Letters to the Editors

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

In a recent letter with the above title (*1*), Deutsch has criticized the interpretation of experiments (*2, 3*) in which the thermal utilization, f, is inferred from foil activations measured over the fuel and moderator regions of a lattice. Deutsch makes two main points: first, that in the analysis of such measurements, no account is taken of the detailed energy and space dependence of the neutron flux in the cladding and structural materials in the lattice (this leads to admittedly negligible errors if the cladding is aluminum); second, and much more serious, that the difference in effective neutron temperature between the fuel and the moderator is not taken into account, that the temperature difference may be large, and that this neglect, while not affecting f, may lead to large errors in the flux disadvantage factors inferred from the foil activations. We would like to comment on the second point.

The corrections to be applied to the measured activation ratios are approximately proportional to the square root of the ratio of the fuel and moderator Maxwellian temperatures. Deutsch gives arguments that lead to an estimate of a fuel-moderator temperature difference of as much as 200°C; thus the correction to be applied to the disadvantage factor is, according to him, approximately $(500/300)^{1/2}$, or 30%.

We wish to point out that although Deutsch's criticism (as applied to flux disadvantage factors) is basically correct, his estimate of the temperature difference seems to be erroneous, and that in fact the temperature differences are probably much smaller and lead to correspondingly smaller corrections.

The 200° difference is arrived at by extrapolation of the temperature dependence of Pu^{239}/U^{235} fission ratios measured in pure Maxwellian spectra (originating in a thermal column) in water (*4*) so as to fit the high values of similar ratios measured in lattices (*5*). The extrapolation therefore neglects the large slowing-down component of the flux in the lattices, and it is this slowing-down flux, rather than the Maxwell distribution, that mainly accounts for the high Pu^{239}/U^{235} fission ratios in the lattices, because of the Pu^{239} resonance at 0.3 ev.

The lattices are in fact very poorly thermalized, with r values (*6*) of the order of 0.1 to 0.3. One can roughly calculate the effective Pu^{239}/U^{235} fission ratio, using the Westcott convention (*6*). It is recognized that the Westcott convention has limited validity in poorly thermalized lattices, nevertheless the calculations reproduce Klein's measured values, cited by Deutsch, and indeed indicate that the fission ratios are quite insensitive to the assumed neutron temperature, changing by only 5% as the temperature is varied from 20° to 200°C. Klein, in ref. (*5*) has performed a calculation, using a better approximation to the spectrum than a pure Maxwellian, and has reproduced his experimental results reasonably well. The concept of neutron temperature was not used in his analysis. It is our belief that Klein's values of the Pu^{239}/U^{235} fission ratios shed no light on the neutron temperature, and certainly do not imply such a large temperature difference between fuel and moderator as that used by Deutsch.

The calculations of Honeck (*7*), briefly referred to at the end of Deutsch's paper, show that for the BNL-Bettis lattices the average temperature differences are of the order of 50°C. Experimental results on similar lattices by Campbell *et al.* (*8*) generally tend to support these lower values of the fuel-moderator temperature differences.

In the original interpretation of the f measurements (*2*) the BNL group assumed a temperature difference of 40° and showed that this led to negligible errors in f. The Bettis group (*3*) showed that even larger assumed temperature differences led to negligible errors in f. The reason is that the technique is meant to measure the disadvantage factor of capture density rather than flux, and inferences about capture density are not very sensitive to the neutron temperature. The corrections to the flux disadvantage factors, not made by either group, are actually of the order of 10% for an assumed temperature difference of about 50°C.

It is worth noting, however, that the published disadvantage factors can be correctly interpreted as ratios of average neutron density or capture density rather than flux. One should in fact calculate the capture density as this is what is measured. This was done by Honeck and Kaplan (*7*) in comparing their calculations with the BNL measurements.

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Received September 13, 1961

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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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Received October 24, 1961

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.