



## A COMPARATIVE STUDY OF EFFECTIVE DELAYED NEUTRON FRACTION

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**Abstract** - Influence of various sets of evaluated six-group delayed neutron parameters on kinetic parameters  $\beta_{eff}$  and  $\beta_{eff}/\Lambda$  is analyzed. The comparative study of the kinetic parameters is made for the coupled fast-thermal system HERBE at the Vinča Institute. Methods of calculation and measurement techniques used in this analysis are described. The results of calculations are compared with measurements.

### 1. INTRODUCTION

In the past decade, the accuracy of data for delayed neutron emission probabilities and neutron emission spectra for individual nuclides has greatly improved. Those data are used to provide delayed neutron yields, half-lives and spectra in classical six-group representation. The accuracy of these six-group parameters is difficult to quantify (Brady and England (1989)). On the other hand, some experts (Lammer and Schwerer (1991)) suggest that greater significance be given to the experimental data (e.g., data of Keepin (1965)) than to the summation calculation presented by Brady and England (1989).

Therefore, a comparative analysis of the effective delayed neutron fraction at the coupled fast-thermal system HERBE (Pešić et al. (1991)) is performed. Results of calculations based on the evaluated experimental data and summation calculation data for the six-group delayed neutron representations are compared to our experimental results. Methods applied in these calculations and measurements are described.

### 2. CALCULATIONS

The coupled fast-thermal system HERBE is used for experimental verification of reactor design-oriented methods. Complexity of the HERBE system arises from strong heterogeneous effects related to its design. The fast core of the HERBE system is loaded with the natural uranium fuel elements. It is surrounded by a neutron filter and a neutron converter. The neutron filter contains a thin cadmium layer and natural uranium fuel elements. The thermal core is composed of the 80 % enriched  $UO_2$  fuel elements arranged in the 12 cm square lattice, moderated and reflected by heavy water.

In kinetic calculation of the HERBE system, the neutron flux and adjoint neutron flux are obtained by applying the following:

- multigroup two-dimensional assembly spectrum code VEGA (Milošević (1990)) based on the collision probability method;
- one-dimensional reactor code AVERY (Milošević and Pešić (1989)), based on the multigroup integral transport equation, and
- few group two-dimensional diffusion code GALER (Milošević (1989)), based on the Galerkin's method.

Value of the effective delayed neutron fraction  $\beta_{eff}$  is calculated on the basis of the perturbation theory, while the value of prompt neutron generation time  $\Lambda$  is obtained as reactivity response to "1/v absorber" (Keepin (1965)).

Seven sets of evaluated six-group delayed neutron parameters, based on integral experiments and summation calculations, are chosen to compare with our measurement results of  $\beta_{eff}$  and  $\beta_{eff}/\Lambda$ . Each set consists of the total number of delayed neutrons per fission  $\bar{\nu}_d$  (Table I), normalized group strength  $a_i$ , effective decay constants  $\lambda_i$ , and delayed neutron spectra  $\chi_{d,i}(E)$  for  $i$ 'th group of delayed neutron precursors.

Table I. The total number of delayed neutrons per fission  $\bar{\nu}_d$

Data set	Total number of delayed neutron per fission	
	$^{235}\text{U}$ (thermal fission)	$^{238}\text{U}$ (fast fission)
1. ENDF/B-IV	0.0167	0.0450
2. ENDF/B-V	0.0167	0.0430
3. ENDF/B-VI	0.0167	0.0430
4. JENDL3.1	0.0160	0.0466
5. BE	$0.0178 \pm 0.0010$	$0.0405 \pm 0.0029$
6. V1	$0.0158 \pm 0.0005$	$0.0412 \pm 0.0017$
7. V2	$0.0165 \pm 0.0004$	$0.0451 \pm 0.0006$

The chosen data sets are taken from the following experiments or evaluations:

1. ENDF/B-IV, based on data evaluated by Cox (1974);
2. ENDF/B-V, based on measured data given by Kaiser and Carpenter (Cox (1974));
3. ENDF/B-VI, based on data given by England et al. (1988) and Brady (1988);
4. JENDL3.1, based on data given by Keepin (1965) and Evans and Thorpe (1973) for parameters  $\bar{\nu}_d$ ,  $a_i$ ,  $\lambda_i$ , and by Saphier et al. (1977) for  $\chi_{d,i}(E)$ ;
5. BE, based on data by Brady and England (1989) for  $\bar{\nu}_d$ ,  $a_i$ ,  $\lambda_i$  parameters, while the  $\chi_{d,i}(E)$  are taken from ENDF/B-VI set;
6. V1, based on data given by Keepin (1965) for  $\bar{\nu}_d$ ,  $a_i$ ,  $\lambda_i$  parameters, and Saphier et al. (1977) for  $\chi_{d,i}(E)$ , and
7. V2, based on data by Tuttle (1975) for  $\bar{\nu}_d$ , Brady and England (1989) for  $a_i$  and  $\lambda_i$  parameters, while the  $\chi_{d,i}(E)$  are taken from ENDF/B-VI.

