MCNP™ and 3rd IAEA INTERCOMPARISON EXPERIMENT on NUCLEAR ACCIDENT DOSIMETRY

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Abstract

The neutron and gamma ray spectra and corresponding dose quantities calculated by using well-known MCNP™ code are presented in this paper and compared with the measured values during the Third International Intercomparison Experiment on Nuclear Accident Dosimetry at RB nuclear reactor at Vinča Institute. Discrepancies in the correlation RB reactor power - dose rate are found indicating that the RB reactor was operated, during the Experiment, at higher power than reported.

Introduction

In 1969 the International Atomic Energy Agency (IAEA) established an international coordinated research programme on Nuclear Accident Dosimetry with the aim of improving the performance of criticality dosimetry systems. Within the framework of this programme, four international multi-laboratories intercomparison experiments have been organised and the response of variety of dosimeters examined in different neutron spectra under simulated accident conditions.

The Third International Intercomparison Experiment on Nuclear Accident Dosimetry was organised by the IAEA in collaboration of the Government of Yugoslavia and took place during period May 14-25, 1973, at the RB research reactor in the Institute of Nuclear Sciences 'Boris Kidrič' (now: 'Vinča'), Belgrade. The results of the Experiment are reported elsewhere [1-3].

The unshielded RB nuclear reactor at Vinča was used as a source of mixed neutron-gamma ray radiation for the Experiment. For that purpose, an appropriate unreflected core configuration (#5/1973) in the reactor tank was selected (Figure 1) in order to obtain as high as possible ratio of fast to slow neutrons in the leakage spectrum. Topography of the radiation field was measured around the RB tank, and it was concluded to be uniform within ± 5 % at distance at least 3 m from the tank surface.

As a very important component of the Experiment, the leaking neutron spectrum at 3 m distance from the RB reactor core is measured by using Bonner spheres and activation foils [3, 4]. Gamma ray spectrum was not measured. Then, 25 years ago, calculations of the neutron and gamma ray leakage spectra from the reactor core were not too reliable by using available transport codes. So, only few neutron and gamma ray transport calculations, under large assumptions, were carried out for estimating of the neutron and gamma ray spectra obtained in the Experiment. In the last decade, reliable radiation transport codes have become readily available, among them, the Monte Carlo based MCNP™ code [5] with modern continuous energy cross section libraries. It was the reason to re-evaluate these new calculations and the reactor power reported data of the 3rd International Intercomparison Experiment.

Figure 1. RB reactor core #5/1973
Experiment

The reactor was prepared to operate under conditions convenient for intercomparison experiments. The irradiation position is placed at 3 m distance from the edge of the reactor tank, where the neutron flux was measured with uniformity within ± 5 %. From preliminary measurements it was concluded that the reactor integrated power (generated fission energy) should be high than 500 Wh. Two irradiations are carried out, each in 30 minutes: the 'low level' (EXP-1, declared maximal power of $P_{\text{max}} = 1.05 \text{ kW}$, total generated fission energy $E_r = 590 \text{ Wh}$) and the 'high level' (EXP-2, $P_{\text{max}} = 6.7 \text{ kW}$, $E_r = 3.3 \text{ kWh}$).

Neutron spectrum was measured using the Bonner spheres and foil sets (activation and threshold) by 18 participating study groups [2-3]. The average neutron fluence in six energy groups was determined from results of measurements of the irradiated foils. It was used for neutron dose determination, beside the phantoms filled with solution of the NaCl. Gamma ray dose rate at irradiation position was determined by using the KAKTUS I L ionization chamber and associated electronic equipment [2].

Only the results of the EXP-1 are shown for the comparison with the calculations, because the EXP-2 has had a 'perturbed run', due to an unexpected shut down of the reactor during power raising [2]. That time dependence of the reactor power was not possible to model by the MCNP code appropriately in this study.

Calculation

The MCNP™ is well-known N-particle (neutron/photon/electron) transport code, verified at huge set of different radiation transport and criticality safety benchmark problems. The code version 4B (distributed by summer 1997) with continuous energy cross sections libraries for neutron, gamma rays and electrons is considered, nowadays, as the reference one. It is applied to calculate the neutron and gamma ray spectra at dosemeters' position during the steady state operation of the reactor in the first experiment. Neutron cross section library ENDF601 [6], based on evaluated data from the ENDF/B-VI file in energy range: 1 meV - 20 MeV, and gamma ray cross section library MCPLIB1 [6], in energy range: 10 keV - 100 MeV, are used. The $S(\alpha,\beta)$ scattering law for neutron scattering at D atoms connected in heavy water molecules at thermal energies is applied according to the TMCCS1 library [6]. Full RB reactor core and coupled neutron-gamma rays transport are modelled in the three dimensional (3D) geometry of the code as close as possible to the real ones [7]. The KCODE option (i.e., the $k_{\text{en}}$ calculation) is used with the total 3 million neutron histories (4000 neutron histories in 750 active cycles, after the 15 initial ones). Starting neutron source for the code is located in each fuel slug and fuel element placed in the reactor core. Neutron flux, calculated by the code in 55 energy groups (from 1 meV to 10 MeV), is normalised to 1.05 kW, the RB reactor's reported power level in the EXP-1 carried out on May 16, 1973. Corresponding gamma ray flux is simultaneously calculated in the group structure used in well-known VITAMIN-E dosimetry library (35 energy groups, in range from 10 keV to 15 MeV, [8]).

