INTERMEDIATE AND FAST NEUTRON ABSORBED DOSES IN FAST NEUTRON FIELD AT THE RB REACTOR

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ABSTRACT

The experimental fuel channel EFC is created as one of the fast neutron fields at the RB reactor. The intermediate and fast neutron spectrum in EFC are measured by activation technique. The intermediate and fast neutron absorbed doses are computed on the basis of these experimental results. At the end the obtained doses are compared.

INTRODUCTION

The RB is a zero power nuclear facility in the Boris Kidrič Institute of Nuclear Sciences /1/. It is possible to create the fast neutron fields by 80% enriched uranium fuel elements. The possibilities to obtain different neutron spectra are investigated using the converter and additional nonfissionable screens /2/. The modified experimental fuel channel EFC /3/ and convertors of neutrons inside the RB reactor CFTS-1 and CFTS-2 were created in continuing of this investigation /4/. The devices give well defined neutron and gamma fields.

EXPERIMENTAL FUEL CHANNEL (EFC)

EFC is realised by modification of the 80% enriched UO_2 RB reactor fuel elements. EFC is formed inside of the standard RB reactor fuel channel, an aluminium tube 41/43 mm diameter and bottom hermetically sealed, and filled with 10 modified 80% enriched UO_2 fuel segments (Fig.1). During fuel segment modification, central aluminium caliber with outermost "stars" are taken of, and rest of fuel segments is slipped on an aluminium tube 27/28 mm diameter, one to another, as closely as possible. All the arrangement is placed inside the standard fuel channel. Movable aluminium tube is placed inside the EFC 25/27 mm diameter, filled with aluminium expellers and sample supporters. Thus, the samples or detectors can be easily placed in the reactor or taken out. The EFC is placed in the RB reactor core at predetermined position (Fig.2).

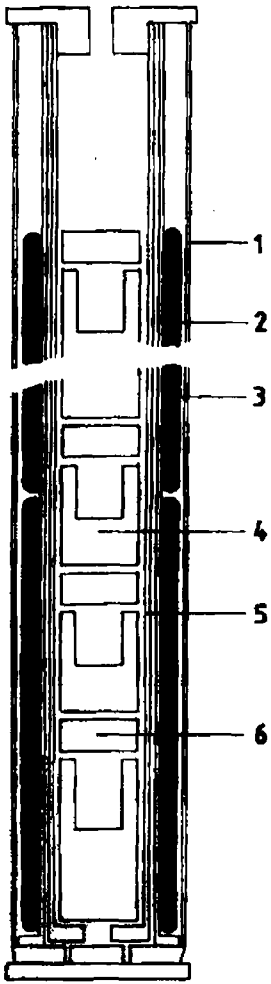

Fig.1. Experimental fuel channel

1 - Al fuel channel 43/41

2 - 80% enriched UO_2 fuel

3 - supporting Al tube 29/28

4 - Al caliber and sample supporters

5 - movable experimental Al tube 27/25

6 - Cd box

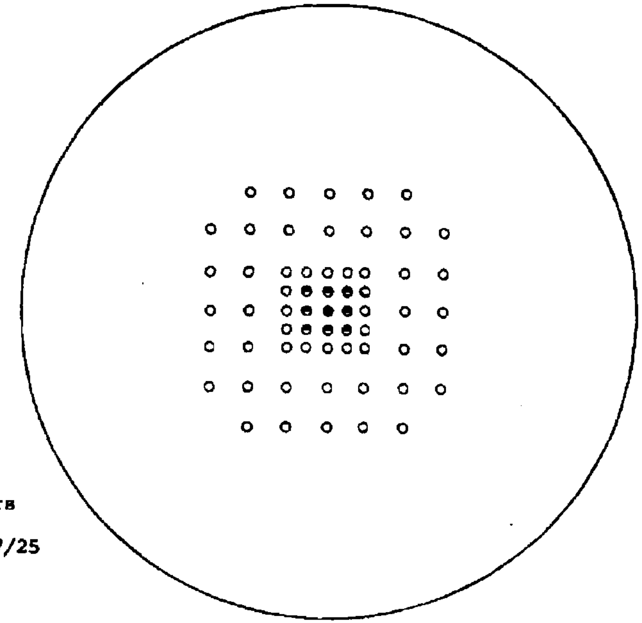

Fig.2. The RB core configuration with EFC postition

- fuel channels with 2% enriched U, lattice pitch 140 mm
- fuel channels with 80% enriched UO_2, without D_2O, lattice pitch 70 mm
- experimental fuel channel, EFC

