**Cross Sections for the Nb(n,xn)
and Nb(n,x γ) Reactions
Between 1 and 20 MeV**

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24

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ABSTRACT

Differential cross sections for the production of secondary neutrons and gamma rays from neutron interactions in niobium have been measured at 129 deg. (lab) for incident neutron energies in the range 1 to 20 MeV. An electron linac was used as a white neutron source. Incident neutron energies were determined using time-of-flight techniques for a source-to-sample distance of 48 m. Secondary spectra were determined by unfolding the pulse-height distributions observed in a NE-213 scintillation counter. The results are compared to the current evaluated data file (ENDF/B-IV, MAT 1189).

INTRODUCTION

In recent years considerable interest has been shown in using niobium as a structural material in practical fusion reactors. Most conceptual design studies (see for example Ref. 1) use the evaluated nuclear data files (e.g. ENDF/B) as the source of basic microscopic cross-section data.

The interaction of neutrons in a structural material can profoundly affect the neutron spectrum and can give rise to secondary gamma radiation. Thus the accuracy of the prediction of design criteria such as tritium breeding ratios, radiation damage, and neutron-induced heating is dependent on the accuracy of the evaluated data for energies up to the primary energy of 14 MeV.

The existing experimental data on niobium cross sections have been assembled, evaluated, and supplemented with model calculations by Smith, Guenther, and Whalen^{2,3)} to form the most recent data file for this nuclide (ENDF/B-IV, MAT 1189). The purpose of the measurements reported here was to provide data to test some aspects of this evaluated cross-section file in the MeV region. The work was performed as part of a program at the Oak Ridge National Laboratory⁴⁻¹⁴⁾ to check evaluated files for those elements that are important in applied problems dealing with weapons radiation, reactor shielding, and fusion technology.

The technique used here has the global testing power of an integral experiment, yet retains a sufficiently differential character to allow localization of discrepancies. The data presented here provide some important tests of evaluated cross sections over wide ranges of incident

and secondary energies. Of particular value for niobium are the data on the large cross sections for the production of low-energy secondary neutrons produced by incident neutrons with energies above 8 MeV.

Measurements with the more conventional neutron sources such as Van de Graaff accelerators are difficult, if not impossible, in this region (except near 14 MeV).

Differential cross sections are presented for the $\text{Nb}(n,xn)$ and $\text{Nb}(n,x\gamma)$ reactions as a function of the incident neutron energy at an average scattering angle of 129 deg. (lab). The data cover the incident-neutron energy range from 1 to 20 MeV. Comparisons are presented with cross sections obtained from the evaluation.

EXPERIMENTAL PROCEDURE

The experimental procedure used in this work has been described in detail elsewhere¹⁵⁾ and is only briefly outlined here. The Oak Ridge Electron Linear Accelerator (ORELA) was used as a white neutron source. An electron-beam pulse width of 10 nsec at a repetition rate of 800 sec^{-1} was employed. A beam time of 200 h was necessary for completion of the measurement.

The scattering sample was located 47.8 m from the neutron source and consisted of 2678 g of niobium (>99.85% purity) formed in an annular ring about 25 cm in o.d., 15 cm in i.d., and with a thickness of 0.95 cm ($= 0.0535 \text{ atom/b}$). Details of the sample-detector geometry are shown in Fig. 1. Secondary radiation from the sample was detected by an NE-213 scintillation counter¹⁶⁾ located on the axis of the ring at a position corresponding to an average scattering angle of 129 deg. 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 neutron-induced radiation from the sample was detected. Each event in the detector was identified as either a neutron or gamma ray by pulse-shape discrimination techniques. The event was then recorded in a two-parameter array of pulse height versus time-of-flight (one each for secondary neutrons and secondary gamma rays).

Backgrounds were determined by measurement with the sample removed. For secondary gamma-rays the background was everywhere less than 10%. For secondary neutrons the background was worst at the highest incident neutron energies where it varied from 10% at the lowest pulse heights to

about 50% at the highest pulse heights (corresponding to elastically scattered neutrons).

The neutron flux incident on the sample was measured with the NE-213 detector using one-parameter time-of-flight techniques. The sample and shadow bar were not in place for these measurements. A flux monitor consisting of a small plastic scintillator located in the edge of the neutron beam 30 m from the source was used to normalize the flux measurement to the actual cross-section measurement. The flux determination had an absolute uncertainty of $\sim 10\%$ and is the largest source of systematic error in the cross sections reported here.

