Dosimetry of Fission Neutrons in a 1-W Reactor, UTR-KINKI

SATORU ENDO1*, EIJII YOSHIDA1, YUSUKE YOSHITAKE1, TETSUO HORIGUCHI2, WENYI ZHANG3, KAZUO FUJIKAWA2, MASAHARU HOSHI3, TETSUO ITOH2, MASAYORI ISHIKAWA3 and KIYOSHI SHIZUMA1

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The energy spectrum of fission neutrons in the biological irradiation field of the Kinki University reactor, UTR-KINKI, has been determined by a multi-foil activation analysis coupled with artificial neural network techniques and a Au-foil activation method. The mean neutron energy was estimated to be 1.26 ± 0.05 MeV from the experimentally determined spectrum. Based on this energy value and other information, the neutron dose rate was estimated to be 19.7 ± 1.4 cGy/hr. Since this dose rate agrees with that measured by a pair of ionizing chambers (21.4 cGy/hr), we conclude that the mean neutron energy could be estimated with reasonable accuracy in the irradiation field of UTR-KINKI.

INTRODUCTION

The Kinki University nuclear reactor, UTR-KINKI, has ample space for the irradiation of living materials in the central portion of its core, where neutrons and gamma rays are available at comparable dose rates, i.e., 20 cGy/hr during operation at a nominal output of 1 W. Separate dosimetry of neutrons and gamma rays was carried out using a pair of tissue equivalent ionization chambers; other instruments were calibrated with fission neutrons from $^{252}$Cf and gamma rays from a $^{60}$Co source installed at the Research Institute Radiation Biology and Medicine (RIRBM), Hiroshima University1,2,3. With the aid of these studies, UTR-KINKI has been intensively used for biological studies on the relative biological effectiveness (RBE) of fast neutrons; a body of relevant information on the RBE of fast neutrons is already available in the literature4,5,6,7. However, a question remains concerning the mean energy of fission neutrons in the biological irradiation field of UTR-KINKI. A lack of such information hinders not only the interpretation of RBE data in relation to the energy dependency of biological effects of fast neutrons, but also the development of an activation method for the in situ monitoring of the neutron dose. The present study was carried out to estimate the mean energy of neutrons in the biological irradiation field of UTR-KINKI.

To estimate the mean neutron energy, an accurate description of the neutron energy spectrum, especially in the region of fast neutrons, is essential8. The neutron spectrum described in our previous study seems to meet this requirement; it was determined for fast neutrons in the irradiation field of UTR-KINKI by a multi-foil activation method coupled with the artificial neural network (ANN) technique. ANN is a simple model of biological systems applicable to non-linear and pattern-recognition problems. In the present study, the neutron spectrum determination was extended to include thermal neutron fluences, and the resulting...
wide-range spectrum was used to estimate the mean energy of fission neutrons in the biological irradiation field of UTR-KINKI. The neutron dose rate was determined from the estimated mean energy and neutron fluence rate measured by the foil activation method. If the mean neutron energy had been accurately estimated, the resulting dose rate would agree with dose rate measured by a pair of ionizing chambers. This was actually the case. The results pertaining to this contention are described and discussed in this report.

MATERIALS AND METHODS

Irradiation field of UTR-KINKI

UTR-KINKI has two fuel tanks separated by 46 cm of internal graphite; each tank contains $^{235}\text{U}$ enriched uranium fuels immersed in a small quantity of light water, as shown in Fig. 1. At the center of the internal graphite, a graphite stringer of $9.6 \times 9.6 \text{ cm}^2$ square and 122 cm long can be withdrawn to provide a cavity for sample irradiation. The center of the core height in the cavity is used as an irradiation field for biological samples. In the present study, neutron detectors were exposed to mixed neutron-gamma radiations in the biological irradiation field of UTR-KINKI operated at 1 W.

Determination of the neutron spectrum using an activation method coupled with ANN

To estimate the fast neutron fluence rate, we used the threshold reactions of $^{115}\text{In}(n, n')^{115m}\text{In}$, $^{58}\text{Ni}(n, p)^{58}\text{Co}$, $^{54}\text{Fe}(n, p)^{54}\text{Mn}$, $^{47}\text{Ti}(n, p)^{47}\text{Sc}$ and $^{27}\text{Al}(n, \alpha)^{24}\text{Na}$. The weights of $1 \times 1 \text{ cm}^2$ foil samples of Fe, Ni, Ti, Al and In were 2.12, 2.27, 0.47, 0.25 and 1.09 g, respectively. To monitor the thermal neutron fluence, a set of bare and 0.5 mm-thick Cd-covered Au foils ($1 \times 1 \text{ cm}^2$ and 5 $\mu\text{m}$ thickness of each) were irradiated. The induced activities of the irradiated samples were measured with a low-background germanium spectrometer having a crystal volume of 77 $\text{cm}^3$ (Canberra, GC1520) and a relative efficiency of 15% shielded by 10 cm-thick lead and 5 mm-thick copper walls. The gamma-ray detection efficiency of the spectrometer was calibrated using standard gamma-ray sources.

