INITIAL INVESTIGATIONS FOR SAD PROJECT IN INRNE-BAS

M. Todorov, N. Janeva, L. Kostov

INRNE-BAS, Tzarigradsko Shaussee Blvd, №72, 1784 Sofia, Bulgaria
todoroff@inrne.bas.bg

Abstract

The scientific and technical potential of JINR is an important advantage of the project SAD [1], which proposes creation of experimental installation on the basis of the existing proton accelerator "Phazotron" and subcritical blanket with Uranium-Plutonium fuel from existing and modernized fuel elements of "BN-600" reactor type. We report some preliminary result on methods for calculating group constants adapted to the facility SAD, ^{232}Th resonance cross-section evaluation. The most important achievements are computer codes and something about nuclear data. Initial research program for SAD system is measuring the spallation products yield. This is done by γ-spectroscopy method in JINR with participation of Bulgarian scientists.

Introduction

The contemporary concept of resolving the spent fuel problem is that of nuclear transmutations. Transuranium nuclides are transformed by fission and long-lived fission products after neutron capture into stable or short-lived nuclei.

There exist a large number of projects for reactor systems driven by the accelerator, but still long way towards realization. Of great interest in this field is the great European project MYRHA[2] - future of this innovative nuclear technology -, and the other very attractive, approachable subcritical assembly in Dubna (SAD)[1] at JINR.

Nowadays, SAD program for research and developing proposes to improve neutron characteristics of the system, to study spallation products yields, methods and codes for modeling system of such kind and their isotopic structure and geometry (benchmark, experiments for computer codes and nuclear data validation). At the same time, we consider enabling ADS with Thorium fuel.

Methods for evaluation of group cross-sections adapted to SAD

The method for calculation neutron spatial-energy distribution is presented. We choose the integral neutron transport equation with integrated over angle flux in the zone (cell) j and neglect the effects of anisotropy in scattering:

$$F_j(\vec{r},E)=\sum_i\int_{v_c} d\vec{r}\,'q_i(\vec{r}\,',E)K(|\vec{r}-\vec{r}\,'|,E), \tag{1}$$

where

$$q_i(\vec{r},E)=Q_i(\vec{r},E)+\int dE'F_i(\vec{r},E')\Sigma_{si}(E')W(E'\to E). \tag{2}$$

The sum in (1) is related to all zones of the system chosen by isotopic homogeneity; q_i is the sum of neutron sources generated by fission, inelastic scattering and $(Q_i(\vec{r},E))$ is moderation integral from elastic scattering.

$$K(|\vec{r}-\vec{r}'|,E)=\frac{1}{4\pi|\vec{r}-\vec{r}'|^2}\exp\left[-\sum_K n_k(\vec{r}-\vec{r}')\sigma_k(E)\right] \quad (3)$$

is so-called kernel of the Peierls equation, where the sum by k concern all elements in cylindrical channel on single section to direction $\vec{r}-\vec{r}'$ and length $|\vec{r}-\vec{r}'|$; $n_k|\vec{r}-\vec{r}'|$ - is nuclei number of *k*-th element in this channel[3,4,5,].

The group presentation follows from averaging of the equation (1) over the corresponding energy intervals $(\langle ... \rangle_\lambda)$. It is supposed moreover that as a first approximation the moderation integral $q_i(\vec{r},E)$ (2) depends smoothly on energy, so that [4,6]:

$$q_i(\vec{r},E)\approx Q_i(\vec{r},E)+\int dE'\langle F_i(\vec{r},E')\Sigma_{si}(E')\rangle_k W(E'-E), \quad (4)$$

$$\langle F_i(\vec{r},E)\Sigma_{si}(E)\rangle_k \approx \sum_\ell \int_{V\ell} d\vec{r}' q_\ell(\vec{r}',E)\langle \Sigma_{si}(E)K(|\vec{r}-\vec{r}'|,E)\rangle; \quad (5)$$

$$q_i(\vec{r},E)\approx Q_i(\vec{r},E)+\sum_\ell \int_{V\ell} d\vec{r}' \int dE' q_\ell(\vec{r}',E')W(E'-E)\langle \Sigma_{si}(E')K(|\vec{r}-\vec{r}|,E')\rangle. \quad (6)$$

