Secondary Neutron Spectra from Neutron Interactions in a Thick Carbon Sample

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Neutron Physics Division

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ABSTRACT

The spectra of secondary neutrons produced by neutron interactions in a thick (~ 3 mean free path) sample of carbon have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. A linac (ORELA) was used as a white neutron source with a 48-m flight path. Incident energy was determined by time-of-flight, while secondary spectra were obtained through pulse-height unfolding techniques. The results of the measurement are compared to calculations based on the current evaluated data file (ENDF/B 1274).

I. INTRODUCTION

Measurements have been made of the spectra of scattered neutrons from a thick carbon sample for incident neutron energies ranging from 1 to 20 MeV. This experiment was performed as part of a program¹⁻¹⁰ at ORNL to provide data suitable for testing the evaluated neutron data files for those elements which are important in the analysis of radiation transport problems.

The experimental technique is capable of sufficient resolution in the energy and angle of both incident and secondary neutrons to provide data with localized sensitivity. Thus, although it is necessary to compare the evaluated cross-section file to the experiment by means of a calculation to include the finite-sample effects, there is still an essentially direct differential relation between the calculated secondary spectra and the cross-section data.

This technique offers the additional advantage of testing the cross-section sets in the same context as they are used in applied transport problems. That is, the process by which microscopic evaluated cross sections are converted to tailored group constants is also checked.

Results of measurement and calculation based on the current evaluation, ENDF/B-IV MAT 1274,¹¹ are presented in graphical form.

II. EXPERIMENTAL PROCEDURE

The experimental procedure used in this work has been described in detail elsewhere¹² and will be only briefly outlined here. The Oak

Ridge Electron Linear Accelerator was used as the neutron source. An electron beam pulse width of 5 ns at a repetition rate of 800 sec^{-1} was employed.

The scattering sample was located 47.8 m from the neutron source and consisted of 8498 grams of reactor grade graphite formed in an annular cylinder. The cylinder was 25.40 cm in outer diameter, 15.24 cm in inner diameter, with a length of 15.30 cm (see Fig. 1 for details of the sample-detector geometry).

Secondary neutrons from the sample were detected by an NE-213 scintillation counter located on the axis of the ring at a position corresponding to scattering angles between about 25° and 75° . A shadow bar shielded the detector from the direct neutron beam.

The energy of the incident neutron was determined by the time after the electron pulse at which the secondary neutron from the sample was detected. Each event in the detector was identified as either a neutron or gamma ray by pulse-shape discrimination techniques. Neutron events were then recorded in a two-parameter array of pulse-height versus time-of-flight.

Backgrounds were due to air scattering in the region near the detector and were determined by measurement with the sample removed. Overall background was less than 10%.

The neutron flux incident on the sample was determined using the same detector employed for the secondary radiation. This measurement was made with the sample and shadow bar removed and utilized one-parameter time-of-flight techniques. The incident flux is shown in Fig. 2 and tabulated in Table 2.

The spectra of secondary neutrons were obtained by unfolding the background-corrected pulse-height distributions resulting from integration of the two-parameter data over ranges of time-of-flight. The extent of these time-of-flight intervals was determined by the need to get adequate statistical accuracy in the pulse-height spectra. The widths of the corresponding incident energy intervals range from $\Delta E = 1$ MeV for $E_{inc} < 10$ MeV to $\Delta E = 2.5$ MeV at $E_{inc} = 20$ MeV. Thus while these data provide detailed information about the energy distribution of the secondary radiation, they represent averages over rather wide intervals of the incident energy.

The resulting secondary neutron spectra still retain the effects of the finite pulse-height resolution of the detector. The resolution of the detector is a gaussian function whose width is a function of energy and follows the relation

$$R = \sqrt{250 + 670/E}$$

where R = FWHM in percent and E is neutron energy.

Units for the secondary spectra are neutrons/cm²/MeV averaged over the detector. These results have been normalized to unit incident neutron using the integral of the measured incident flux over the same incident energy range from which the pulse-height distribution was formed. The results are tabulated in Table 1 and are shown in Figs. 3-14. The uncertainties shown in the table and figures do not include a 10% uncertainty in overall normalization.

III. CALCULATIONS

As a test of the current evaluation for carbon (ENDF/B 1274) the experimentally determined secondary spectra were compared to the results of a simple Monte Carlo calculation using the evaluated data. This calculation used group cross sections generated with the AMPX code¹³ system. The group structure was as follows:

<u>Energy Range</u>	<u>Group Width</u>
0.1-4.0 MeV	0.100 MeV
4.0-6.0	0.125
6.0-10.0	0.200
10.0-20.0	0.625

A P_{15} expansion of the angular dependence was used.

For a given incident energy range the neutrons were started randomly from the energy groups covering that range, each group being assigned a relative weight determined by the measured incident flux in that group (see Fig. 2).

In tracking through a given history, if the neutron spent sufficient time in the sample that its time-of-arrival at the detector would cause it to be assigned to a lower incident energy, it was flagged to be included in the calculation for the next lower incident energy bin rather than the bin in which it originated. Simple estimates based on the 48-m flight path show that the probability of remaining in the sample long enough to be assigned to any but the next lowest incident energy bin was negligible. This type of time slewing was largest at the high incident energies where it was about 10%.

After completion of the calculation for a given incident energy range, the resulting secondary spectra were smeared by the energy resolution of the detector. In each case sufficient histories were run that the statistical standard deviation (as calculated by the batch method) was less than 10% for any neutron group. Smoothing by the instrumental resolution further reduces statistical variations. Because of the difficulty in propagating the statistical error through the resolution smearing, no uncertainties are indicated for the final result of the calculation; however, the uncertainty should be less than 10% for the smoothed curves shown in Figs. 3-14. Results are not shown for the 17.5-20.0 MeV group since the evaluated data do not go beyond 20 MeV and the effects of time slewing cannot be calculated correctly for this incident energy bin.

