IOSR Journal of Applied Physics (IOSR-JAP)
ISSN: 2278-4861. Volume 2, Issue 1 (Sep-Oct. 2012), PP 30-33
www.iosrjournals.org

Capture Cross Section Evaluations of 18 Stable Fission Products

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Abstract: *A comparison of the capture cross sections in the energy range of 0.1-20 MeV of some 18 stable fission products using the main international libraries have been calculated. A weighted average values for the 18 isotopes was found. Curves of all considered data are presented.*

Key words: *Stable fission products, ^{109Ag}, ^{81}Br-81, ^{111}Cd, ^{140}Ce, ^{142}Ce, ^{133}Cs, ^{155}Gd, ^{143}Nd, ^{145}Nd, ^{134}Xe, ^{146}Nd, ^{108}Pd, ^{141}Pr, ^{101}Ru, ^{94}Zr, ^{154}Sm, ^{89}Y, ^{92}Zr, weighted average*

I. Introduction

The fission in a nuclear reactor produces hundreds of different products of radionuclide having neutron to proton ratio of over 1.5, these unstable radio nuclides decays toward stability mainly by a series of beta emission [1]. Some of the fission fragments are nuclides having a large cross sections for neutron capture, even in a small quantities and/or small half life, they can remove from the reactor a number of neutrons which, otherwise, could be contribute to the chain reaction[2]. Evaluation of neutron capture cross sections is one of the most important subject needed to the save work of reactors [3]. The fission products are needed too in other fields of calculations such as understanding the nature of fission process, determination of the fuel burn- up, performing shielding calculation, calculation of decay heating power, estimation of the amount of gas production, nuclear transmutation of fuel in different types of nuclear reactors, estimation of radiation damage of all reactor material and components, neutron dosimetry of produced nuclides and other special applications [4]. A great effort has been devoted to the cross section evaluation of major isotopes pertinent to reactor application including particularly the Uranium and Plutonium isotopes, however, the effort to evaluate fission product data are less than the effort given for the actinide data. The nuclear fission of a fissile material is a complicated process in which more than 500 different products of radio nuclides of about 40 elements of the periodic table (ranged from Z=31 to 68) are produced.Fission products belonging to about 90 important mass chains [5], they are ranging from Nickel to Erbium. Only about 25% of them are creating in both ground and isomeric states. Besides this, but with a very little probability, another contribution comes from light fission products (tritium, helium) generated in the ternary nuclear fission.

Most of these fission products may be formed in different ways, as a primary event or as they have extra neutrons, they tend to decay to more stable isotopes through beta emission constituting the fission chains. Each fission chain is formed of a certain number of fission products which are, in general, radioactive such as:[6]

$$^{85}_{31}Ga \xrightarrow[0.09\ s]{} {}^{85}_{32}Ge \xrightarrow[0.54\ s]{} {}^{85}_{33}As \xrightarrow[2.03\ s]{} {}^{85}_{34}Se \xrightarrow[32\ s]{} {}^{85}_{35}Br \xrightarrow[2.9\ m]{} {}^{85}_{36}Kr \xrightarrow[10.7\ y]{} {}^{85}_{37}Rb\ (stable)$$

$$^{131}_{50}Sn \xrightarrow[51\ s]{} {}^{131}_{51}Sb \xrightarrow[23\ m]{} {}^{131}_{52}Te \xrightarrow[25\ m]{} {}^{131}_{53}I \xrightarrow[8.02\ d]{} {}^{131}_{54}Xe(*) \xrightarrow[11.9\ d]{} {}^{131}_{54}Xe\ \ (stable)$$