DATA REDUCTION

Two methods of data reduction were employed, each intended to make use of the available counting statistics to provide a maximum of information about a particular aspect of the cross section. In the first, the neutron or gamma-ray pulse-height spectra were integrated over intervals of neutron time-of-flight to form pulse-height spectra for specific incident neutron energy ranges. These intervals ranged in width from $\Delta E_{\text{inc}} = 0.25 \text{ MeV}$ at $E_{\text{inc}} = 1 \text{ MeV}$ to $\Delta E_{\text{inc}} = 2.5 \text{ MeV}$ at $E_{\text{inc}} = 20 \text{ MeV}$. The pulse-height spectra so formed were then converted to neutron or gamma-ray energy distributions by unfolding with the FERD code¹⁷⁾ using the measured neutron or gamma-ray response functions of the NE-213 detector. These distributions were then normalized to cross sections using the measured neutron flux, average sample-to-detector distance, and sample thickness.

The neutron data were also corrected for finite sample effects. These corrections consisted of two terms: 1) the average transmissions of the incident and scattered neutron fluxes and 2) the fraction of neutrons reaching the detector after scattering only once. The magnitudes of the corrections were calculated using a Monte Carlo method with group cross sections generated from the evaluated data by the AMPX code system.¹⁸⁾

Forty neutron groups were used in this calculation covering neutron energies from 0.5 to 20 MeV. Group widths were varied so as to maintain about a 10% relative width. Calculated transmissions depend only on the total cross sections and varied from about 0.66 for 1-MeV secondary neutrons at $E_{inc} = 1$ MeV to 0.79 for 20-MeV secondary neutrons at $E_{inc} = 20$ MeV. Accuracy for this part of the correction is better than 3% since the total cross section is accurately known.^{2,3)} Estimates for the fraction of singly-scattered neutrons in the flux at the detector for the two cases cited above were 54% and 90% respectively. The uncertainty in the multiple scattering component is dependent on how well the evaluated data describe the neutron interactions; therefore to allow for any inaccuracies in the evaluation an uncertainty of 10% was assigned the multiple scattering component and this error was then propagated through the succeeding analysis.

The gamma-ray data were corrected for self-absorption in the sample (this correction ranged from ~35% at $E_\gamma = 0.7$ MeV to ~20% at $E_\gamma = 10$ MeV) and for attenuation of the incident neutron beam (<5%, calculated using the non-elastic cross section). Neutron multiple scattering was not calculated, however, since most multiple scattering involves elastic scattering which does not appreciably degrade the neutron energy; using

the non-elastic cross in the beam attenuation should approximately allow for finite sample effects. Previous investigations by other workers^{19,20)} have shown this approximation to yield results accurate to about 5% for sample thicknesses up to $\sim 1/2$ mean-free-path (our sample thickness was $\sim 1/4$ mfp).

The unfolded data described above provide detailed information about the secondary neutron and gamma-ray spectra, but because the unfolding technique requires good statistical accuracy, the data must be binned over rather large incident energy intervals. Therefore, a second type of data reduction, pulse-height weighting,^{15,21)} was also used. This technique provided only integral information about the secondary spectra (i.e., total yield and average energy), but because the demands on statistical accuracy are less severe, it allowed better resolution in the incident neutron energy. For this work, the pulse-height weighting analysis was applied to spectra formed by integration over time-of-flight intervals corresponding to $\Delta E_{inc} = 0.05$ MeV at $E_{inc} = 1$ MeV and increasing to $\Delta E_{inc} = 0.5$ MeV at $E_{inc} = 20$ MeV. The finite sample effects described above were also applied to these data. The final results of this analysis are total yield of secondary neutrons and secondary gamma rays and their average energies as a function of incident neutron energy.

DISCUSSION

The work reported here provides data to test the evaluated secondary cross sections over a wide range of energies. The experimental resolution in the incident energy (~ 0.4 nsec/m contributed by electron pulse width and sample-to-detector time lag) is very good, so resolution in the final

data is limited only by the bin widths necessary for adequate statistical accuracy. Energy resolution for the secondary spectra is due to the NE-213 detector's pulse-height resolution and is rather poor. This resolution does not permit unique identification of the various reactions contributing to the secondary spectra. Because of this inability to completely resolve the different contributions to the secondary spectra, results have been obtained only as laboratory-system cross sections. These data are shown in Figs. 2-16. The errors shown do not include an estimated 10% uncertainty in overall normalization.

