuclear reactor so safe that even with the sudden withdrawal of all the control rods the reactor will reach the steady state without damaging the fuel (Dyson, 1979); this goal was accomplished in the design of the reactor fuel.

The fuel used in the TRIGA reactors is hydrogen-containing uranium–zirconium alloy, UZrH, where U is 8% enriched with 20% of ^{235}U. The H in the UZrH array makes it inherently safe (GD, 1963).

The moderator, the fuel features and its geometrical array in the core characterize the neutron spectrum in the core and along the beam ports. In order to use the beam ports the neutron spectra must be characterized (Marek and Viererbi, 2011; Nakamura et al., 2011; Nascimento et al., 2007).

The ININ's Nuclear Center Dr. Nabor Carrillo has a TRIGA Mark III reactor; on April 24th, 1964, it became critical. Since then the reactor has been used for research, training, teaching and for radioisotopes production. Due the ^{235}U enrichment, the nuclear fuel is defined as high enriched uranium, HEU.

In order to fulfill the international agreement to reduce the nuclear risk worldwide it was decided to change the HEU with a core of low enriched uranium, LEU (US, 2010). This change could

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modify the features of neutron spectra in the irradiation facilities around the reactor; thus before the fuel is changed it was necessary to determine the neutron spectra with the core with HEU. For different applications it is important to determine the neutron features in the beam ports (Nakamura et al., 2011; Nascimento et al., 2007).

Twenty years ago the neutron flux profiles and the doses were determined in two sites inside the reactor beam ports using CR39 and gold foils (Balcazar et al., 1993). In the beam ports the radiation field is a mixture of neutrons and gamma-rays whose intensity depends upon reactor power; thus to measure the neutron spectrum the spectrometer should use a passive detector, because if an active detector is used the reactor must work at a low power to avoid large dead times and pulse pileup (Marek and Viererbi, 2011).

The aim of this work was to determine the neutron spectrum in two beam ports running the reactor at 5 and 10 W with HEU. The spectra were measured in a radial beam port and a tangential beam port using a Bonner sphere spectrometer, BSS.

The neutron spectra were used to calculate the integral features such as the total neutron fluence rates, the neutrons mean energy and the ambient dose equivalent values.

2. Materials and methods

A Bonner sphere spectrometer, BSS, with a ^{6}LiI(Eu) scintillator was used to obtain the neutron spectra. The scintillator is 0.4 cm in diameter and 0.4 cm in length where 96% of Li is ^{6}Li that has a large cross section for ^{6}Li(n, α)^{3}H reaction to take place induced by thermal neutrons. In order to increase the response to neutrons with larger energies the detector is inserted in high-density polyethylene spheres with different diameters. In this work the diameters of the spectrometer spheres were 5.08, 7.62, 12.7, 20.32, 25.4, and 30.48 cm. In neutron spectrometry the spheres are named by their diameter in inches; thus in this study the spectrometer sphere has 0, 2, 3, 5, 8, 10, and 12 in. diameter; here 0″ sphere means the bare detector. Each sphere has a response function for 10^{-8} up to 100 MeV neutrons having an overall efficiency from 0.07 to 0.84 (Vega-Carrillo et al., 2009).

The commercial version of the BSS has a single channel, and in this study the spectrometer with a multichannel analyzer was used simplifying the selection of the region of interest of the α-peak in the pulse height spectrum; this is important because radiation coming out of the reactor beam ports have a large contribution due to gamma-rays. The α-peak in the pulse-height spectrum accounts for the amount of the (n, α) reactions occurring in the scintillator.

Measurements were carried out in two beam ports, one radial and one tangential. The BSS was located 120 cm from the beam ports output as is shown in Fig. 1.

In the radial beam port the spectrum was measured twice with the reactor running at 5 and 10 W; measurements at the tangential beam port were carried out using 10 W. Low powers were selected to reduce the pulse pileup and the detector dead time (Marek and Viererbi, 2011).