V. RESULTS

The comparison between calculation and experiment is shown in Figs. 3-14. Several aspects of the interaction of neutrons with carbon are relevant: below 5 MeV only elastic scattering can occur. Above 5 MeV inelastic scattering to the 4.43-MeV level becomes possible and above about 8 MeV there is inelastic scattering to various levels which decay by alpha particle breakup.

Insight into the extent to which the experiment can be considered to have a differential nature can be gained from the calculation. For instance, in Fig. 3 ($E_{inc} = 1-2$ MeV) the peak in the secondary spectrum (≈ 1 MeV) is about 50% first collided events, the rest multiple scattering. At $E_{inc} = 4-5$ MeV (Fig. 6) the elastic peak (≈ 4.2 MeV) is about

70% first collided; however the secondary peak at 2.4 MeV is all multiple scattering. For $E_{inc} = 8-9$ MeV (Fig. 10) the elastic peak (≈ 8.1 MeV) is 75% single scattering, the inelastic peak (≈ 3.7 MeV) is 65% first collided while the small peak at 2 MeV is 100% multiple scattering. In Fig. 13, $E_{inc} = 12.5-15.0$ MeV, the percentage of first collided events varies from 25% near 1 MeV to about 75% for the elastic peak.

The comparisons indicate the evaluated data for elastic scattering are very good up to incident energies of about 8 MeV. Above 8 MeV there are discrepancies of the order of 20-40%. For the angular range of this measurement, the evaluated inelastic scattering to the 4.43-MeV level appears to be 10-20% too low for $E_{inc} < 10$ MeV, but rather good above $E_{inc} = 10$ MeV. This conclusion agrees with a comparison presented in Ref. 12 for backangle scattering from a thin carbon ring.

Figure 12, $E_{inc} = 10-12.5$ MeV, indicates the evaluated data for the $(n,n'3\alpha)$ reaction may be somewhat too low, but agreement is reasonably good for higher incident energies.

VI. SUMMARY

The results reported here indicate that the current evaluation for carbon is generally good, with small improvements possible in the higher energy region. The conclusions reached in this work are generally in agreement with the results of a similar experiment given in Ref. 7. The major difference is that in Ref. 7, the evaluated inelastic scattering was found to be high rather than low. However, this work covers a somewhat different angular range which may explain this difference.

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Fig. 1. Details of the geometry for scattering sample and detector.

Fig. 2. Energy dependence of the neutron flux incident on the scattering sample.

5-DEC-76

Fig. 3. Measured and calculated secondary neutron spectrum for incident neutron energies from 1 to 2 MeV.

5-DEC-76

Fig. 4. Measured and calculated secondary neutron spectrum for incident neutron energies from 2 to 3 MeV.

5-OEC-76

Fig. 5. Measured and calculated secondary neutron spectrum for incident neutron energies from 3 to 4 MeV.

Fig. 6. Measured and calculated secondary neutron spectrum for incident neutron energies from 4 to 5 MeV.

5-DEC-76

Fig. 7. Measured and calculated secondary neutron spectrum for incident neutron energies from 5 to 6 MeV.

Fig. 8. Measured and calculated secondary neutron spectrum for incident neutron energies from 6 to 7 MeV.

Fig. 9. Measured and calculated secondary neutron spectrum for incident neutron energies from 7 to 8 MeV.

Fig. 10. Measured and calculated secondary neutron spectrum for incident neutron energies from 8 to 9 MeV.

Fig. 11. Measured and calculated secondary neutron spectrum for incident neutron energies from 9 to 10 MeV.

Fig. 12. Measured and calculated secondary neutron spectrum for incident neutron energies from 10 to 12.5 MeV.

Fig. 13. Measured and calculated secondary neutron spectrum for incident neutron energies from 12.5 to 15.0 MeV.

Fig. 14. Measured and calculated secondary neutron spectrum for incident neutron energies from 15.0 to 17.5 MeV.

TABLE 1.

SECONDARY NEUTRON SPECTRA FOR SCATTERING FROM A CYLINDRICAL SAMPLE OF CARBON. THE VALUES GIVEN ARE THE LOWER AND UPPER LIMITS OF THE 68% CONFIDENCE INTERVAL FOR THE FLUX AT THE DETECTOR POSITION. THEY REPRESENT THE SECONDARY FLUX AS AVERAGED OVER THE AREA OF THE PROJECTION OF THE DETECTOR IN THE DIRECTION OF A POINT ON THE SCATTERING RING. THE DATA ARE NORMALIZED TO UNIT INCIDENT NEUTRON.

INCIDENT NEUTRON ENERGY RANGE = 17.485 TO 19.952 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	2.643E-05	3.807E-05	3.494	7.489E-06	1.398E-05	9.733	3.423E-06	8.055E-06
1.008	2.464E-05	3.523E-05	3.700	4.609E-06	1.512E-05	10.250	2.863E-06	7.203E-06
1.108	1.605E-05	3.756E-05	3.907	4.203E-06	1.564E-05	10.767	6.761E-07	5.622E-06
1.206	2.065E-05	3.837E-05	4.154	7.568E-06	1.388E-05	11.235	8.224E-07	5.392E-06
1.311	2.502E-05	3.984E-05	4.454	9.514E-06	1.550E-05	11.738	3.991E-06	8.581E-06
1.413	1.603E-05	3.806E-05	4.752	9.828E-06	1.507E-05	12.432	1.273E-05	1.783E-05
1.512	1.192E-05	2.807E-05	5.042	4.789E-06	1.109E-05	13.238	1.831E-05	2.301E-05
1.611	1.230E-05	2.522E-05	5.346	5.045E-06	1.104E-05	14.001	1.445E-05	1.904E-05
1.711	1.501E-05	2.709E-05	5.645	9.983E-06	1.414E-05	14.770	1.395E-05	1.448E-05
1.814	1.814E-05	2.751E-05	5.937	1.043E-05	1.531E-05	15.604	2.874E-05	3.391E-05
1.934	1.810E-05	2.836E-05	6.253	6.552E-06	1.142E-05	16.511	6.333E-05	6.823E-05
2.101	1.675E-05	2.636E-05	6.554	3.916E-06	8.313E-06	17.501	4.530E-05	9.924E-05
2.298	1.215E-05	2.233E-05	6.837	4.883E-06	1.025E-05	18.485	8.993E-05	9.331E-05
2.495	1.229E-05	2.233E-05	7.232	8.639E-06	1.434E-05	19.459	5.794E-05	6.030E-05
2.699	1.564E-05	2.267E-05	7.733	9.713E-06	1.453E-05	20.512	2.516E-05	2.689E-05
2.900	1.560E-05	2.424E-05	8.232	8.957E-06	1.454E-05	21.531	8.511E-06	9.445E-06
3.049	9.278E-06	2.233E-05	8.752	8.276E-06	1.245E-05	22.507	2.299E-06	2.932E-06
3.298	7.008E-06	1.441E-05	9.254	4.444E-06	9.626E-06	23.552	2.401E-07	6.122E-07