The incident neutron energy spread, the secondary energy resolution, and the angular spread are much broader in this measurement than in the more standard types of differential cross-section measurements. This has the advantage of allowing a comprehensive check of the cross sections in a mode that is compatible with the averaging generally used to produce group cross sections for practical applications. However, it complicates and sometimes makes impossible the comparison of the present results with previous measurements. To correctly average over the experimental spreads, one needs a complete prescription for cross sections as a function of incident energy, scattering angle, Q-value, etc. The evaluated file provides this information and reflects previous measurements to the extent that the evaluators base their results on existing data.

Secondary neutron and gamma-ray spectra were generated from the evaluation by suitably weighted averages over the incident energy interval and angular spread. Kinematics were treated exactly along with the angular dependence specified in the evaluation. The resulting secondary spectra were then smeared by a Gaussian function corresponding to the resolution of the detector.

Curves for comparison to the pulse-height weighting results were produced by integration of secondary neutron or gamma-ray spectra calculated as described above for the smaller incident energy intervals used in the pulse-height weighting analysis.

Several cautions are in order before proceeding to a detailed comparison of evaluation and measurement. The gamma-ray emission specified in the evaluation is isotropic. For a medium-weight nuclide such as niobium this is undoubtedly very close to the real angular dependence. Thus a comparison of the evaluation with differential measurements taken at 129 deg. checks the gamma-ray production data of the evaluation rather well. In the case of the neutron emission spectra, the test may not be as rigorous, especially with respect to the elastic scattering cross sections. Because of the strong anisotropy of the elastic angular distribution with its energy dependent diffraction pattern, a single differential measurement covering a large angular range (the angular resolution function is approximately triangular with a base 28 deg. in width) provides little or no check on the validity of the elastic scattering cross section in the evaluation. However, for neutron emission through processes other than elastic scattering, the situation is much better since most such emissions are only weakly anisotropic, particularly when averaged over the broad incident neutron energies of this experiment.

Figures 2 and 3 show the pulse-height weighting results for the total yield and average energy of gamma rays above $E_\gamma = 0.7$ MeV. Figures 4 through 8 show detailed secondary gamma-ray spectra for the wider intervals in incident neutron energy. Figure 2 indicates that the evaluated total yield is generally low, particularly for $E_n < 10$ MeV. Figure 3 shows the

average energy from the evaluation to be 20-30% high over the whole range studied. Reference to Figs. 2-8 indicates that the measured and evaluated spectra generally agree except at the low gamma-ray energies where the differences are pronounced. These comparisons pretty well explain the variation between the total yields (Figs. 2 and 3) as due to discrete gamma-ray transitions at about 0.95 and 1.5 MeV (remember that the resolution of the NE-213 detector smears sharp lines into rather broad humps). Reference to the evaluation³⁾ indicates that for $E_n > 100$ keV the gamma-ray production is based on calculations using the formalism of Howerton and Plechaty²²⁾ as modified by Perkins, Haight and Howerton.²³⁾ This is an empirical formalism based on systematics for $A > 20$. As such it appears to do a rather good job of estimating the continuum components of the gamma-ray spectra in Figs. 2-8. However, the inclusion of discrete gamma-ray transitions would improve the evaluation.

Another measurement of the $\text{Nb}(n, \gamma)$ cross section has recently been made at this laboratory by Dickens *et al.*²⁴⁾ using a NaI crystal as the gamma-ray detector and a different sample and geometry. Agreement, as illustrated for two typical spectra in Figs. 9 and 10, is very good. However, it should be pointed out that there are sources of systematic uncertainty (such as measurement of incident neutron flux and calibrated gamma-ray sources used to measure detector efficiency) that are common to both experiments. Thus this comparison may not show some systematic errors in the measurements. The results from Ref. 24 have been smeared to the same resolution as the present measurement.