The activity data from the multi-foil method were processed to neutron fluence data using the ANN technique. ANN is a simple model of a biological neural network system. A schematic view of ANN used is shown in Fig. 2. ANN is composed of an input layer, a hidden layer and an output layer. Each layer contains neurons, which are connected to all neurons of the successive layer and have their own weights; the weighted sum of data entering each neuron is processed with a transformation function of the sigmoid function. The weight of each neuron can be obtained by a training procedure using supervisor data. The supervisor data for training were prepared using the Monte Carlo N-Particle transport code (MCNP4B), after checking that the reproducibility of the calculated activities could be verified to be within 15% of the experimentally determined activities. Several neutron sources (bare fission neutrons, water-moderated neutrons and fusion-neutrons) were generated and transported to foil-detectors using MCNP4B. The induced activities according to the threshold reactions were calculated using all individual reaction cross sections, which were taken from the JENDL 3.2 cross section library. The number of units for the hidden layers was selected to be 5, which was the lowest value under the conditions, with residual errors of less than 1%.
this study, a back-propagation algorithm was used for training. Training of ANN was carried out using the supervisor data for 50000 iterations. After training, the neutron spectrum in UTR-KINKI was calculated from the measured specific activities using the determined weights in ANN\(^9\).

The thermal and epithermal neutron fluences were estimated using the Cd cutoff method. The epithermal neutron fluence was determined from the activity induced in the Cd-covered Au foils and the resonance cross section at an energy of 4.9 eV. The thermal neutron fluence was determined from the activity in bare Au foils corrected with that in Cd-covered Au foils. For this estimation, the thermal neutron shielding factor of the 25 \(\mu\)m-thick Au foils and the 0.5 mm-thick Cd-covered correction factor for the Au foils were assumed to be 1.25 and 1.1, respectively\(^{12}\).

**Transport calculation of the neutron spectrum**

For a comparison with the measured neutron spectrum, the calculated spectrum was obtained for fission neutrons in the irradiation field of UTR-KINKI using MCNP4B. For the transport calculation, the input data were taken from the cross-section library of the ENDF B/VI, and the core geometry of UTR-KINKI (Fig. 1), was simplified as a core filled with a carbon reflector, except for the central cavity and the portions occupied by the two fuel tanks.

**Estimation of the mean neutron energy**

The following equation was used to estimate the mean neutron energy \(\langle E \rangle\) as the tissue kerma factor weighted mean of the neutron spectrum:

\[
\langle E \rangle = \frac{\int_0^\infty \phi(E)K_f(E)E \, dE}{\int_0^\infty \phi(E)K_f(E) \, dE},
\]

where \(\phi(E)\) and \(K_f(E)\) are the neutron fluence and the neutron kerma factor at energy \(E\), respectively. The values of the neutron tissue kerma factor as a function of the energy were taken from the ICRU report\(^{13}\).

**Dosimetry of neutrons by a pair of ionizing chambers**

The ionizing chambers used were an A150-walled, tissue equivalent gas-flowed chamber (TE-TE) and a graphite walled, carbon dioxide gas-flowed chamber (C-CO\(_2\)), which are named IC-17 and IC-17G (Far West Tech. Inc., California), respectively. Their sensitivities were calibrated with \(^{60}\)Co gamma rays at the Research Institute for Radiation Biology and Medicine, Hiroshima University. Several corrections (general recombination, initial recombination, and atmospheric pressure) were made for the chamber readings. The detailed procedure has been described in references\(^{14, 15}\). The neutron sensitivities of the TE-TE \((k_T)\) and the C-CO\(_2\) chambers \((k_U)\), which are the responses to neutrons of the chambers relative to gamma-rays, are assumed to be the same as those for \(^{252}\)Cf-fission neutrons \((k_T = 0.98\) and \(k_U = 0.08\), respectively)\(^{16}\). The neutron and gamma-ray doses are given by

\[
D_n = \frac{R_T - R_U}{k_T - k_U},
\]

\[
D_{\gamma} = \frac{k_T R_U - k_U R_T}{k_T - k_U},
\]
where $R_T$ and $R_U$ are the responses of the TE-TE and the C-CO$_2$ chambers, respectively. From these equations, the neutron and gamma-ray doses were obtained, independently. The uncertainty of this method is mainly caused by an assumption of the neutron sensitivity ($k_U = 0.08$) of the C-CO$_2$ chambers. The uncertainty of the neutron dose can be estimated by

$$\frac{\Delta D_n}{D_n} = \frac{\Delta k_U / k_U}{(1 / k_U) - (1 + \Delta k_U / k_U)},$$

(4)
as discussed in the ICRU report.$^{17}$ The uncertainty of the sensitivity is less than 20% from the neutron energy spectrum. The uncertainties for the neutron and gamma-ray doses are estimated to be less than 2%.