Results

Calculated neutron and gamma ray spectra at 3 m distance from the RB reactor tank (at height from 74 cm to 106 cm), for the reported reactor power of 1.05 kW are given at Figures 2-3. Uniformity within ± 1.5 % of the axial space distribution of the neutron flux at the irradiation place (at 3 m distance from the RB reactor tank (at height from 36 cm to 142 cm) is confirmed by the results of the MCNP calculations.

Six group average neutron fluences at the irradiation position, determined by the foils and the MCNP code for the RB reactor reported power (1.05 kW) are given in Table I (1σ errors are given). Great discrepancy could be seen. But, if the calculated total neutron fluence is normalised to the measured one, the 2.597 ratio factor of the measured to calculated fluence is found, suggesting that the fission power of the RB reactor was higher by the same figure. The six group neutron spectra, determined in the EXP-1 and calculations (normalised to the reactor 'new' power of 2.63 kW) are compared at Figure 4, showing an excellent agreement. Further, the measured neutron spectra by the Bonner spheres and the calculated one by the MCNP code, divided by width of the lethargy for each energy group, are shown at Figure 5. Again, very well agreement is obtained.

Measured and calculated absorbed dose in air and corresponding dose rates due to gamma rays and neutron at the irradiation position are given in Table II (1σ errors are given). Intensity of the neutron and gamma ray flux is converted to the absorbed dose rate by applying the flux-to-dose rate conversion factors taken from Ref. [5, 9-10].
Figure 2. MCNP: Neutron spectrum at 3 m from the RB reactor

Figure 3. MCNP: Gamma spectrum at 3 m from the RB reactor
Table I. Measured and calculated neutron fluence at the irradiation position in the EXP-1

<table>
<thead>
<tr>
<th>$\Delta E$ [MeV]</th>
<th>Foils: $F_\text{n} \left[10^{9} \text{n/cm}^2\right]$</th>
<th>MCNP: $F_\text{n} \left[10^{9} \text{n/cm}^2\right]$</th>
<th>MCNP: $F_\text{n} \left[10^{9} \text{n/cm}^2\right]$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$1.0 \times 10^6$ - $0.4 \times 10^6$</td>
<td>$3.49 \pm 0.30$</td>
<td>$1.3318 \pm 0.0065$</td>
<td>$3.3391 \pm 0.0163$</td>
</tr>
<tr>
<td>$0.4 \times 10^6$ - $1.0 \times 10^6$</td>
<td>$1.53 \pm 0.33$</td>
<td>$0.5969 \pm 0.0105$</td>
<td>$1.4966 \pm 0.0263$</td>
</tr>
<tr>
<td>$1.0 \times 10^6$ - $0.75 \times 10^6$</td>
<td>$0.84 \pm 0.17$</td>
<td>$0.3971 \pm 0.0127$</td>
<td>$0.9936 \pm 0.0318$</td>
</tr>
<tr>
<td>$0.75 \times 10^6$ - $1.5 \times 10^6$</td>
<td>$0.10 \pm 0.05$</td>
<td>$0.0315 \pm 0.0450$</td>
<td>$0.0789 \pm 0.1128$</td>
</tr>
<tr>
<td>$1.5 \times 10^6$ - $2.5 \times 10^6$</td>
<td>$0.104 \pm 0.015$</td>
<td>$0.0402 \pm 0.0396$</td>
<td>$0.1007 \pm 0.0993$</td>
</tr>
<tr>
<td>$2.5 \times 10^6$ - $10$</td>
<td>$0.096 \pm 0.015$</td>
<td>$0.0594 \pm 0.0321$</td>
<td>$0.1489 \pm 0.0805$</td>
</tr>
<tr>
<td>Total:</td>
<td>$6.16 \pm 0.32$</td>
<td>$2.4569 \pm 0.0005$</td>
<td>$6.1599 \pm 0.0125$</td>
</tr>
</tbody>
</table>