A number of different institutions have been engaged in considerable theoretical and experimental efforts to build their national nuclear data libraries. We select 18 of the stable fission products produced in LWR which represent 34% of the total of fission products yield, this number is determined in the base of their yields, half-life and in particularly their capture cross sections. The most recent evaluated neutron nuclear data libraries concerning the fission products are actually JENDL-3[7] and JENDL-4[8] from Japan, ENDF/B-V1 release 8 from United States[9], JEFF-3.0 and JEF-2.2 from Europe[10] , BROND-2 from Russia[11] and CENDL-3.0 from China[12] . These libraries are considered now as containing the most up-to-date nuclear data of evaluated (recommended) cross sections, spectra, angular distributions, fission product yields, thermal neutron scattering, photo-atomic reactions, and other data which are important in neutron-induced reactions relevant to reactor calculations[13,14]. They contain too cross sections of numbers of fission products nuclides according to their yields, half-lives, capture, elastic, inelastic and absorption cross sections. In spite of all these voluminous and important data the International Atomic Energy Agency (IAEA), point out that, there are still considerable discrepancies among the evaluated data sets [15]. The data of some of the isotopes given by the libraries were obtained by a special numerical evaluation method based on the fitting of nuclear fragments of mass distributions by several Gaussian functions [16,17]. The generation of nuclear data based on theoretical physic models is frequently used when no experimental data are available [18]. In particular the total cross sections above the resonance region are generally evaluated using the optical model by fitting the data to the measured total cross sections. This method allows filling the gaps in the data of experimental results.These libraries contains all kinds of nuclear information principally the capture cross sections of fission nuclides which accounts for more than 40% of the total neutron absorption cross section of materials constituting the reactor core[19], and it contribute approximately to about 9% of the reactor reactivity (as their values are proportional to the fuel burn-up or the neutron flux). Using a weighted average formula, we obtain an average value for each of the isotopes. Results of the calculated capture cross sections in reduced energy intervals with the original data are presented in figures.

II. Results

The capture cross section of twenty stable fission products (Ag-109, Br-81, Cd-111, Ce-140, Ce-142, Cs-133, Gd-155, Nd-143, Nd-145, Xe-134, Nd-146, Pd-108, Pr-141, Ru-101, Zr-94, Sm-154, Y-89, Zr-92), are plotted using the published data of the main libraries , and an average evaluated value for each isotope is calculated using the following weighted mean formula[20] ,

$$Y = \frac{\sum w_i \ y_i}{\sum w_i}$$

Where: $w_i = 1/\sigma_i^2$ and σ_i =standard deviation of sample i

Some libraries declare the uncertainty of the capture cross section values which we considered, otherwise we used 10% for the unmentioned error.

III. Conclusion

Figures (1-18) shows the capture cross sections of eighteen stable fission products. The evaluations are based on the main international data libraries using an averaging formula. The cross sections are probabilistic events, two important types of experimental errors known as systematic and random errors exist, thus there is always some uncertainty in all measurements. It's clear that the average weighted values eliminate a large deviation of values or of errors. This method is very important for some problems such as neutron dosimetry, fuel burn-up and determination of isotope production.

Fig. 2: Capture Cross Section of Br-81

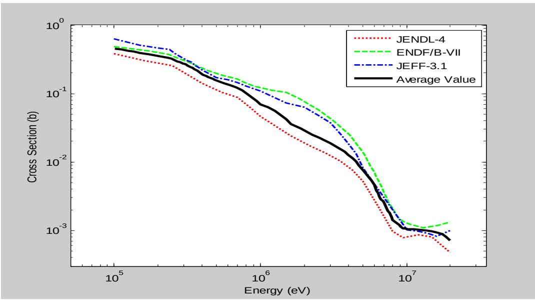


Fig. 1: Capture Cross Section of Ag- 109

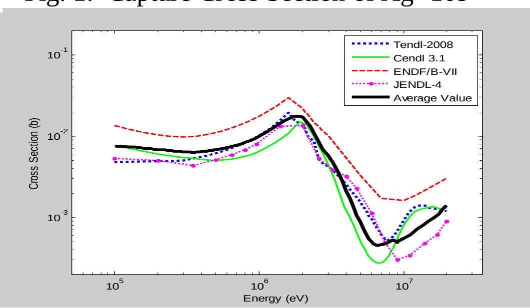
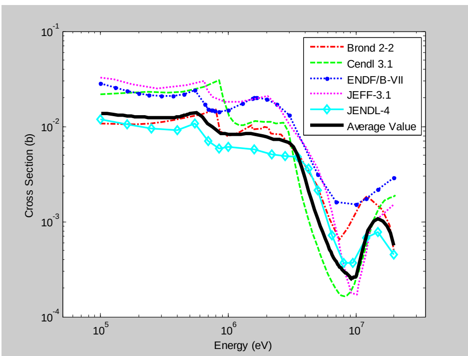