Using the "narrow resonance" approximation in resonance region q_i are given in the following form:

$$q_i(\vec{r},E)\approx Q_i(\vec{r},E)+\varphi_0\Sigma_{ps}, \quad (7)$$

which goes after the equation (2); for energies above the resonance integral term is replaced with $F_i\Sigma_{si}\approx\varphi_0\Sigma_{pi}=const$, and also from equation (6) at $q_\ell\approx\varphi_0\Sigma_{p\ell}$, $\Sigma_{si}\approx\Sigma_{pi}$ accounting normalization of the Peierls equation [5,7] kernel:

$$\sum_\ell \int_{V_\ell} d\vec{r}\Sigma_\ell K(|r-r'|,E)=1.$$

The scheme continues with consecutive approximations accounting energy dependence $q_i(\vec{r},E)$ (6). In our methodology for resonance averaging in energy group the neutron flux is given by:

$$\langle F_j(\vec{r},E)\rangle=\sum_i \int_{Vi} d\vec{r}' q_i(\vec{r}',E)\langle K(|\vec{r}-\vec{r}'|,E)\rangle, \quad (8)$$

resonance absorption distribution is given with:

$$\langle \Sigma_{aj}(E)F_j(\vec{r},E)\rangle=\sum_i \int_{Vi} d\vec{r}' q_i(\vec{r}',E)\langle \Sigma_{aj}(E)K(|\vec{r}-\vec{r}'|,E)\rangle, \quad (9)$$

where basic characteristic of elementary interaction are included in the next average functional on energy groups of neutron cross section:

$$\langle \exp[-n(\vec{r}-\vec{r}')\sigma]\rangle_k,\ \langle \sigma_a \exp[-n(\vec{r}-\vec{r}')\sigma]\rangle_k,\ \langle \sigma_S \exp[-n(\vec{r}-\vec{r}')\sigma]\rangle_k. \quad (10)$$

In the case of source φ_0 regularly distributed on the block surface S in narrow resonance approximation, the equation of neutron flux (1) can be written as [4,7]:

$$F(\vec{r},E)=\varphi_0\left[\frac{\Sigma_p}{\Sigma(E)}J_V(\vec{r},E)+J_S(\vec{r},E)\right], \quad (11)$$

where

$$J_V(\vec{r},E)=\Sigma(E)\int_V d\vec{r}'K(|\vec{r}-\vec{r}'|,E) \quad (12)$$

$$J_S(\vec{r},E)=\int_S d\vec{r}_S'K(|\vec{r}-\vec{r}_S|,E)\left(\frac{\vec{r}-\vec{r}_S}{\vec{r}-\vec{r}_S}\vec{n}\right) \quad (13)$$

$\vec{n}$ is normal vector toward the surface of block in point $\vec{r}_S$. Volume integral J_V and surface integral J_S are connected by the reciprocity condition:

$$J_S(\vec{r},E)=1-J_V(\vec{r},E). \quad (14)$$

Averaging in energy intervals in resonance region (in relevant energy groups[8]), expression for spatial distribution of group flux and absorption distribution is given as:

$$\langle F(\vec{r},E)\rangle=\varphi_0\left|\left\langle\frac{\Sigma_p}{\Sigma}J_V(\vec{r},E)\right\rangle+\langle J_S(\vec{r},E)\rangle\right. \quad (15)$$

$$\langle\Sigma_a(E)F(\vec{r},E)\rangle=\varphi_0\left[\left\langle\frac{\Sigma_p}{\Sigma}\Sigma_a J_V(\vec{r},E)\right\rangle+\langle\Sigma_a J_S(\vec{r},E)\rangle\right], \quad (16)$$

with accounting K(3) in (12):

$$\left\langle\frac{\Sigma_p}{\Sigma}J_V(\vec{r},E)\right\rangle=\Sigma_p\int_V\frac{d\vec{r}'}{4\pi|\vec{r}-\vec{r}'|^2}\left\langle e^{-n|\vec{r}-\vec{r}'|\sigma(E)}\right\rangle \quad (17)$$

$$\left\langle\frac{\Sigma_p}{\Sigma}\Sigma_a J_V(\vec{r},E)\right\rangle=\Sigma_p\int_V\frac{d\vec{r}'}{4\pi|\vec{r}-\vec{r}'|^2}\left\langle\Sigma_a e^{-n|\vec{r}-\vec{r}'|\sigma(E)}\right\rangle \quad (18)$$

and by the similar manner in surface integrals in (15), (16).

Obviously, for accomplishing this method it is necessary to define average transmission by the resonances in every energy group $\langle\exp[-n\sigma]\rangle$ and self-indication $\langle\sigma_a\exp(-n\sigma)\rangle$ for possible value of n. In the experimentally resolved resonance region, energy average is immediately obtained, but there isn't any direct information for the resonance structure in unresolved region. In that case, we need to use structure models of that kind or results from transmission experiments or self-indication cross-section.

