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A Warning on Fission Resonance Integrals – *Caveat Utor!**

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Abstract

A common error is made in defining the resonance integral in most tabulations and handbooks. Although it has a minor effect on the capture resonance integral and on the fission resonance integral for the fissile nuclides, it leads to gross errors in the fission resonance integral for the fertile nuclides. The errors in the fission resonance integral for fertile nuclides of the elements from thorium through curium in the ENDF/B-V library will be presented. *Let the user beware.*

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Summary

Experimenters utilize tables of neutron cross sections and resonance integrals to provide a preliminary estimate of what they might anticipate as the result of a given experiment. If the tabulation indicates a nuclide's value where none exists or it grossly overestimates the value, the experimenter will be misled and an experiment may be performed uselessly. Such a problem exists for fertile nuclide fission resonance integrals and will be examined in this paper.

In reactor design, a usual practice has been to analyse the neutron flux into two components. One component is a thermalized flux with an assumed Maxwellian energy distribution and the other is an epithermal or resonance flux whose energy distribution is assumed to be inversely proportional to the neutron energy, E.

Marshak[1] showed that for neutrons slowing down in a moderator without absorption, the neutron spectrum varies as $\Delta E/E$. If a dimensionless variable, the lethargy u, is introduced as $u = \ln E_0/E$, where E_0 is a fixed energy (usually 10 Mev), the intermediate neutron energy distribution becomes a region of constant lethargy. Although the intermediate energy range varies for different reactors, the lethargy is reasonably constant from an energy of a few tenths of an eV up to approximately a tenth of an MeV for a light-water-moderated reactor.

For a small amount of a material, so-called infinite dilution, the reaction rate is

$$R = \varphi_{th}\, \sigma_{th} + \varphi_r\, I,$$

where φ_{th} is the thermal neutron flux, φ_r is the epithermal (resonance) neutron flux per unit lethargy, σ_{th} is the thermal neutron cross section for the reaction and I is the infinitely dilute resonance integral for the reaction. The problems of resonance self-shielding and flux depression are avoided by the infinite dilution assumption.

The resonance integral, I, is given by the expression

$$I = \int \sigma(E)\, \Delta E/E.$$

The integration is performed over the entire 1/E flux region, from a lower limit of E_L to an upper limit of E_U. There are several advantages to the concept of the resonance integral. It is independent of the temperature and the reactor system. It only requires that the neutron energy spectrum vary as 1/E in the defined energy interval from E_L to E_U. Also, a resonance integral can be directly estimated experimentally with the use of a high pass filter, such as a cadmium cover.

The integration limits of the resonance integral are important. The lower bound, E_L, has been taken by various authors to be from 0.4 ev to 0.625 eV, where the thickness and the geometry of the cadmium cover leads to the various choices. In the USA, the most common lower limit is 0.5 eV. Unless there is a resonance in the reaction near this lower bound, the final choice is not critical. It is the upper boundary that is of interest.

As mentioned earlier, the region of constant lethargy extends upward in energy to about a tenth of an MeV. The upper limit on the resonance integral should then be in the range from 50 keV to about 200 keV. In the case of exoergic reactions, such as neutron capture, the major contribution to the integral comes from the region of the lower limit. Since the background cross section (in the absence of resonances) will vary as $E^{-½}$ and the flux as E^{-1}, the contribution in the tenths of an eV energy range will be about three orders of magnitude larger than the contribution in the tenths of an MeV range. The effects of resonances at low energies will greatly enhance this ratio. As a result, whether the upper limit, E_U, is 100 keV, 10 MeV, or ∞, any difference would be well within the experimental uncertainty of a measurement. Many standard tabulations of these resonance integrals, such as are found in BNL-325[2], the IAEA Handbook[3], and the ENDF/B-V guidebook[4] make the substitution of ∞ for this upper limit. This substitution is an implicit assumption that the 1/E neutron flux distribution extends to ∞, or at least to the current ENDF/B upper limit of 20 MeV, which is in contr..ction of the facts.

For an endoergic reaction, such as neutron fission in a fertile nuclide, the threshold for the reaction is often at an energy which is higher than the constant lethargy region. In such a case, the indiscriminate integration of the cross section with a 1/E flux to the upper limit of available data, *i.e.*, 20 MeV, results in a calculated resonance integral when, in fact, none exists.

Even when the reaction threshold does not exceed the integral's upper limit, the calculated value is still much larger than it should be and could mislead the user into thinking that there is some small cross section in the eV and the low keV energy range. Table I compares the resonance integrals as reported in BNL–325, the IAEA handbook, and the ENDF/B–V guidebook with the values calculated using the proper upper limit of 100 keV. It demonstrates that large differences exist, depending upon the upper limit, E_U used. In searching tabulations, the user should carefully check the textual material to insure that the data presented is the data that is required.

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Table I. Fission Resonance Integral Comparison

Nuclide	BNL–325 Value[2] (barns)	IAEA Value[3] (barns)	ENDF/B–V Value[4] (barns)	E_U = 100 kev Value (barns)
^{232}Th	no value quoted	0.0746 ± 0.0016	0.6185	0.0
^{233}Pa	3.0	no value quoted	2.947	0.0
^{236}U	7.8 ± 1.6	2.	7.768	4.38
^{238}U	0.00154 ± 0.00015	0.0013 ± 0.0002	2.0324	0.00153
^{237}Np	6.9 ± 1.0	6.5 ± 1.2	6.870	0.317
^{240}Pu	8.8	5.	8.830	2.41
^{242}Pu	no value quoted	5.	5.568	0.23
^{243}Am	6.151	13. ± 2.5	6.151	0.056
^{244}Cm	12.5 ± 2.5	13.4 ± 1.5	18.697	10.8

References

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