Fig. 4 : Capture Cross Section of Ce-140

Fig. 3 : Capture Cross Section of Cd-111



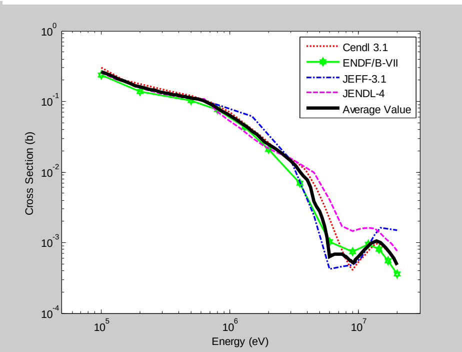


Fig 6 : Capture Cross Section of Cs-133

Fig 5 : Capture Cross Section of Ce-142

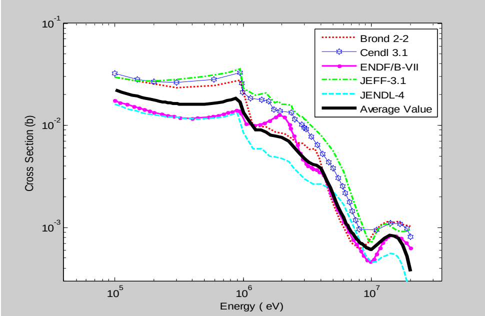


Fig 8: Capture Cross Section of Gd-155

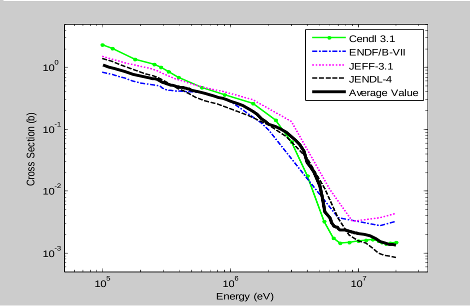


Fig 7: Capture Cross Section of Zr-92

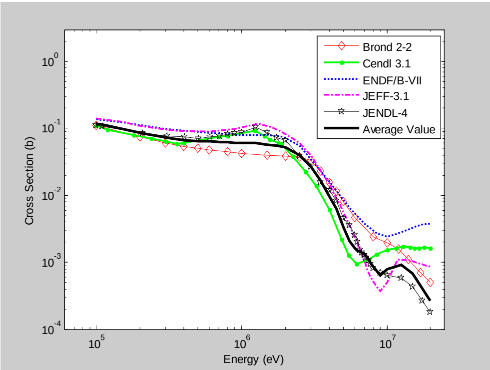


Fig 10: Capture Cross Section of Nd-145

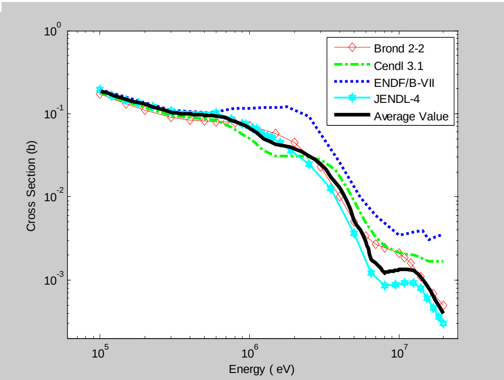


Fig 9: Capture Cross Section of Nd-143

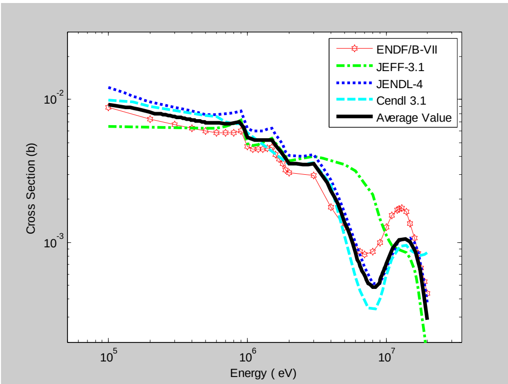


Fig 12: Capture Cross Section of Xe-134