Figure 4. Normalised neutron fluence at 3 m from the RB reactor at $P = 2.63$ kW

Continuously measured gamma ray dose rate in air [3], by the KAKTUS system, at the irradiation position during the EXP-1 at the reactor steady state power was 0.773 Gy/h, i.e., authentically: 88.2 R/h. The conversion factor 1 R = 8.764 mGy for absorbed gamma dose in air is applied, [11]. It is 14.3 % higher gamma ray absorbed dose rate in air than measured by other methods (TLD/PLD/F*) and 41.6 % higher than the calculated one (Table II). These differences can be explained by: (1) contribution of the $\alpha, \gamma$ reaction at double walls of the gamma ray chamber, which is not included in the calculated or measured (TLD/PLD/F*) absorbed gamma ray dose rate in air, and (2) by the fact that the MCNP code does not generate delayed gamma rays from fission process [5]. The calculated neutron absorbed dose in air at the irradiation position is 26.5 % higher than the measured one, which is determined as the sum of the contribution of the recoils and protons and $\text{H(n,\gamma)}$ reaction in the dosimeters [2-3]. At the same time, the calculated neutron flux in the highest energy group is higher than measured one (Figure 4 and 5), contributing more to the absorbed dose.

1 TLD = Thermoluminescent dosimeter; PLD = Photoluminescent glass dosimeter; F = Film dosimeter

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Table II. Measured and calculated absorbed dose in air and corresponding dose rate at the irradiation position in the EXP-1

<table>
<thead>
<tr>
<th>Quantity Determined by</th>
<th>Measurement, [3]</th>
<th>MCNP at P = 2.63 kW</th>
</tr>
</thead>
<tbody>
<tr>
<td>$D_n$ [Gy]</td>
<td>$0.378 \pm 0.053$</td>
<td>$0.478 \pm 0.012$</td>
</tr>
<tr>
<td>$\dot{D}_n$ [Gy/h]</td>
<td>$0.708 \pm 0.094$</td>
<td>$0.851 \pm 0.021$</td>
</tr>
<tr>
<td>$D_T$ [Gy]</td>
<td>$0.38 \pm 0.04$</td>
<td>$0.307 \pm 0.008$</td>
</tr>
<tr>
<td>$\dot{D}_T$ [Gy/h]</td>
<td>$0.676 \pm 0.071$</td>
<td>$0.546 \pm 0.015$</td>
</tr>
</tbody>
</table>

Conclusion

In this paper, the results of the calculations of the neutron and gamma ray spectra and corresponding dose quantities, for steady state reactor power during the 3rd International Intercomparison Experiment on Nuclear Accident Dosimetry, are presented and compared with the measured ones. Very good agreements are found in the measured and calculated neutron spectra at the irradiation position. For the first time, the calculated gamma ray spectrum at the same location is given.

Discrepancies in correlations: the reactor reported power - calculated dose rate and intensity of the neutron fluence with the measured ones are found. These disagreements have led to deduction that the RB reactor was operated at higher power than reported. This conclusion is supported also by few facts, including following:

(a) The absolute fission power of the RB reactor was not measured for the actual core (#5/1973) used. Instead, the power calibration factor for the regular RB cores was applied during the Experiment. It was: $2.4 \text{ V} \times 10 \text{ W}$ for $R_n$ at the linear channel 'K6' with the preamplifier PCC-32 K6: $R_e = 10^9 \Omega$, $R_n = 10^7 \Omega$. 

![Figure 5. Measured and calculated neutron spectrum at 3 m from the RB reactor](image-url)
(b) All neutron ionisation chambers of the RB reactor, except one ('K4') that are used for the reactor fission power monitoring, were either covered by cadmium foil or moved far away from their regular positions (nearby the reactor tank wall) to avoid their saturation at the required (high) operating reactor power. Intercomparison of reading of all neutron ionisation chambers was then possible (and has been performed) only at the reactor low power (10 W). The reading of the K4 neutron chamber at full operation power was not possible: it was out of range;

c) The results of the MCNP calculations were verified for determinations of the RB criticality and relations: the absolute neutron flux - fission power (within ± 10%) for different RB reactor's benchmark cores [7], including ones in which the absolute neutron flux measurements by activation of gold foils were carried out [12].

Acknowledgements

This work is part of the Project 08M06 supported by the Ministry of Science and Technology of Republic of Serbia.

References

7. M. Pešić, RB Reactor Benchmark Cores, paper presented at the 2nd Yugoslav Nuclear Society International Conference - YUNSC'98, Belgrade (September 28 - October 1, 1998)
10. The ICRP Committee 3 Task Group, P. Grant, M.C. O'Riordan, *Data for Protection Against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15*. ICRP-21, ICRP, Pergamon Press (April 1971)