Each of these seven data sets is supplemented with an additional set of photoneutron nine-group data given by Keepin (1965). The photoneutron spectra are evaluated by a simplified method based on isotropic emission of neutron and proton with equal energies.

### 3. EXPERIMENT

For the purpose of comparison, two experiments at the HERBE system are carried out. In the first experiment, the effective delayed neutron fraction  $\beta_{eff}$  is measured by a method based on perturbation theory (Carpenter et al. (1972)). In the second experiment, the kinetic quantity  $\beta_{eff}/\Lambda$  is obtained by using the Feynman- $\alpha$  method (Feynman et al. (1956)).

#### 3.1 Measurement of $\beta_{eff}$

The effective delayed neutron fraction  $\beta_{eff}$  in the HERBE coupled fast-thermal system is determined by measurement of fission rate, adjoint neutron flux and reactivity worth of a calibrated  $^{241}\text{Am}$ -Be neutron source, according to the following relation (Carpenter et al. (1972)):

$$\beta_{eff} = \frac{S \Phi_x^*(\vec{r})}{(\rho/\beta_{eff}) \bar{\nu} \cdot F} f(\vec{r})$$

where:

$S$  is the absolute neutron source strength,

$\Phi_x^*(\vec{r})$  is the average importance of a fission neutron at the source position ( $\vec{r}$ ),

$\rho/\beta_{eff}$  is the apparent reactivity effect of the source neutrons,

$\bar{\nu}$  is the average number of neutrons per fission,

$F$  is the total fission rate in the system, and

$f(\bar{r})$  is a correction factor, calculated as the ratio of the average importance for the source and fission neutrons.

Absolute intensity of the  $^{241}\text{Am}$ -Be neutron source is validated by the flux density method. The two group neutron flux distribution in the HERBE subcritical system with the  $^{241}\text{Am}$ -Be neutron source is measured by irradiation of the gold foils with and without cadmium covers. Absolute gamma activity of the irradiated gold foils is measured by a calibrated coaxial germanium detector. Gamma spectra of the irradiated gold foils are measured long enough to obtain sufficient statistical data for automatic peak search and fit ( $E_\gamma=411.80$  keV, yield=95.57%), while the absolute activity is evaluated by the APOGEE computer code (Canberra Industries Inc. (1986a)). These foil activities are used for absolute calibration of an automatic beta counting system with a flow gas (P-10) counter of high efficiency (Canberra Industries Inc. (1986b)) in order to measure activities of gold foils used in mapping the neutron flux distribution.

In this experiment two cases of local perturbation of neutron flux occurred due to following: (1) insertion of experimental aluminum channels and supports of activation foils, and (2) insertion of the foils and cadmium covers. Influence of the first effect is calculated by VEGA code. A new computer program NSD (Milošević (1992)), based on the collision probability method in slab geometry, is developed to account for the second effect. It uses an enhanced method only in resonance groups of foil material. It is assumed that the remaining nuclides in these groups behave as narrow-resonance scatterers with effective cross sections equal to their multigroup shielded cross sections. This assumption allows the flux and cross sections for foil material to be calculated by a numerical slowing-down method with ultrafine group structure. The results of neutron flux calculations and measurements in the HERBE subcritical system with the  $^{241}\text{Am}$ -Be neutron source are shown in Figure 1.

Relative fission rates are obtained by measurement of space and energy distribution of the absolute neutron flux (at a power level where the effect of the external source is much enhanced) and by calculated fission cross-sections. A calibrated coaxial germanium detector is used to measure gamma-ray activity of  $^{140}\text{La}$  nuclide from an irradiated  $\text{UO}_2$  sample and evaluated by APOGEE code in order to determine absolute fission rate density. Nuclear data for  $^{140}\text{Ba}$  and  $^{140}\text{La}$  nuclides are taken from the ENDF/B-V files. Corrections for local flux perturbations in this experiment, due to insertion of the experimental channels, aluminum supports, gold foils, cadmium covers and  $\text{UO}_2$  samples, are calculated by the NSD code. The results of the calculations and measurements for the fission rate density in the HERBE system are shown in Figure 2.