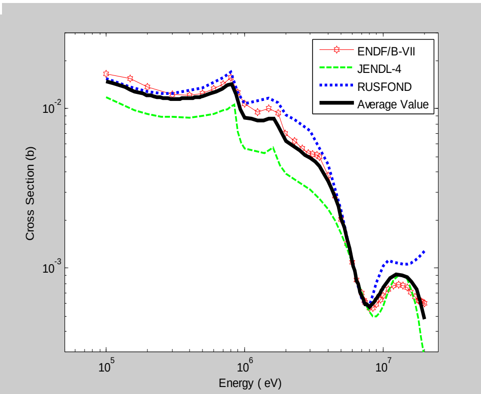


Fig 11: Capture Cross Section of Y-89

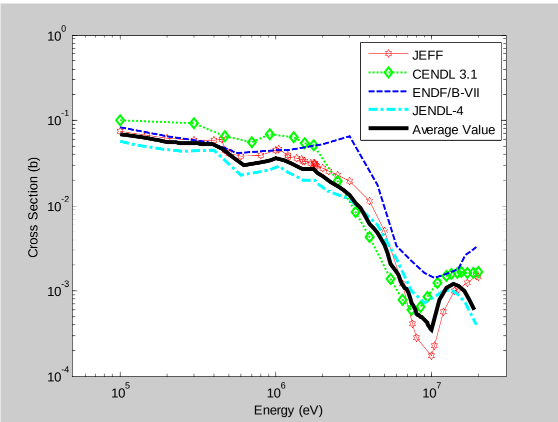


Fig 14: Capture Cross Section of Pd-108

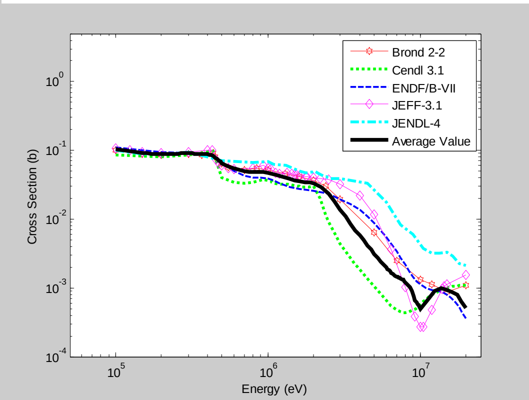


Fig 13: Capture Cross Section of Nd-146

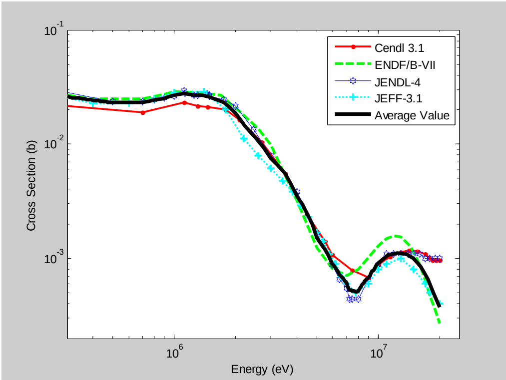


Fig 16 : Capture Cross Section of Ru-101

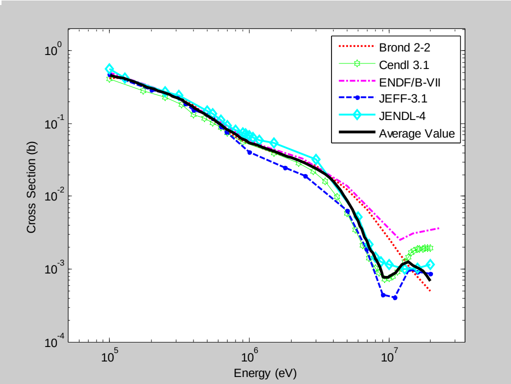


Fig 15 : Capture Cross Section of Pr-141

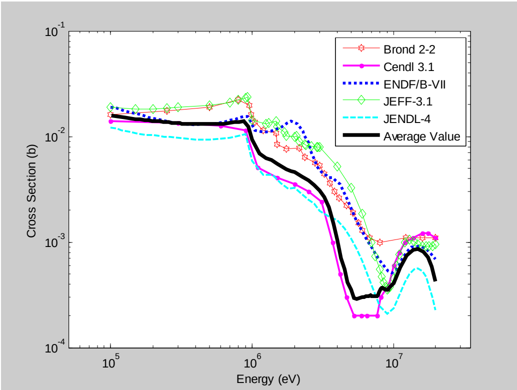


Fig 18: Capture Cross Section of Sm-145

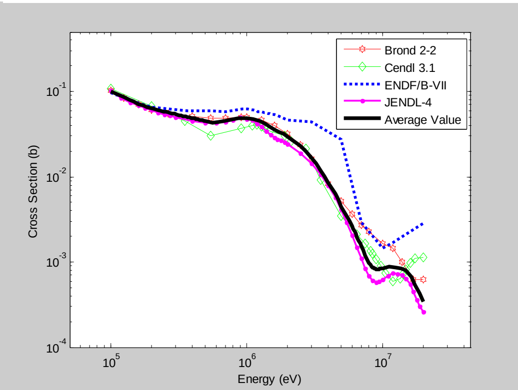


Fig 17 : Capture Cross Section of Zr-94

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