THERMAL NEUTRON FLUX GENERATION BY HIGH-ENERGY PROTONS IN THICK URANIUM TARGETS*

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For several applications, e.g., in designing facilities to produce an intense source of low-energy neutrons by using medium-energy protons and for studies of the feasibility of converting fertile-to-fissile material using medium-energy protons, it is necessary to carry out calculations of the transport of medium- and low-energy nucleons and pions through fissionable material. The high-energy transport code HETC has often been used to carry out such transport calculations, but because this code did not take into account high-energy fission the results were very approximate. Recently, a fission channel has been added to the intranuclear-cascade-evaporation model of nuclear reactions and this revised cross section model has now been incorporated into the transport code, HETC, so that medium-energy nucleon and pion transport calculations in fissionable material may be carried out. The transport code, HETC, used here is in all respects the same as that described in Ref. 3, except that when a nucleon-nucleus or pion-nucleus nonelastic collision occurs the probability that fission occurs and the particle production from the excited residual nuclei following fission are taken into account.

To test the validity of the revised code, calculations have been carried out and compared with the experimental data of J. S. Fraser et al. In the experiment protons of various energies
were incident on a depleted uranium target (10.2 cm diameter x 60 cm long) surrounded by a large tank of water, and the number of thermal neutron captures in the water per incident proton was measured. The calculations were carried out using the geometry of the experiment and the number of thermal neutron captures in the water was calculated. The parameter, $B_0$, that occurs in the level density expression in the intranuclear-cascade-evaporation model was taken to be 10 MeV. The low-energy ($\leq 15$ MeV) neutron transport calculations were carried out with MORSE$^7$ using cross section data based on ENDF/B-IV.$^8$

The comparisons between the calculated results and the experimental data of J. S. Fraser et al.$^1$ are shown in Table 1. The calculated error bars are statistical only and indicate one standard deviation in the Monte Carlo calculations. At the proton energies of 540 and 960 MeV the calculated and experimental results are in very good agreement, but for incident protons of 1470 MeV the calculated results are larger than the experimental data.

Transport calculations similar to those considered here have also been carried out by V. S. Barashenkov et al.$^9$ for protons incident (at the center) of an infinite natural uranium system. In Table 1 the results from Ref. 9 are compared with similar results obtained here for the case of 1 GeV incident protons.
The quantities compared are the total number of fissions (i.e., fissions in both $^{235}\text{U}$ and $^{238}\text{U}$) per incident proton and the total number of neutron captures per incident proton. Comparisons similar to those in Table 1 with a variety of other experimental data will also be presented.
Table 1

Comparison of Calculated Results With Experimental Data of J. S. Fraser et al. and With Calculated Results of V. S. Barashenkov et al.\textsuperscript{9}

Proton Incident on a Depleted Uranium Target Surrounded by Water

<table>
<thead>
<tr>
<th>Incident Proton Energy (MeV)</th>
<th>Thermal Neutron Captures in $\text{H}_2\text{O}$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Experiment\textsuperscript{6}</td>
</tr>
<tr>
<td>540</td>
<td>$15.1 \pm .8$</td>
</tr>
<tr>
<td>960</td>
<td>$32.3 \pm 1.6$</td>
</tr>
<tr>
<td>1470</td>
<td>$44.8 \pm 2.2$</td>
</tr>
</tbody>
</table>

1 GeV Protons Incident at the Center of an Infinite Natural Uranium System

<table>
<thead>
<tr>
<th>Calculated Barashenkov et al.\textsuperscript{9}</th>
<th>Calculated Here</th>
</tr>
</thead>
<tbody>
<tr>
<td>No. of Fissions per Incident Proton</td>
<td>28</td>
</tr>
<tr>
<td>No. of Captures per Incident Proton</td>
<td>102</td>
</tr>
</tbody>
</table>
References


8. ORNL/RSIC-37, Radiation Shielding Information Center, Oak Ridge National Laboratory (1975).