Comparisons of the measured and evaluated neutron emission data are shown in Figs. 11-17. A few points about the evaluation will facilitate

the comparison. Below 2.5 MeV the neutron emission is only through elastic scattering and inelastic scattering to discrete levels up to 2.5 MeV excitation energy. Above 2.5 MeV continuum inelastic scattering is added, gradually increasing to become the total inelastic cross section above 6 MeV. Reactions for $(n,2n)$ and $(n,3n)$ have thresholds occurring at 8.91 and 16.90 MeV respectively. The only other neutron emission cross section is the $Nb(n,n'\alpha)$ reaction which has a threshold of 1.97 MeV, but whose magnitude is negligible compared to other cross sections.

Below 4 MeV the elastic and discrete inelastic scattering of the evaluation are based on very extensive measurements of Smith, Guenther and Whalen.²⁾ As indicated by Figs. 11-17, agreement between the present data and the evaluation is good up to 3 MeV. Agreement from 3 to 4 MeV for the inelastic scattering is not as good (see Fig. 15). Our data are compared in Fig. 18 with results for the complete evaluation and the evaluation with continuum inelastic scattering deleted. This comparison indicates that the discrete inelastic scattering is almost sufficient and less continuum component in the 3 to 4 MeV region would improve agreement.

Above 4 MeV experimental data on the various reactions become sparse and the evaluators rely more heavily on supplementation by theoretical models. In general, the evaluated neutron emission spectra are in fairly good agreement with the measurements. The worst disagreements are above $E_{inc} = 6$ MeV (see Figs. 16 and 17) and occur in the region of low secondary energies where the evaluated spectra are systematically low and decrease with energy more rapidly than the measured spectra.

Since the evaluation was prepared several new sets of data²⁵⁻²⁷⁾ on emission spectra have been reported. Figs. 19 and 20 show comparisons

between the present results, ENDF/B-IV, and curves taken from Refs. 25, 26 and 27. Note that our data and the evaluated curve are averages over broad incident energy ranges while the data of Refs. 25, 26 and 27 have small spreads on the incident energies. In spite of these differences the comparisons are informative. The measured data sets are generally consistent and show slopes for the non-elastic emission spectra which are distinctly different from the evaluation.

CONCLUSIONS

The evaluation for niobium in ENDF/B-IV has been found to give generally satisfactory agreement with the measurement. Some improvements are suggested by the present work. The gamma-ray production cross sections could be improved by the inclusion of discrete lines, or better yet by the direct inclusion of measured cross sections such as those reported in Ref. 24. For neutrons, less continuum inelastic scattering for incident energies from 3 to 4 MeV would improve agreement in this region. At higher neutron energies a less pronounced slope and larger magnitude on the non-elastic emission spectra would better represent existing measurements. For the reasons listed previously the present data do not provide a valid test of the elastic scattering.

Numerical values of the differential cross sections measured in this work are given in the tables of the Appendix.

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Fig. 1. Schematic diagram showing details of the sample-detector geometry.

Fig. 2. Total yield for the Nb(n,x γ) reaction for $E_\gamma \geq 0.7$ MeV as a function of the incident neutron energy. Error bars shown in this and succeeding figures do not include a 10% uncertainty in absolute normalization.

Fig. 3. Average secondary gamma-ray energy as a function of incident neutron energy.

Fig. 4. Secondary gamma-ray spectra for incident neutron energies between 1 and 2 MeV.

Fig. 5. Secondary gamma-ray spectra for incident neutron energies between 2 and 3 MeV.

Fig. 6. Secondary gamma-ray spectra for incident neutron energies between 3 and 5 MeV.

Fig. 7. Secondary gamma-ray spectra for incident neutron energies between 5 and 10 MeV.

Fig. 8. Secondary gamma-ray spectra for incident neutron energies between 10 and 20 MeV.

Fig. 9. Gamma-ray spectra for incident neutron energies between 7 and 8 MeV. The points represent the present work while the two curves are the 68% confidence limits for the data of Ref. 24.

Fig. 10. Comparison of gamma-ray data from the present work for $E_{\text{inc}} = 12.5\text{-}15.0$ MeV and data from Ref. 24 for $E_{\text{inc}} = 12.0\text{-}14.0$ MeV.

Fig. 11. Total yield at 129° for the Nb(n,xn) reaction as a function of incident neutron energy for secondary neutron energies of 0.76 MeV and above.

Fig. 12. Average secondary neutron energy as a function of the incident neutron energy.