The proposed method is convenient for calculation of heterogeneous effect in ADS accounting resonance self-shielding. Likewise, the method is good for benchmark in unresolved resonance region using the experimental results for transmission and self-indication as group constants.

Methods for experimental assignment of spallation products

Methods for experimental investigation of spallation products and high-energy ions with material interaction modeling were accomplished by collaboration with JINR. These research includes the following tendencies:

- Residual nuclei formation in reactions induced by protons with energy 660 MeV, which interact with radioactive targets such as 232Th, 235U and 238Pu;
- Transmutation of long-lived radioactive nuclei such as 129I, 237Np, 241Am, 238U, 239Pu, 232Th, 235U and 238Pu;
- Study of neutron spectra created by heavy targets (Pb, Pb+Bi) irradiation with medium energy particles.

Mathematical formalism, respectively method for assignment of values *B(A)* obtained from measured gamma-lines an area that describes the generated isotopes in activation detectors or transmuted targets. *B(A)* values describes the generation speed of obtaining new isotopes (A) as transmutation cross-section. Software for computer realization on this formalism and program for gamma-spectra processing, DEIMOS, were offered by the Djelepov Nuclear Problems Laboratory of JINR. At the present time this software is adopted and studied in INRNE.

We participated in deuteron irradiation experiments with energy 700 MeV on "Energy plus transmutation" experimental device at Nuclotron (Fig. 1) in the laboratory high-energy – JINR. Large amount on experimental γ-lines of spallation products were processed in Sofia. "Nuclear Spectroscopy" laboratory in INRNE utilize software for γ-lines and assignment programs for (n,xn), (n,α), (n,γ) reaction cross-section analysis, which are important in reaction of spent nuclear fuel transmutation

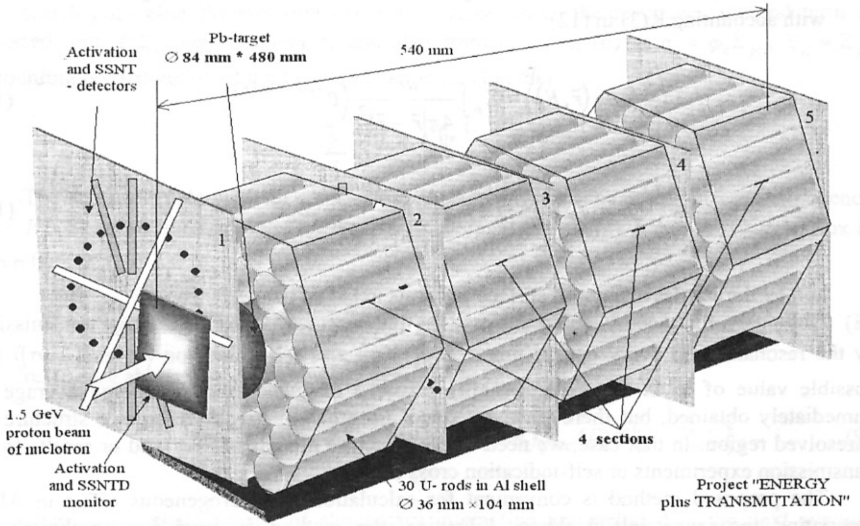

Fig. 1: Technical design of four sections U/Pb assembly with uranium blanket around a massive lead target.[9]

Neutron data for SAD

Much of the nuclear data currently needed for ADS have become available with sufficient accuracy, which allowed an initial assessment of the concepts. As the design becomes more precise and sophisticated, a considerable amount of more accurate data is required. The data needs range from general reaction cross-sections, radiation as well as transmutation reaction to radiation damage and activation. Similar data necessary not only for protons and neutrons, but also for specific elements and nuclides take part in the system design. Presently, neutron data below 20 MeV have been developed extensively and appreciably high accuracy is assured through the exercises in fission reactor programs. Thus, major focus of data need research has been at the high-energy region above 20 MeV up to several GeV. Presently neutron data below 20 MeV have been developed extensively and high accuracy is assured through the exercise in fission reactor programs. [10]

Subcritical reactors systems driven by accelerator with Th fuel are interesting proposal for waste burning from usual power plants. The ^{232}Th (n,γ) neutron capture cross-section is of great importance for ADS based on the Th-U fuel cycle that have noticeable advantages. It can be easily shown that a ± 10% uncertainty on σ_c of the ^{232}Th can produce an uncertainty of approximately 30% on the proton current requirement if the system has to be operated at a subcritical level of $K_{eff} \approx 0, 97$.[11]

Measurements of neutron resonance capture cross section in the energy region 4 -140 KeV were performed in Geel (Belgum) by n-TOF method [12]. The achieved results are with high accuracy. Systematical errors from normalization are under 1% and addition of 0.5% that are induced from the self-shielding and multiple scattering in the target. Resonance analysis of obtained unique experimental data was done on the basis of the existing statistical models and our method of the characteristic function at terms of R-matrix elements distribution.

