ON THE NUMERICAL DIFFERENCE BETWEEN HOMOGENEOUS AND HETEROGENEOUS CALCULATIONS OF URANIUM 238 CAPTURE RATE RATIOS IN LIGHT WATER REACTORS

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Numerische Differenz von homogenen und heterogenen Berechnungen von U-238-Einfangraten in Leichtwasserreaktoren

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In this work, we briefly try to illustrate the effect of the space-dependent heterogeneous treatment on the uranium 238 reaction rate calculations and these results are compared with homogeneous calculations. In Brookhaven National Laboratory, USA, an extensive experimental program has been completed by R. L. Hellen and H. C. Honeck [1] to measure and calculate the behavior of neutrons in slightly enriched uranium metal fuelled reactor lattices moderated by light water. The intent of this work has been to provide precise data which can be used to find the capabilities and limitations of the theoretical methods used in reactor design. The assemblies studied have been limited to simple cylindrical arrangements of metal fuel rods in a triangular lattice without the presence of control rods or other unnecessary perturbations. Three batches of fuel were employed in the measurements having the enrichments of 1.027 V*, 1.143 V*, 1.299 V* (by weight) in U 235. The fuel rods were encased in 0.028 inch 2 S aluminum tubing with a nominal 0.005 inch air gap between the fuel and the inside of the tube.

In Fig. 1 the spectra generated by the multigroup calculations for two lattices are given [1] as a function of lethargy. The high energy fluctuations in the flux are caused by the oxygen resonances; however, the three small dips on the low side of the fission peak are spurious and arise from the use of difference equations to represent the heavy element slowing down. Similar spurious detail is seen below the 6.7 eV resonance of U 238. It is stated that these effects were observed in calculations with hydrogen moderation alone.

The cross-sections used in the MUFT calculations [12] have been tabulated by Henry [2]. The thermal group cross-sections were taken from BNL-325 [13], except for the U 235 cross-section which was taken from Safford and Havens tabulation [3].

The thermal neutron distributions are computed from the integral form of the transport equation which is solved numerically by a code called THERMOS [4, 5].

In this short work we are concentrating on 0.600 inch diameter, 1.0 V* enriched uranium metal rod lattice with volume ratio (water to uranium) W/U = 4.0. At BNL [1] calculations and measurements have been made of $\rho_m$, the ratio of epicadmium-to-subcadmium captures in U 238 foils placed in a typical cross-section of the lattice fuel rod. In Table 1 the theoretical and experimental values of $\rho_m$ are tabulated. As it is seen for the 0.600 inch diameter, 1.0 V* enriched uranium metal rod lattice with volume ratio W/U = 4.0, the $\rho_m$ value theoretically found by heterogeneous treatment is $\rho_m = 0.624$. The BNL experimental value is $\rho_m \pm 0.05$.

<table>
<thead>
<tr>
<th>$%$ U 235</th>
<th>W/U</th>
<th>Theory</th>
<th>BNL exp.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.00</td>
<td>2.0</td>
<td>2.294</td>
<td>2.0 ± 0.2</td>
</tr>
<tr>
<td>1.5</td>
<td>1.520</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2.0</td>
<td>1.151</td>
<td>1.05 ± 0.14</td>
<td></td>
</tr>
<tr>
<td>3.0</td>
<td>0.793</td>
<td>0.79 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>4.0</td>
<td>0.624</td>
<td>0.60 ± 0.05</td>
<td></td>
</tr>
</tbody>
</table>

If we take definition of $\rho_m$ as follows:

$$\rho_m = \frac{\int_0^{0.625 eV} \sigma_0(\epsilon) \phi(\epsilon) d\epsilon}{\int_0^{\infty} \sigma_0(\epsilon) \phi(\epsilon) d\epsilon}$$

taking the neutron energy spectrum from Fig. 1 for the lattice in question and taking the cross-section values for neutrons below 0.625 eV energy from TEMPEST library and for neutrons above 0.625 eV energy from MUFT library we have calculated the $\rho_m$ value and found it to be $\rho_m = 0.249$ (taking only smoothed cross-sections from [12]). If we take resonance integral into account and suppose $U/E$ shape for resonance region (for Uranium-238) $\rho_m$ value is found to be $\rho_m = 0.755$.

The experimental setup for $\rho_m$ measurements can be found in many references [6-9]. Natural or preferably depleted uranium foils are placed in a typical cross-section of the lattice fuel rod.

Although Fig. 1 shows a very sophisticated calculation of neutron energy distribution it does not specifically represent the space-dependent neutron energy distribution on the fuel rod surface or inside the fuel rod or outside the fuel rod (i.e. in the moderator). In light water lattices there are very severe energy spectrum changes especially near and inside the fuel element as stated by H. Honeck [14]. So, reaction rate calculations (especially for Uranium 238) of this sort need special and very sophisticated ways of space and energy-dependent calculations.
In short, we have illustrated the great difference between the two different calculations if one treats the reaction rate calculation as a straightforward numerical integration knowing the energy spectrum properly. In reactor design, if a theory predicts well the material buckling of the system, it does not necessarily mean that the energy spectrum of the reactor can be predicted correctly.

As it is stated by Strawbridge [10], "In addition to buckling uncertainties other effects that increase the standard deviation of the calculated results include impurities in the fuel, clad and moderator (impurities were neglected in the calculations), and uncertainties in physical parameters such as dimensions densities and enrichments."

In order to check the energy spectrum of the reactor, one has to make reaction rate ratio measurements and spectral indices measurements, and these results must be compared with sophisticated reaction rate calculations including perturbations and heterogeneities of the system. A proper recipe for a \( \phi^2 \) calculation of this kind is given in the Appendix and given by D. R. Oden Jr. [11].

Appendix
The heterogeneous calculation of \( \phi^2 \) values were generated from fast neutron energy spectrum codes by fast group constants and thermal energy spectrum codes by thermal group constants via the standard two group formula:

\[
\phi^2 = \left( \frac{\Sigma_{\text{eff}}}{\Sigma_{\text{c}}^2} \right)^2 \left( \frac{\phi_f}{\phi_i} \right)
\]

where

\[
\frac{\phi_f}{\phi_i} = \frac{\Sigma_{\text{eff}}}{\Sigma_{\text{c}} + D \cdot B^2}
\]

other parameters are:

- \( \langle \sigma_{\text{c}} \rangle_{\phi} \) microscopic fast capture cross-section for U 238 from fast codes
- \( \langle \sigma_{\text{abs}} \rangle_{\phi} \) microscopic thermal absorption cross-section for U 238 from thermal code
- \( \Sigma_{\text{eff}} \) macroscopic cell averaged down scattering cross-section from fast to thermal group obtained from fast code results
- \( D \) cell averaged thermal diffusion coefficient from thermal code
- \( B^2 \) geometrical buckling

References