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Re: "Spatial Dependence of Thermal-Neutron Spectra and the Interpretation of Thermal Utilization Measurements"

In a previous letter (*1*), it was pointed out that the values quoted for the experimentally determined thermal-neutron disadvantage factors have not been corrected for the spatial dependence of neutron spectra, and this correction could substantially change the magnitude of these quantities. In reference to their experiments, Sher, Kouts, and Klein agree that such a correction should be applied to the measured disadvantage factors (*2*). However, their consideration of the magnitude of the correction deserves further discussion.

In ref. *1*, data are presented which show that the neutron temperature in a fuel rod in a light-water lattice is much larger than the physical temperature. For moderator-to-fuel ratios used in the light-water experiments, the moderator neutron temperature is also undoubtedly larger than its physical temperature but lower than the fuel temperature; therefore, a correction must be applied to the experimentally determined disadvantage factor. To illustrate that such a correction factor is indeed significant, an admittedly extreme example was chosen in which the fuel temperature was assumed to be 200°C and the moderator temperature 20°C, and the calculated correction factor is approximately 30%. This example is not meant to imply that a 30% correction factor should be applied to all measurements or that a 200°C difference in temperature exists between the fuel and moderator regions of the Sher, Kouts, and Klein experiments. In fact, the determination of a specific correction factor is dependent upon the physical properties of the lattice. Sher, Kouts, and Klein give evidence that for some of their lattices with ~1.3 wt.% U^{235} enrichment the neutron temperature difference between fuel and moderator is approximately 50°C, and about a 10% correction to the presently quoted experimental disadvantage factors is required (*2*). It would appear that for similar lattices with higher U^{235} enrichments, the neutron temperature difference between fuel and moderator would be greater, and could be 200°C for the appropriate fuel enrichment.

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A Note on the Measurement of Diffusion Parameters by the Pulsed-Neutron Source Technique

The pulsed-neutron technique has been used extensively to measure diffusion parameters in a variety of moderators. A description of the technique and an excellent summary of the present status of experiments may be found in a recent review by Beckurts (*1*). It may be seen from Beckurts' review that the values of the absorption cross section and of the diffusion constants obtained by the pulsed-neutron technique for the various moderators are quite consistent. However, some very puzzling discrepancies are observed between various measurements of the diffusion cooling constant, especially for crystalline moderators such as beryllium and graphite.

Beckurts proposes the following possible causes for these discrepancies: (1) the role of B^6 terms, (2) the effect of higher harmonics, and (3) the importance of the data-evaluation schemes. For the case of beryllium, at least, different laboratories measure different decay constants for the same value of the buckling (*2*, *3*). Such discrepancies cannot be blamed on B^6 terms. It is also difficult to see how the effect of harmonics may not be properly accounted for since, for a small cube of a moderator with low absorption, the first spatial harmonic decays almost twice as rapidly as the fundamental mode.

The purpose of this note is to propose another possible cause for the observed discrepancies in the measurements of decay constants. It appears that under certain conditions the decay of the neutron population out of a moderating assembly may never be strictly exponential. In this case, the "asymptotic decay constant" is not directly measurable and the diffusion cooling constant is not a well defined concept. The argument will be presented in some detail for the case of beryllium; however, the general conclusions should certainly be valid for crystalline moderators like graphite and beryllium oxide, and perhaps other materials as well.

The elastic transport cross section of beryllium has been computed by Bhandari (4) and is shown in Fig. 1. It changes very rapidly with energy and reaches its maximum value where the neutron wavelength is just equal to the distance between parallel planes of Miller's index (1, 0, 1). This corresponds to a neutron energy of 6.85 mev (millielectron volt) and to a velocity of 1.142×10^5 cm/sec. At this energy the transport cross section is about 18 barns and changes very little with moderator temperature. Below this energy the cross section drops stepwise to a value of 5.5 barns.