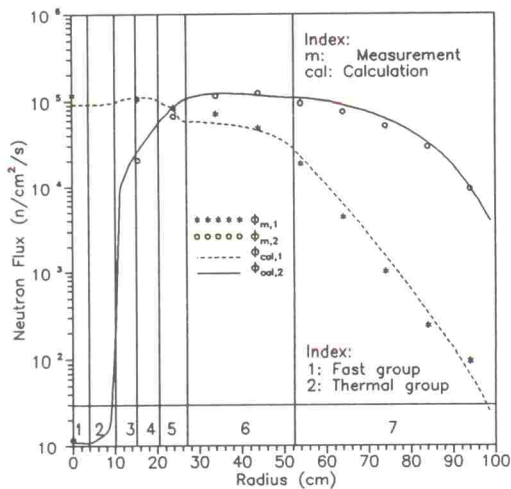


Fig.1. Measured and calculated neutron flux distribution

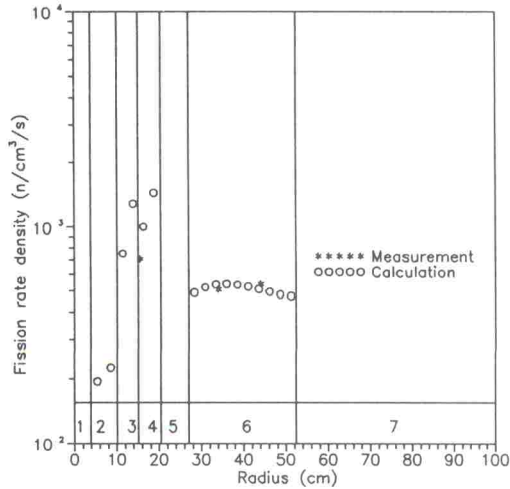


Fig.2. Measured and calculated fission rate density

- Legend:
- 1) Vertical channel
  - 2) Fast core
  - 3) Neutron filter
  - 4) Neutron converter
  - 5) Inner reflector
  - 6) Thermal core
  - 7) Outer reflector

Value of the  $\phi_x^+(\vec{r})$  is determined by measurement of two group adjoint neutron flux distributions (Profio (1976)). These distributions are derived from measurements of the neutron flux densities and reactivity worths of an absorbing material (cadmium) and a fission sample ( $\text{UO}_2$  fuel; 80 % enriched  $^{235}\text{U}$ ). The results of the adjoint neutron flux calculations and measurements for the HERBE system are given in Figure 3.

The measured reactivity of the  $^{241}\text{Am}$ -Be neutron source ( $\rho_m$ ) in the HERBE system originates from two effects (Carpenter (1972)). The first one is neutron interactions in the source materials ( $^{241}\text{Am}$ , Be and the source container) producing reactivity  $\rho_0$ . The second effect is the emission of neutrons from the source contributing to apparent reactivity  $\rho$ . The reactivity  $\rho_0$  is independent of the neutron flux, while the apparent reactivity varies inversely to the neutron flux. Accordingly, the  $\rho$  is obtained by measuring the reactivity at several different reactor power levels (Figure 4). For each measured value of  $\rho_m$ , a constant reactor power level is maintained, the reactor being subcritical for an amount required to compensate the effect of the neutrons emitted from the external source.

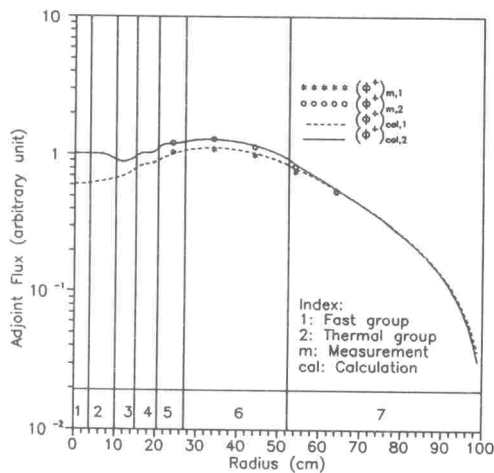


Fig.3. Measured and calculated adjoint neutron flux (The same Legend as for Figs. 1. and 2.)