INCIDENT NEUTRON ENERGY RANGE = 15.034 TO 17.485 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	3.510E-05	4.512E-05	3.494	1.002E-05	1.453E-05	9.733	1.631E-05	2.014E-05
1.008	3.114E-05	3.992E-05	3.700	7.519E-06	1.294E-05	10.250	1.957E-05	2.318E-05
1.108	1.458E-05	4.152E-05	3.907	7.715E-06	1.418E-05	10.767	1.574E-05	1.988E-05
1.206	1.368E-05	3.019E-05	4.154	1.190E-05	1.600E-05	11.235	1.208E-05	1.615E-05
1.311	2.082E-05	3.377E-05	4.454	1.435E-05	1.893E-05	11.738	1.276E-05	1.653E-05
1.413	1.838E-05	3.975E-05	4.752	1.475E-05	1.841E-05	12.432	1.731E-05	2.139E-05
1.512	1.795E-05	3.185E-05	5.042	1.141E-05	1.772E-05	13.238	2.810E-05	3.161E-05
1.611	1.671E-05	2.781E-05	5.346	1.189E-05	1.708E-05	14.001	5.407E-05	7.749E-05
1.711	1.861E-05	2.909E-05	5.645	1.512E-05	1.832E-05	14.770	8.766E-05	9.376E-05
1.814	2.335E-05	3.000E-05	5.937	1.531E-05	1.916E-05	15.604	9.895E-05	1.019E-04
1.934	2.387E-05	3.063E-05	6.253	1.231E-05	1.615E-05	16.511	7.322E-05	7.555E-05
2.101	2.194E-05	2.780E-05	6.554	8.427E-06	1.193E-05	17.501	3.539E-05	3.687E-05
2.298	1.985E-05	2.620E-05	6.837	4.507E-06	9.271E-06	18.485	1.191E-05	1.283E-05
2.495	2.042E-05	2.508E-05	7.232	1.998E-06	6.961E-06	19.459	2.719E-06	3.264E-06
2.699	1.673E-05	2.184E-05	7.733	3.218E-06	7.265E-06	20.512	3.064E-07	6.649E-07
2.900	1.332E-05	1.809E-05	8.232	2.120E-06	6.819E-06	21.531	1.006E-07	3.804E-07
3.049	1.340E-05	1.630E-05	8.752	3.577E-06	7.043E-06	22.507	7.659E-09	2.390E-09
3.298	1.205E-05	1.647E-05	9.254	9.388E-06	1.338E-05	23.552	-1.437E-07	2.196E-09

INCIDENT NEUTRON ENERGY RANGE = 12.532 TO 15.034 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	2.558E-05	3.393E-05	3.494	1.303E-05	1.646E-05	9.733	2.102E-05	2.417E-05
1.008	3.048E-05	3.746E-05	3.700	1.181E-05	1.545E-05	10.250	1.879E-05	2.184E-05
1.108	2.940E-05	4.449E-05	3.907	9.772E-06	1.423E-05	10.767	1.806E-05	2.146E-05
1.206	2.392E-05	3.874E-05	4.154	9.615E-06	1.266E-05	11.235	2.732E-05	3.060E-05
1.311	1.967E-05	3.024E-05	4.454	9.114E-06	1.309E-05	11.738	5.006E-05	5.374E-05
1.413	1.360E-05	3.344E-05	4.752	7.660E-06	1.066E-05	12.432	8.891E-05	9.145E-05
1.512	1.776E-05	3.033E-05	5.042	4.393E-06	1.021E-05	13.238	1.024E-04	1.045E-04
1.611	1.981E-05	2.809E-05	5.346	3.759E-06	8.494E-06	14.001	7.775E-05	7.942E-05
1.711	1.455E-05	2.795E-05	5.645	3.945E-06	6.730E-06	14.770	4.360E-05	4.472E-05
1.814	2.181E-05	2.664E-05	5.937	2.516E-06	7.282E-06	15.604	1.793E-05	1.881E-05
1.934	2.234E-05	2.704E-05	6.253	4.637E-06	9.165E-06	16.511	5.024E-06	5.574E-06
2.101	2.216E-05	2.594E-05	6.554	5.051E-06	8.210E-06	17.501	6.808E-07	1.033E-06
2.298	2.109E-05	2.517E-05	6.837	5.025E-06	9.494E-06	18.485	-1.423E-07	1.002E-07
2.495	1.986E-05	2.320E-05	7.232	1.029E-05	1.495E-05	19.459	-2.262E-07	-3.697E-08
2.699	1.817E-05	1.944E-05	7.733	7.716E-06	3.540E-05	20.512	-3.393E-07	-7.640E-09
2.900	1.607E-05	1.940E-05	8.232	2.924E-05	3.356E-05	21.531	-1.830E-07	-6.658E-08
3.049	1.552E-05	1.955E-05	8.752	2.791E-05	3.079E-05	22.507	-1.144E-07	-2.446E-08
3.298	1.349E-05	1.803E-05	9.254	2.273E-05	2.614E-05	23.552	-4.949E-08	-4.620E-09