Fig. 13. Secondary neutron spectra for incident neutron energies between 1 and 2 MeV.

Fig. 14. Secondary neutron spectra for incident neutron energies between 2 and 3 MeV.

Fig. 15. Secondary neutron spectra for incident neutron energies between 3 and 5 MeV.

$\text{Nb}(n, xn) \quad \theta = 129^\circ$

- ENDF/B IV

Fig. 16. Secondary neutron spectra for incident neutron energies between 5 and 10 MeV.

$\text{Nb}(n, xn) \theta = 129^\circ$

ENDF/B IV

Fig. 17. Secondary neutron spectra for incident neutron energies between 10 and 20 MeV.

ORNL-DWG 77-3868

Fig. 18. Secondary neutron spectrum for incident neutrons from 3.0 to 3.5 MeV. The solid curve is the complete evaluation while the dashed curve results from deletion of the continuum inelastic component.

Fig. 19. Secondary neutron spectrum from this work ($E_{\text{inc}} = 9\text{-}10 \text{ MeV}$), data from Ref. 25 ($E_{\text{inc}} = 9.1 \text{ MeV}$), and from ENDF/B-IV ($E_{\text{inc}} = 9\text{-}10 \text{ MeV}$). The curve for Ref. 25 is their value of the total cross section divided by 4π .

Fig. 20. Secondary neutron spectrum from this work ($E_{\text{inc}} = 12.5\text{-}15 \text{ MeV}$), data from Ref. 26 ($E_{\text{inc}} = 14.4 \text{ MeV}$), Ref. 27 ($E_{\text{inc}} = 14.6 \text{ MeV}$) and from ENDF/B-IV ($E_{\text{inc}} = 12.5\text{-}15 \text{ MeV}$). The curves for Refs. 26 and 27 are their values divided by 4π .

TABLE 1. $\text{Nb}(\text{H}, \text{X})$ LABORATORY CROSS SECTIONS AT 12° DEGREES. THE NUMBERS IN PARENTHESES ARE STATISTICAL UNCERTAINTIES (IN THE SAME UNITS AS THE DATA). THE OVERALL UNCERTAINTY IN NORMALIZATION IS ABOUT 10%.