The combined analysis of the total and radiation cross-sections determinate energy averaged resonance parameters of the ^{232}Th.(Fig. 2)

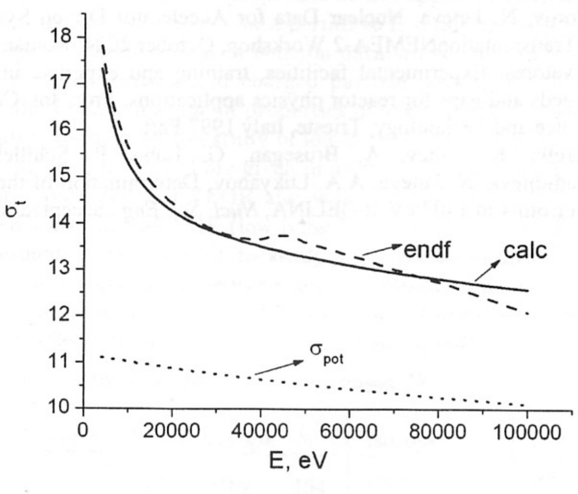

Fig. 2: Fitting of ^{232}Th total cross section in the energy interval 4 – 100 keV.

Conclusions

We propose a method that evaluates heterogeneous effects in SAD system with resonance self-shielding accounting. Benchmarking for the unresolved resonance region by using the results from oversight experiments (group constants description) is assessed.
The significance for reactor systems driven by accelerator analysis of energy dependence at ^{232}Th radiation capture cross-section in unresolved resonance region was made using the experimental data from JINR.

References

1. V.N. Shvetsov: SAD Project at Glance. Joint Institute for Nuclear Research, 141980, Joliot-Curie 6, Dubna, Russia
2. H. Aıt Abderrahim, P. Kupschus, E. Malambu, Ph. Benoit, K. Van Tichelen, B. Arien, F. Vermeersch, P. D'hondt, Y. Jongen, S. Ternier, D. Vandeplassche: MYRRHA: A multipurpose accelerator driven system for research & development. *SCK_CEN, Boeretang 200, B-2400 Mol, Belgium IBA, Chemin du Cyclotron 3, B-1348Lou vain-la-Neuve, Belgium.*
3. Г.И. Марчук, „Методы расчета ядерных реакторов", М., Госатомиздат, 1961
4. А.А. Лукьянов, „Замедление и поглощение резонансных нейтронов", М., Атомиздат, 1974
5. Б. Дэвисон, „Теория переноса нейтронов", М., Атомиздат,1960
6. В.В. Орлов, Труды ФЭИ, М., Атомиздат, 1974, стр. 157
7. R. Bonalumi, Energia Nucl., 1961, V. 8, p. 326
8. Л.П. Абагян и др. „Групповые константы для расчета реакторов и защиты", М., Энергоиздат, 1981
9. M.I Krivopustov et al., Investigation of neutron spectra and transmutation of 129I, 237Np and other nucleids 1.5 GeV protons from the Dubna nuclotron using the electronuclear setup "Energy plus transmutation", JINR Preprint E1-2004-79, Dubna, 2004
10. M.Todorov, N. Janeva, Nuclear Data for Accelerator Driven System and Nuclear Waste TransmutationNEMEA-2 Workshop, October 2004, Romania, in print.
11. M. Salvatores: Experimental facilities, training and expertise in the nuclear data field: needs and gaps for reactor physics applications. Proc. Int. Conf. Nuclear Data for Science and Technology, Trieste, Italy 1997 Part
12. A. Borella, K. Volev, A. Brusegan, G. Lobo, P. Schillebeeckx, F.Corvi, N.Koyumdjieva, N. Janeva, A.A. Lukyanov, Determination of the $^{232}Th(n,\gamma)$ Cross Section from 4 to 140 keV at GELINA, *Nucl. Sci. Eng.,* accepted for publication.