INCIDENT NEUTRON ENERGY RANGE = 9.993 TO 12.532 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	3.317E-05	3.953E-05	3.494	3.501E-06	6.393E-06	9.733	6.853E-05	7.076E-05
1.008	3.545E-05	3.996E-05	3.700	4.510E-06	7.560E-06	10.250	9.362E-05	9.588E-05
1.108	3.129E-05	4.213E-05	3.907	5.847E-06	9.444E-06	10.767	9.940E-05	1.012E-04
1.206	3.027E-05	4.021E-05	4.154	8.126E-06	1.067E-05	11.235	8.885E-05	9.044E-05
1.311	2.813E-05	3.424E-05	4.454	1.136E-05	1.445E-05	11.738	6.732E-05	6.856E-05
1.413	2.003E-05	3.283E-05	4.752	1.724E-05	1.989E-05	12.432	3.563E-05	3.664E-05
1.512	2.016E-05	2.880E-05	5.042	2.489E-05	2.916E-05	13.238	1.228E-05	1.283E-05
1.611	2.074E-05	2.548E-05	5.346	3.557E-05	3.912E-05	14.001	3.605E-06	3.915E-06
1.711	1.913E-05	2.398E-05	5.645	4.293E-05	4.532E-05	14.770	9.938E-07	1.214E-06
1.814	1.781E-05	2.077E-05	5.937	4.353E-05	4.828E-05	15.604	1.390E-07	1.397E-07
1.934	1.556E-05	1.871E-05	6.253	4.501E-05	4.778E-05	16.511	-1.710E-07	-2.991E-08
2.101	1.461E-05	1.723E-05	6.554	4.018E-05	4.265E-05	17.501	-1.709E-07	-6.599E-08
2.298	1.301E-05	1.596E-05	6.837	3.258E-05	3.578E-05	18.485	-1.044E-07	-2.438E-08
2.495	1.226E-05	1.472E-05	7.232	2.293E-05	2.630E-05	19.459	-6.978E-08	-2.438E-08
2.699	1.198E-05	1.463E-05	7.733	1.585E-05	1.850E-05	20.512	-5.598E-08	-1.991E-08
2.900	1.087E-05	1.383E-05	8.232	1.221E-05	1.539E-05	21.531	-3.107E-08	-4.797E-09
3.049	8.769E-06	1.181E-05	8.752	1.694E-05	1.921E-05	22.507	-1.217E-08	-4.937E-09
3.298	5.147E-06	8.007E-06	9.254	3.000E-05	4.032E-05	23.552	4.338E-09	1.489E-08

TABLE 1. CONTINUED

INCIDENT NEUTRON ENERGY RANGE = 8.973 TO 9.99 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	9.313E-06	1.480E-05	3.494	1.278E-05	1.656E-05	9.733	8.030E-05	8.185E-05
1.008	1.060E-05	1.458E-05	3.700	1.506E-05	1.913E-05	10.250	4.183E-05	4.289E-05
1.108	1.196E-05	1.810E-05	3.907	2.395E-05	2.785E-05	10.767	1.700E-05	1.769E-05
1.206	1.558E-05	2.025E-05	4.154	4.407E-05	4.717E-05	11.235	6.487E-06	6.971E-06
1.311	1.679E-05	2.040E-05	4.454	6.768E-05	7.110E-05	11.739	2.063E-06	2.424E-06
1.413	1.543E-05	1.895E-05	4.752	7.444E-05	7.728E-05	12.432	2.730E-07	5.622E-07
1.512	1.261E-05	1.620E-05	5.042	5.951E-05	6.276E-05	13.238	-5.333E-10	1.830E-09
1.611	1.028E-05	1.405E-05	5.346	3.716E-05	4.043E-05	14.001	4.922E-08	1.952E-07
1.711	1.065E-05	1.392E-05	5.645	1.944E-05	2.219E-05	14.770	3.696E-08	1.412E-07
1.814	1.290E-05	1.649E-05	5.937	7.996E-06	1.126E-05	15.604	-1.891E-06	5.106E-06
1.934	1.711E-05	2.117E-05	6.253	2.737E-06	6.120E-06	16.511	-3.803E-08	3.238E-09
2.101	2.349E-05	2.719E-05	6.554	1.963E-06	5.054E-06	17.501	-2.386E-08	-5.382E-09
2.298	2.454E-05	2.865E-05	6.837	2.858E-06	5.638E-06	18.485	-1.262E-08	-1.524E-09
2.495	2.211E-05	2.576E-05	7.232	5.508E-06	8.566E-06	19.459	-1.134E-08	1.727E-09
2.699	1.769E-05	2.191E-05	7.733	1.975E-05	2.295E-05	20.512	-1.218E-08	2.423E-09
2.900	1.417E-05	1.792E-05	8.232	5.954E-05	6.244E-05	21.531	-1.299E-08	2.589E-09
3.099	1.205E-05	1.582E-05	8.752	1.049E-04	1.073E-04	22.507	-1.235E-08	4.505E-09
3.298	1.165E-05	1.573E-05	9.254	1.101E-04	1.121E-04	23.552	-3.706E-09	7.749E-09