INCIDENT ENERGY	17.46-19.92	15.01-17.46	12.52-15.01	9.98-12.52	9.02- 9.98	7.99- 5.02	6.99- 7.99
SECONDARY ENERGY (MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)
0.907	109.3 (6.3)	102.1 (5.7)	107.1 (5.2)	54.7 (5.0)	51.4 (4.3)	56.4 (4.3)	62.2 (4.3)
1.007	107.8 (5.5)	98.4 (5.0)	101.8 (4.5)	61.0 (4.3)	49.2 (3.6)	55.5 (3.6)	50.4 (3.9)
1.107	100.4 (5.1)	88.9 (4.5)	94.6 (4.0)	42.9 (3.7)	47.3 (3.1)	55.7 (3.2)	53.7 (3.3)
1.205	52.6 (4.5)	82.6 (3.9)	88.5 (3.6)	75.6 (3.2)	46.4 (2.8)	53.5 (2.8)	51.0 (2.4)
1.310	88.3 (4.3)	74.8 (3.5)	85.0 (3.1)	70.6 (2.7)	46.4 (2.5)	51.2 (2.6)	51.1 (2.7)
1.412	86.1 (4.0)	77.2 (3.4)	81.4 (3.1)	66.3 (2.5)	46.4 (2.1)	49.2 (2.4)	50.7 (2.5)
1.511	63.1 (3.6)	75.4 (3.1)	74.7 (2.9)	61.1 (2.2)	42.0 (2.1)	46.4 (2.2)	48.2 (2.2)
1.610	78.9 (3.5)	72.3 (2.8)	67.9 (2.4)	56.0 (1.9)	38.9 (1.9)	42.3 (1.5)	44.1 (1.9)
1.709	74.4 (3.4)	67.7 (2.7)	63.2 (2.1)	51.5 (1.7)	37.7 (1.4)	38.7 (1.7)	40.8 (1.7)
1.813	70.2 (3.3)	62.9 (2.6)	60.6 (2.0)	47.5 (1.6)	34.1 (1.4)	37.2 (1.7)	39.5 (1.7)
1.932	66.0 (3.1)	58.2 (2.3)	57.4 (1.9)	44.1 (1.5)	37.1 (1.7)	37.0 (1.7)	39.1 (1.6)
2.100	60.6 (3.0)	53.4 (2.1)	50.4 (1.7)	39.6 (1.3)	31.2 (1.6)	33.6 (1.6)	35.9 (1.5)
2.296	52.7 (2.9)	47.9 (1.9)	42.3 (1.6)	32.7 (1.2)	27.0 (1.4)	29.0 (1.4)	29.1 (1.3)
2.499	44.9 (2.5)	40.5 (1.7)	35.5 (1.3)	27.4 (1.0)	25.9 (1.2)	27.0 (1.2)	24.9 (1.1)
2.703	39.3 (2.2)	35.1 (1.6)	30.9 (1.2)	24.2 (0.9)	24.6 (1.2)	24.7 (1.2)	23.1 (1.2)
2.898	34.7 (2.2)	31.3 (1.6)	26.2 (1.2)	21.4 (0.9)	22.3 (1.2)	21.7 (1.2)	20.5 (1.2)
3.097	31.1 (2.2)	26.6 (1.6)	21.5 (1.2)	17.8 (0.9)	19.2 (1.2)	17.6 (1.2)	16.3 (1.2)
3.296	27.7 (1.9)	22.6 (1.3)	18.7 (1.0)	14.6 (0.8)	13.5 (1.0)	13.5 (1.0)	13.1 (1.1)
3.501	23.8 (2.0)	19.9 (1.4)	17.1 (1.1)	12.9 (0.7)	10.0 (1.1)	11.1 (1.1)	11.8 (1.2)
3.707	19.7 (1.7)	17.1 (1.2)	14.8 (0.9)	11.8 (0.7)	8.9 (1.0)	9.6 (1.0)	11.3 (1.1)
3.906	16.9 (1.6)	14.1 (1.1)	12.6 (0.8)	10.5 (0.6)	8.3 (0.9)	8.3 (0.9)	10.1 (1.0)
4.153	15.9 (1.5)	12.1 (1.0)	10.8 (0.8)	9.0 (0.7)	7.7 (0.9)	7.2 (0.9)	7.6 (1.0)
4.454	14.5 (1.3)	11.8 (0.9)	8.6 (0.7)	7.5 (0.6)	7.0 (0.8)	6.2 (0.9)	6.0 (1.0)
4.752	12.5 (1.1)	10.4 (0.8)	6.8 (0.7)	6.2 (0.6)	5.9 (0.9)	5.9 (0.8)	5.5 (0.9)
5.042	10.9 (1.1)	8.6 (0.8)	6.2 (0.7)	5.0 (0.6)	5.0 (0.8)	6.3 (0.9)	5.1 (0.9)
5.346	10.0 (1.0)	6.8 (0.8)	5.9 (0.6)	4.1 (0.5)	4.7 (0.9)	5.5 (0.8)	4.2 (0.9)
5.645	8.6 (1.1)	4.8 (0.8)	5.0 (0.7)	3.7 (0.6)	4.0 (0.8)	4.1 (0.9)	4.9 (1.0)
5.938	6.8 (1.1)	3.5 (0.7)	4.2 (0.7)	3.1 (0.6)	3.5 (0.9)	3.6 (0.9)	3.9 (1.0)
6.254	5.6 (1.0)	3.3 (0.7)	3.7 (0.7)	3.0 (0.6)	3.5 (0.9)	3.6 (0.9)	3.9 (1.0)
6.556	5.0 (0.9)	4.8 (0.8)	3.4 (0.7)	3.6 (0.6)	3.9 (0.8)	3.5 (0.9)	17.3 (0.9)
6.839	4.4 (0.9)	4.9 (0.8)	3.2 (0.6)	3.9 (0.5)	4.0 (0.8)	5.6 (0.7)	20.6 (0.7)
7.235	3.3 (1.0)	3.7 (0.9)	3.2 (0.7)	3.1 (0.6)	3.6 (0.9)	10.1 (0.8)	18.8 (0.6)
7.736	2.5 (0.9)	2.3 (0.8)	2.3 (0.7)	1.6 (0.6)	0.6 (0.9)	13.4 (0.8)	9.9 (0.4)
8.236	3.3 (0.9)	1.9 (0.8)	0.9 (0.7)	1.8 (0.6)	7.2 (0.8)	11.6 (0.6)	3.6 (0.4)
8.755	3.3 (0.9)	2.7 (0.8)	0.6 (0.7)	3.2 (0.5)	8.5 (0.6)	6.8 (0.8)	1.3 (0.1)
9.261	1.4 (0.9)	3.3 (0.8)	1.1 (0.7)	4.4 (0.5)	1.1 (0.7)	6.5 (0.8)	2.9 (0.4)
9.741	0.0 (0.9)	2.7 (0.8)	1.6 (0.7)	4.6 (0.5)	3.9 (0.7)	1.1 (0.2)	0.6 (0.1)
10.259	0.3 (0.9)	1.4 (0.8)	1.9 (0.7)	5.0 (0.4)	1.7 (0.2)	0.5 (0.1)	
10.777	1.0 (0.9)	0.7 (0.8)	2.5 (0.7)	4.6 (0.4)	0.6 (0.1)	0.2 (0.1)	
11.247	1.5 (0.9)	0.6 (0.8)	3.3 (0.7)	3.6 (0.3)	0.2 (0.1)		
11.751	1.9 (1.0)	0.7 (0.8)	4.2 (0.6)	2.2 (0.2)	0.1 (0.0)		
12.393	2.1 (1.0)	0.7 (0.8)	4.5 (0.6)	0.9 (0.1)			
13.201	2.4 (1.0)	0.9 (0.8)	3.1 (0.3)				
14.021	2.8 (1.0)	2.1 (0.7)	2.1 (0.3)				
14.792	2.1 (1.0)	4.1 (0.6)	0.9 (0.2)				
15.629	1.5 (1.0)	3.4 (0.5)	0.2 (0.1)				
16.539	2.2 (0.8)	1.6 (0.3)					
17.533	2.7 (0.6)	0.5 (0.2)					
18.521	1.9 (0.5)						
19.498	0.7 (0.3)						
20.556	0.1 (0.2)						