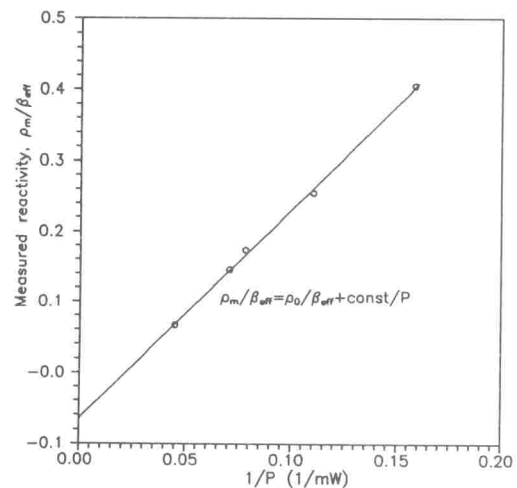


Fig.4. Reactor response to the  $^{241}\text{Am}$ -Be neutron source

The value of the average number of neutrons per fission ( $\bar{\nu}$ ) is determined by few multigroup two-dimensional ( $r$ - $z$ ) calculations carried out by GALER code. The value of the correction factor  $f(\vec{r})$  is calculated by VEGA and AVERY codes. The  $^{241}\text{Am}$ -Be neutron source spectrum, used in the calculations, is taken from the International Atomic Energy Agency Technical Report Series' Publication (IAEA (1990)).

The measured quantities used in determination of the effective delayed neutron fraction  $\beta_{eff}$ , mentioned above, and estimated uncertainties are given in Table II.

Table II. Measured quantities for determination of the  $\beta_{eff}$

Quantity	Measured Value	Estimated Error [%]
$S$ [neutrons/s]	$1.79 \cdot 10^6$	$\pm 2.0$
$\rho / \beta_{eff}$	0.2256	$\pm 1.5$
$\bar{\nu}$	2.442	$\pm 0.5$
$F$ [fissions/s]	$4.215 \cdot 10^8$	$\pm 2.0$
$\phi_x^+(\vec{r}_0)$	1.057	$\pm 1.0$
$f(\vec{r}_0)$	0.970	$\pm 1.0$

3.2 Measurement of  $\beta_{eff}/\Lambda$

The Feynman- $\alpha$  method, known also as the "variance-to-mean" method, is applied to determine the ratio of the effective delayed neutron fraction to the prompt neutron generation time ( $\beta_{eff}/\Lambda$ ) in the HERBE system, by measuring the prompt neutron decay constant  $\alpha$ .

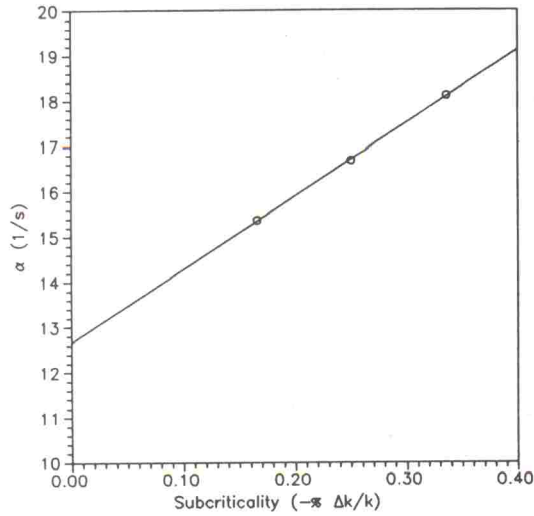


Fig. 5. Measured prompt neutron decay constant as function of reactivity

The neutrons are detected by two BF<sub>3</sub> pulse-type detectors, and  $\alpha$  is determined by measuring the mean ( $\bar{c}$ ) and variance values ( $\overline{c^2} - \bar{c}^2$ ) of the neutron count rate in a certain time interval ( $\tau$ ). In the space-independent reactor kinetics with n-group delayed neutrons, the formula used in the variance-to-mean method is

$$Y = \frac{\overline{c^2} - \bar{c}^2}{\bar{c}} - 1 = \sum_{i=1}^{n+1} K_i \left[ 1 - \frac{1 - \exp(-\alpha_i \tau)}{\alpha_i \tau} \right]$$

where parameters  $K_i$  and  $\alpha_i$  are determined by a fitting procedure (Zavaljevski et al. (1994)). The constant  $\alpha = [\alpha_1]$  is defined as:

$$\alpha = (\beta_{eff} - \rho) / \Lambda$$

where  $\rho$  is the reactivity of the system. In a series of these experiments, the ratio of  $\beta_{eff}/\Lambda$  is obtained by extrapolating the measured  $\alpha$  values in a few subcritical states to the critical one (Figure 5).