INCIDENT NEUTRON ENERGY RANGE = 7.996 TO 8.973 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	1.505E-05	1.902E-05	3.494	6.173E-05	6.525E-05	9.733	1.982E-05	2.055E-05
1.008	1.305E-05	1.673E-05	3.700	5.931E-05	6.311E-05	10.250	6.968E-06	7.442E-06
1.108	9.465E-06	1.354E-05	3.907	4.981E-05	5.344E-05	10.767	2.664E-06	2.992E-06
1.206	1.040E-05	1.400E-05	4.154	6.320E-05	3.899E-05	11.235	1.203E-06	1.466E-06
1.311	1.520E-05	1.864E-05	4.454	1.828E-05	2.156E-05	11.739	5.304E-07	7.445E-07
1.413	1.972E-05	2.315E-05	4.752	9.065E-06	1.190E-05	12.432	2.086E-07	3.463E-07
1.512	2.158E-05	2.509E-05	5.042	7.157E-06	1.053E-05	13.238	1.099E-07	2.067E-07
1.611	2.288E-05	2.663E-05	5.346	4.720E-06	8.428E-06	14.001	6.335E-08	1.303E-07
1.711	2.625E-05	2.962E-05	5.645	3.299E-06	6.456E-06	14.770	2.195E-08	6.652E-08
1.814	3.026E-05	3.399E-05	5.937	1.411E-06	5.187E-06	15.604	-3.310E-09	5.502E-08
1.934	3.328E-05	3.740E-05	6.253	5.920E-07	4.498E-06	16.511	-8.223E-09	6.193E-09
2.101	3.256E-05	3.646E-05	6.554	6.342E-06	9.876E-06	17.501	-5.104E-09	1.454E-10
2.298	2.497E-05	2.915E-05	6.837	2.326E-05	2.633E-05	18.485	-2.631E-09	-1.050E-09
2.495	1.865E-05	2.238E-05	7.232	7.524E-05	7.843E-05	19.459	-1.762E-09	-7.418E-10
2.699	1.867E-05	2.302E-05	7.733	1.545E-04	1.574E-04	20.512	-1.587E-09	-4.925E-10
2.900	2.610E-05	2.992E-05	8.232	1.644E-04	1.669E-04	21.531	-1.587E-09	-4.128E-10
3.099	3.989E-05	4.360E-05	8.752	1.065E-04	1.081E-04	22.507	-1.461E-09	-2.079E-10
3.298	5.400E-05	5.791E-05	9.254	4.963E-05	5.071E-05	23.552	-1.181E-09	1.263E-10

INCIDENT NEUTRON ENERGY RANGE = 6.995 TO 7.996 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	1.680E-05	2.085E-05	3.494	1.184E-05	1.530E-05	9.733	2.372E-06	2.794E-06
1.008	2.107E-05	2.484E-05	3.700	6.752E-06	1.059E-05	10.250	6.548E-07	9.337E-07
1.108	3.006E-05	3.406E-05	3.907	6.193E-06	1.067E-05	10.767	3.049E-07	5.660E-07
1.206	2.956E-05	3.312E-05	4.154	1.124E-05	1.425E-05	11.235	2.862E-07	5.602E-07
1.311	3.210E-05	3.553E-05	4.454	1.110E-05	1.468E-05	11.739	2.687E-07	4.600E-07
1.413	3.342E-05	3.689E-05	4.752	4.658E-06	7.872E-06	12.432	1.728E-07	3.222E-07
1.512	3.296E-05	3.649E-05	5.042	1.638E-06	5.339E-06	13.238	9.612E-08	1.817E-07
1.611	3.205E-05	3.567E-05	5.346	5.418E-06	9.488E-06	14.001	4.616E-08	1.059E-07
1.711	3.217E-05	3.584E-05	5.645	1.447E-05	1.804E-05	14.770	1.216E-08	6.084E-08
1.814	3.319E-05	3.680E-05	5.937	3.209E-05	3.621E-05	15.604	7.441E-10	2.705E-08
1.934	3.443E-05	3.800E-05	6.253	7.165E-05	7.573E-05	16.511	-4.685E-09	1.042E-08
2.101	3.593E-05	3.961E-05	6.554	1.290E-04	1.324E-04	17.501	-3.184E-09	4.713E-09
2.298	4.642E-05	5.038E-05	6.837	1.011E-04	1.838E-04	18.485	-1.627E-09	3.723E-09
2.495	5.533E-05	5.873E-05	7.232	2.047E-04	2.071E-04	19.459	-2.023E-09	4.120E-09
2.699	4.713E-05	5.090E-05	7.733	1.031E-04	1.449E-04	20.512	-2.614E-09	4.379E-09
2.900	3.475E-05	3.815E-05	8.232	6.440E-05	6.569E-05	21.531	-2.689E-09	4.948E-09
3.099	2.696E-05	3.034E-05	8.752	2.162E-05	2.234E-05	22.507	-2.476E-09	5.499E-09
3.298	1.958E-05	2.323E-05	9.254	7.136E-06	7.639E-06	23.552	-1.894E-09	6.444E-09

INCIDENT NEUTRON ENERGY RANGE = 5.997 TO 6.995 MEV.

NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLUX(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	4.785E-05	5.179E-05	3.494	9.807E-06	1.298E-05	9.733	1.013E-06	1.274E-06
1.008	4.790E-05	5.175E-05	3.700	1.134E-05	1.483E-05	10.250	7.206E-07	9.262E-07
1.108	5.059E-05	5.457E-05	3.907	1.264E-05	1.591E-05	10.767	4.905E-07	6.454E-07
1.206	5.749E-05	6.089E-05	4.154	1.456E-05	1.730E-05	11.235	3.250E-07	4.492E-07
1.311	6.593E-05	6.918E-05	4.454	1.529E-05	1.843E-05	11.739	2.158E-07	3.036E-07
1.413	7.456E-05	7.771E-05	4.752	1.554E-05	1.837E-05	12.432	1.153E-07	1.752E-07
1.512	8.006E-05	8.314E-05	5.042	2.409E-05	2.727E-05	13.238	3.139E-08	7.033E-08
1.611	8.087E-05	8.404E-05	5.346	5.759E-05	6.081E-05	14.001	-1.919E-09	2.401E-08
1.711	7.864E-05	8.140E-05	5.645	1.089E-04	1.114E-04	14.770	-7.924E-09	8.812E-09
1.814	7.315E-05	7.610E-05	5.937	1.464E-04	1.493E-04	15.604	-4.873E-09	7.524E-09
1.934	6.287E-05	6.618E-05	6.253	1.476E-04	1.503E-04	16.511	-4.104E-09	9.342E-09
2.101	4.463E-05	4.741E-05	6.554	1.157E-04	1.174E-04	17.501	-4.702E-09	1.058E-08
2.298	2.292E-05	2.626E-05	6.837	7.703E-05	7.831E-05	18.485	-5.194E-09	1.062E-08
2.495	1.067E-05	1.345E-05	7.232	3.484E-05	3.570E-05	19.459	-5.668E-09	1.150E-08
2.699	7.043E-06	1.030E-05	7.733	9.748E-06	1.038E-05	20.512	-6.224E-09	1.194E-08
2.900	6.048E-06	9.056E-06	8.232	3.531E-06	4.045E-06	21.531	-6.592E-09	1.258E-08
3.099	6.387E-06	9.307E-06	8.752	2.377E-06	2.758E-06	22.507	-6.326E-09	1.375E-08
3.298	7.826E-06	1.102E-05	9.254	1.540E-06	1.842E-06	23.552	-5.131E-09	1.514E-08

TABLE 1. CONTINUED

INCIDENT NEUTRON ENERGY RANGE = 5.012 TO 5.997 MEV.

NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	5.592E-05	5.910E-05	3.494	2.428E-05	2.781E-05	9.733	4.422E-07	5.747E-07
1.008	5.120E-05	5.630E-05	3.700	2.180E-05	2.546E-05	10.250	3.085E-07	4.078E-07
1.108	4.567E-05	4.971E-05	1.907	2.326E-05	2.672E-05	10.767	2.247E-07	3.011E-07
1.206	3.777E-05	4.046E-05	4.154	3.976E-05	4.297E-05	11.235	1.402E-07	2.014E-07
1.311	2.867E-05	3.141E-05	4.454	9.532E-05	9.841E-05	11.738	5.917E-08	1.042E-07
1.413	1.965E-05	2.237E-05	4.752	1.710E-04	1.737E-04	12.432	3.107E-09	3.470E-08
1.512	1.233E-05	1.515E-05	5.042	2.185E-04	2.210E-04	13.238	-1.433E-08	1.618E-08
1.611	7.361E-06	1.038E-05	5.346	2.083E-04	2.103E-04	14.001	-1.002E-08	1.305E-08
1.711	5.389E-06	8.186E-06	5.645	1.521E-04	1.537E-04	14.770	-1.902E-09	1.409E-08
1.814	5.944E-06	9.076E-06	5.937	8.959E-05	9.111E-05	15.604	-2.152E-09	1.587E-08
1.934	6.630E-06	1.212E-05	6.253	4.140E-05	4.239E-05	16.511	-3.211E-09	1.808E-08
2.101	1.246E-05	1.579E-05	6.554	1.703E-05	1.771E-05	17.501	-4.626E-09	1.859E-08
2.298	1.177E-05	1.560E-05	6.837	6.958E-06	7.410E-06	18.485	-5.583E-09	1.960E-08
2.495	1.389E-05	1.726E-05	7.232	2.282E-06	2.666E-06	19.459	-5.032E-09	2.000E-08
2.699	2.581E-05	2.952E-05	7.733	1.180E-06	1.519E-06	20.512	-4.031E-09	2.259E-08
2.900	3.290E-05	3.654E-05	8.232	1.176E-06	1.473E-06	21.531	-3.625E-09	2.347E-08
3.099	3.040E-05	3.396E-05	8.752	8.752E-07	1.277E-06	22.507	-3.115E-09	2.610E-08
3.298	2.699E-05	3.070E-05	9.254	7.135E-07	8.804E-07	23.552	-2.244E-09	2.557E-08

INCIDENT NEUTRON ENERGY RANGE = 3.998 TO 5.012 MEV.

NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	6.804E-06	9.868E-06	3.494	9.959E-05	1.029E-04	9.733	-6.269E-09	8.149E-09
1.008	3.183E-06	6.391E-06	3.700	1.711E-04	1.763E-04	10.250	-6.305E-09	8.211E-09
1.108	5.247E-05	1.148E-05	1.907	2.525E-04	2.554E-04	10.767	-4.009E-09	3.444E-09
1.206	1.704E-05	1.551E-05	4.154	1.067E-04	1.092E-04	11.235	-2.333E-09	4.279E-09
1.311	1.420E-05	1.735E-05	4.454	2.801E-04	2.820E-04	11.738	-2.183E-09	4.344E-09
1.413	1.491E-05	1.805E-05	4.752	1.830E-04	1.843E-04	12.432	-2.341E-09	4.349E-09
1.512	1.955E-05	2.277E-05	5.042	9.160E-05	9.249E-05	13.238	-3.617E-09	4.158E-09
1.611	2.513E-05	2.847E-05	5.346	3.520E-05	3.576E-05	14.001	-4.487E-09	4.433E-09
1.711	2.680E-05	3.008E-05	5.645	1.187E-05	1.225E-05	14.770	-4.318E-09	4.613E-09
1.814	2.786E-05	3.140E-05	5.937	4.089E-06	4.374E-06	15.604	-2.854E-09	3.046E-09
1.934	3.066E-05	4.182E-05	6.253	1.708E-06	1.949E-06	16.511	-4.017E-10	2.854E-09
2.101	6.207E-05	6.583E-05	6.554	1.054E-06	1.244E-06	17.501	-2.745E-10	3.302E-09
2.298	7.727E-05	8.138E-05	6.837	7.223E-07	8.632E-07	18.485	-4.598E-10	3.449E-09
2.495	7.351E-05	7.708E-05	7.232	1.474E-07	4.692E-07	19.459	-4.540E-10	3.947E-09
2.699	5.818E-05	6.204E-05	7.733	1.088E-07	1.785E-07	20.512	-4.179E-10	4.310E-09
2.900	4.762E-05	5.152E-05	8.232	3.131E-08	7.493E-08	21.531	-5.260E-10	4.693E-09
3.099	4.779E-05	5.143E-05	8.752	7.843E-09	3.379E-08	22.507	-5.723E-10	4.793E-09
3.298	6.000E-05	6.373E-05	9.254	-2.873E-09	1.228E-08	23.552	-4.776E-10	4.715E-09

INCIDENT NEUTRON ENERGY RANGE = 2.995 TO 3.998 MEV.

NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	1.647E-05	1.961E-05	3.494	2.410E-04	2.427E-04	9.733	-9.735E-10	2.817E-10
1.008	2.116E-05	2.435E-05	3.700	1.812E-04	1.827E-04	10.250	-8.714E-10	4.408E-10
1.108	2.435E-05	2.766E-05	3.907	1.128E-04	1.137E-04	10.767	-7.771E-10	6.377E-10
1.206	2.769E-05	3.077E-05	4.154	5.085E-05	5.144E-05	11.235	-6.967E-10	7.998E-10
1.311	3.521E-05	3.844E-05	4.454	1.475E-05	1.507E-05	11.738	-5.773E-10	9.597E-10
1.413	4.533E-05	4.848E-05	4.752	3.517E-06	3.707E-06	12.432	-3.451E-10	2.110E-09
1.512	5.562E-05	5.881E-05	5.042	8.463E-07	9.706E-07	13.238	-4.054E-11	1.814E-09
1.611	6.816E-05	7.137E-05	5.346	2.409E-07	3.831E-07	14.001	1.068E-11	1.648E-09
1.711	8.320E-05	8.637E-05	5.645	1.114E-07	2.000E-07	14.770	3.242E-11	1.836E-09
1.814	9.670E-05	1.001E-04	5.937	3.443E-08	8.253E-08	15.604	3.540E-11	1.978E-09
1.934	1.034E-04	1.067E-04	6.253	-1.722E-08	1.526E-08	16.511	3.119E-11	2.039E-09
2.101	8.983E-05	9.334E-05	6.554	-2.711E-08	-2.733E-09	17.501	1.902E-11	2.255E-09
2.298	6.359E-05	6.694E-05	6.837	-2.152E-08	-3.610E-09	18.485	5.635E-12	2.067E-09
2.495	6.049E-05	6.368E-05	7.232	-1.007E-08	-9.989E-10	19.459	-9.314E-12	2.145E-09
2.699	5.998E-05	1.030E-04	7.733	-3.642E-09	-2.102E-10	20.512	-5.302E-11	2.050E-09
2.900	1.735E-04	1.765E-04	8.232	-1.508E-09	-9.447E-11	21.531	-1.119E-10	1.824E-09
3.099	2.394E-04	2.419E-04	8.752	-1.095E-09	9.914E-11	22.507	-1.459E-10	1.356E-09
3.298	2.636E-04	7.658E-04	9.254	-1.033E-09	1.093E-10	23.552	-1.345E-10	1.186E-09

INCIDENT NEUTRON ENERGY RANGE = 2.003 TO 2.995 MEV.

NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOX(N/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	5.648E-05	5.928E-05	3.494	3.497E-06	3.642E-06	9.733	-1.403E-05	1.925E-09
1.008	7.588E-05	7.862E-05	3.700	7.616E-07	8.561E-07	10.250	-1.951E-05	1.305E-09
1.108	9.141E-05	9.424E-05	3.907	6.269E-08	1.303E-07	10.767	-2.214E-09	1.294E-09
1.206	1.038E-04	1.064E-04	4.154	-7.302E-08	-2.570E-08	11.235	-2.048E-09	1.549E-09
1.311	1.151E-04	1.178E-04	4.454	-3.653E-08	-4.497E-08	11.738	-1.444E-09	1.477E-09
1.413	1.200E-04	1.226E-04	4.752	-4.886E-09	1.663E-08	12.432	-6.007E-10	2.256E-09
1.512	1.291E-04	1.318E-04	5.042	-4.635E-09	1.993E-08	13.238	-2.719E-10	2.669E-09
1.611	1.249E-04	1.274E-04	5.346	-1.133E-08	8.833E-11	14.001	-2.728E-10	2.966E-09
1.711	1.498E-04	1.522E-04	5.645	-1.002E-08	-1.357E-09	14.770	-3.309E-10	3.224E-09
1.814	1.957E-04	1.981E-04	5.937	-4.076E-09	3.605E-09	15.604	-4.133E-10	3.259E-09
1.934	2.521E-04	2.543E-04	6.253	9.983E-10	8.542E-09	16.511	-4.567E-10	3.341E-09
2.101	2.871E-04	2.891E-04	6.554	1.616E-09	8.826E-09	17.501	-4.268E-10	3.562E-09
2.298	2.571E-04	2.586E-04	6.837	7.404E-10	6.456E-09	18.485	-3.889E-10	3.697E-09
2.495	1.971E-04	1.983E-04	7.232	-1.027E-09	2.980E-09	19.459	-3.622E-10	3.941E-09
2.699	1.316E-04	1.325E-04	7.733	-1.594E-09	2.067E-09	20.512	-3.955E-10	4.191E-09
2.900	7.286E-05	7.344E-05	8.232	-1.125E-09	2.486E-09	21.531	-3.824E-10	4.470E-09
3.099	4.230E-05	4.267E-05	8.752	-7.009E-10	2.672E-09	22.507	-4.450E-10	4.101E-09
3.298	1.159E-05	1.182E-05	9.254	-8.910E-10	2.438E-09	23.552	-5.092E-10	3.673E-09

TABLE 1. CONTINUED

INCIDENT NEUTRON ENERGY RANGE = 0.998 TO 2.003 MEV.