INTERMEDIATE AND FAST NEUTRON SPECTRUM MEASUREMENTS

The intermediate and fast neutron spectrum are measured by activation technique. The method of resonance detectors /5/ for intermediate and threshold detectors /6/ for fast spectrum are used for these measurements. Some of the foils used as resonance detectors are given in Table 1, and those used as threshold detectors are given in Table 2. Foil activities are measured by using scintillation technique and $4\pi/\beta$ absolute counting method. The measuring results were treated with ACT code based on analytical relations accounting all necessary physical and geometrical corrections which returns foil saturated activity and neutron flux density. Intermediate spectrum is obtained with KRIFIT code and fast spectrum with HEFEST code on the basis of experimental results. The codes are based on the minimum mean square method and method of the maximal probability respectively.

Table 1. Some of the foils used for intermediate spectrum determination

Detector	E_R (eV)	RI (10^{-24} cm^2)	$T_{1/2}$
$^{176}Lu(n,\gamma)^{177}Lu$	0.142	900	6.71 d
$^{115}In(n,\gamma)^{116m}In$	1.457	3243	54 m
$^{197}Au(n,\gamma)^{198}Au$	4.906	1565	2.69 d
$^{186}W(n,\gamma)^{187}W$	18.8	350	24.1 h
$^{59}Co(n,\gamma)^{60}Co$	132	77	5.27 g

Table 2. Some of the foils used for fast spectrum determination

Detector	E_{eff} (MeV)	$_{eff}$ (10^{-27} cm^2)	$T_{1/2}$
$^{103}Rh(n,n')^{103m}Rh$	0.80	950	57 min
$^{115}In(n,n')^{115m}In$	1.15	302	4.5 h
$^{32}S(n,p)^{32}P$	2.65	252	14.3 d
$^{58}Ni(n,p)^{58}Co$	2.70	452	72 d
$^{54}Fe(n,p)^{54}Mn$	3.00	370	310 d

INTERMEDIATE AND FAST NEUTRON ABSORBED DOSES

Neutron spectrum in the air hole of the EFC is calculated with VESNA /7/ code. The spectrum is normalised using the mentioned absolute spectrum measurements. The neutron spectrum is calculated in 44 energy groups, condensed and normalised in 25 groups. This spectrum is converted in neutron absorbed dose rates with ADOS code. This code is based on analytical relations using absorbed dose - neutron flux density conversion factors. The neutron absorbed dose rates are condensed in 4 energy macrogroups. The obtained results are given in Table 3.

Table 3. Intermediate and fast neutron absorbed doses in EFC

Macrogroups $g_L - g_U$	ΔE	$\dot{D}a$ (Gy/Wh)
1 - 5	0.8 MeV - 10.5 MeV	0.320
6 - 12	4.65 keV - 800 keV	0.196
13 - 24	0.465 eV - 4.65 keV	0.030
25 - 44	1 meV - 0.465 eV	0.003

CONCLUSION

The intermediate and fast neutron absorbed doses in EFC are determined in this paper. The fast neutron absorbed dose has the highest contribution, what was expected because of the fact that the experimental channel is one of the fast fields at the RB reactor. The EFC will be used for dosimeter testing and callibration, irradiation studies and similar investigations.

REFERENCES

/1/ Popović D., A Bare critical Assembly of Natural Uranium and Heavy Water, Proceeding of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Vol.12, 392, Geneva, 1958.

/2/ Strugar P., Šotić O., Ninković M., Pešić M., Altiparmakov D., Kernenergie 24, 101, 1981.

/3/ Pešić M. Marković H., Šokčić-Kostić M., Mirić I., Prokić M., Strugar P., Kernenergie 27, 461, 1984.

/4/ Pešić M., Kernenergie 80, 142, 1987.

/5/ Šokčić-Kostić M., Pešić M., Antić D., Determining of the Intermediate Neutron Spectrum in Fast Neutron Field at the RB Reactor, XXXI Yougoslav Conference of ETAN, Bled, 1987.

/6/ Šokčić-Kostić M., Pešić M., Antić D., Fast Neutron Spectrum Determination with Threshold Detectors at the RB Reactor, Proceedings of the International Conference on Fast Neutron Physics, 250, Zagreb, 1986.

/7/ Milošević M., VESNA; a Multigroup Lattice Calculation Code with Nuclear Data Library NEDA, to be published.