With the BSS count rates the neutron spectrum, $\Phi_E(E)$, was unfolded using the UTA4 response matrix (Hertel and Davidson, 1985) in the NSDUAZ unfolding code (Vega-Carrillo et al., 2012). The $\Phi_E(E)$ was used to calculate the total fluence rate, ϕ, the neutron mean energy, E_{Av}, and the ambient dose equivalent rate, $H^*(10)$; these calculations were carried out using the following equations respectively.

$$\phi = \int_{E_{min}}^{E_{max}} \Phi_E(E)\, dE \quad (1)$$

$$E_{Av} = \frac{1}{\phi}\int_{E_{min}}^{E_{max}} E\Phi_E(E)\, dE \quad (2)$$

$$H^*(10) = \int_{E_{min}}^{E_{max}} \Phi_E(E) h^*(10, E)\, dE \quad (3)$$

In Eq. (3), $h^*(10, \mathrm{E})$ are the neutron fluence-to-ambient dose equivalent conversion coefficients taken from the ICRP74 (ICRP, 1996).

Using the integral features, ϕ and $H^*(10)$ for each beam port, and for each reactor power, the fluence-to-ambient dose factors were calculated using

$$h^*_\phi(10) = \frac{H^*(10)}{\phi} \quad (4)$$

3. Results and discussion

In Fig. 2, the count rates of each sphere measured in the beam ports are shown; in this plot the count rates were normalized to the count rates measured with the 8″ sphere.

In the radial beam port the relative count rates look alike for both reactor powers; these count rates are quite different from the relative count rates measured in the tangential beam port. In the radial beam port the maximum count rate was obtained with the 5″ sphere, while for the tangential beam port the maximum count rate was measured with the 3″ sphere; this means that the neutron spectrum is harder in the radial beam port while in the tangential beam port the neutron spectrum is softer.

The BSS count rates were used to unfold the neutron spectra using the NSDUAZ unfolding code; in Fig. 3 the neutron spectra measured in the beam ports are shown.

All the spectra have thermal, epithermal and fast neutrons being in agreement with the neutron spectrum in the beam port of a research reactor (Nascimento et al., 2007). For both reactor powers in the radial port the spectra have a peak from 0.1 to 10 MeV produced in the nuclear fission, these neutrons are transported along the beam port colliding with the beam port surface. In these interactions some neutrons are scattered out from the beam port and are absorbed; other neutrons are slowed down and reach the beam port output as epithermal and thermal neutrons.

In the tangential beam port streaming fission neutrons are shifted to lower energies showing a peak from 0.2 to 0.8 MeV due to the interactions between neutrons and the materials and moderator between the reactor core and the beam port pipe; once along the pipe neutrons are slowed down becoming epithermal and thermal neutrons.

For each beam port the total fluence rate, ϕ, the neutrons' mean energy, E_{Av}, and the ambient dose equivalent rate, $H^*(10)$, were calculated. These integral features are shown in Table 1.

The uncertainty in the fluence and the ambient dose equivalent was obtained adding the square of errors in the reactor power, the distance between the BSS and the beam port output, the response matrix and the largest error in the measured count rates.

The total fluence rates per Watt are 215 ± 14 and $209 \pm 14\ \mathrm{cm^{-2}\,s^{-1}\,W^{-1}}$ for the radial port and $54 \pm 4\ \mathrm{cm^{-2}\,s^{-1}\,W^{-1}}$ for the tangential beam port. It can be noticed that in the radial port the normalized fluence rates are statistically the same; this value is approximately four times larger than the normalized fluence rate in the tangential beam port.

The ambient dose equivalent rates were also normalized to the reactor power being 101.0 ± 6.7 and $93.4 \pm 6.2\ \mu\mathrm{Sv\,h^{-1}\,W^{-1}}$ for the radial port and $10.6 \pm 0.7\ \mu\mathrm{Sv\,h^{-1}\,W^{-1}}$ for the tangential port. Thus, in the radial beam port the ambient dose equivalent rate is