NEUT. ENERGY (MEV)	FLOI(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOI(M/CM**2/MEV)		NEUT. ENERGY (MEV)	FLOI(M/CM**2/MEV)	
	LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT		LOWER LIMIT	UPPER LIMIT
0.908	2.693E-04	2.714E-04	3.494	-1.323E-08	-7.805E-12	9.733	-4.435E-10	1.437E-09
1.009	3.093E-04	3.112E-04	3.700	-1.293E-08	-3.675E-10	10.250	-4.551E-10	1.493E-09
1.108	3.151E-04	3.169E-04	3.907	-9.677E-09	1.284E-09	10.767	-7.021E-10	1.827E-09
1.206	3.039E-04	3.054E-04	4.154	-2.691E-09	5.812E-09	11.235	-8.571E-10	2.010E-09
1.311	2.874E-04	2.887E-04	4.454	1.270E-09	9.207E-09	11.738	-8.398E-10	2.232E-09
1.413	2.685E-04	2.696E-04	4.752	2.174E-09	8.202E-09	12.432	-6.296E-10	1.160E-09
1.512	2.454E-04	2.464E-04	5.042	1.859E-09	6.759E-09	13.238	-2.734E-10	1.017E-09
1.611	2.159E-04	2.167E-04	5.346	7.993E-11	5.216E-09	14.001	-2.016E-10	1.122E-09
1.711	1.782E-04	1.789E-04	5.645	-1.006E-09	3.377E-09	14.770	-2.528E-10	1.226E-09
1.814	1.331E-04	1.336E-04	5.937	-5.584E-10	2.572E-09	15.604	-3.474E-10	1.376E-09
1.914	8.169E-05	8.208E-05	6.253	2.796E-10	3.349E-09	16.511	-3.508E-10	1.403E-09
2.101	3.149E-05	3.173E-05	6.554	6.002E-10	3.910E-09	17.501	-3.163E-10	1.561E-09
2.298	7.409E-06	7.528E-06	6.937	3.972E-10	3.656E-09	18.485	-2.063E-10	1.759E-09
2.495	1.347E-06	1.420E-06	7.232	-2.189E-11	2.719E-09	19.459	-8.529E-12	2.111E-09
2.699	1.585E-07	2.053E-07	7.733	-4.957E-10	1.941E-09	20.512	1.877E-10	2.506E-09
2.900	-3.478E-08	-4.033E-09	8.232	-6.829E-10	1.824E-09	21.531	2.156E-10	2.658E-09
3.099	-3.706E-08	-1.625E-08	8.752	-7.502E-10	1.873E-09	22.507	6.198E-11	2.580E-09
3.298	-1.673E-08	-2.692E-09	9.254	-6.336E-10	1.726E-09	23.552	-9.983E-11	2.362E-09

TABLE 2.

RELATIVE ENERGY DEPENDENCE OF THE NEUTRON FLUX INCIDENT ON THE SCATTERING SAMPLE.

ENERGY (MEV)	FLUX (ARB. UNITS)	ENERGY (MEV)	FLUX (ARB. UNITS)	ENERGY (MEV)	FLUX (ARB. UNITS)	ENERGY (MEV)	FLUX (ARB. UNITS)
1.010	13.543	2.010	11.057	3.085	3.671	11.585	0.942
1.030	14.040	2.030	10.908	3.255	3.605	11.755	0.931
1.050	14.416	2.050	10.689	3.425	3.592	11.925	0.921
1.070	14.873	2.070	10.334	3.595	3.534	12.095	0.890
1.090	14.472	2.090	10.167	3.765	3.488	12.265	0.836
1.110	13.867	2.110	9.980	3.935	3.464	12.435	0.792
1.130	14.039	2.130	9.929	4.105	3.414	12.605	0.753
1.150	13.865	2.150	9.442	4.275	3.168	12.775	0.731
1.170	13.797	2.170	9.455	4.445	3.077	12.945	0.722
1.190	14.386	2.190	8.980	4.615	3.013	13.115	0.701
1.210	14.684	2.210	8.877	4.785	2.959	13.285	0.671
1.230	14.542	2.230	8.639	4.955	2.796	13.455	0.644
1.250	14.414	2.250	8.540	5.125	2.665	13.625	0.622
1.270	14.340	2.270	8.263	5.295	2.544	13.795	0.608
1.290	14.260	2.290	8.215	5.465	2.472	13.965	0.604
1.310	13.856	2.310	8.114	5.635	2.384	14.135	0.599
1.330	14.131	2.330	7.920	5.805	2.307	14.305	0.588
1.350	14.167	2.350	7.764	5.975	2.241	14.475	0.574
1.370	14.398	2.370	7.683	6.145	2.193	14.645	0.550
1.390	14.038	2.390	7.430	6.315	2.137	14.815	0.529
1.410	14.345	2.410	7.256	6.485	2.098	14.985	0.506
1.430	14.422	2.430	7.181	6.655	2.031	15.155	0.489
1.450	14.571	2.450	6.853	6.825	1.975	15.325	0.462
1.470	14.923	2.470	6.636	6.995	1.952	15.495	0.462
1.490	14.732	2.490	6.282	7.165	1.873	15.665	0.455
1.510	14.502	2.510	6.007	7.335	1.816	15.835	0.440
1.530	14.551	2.530	5.949	7.505	1.901	16.005	0.429
1.550	14.628	2.550	5.842	7.675	1.778	16.175	0.425
1.570	14.623	2.570	5.609	7.845	1.768	16.345	0.403
1.590	14.013	2.590	5.460	8.015	1.726	16.515	0.396
1.610	14.281	2.610	5.336	8.185	1.644	16.685	0.373
1.630	14.489	2.630	5.022	8.355	1.577	16.855	0.361
1.650	14.222	2.650	4.654	8.525	1.542	17.025	0.353
1.670	14.068	2.670	4.118	8.695	1.513	17.195	0.331
1.690	13.845	2.690	3.731	8.865	1.500	17.365	0.324
1.710	13.538	2.710	3.450	9.035	1.518	17.535	0.313
1.730	12.990	2.730	3.393	9.205	1.494	17.705	0.307
1.750	12.694	2.750	3.409	9.375	1.429	17.875	0.300
1.770	12.721	2.770	3.377	9.545	1.340	18.045	0.293
1.790	12.893	2.790	3.411	9.715	1.317	18.215	0.286
1.810	12.634	2.810	3.450	9.885	1.255	18.385	0.280
1.830	12.787	2.830	3.563	10.055	1.209	18.555	0.268
1.850	12.802	2.850	3.409	10.225	1.173	18.725	0.257
1.870	12.656	2.870	3.453	10.395	1.160	18.895	0.246
1.890	12.232	2.890	3.474	10.565	1.163	19.065	0.243
1.910	11.885	2.910	3.491	10.735	1.144	19.235	0.241
1.930	11.881	2.930	3.430	10.905	1.134	19.405	0.231
1.950	11.672	2.950	3.549	11.075	1.084	19.575	0.224
1.970	11.509	2.970	3.549	11.245	1.020	19.745	0.221
1.990	11.310	2.990	3.592	11.415	0.974	19.915	0.212

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