**RESULTS**

The fast neutron spectrum obtained by the multi-foil activation method coupled with the ANN technique and the thermal neutron fluence rates from the Au-activation method are plotted in Fig. 2. The spectrum from a neutron-transport calculation using MCNP4B is also shown in Fig. 2. By integrating the measured spectra for fast neutrons and the calculated spectrum for the unmeasured energy region (i.e., the epithermal region below 0.1 MeV) by Eq. 1., we estimated the mean neutron energy to be $1.26 \pm 0.05$ MeV. This energy value is smaller by a magnitude of about 9% as compared with the value calculated using only the experimental data.

The activity induced in the In foils by the threshold reaction of $^{115}$In(n, n')$^{115m}$In was $1.93 \pm 0.14$ Bq/mg. From the MCNP calculation, this activity gave a fluence rate of $2.17 \times 10^6$ n/s/cm$^2$ for neutrons with an energy greater than the threshold energy of 0.3 MeV. From this fluence rate and the tissue karma factor at an energy of 1.26 MeV ($= 0.252 \times 10^{-8}$ cGy cm$^2$), we obtained the dose-rate of neutrons to be $19.7 \pm 1.4$ cGy/hr, where the error was estimated from the induced $^{115m}$In activity only. The mean neutron energy and the estimated neutron dose rate are listed in Table 1 together with the results of the same estimation from the neutron spectrum calculated by MCNP4B.

| Table 1. Mean energy of fission neutrons and the neutron dose rates estimated by the three methods. |
|----------------------------------------------------------|
| Mean neutron energy (MeV) | Dose rate (cGy/hr) | neutron | gamma-rays |
|---------------------------|-------------------|---------|-----------|
| Foil activation           | $1.26 \pm 0.05$   | 19.7    | 1.4       |
| MCNP4B                    | $1.56 \pm 0.28$   | 21.7    | 4.2       |
| Paired chambers$^a$       | –                 | 21.4    | 19.6      |

$^a$The systematic uncertainty of the chamber measurement was estimated to be less than 2%.

The dose rates of fast neutrons and gamma rays measured by the paired ionizing chambers were 21.4 cGy/hr and 19.6 cGy/hr, respectively (Table 1). The reproducibility of 10 measurements was 99.9%, and the systematic error caused by the neutron sensitivity was estimated to be 2%.

**DISCUSSION**

As can be seen from Fig. 2, the neutron energy spectrum calculated by MCNP4B nicely fits the spectra obtained by the multi-foil activation method coupled with the ANN technique and the thermal neutron fluence rates from the Au-activation method. This agreement lends further support to the notion that the ANN technique is feasible in determining the energy spectrum of the reactor’s fission neutrons. It also gave credit to using the calculated spectrum for the unmeasured energy region in estimating the mean neutron energy. The estimated mean energy is 1.26 MeV. Based on this energy value, the activity induced in the In foils and other information, a neutron dose rate of $19.7 \pm 1.4$ cGy/hr was obtained. As can be seen from Table 1, this dose rate is nearly equal to the dose rate from the MCNP calculation ($21.7 \pm 4.2$ cGy/hr) and the rate measured by the chambers ($21.4$ cGy/min). We may thus conclude that the mean energy of fission neutrons in the biological irradiation field of UTR-KINKI could be determined with reasonable accuracy.

Knowing the mean energy of neutrons and the neutron energy spectrum in the irradiation field of
UTR-KINKI, we can pursue two issues in further studies. One concerns the LET dependency of the neutrons’ biological effect. Namely, the energy spectrum data allow one to calculate the LET distribution for fission neutrons from the UTR-KINKI; thereby, the data provide a basis for interpreting the biological effect as an integrated effect of neutrons with different LET. The other concerns the development of a simple system for the *in situ* monitoring of fast neutrons for biological studies that utilize an activation method, such as the one used in the present study.

In summary, the present study has confirmed the feasibility of the ANN technique in determining the energy spectrum of the reactor’s fission neutrons and has provided the necessary information for further studies on the biological effects of fission neutrons in the irradiation field of UTR-KINKI by estimating the mean energy from the energy spectrum.

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