4. RESULTS AND CONCLUSION

The comparative study of the kinetic parameters is performed for the coupled fast-thermal system HERBE. The calculated quantities of  $\beta_{eff}$  and  $\beta_{eff}/\Lambda$  are compared to the measured ones in Table III. All the results of the calculations and measurements for  $\beta_{eff}$  and  $\beta_{eff}/\Lambda$ , except the BE set (due to high value of  $\bar{v}_d$ ), agree with the Keepin experimental data set, considering that the standard deviation of Keepin's  $\bar{v}_d$  is 4%.

Table III. Results of calculations and measurements of  $\beta_{eff}$  and  $\beta_{eff}/\Lambda$

I. Calculation based on data set	$\beta_{eff}$	$\beta_{eff}/\Lambda$ [1/s]
1. ENDF/B-IV	0.00813	12.79
2. ENDF/B-V	0.00806	12.68
3. ENDF/B-VI	0.00808	12.71
4. JENDL3.1	0.00782	12.30
5. BE	0.00850	13.36
6. V1	0.00770	12.10
7. V2	0.00803	12.62
II. Measurement	$0.00791 \pm 0.00028$	$12.67 \pm 0.39$

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## REFERENCES

- Brady M.C. (1988) *Evaluation and Application of Delayed Neutron Precursors Data*, PhD Dissertation, Texas A&M University; also Report LA-11534-T (1989), Los Alamos National Laboratory USA
- Brady M.C., England T.R. (1989) *Nucl. Sci. Eng.* **103**, 129
- Canberra Industries Inc. (1986a) *APOGEE Version 1.3, Gamma Spectrum Analysis Code*
- Canberra Industries Inc. (1986b) *Low Level Alpha/Beta 2400F Automatic Counting System*
- Carpenter S.G., Gasidlo J.M., Stevenson J.M. (1972) *Nucl. Sci. Eng.* **49** (2), 236
- Cox S.A. (1974) *Delayed Neutron Data - Review and Evaluating*, Report ANL/NDM-5, Argonne National Laboratory. For ENDF/B-V, these data were updated by R.E.Kaiser and S.G.Carpenter, Argonne National Laboratory-West (unpublished)
- England T.R., Arthur E.D., Brady M.C., LaBauve R.J. (1988) *Background Radiation from Fission Pulses*, Report LA-11151-MS, Los Alamos National Laboratory USA
- Evans A.E., Thorpe M.M. (1973) *Nucl. Sci. Eng.*, **50**, 80
- Feynman R.F., de Hoffmann F., Serber R. (1956) *J. Nucl. Energy* **3**, 64
- IAEA (1990) *Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes*, Technical Report Series No. 318, Vienna Austria
- Keepin G.R. (1965) *Physics of Nuclear Kinetics*, Addison-Wesley Publ. Co., Reading USA
- Lammer M., Schwerer O. (1991) *Handbook of Nuclear Data for Safeguards*, Preliminary Issue, INDC(NDS)-248 IAEA Vienna Austria
- Milošević M., Pešić M. (1989a) *Program AVERY for Coupled Fast-Thermal Reactors Kinetics Parameters Calculation*, Report IBK-NET-27 (in Serbian), Institute of Nuclear Sciences "Vinča", Yugoslavia
- Milošević M. (1989b) *Multigroup Diffusion Code GALER in r-z Geometry Based on Galerkin's Method*, Report IBK-NET-29 (in Serbian), Institute of Nuclear Sciences "Vinča", Yugoslavia
- Milošević M. (1990) *VEGA Assembly Spectrum Code*, Report IBK-NET-30 (in Serbian), Institute of Nuclear Sciences "Vinča", Yugoslavia
- Milošević M. (1992) *Program NSD for Calculation of Reaction Rates in Slab Geometry*, Report Vinča-NET-72 (in Serbian), Institute of Nuclear Sciences "Vinča", Yugoslavia
- Pešić M., Zavaljevski N., Milošević M., Stefanović D., Popović D., Nikolić D., Marinković P., Avdić S. (1991) *Ann. nucl. Energy*, **18**, 413
- Profio A.E. (1976) *Experimental Reactor Physics*, John Wiley & Sons, New York USA
- Saphier D., Ilberg D., Shalev S., Yiftah S. (1977) *Nucl. Sci. Eng.*, **62**, 560
- Tuttle R.J. (1975) *Nucl. Sci. Eng.*, **56**, 37
- Zavaljevski N., Kostić Lj., Pešić M. (1994) *Proc. of ANS 1994 Topical Meeting: 'Reactor Physics Faces the 21st Century'*, Vol. II, pp. 389-398, Knoxville, TN USA