INCIDENT ENERGY	5.99- 6.99	5.01- 5.99	4.50- 5.01	3.99- 4.50	3.51- 3.99	3.00- 3.51	2.75- 3.00
SECONDARY ENERGY (MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)
0.907	65.6 (4.8)	70.3 (4.7)	76.3 (4.7)	75.0 (5.4)	68.2 (5.3)	78.0 (4.5)	108.1 (5.9)
1.007	61.2 (4.1)	70.5 (4.2)	69.8 (4.0)	76.8 (4.9)	64.8 (4.1)	82.6 (4.0)	102.6 (5.0)
1.107	58.9 (3.6)	63.6 (3.4)	65.5 (3.6)	71.9 (4.1)	59.2 (3.2)	81.4 (3.6)	100.1 (4.5)
1.205	56.9 (3.2)	57.2 (2.8)	61.8 (3.3)	64.1 (3.3)	53.8 (2.6)	77.3 (3.1)	91.4 (3.9)
1.310	53.4 (2.8)	53.7 (2.5)	57.8 (2.9)	55.9 (2.7)	51.6 (2.3)	70.5 (2.8)	78.9 (3.5)
1.412	51.7 (2.5)	52.7 (2.4)	55.2 (2.7)	49.4 (2.3)	50.7 (2.2)	63.7 (2.5)	69.6 (3.3)
1.511	50.1 (2.3)	50.5 (2.2)	53.2 (2.5)	45.4 (2.1)	47.3 (2.1)	59.4 (2.4)	65.9 (3.0)
1.610	46.6 (2.0)	45.7 (1.9)	47.4 (2.2)	41.4 (1.9)	49.5 (2.0)	56.7 (2.3)	68.2 (3.0)
1.709	41.2 (1.8)	41.2 (1.7)	45.5 (2.0)	38.0 (1.9)	46.1 (1.9)	54.1 (2.2)	71.0 (3.0)
1.813	37.3 (1.7)	38.9 (1.7)	40.3 (1.9)	35.6 (1.8)	43.4 (1.9)	51.5 (2.1)	66.9 (3.0)
1.932	34.7 (1.6)	38.5 (1.6)	35.6 (1.8)	33.6 (1.8)	40.1 (1.7)	49.7 (2.1)	52.6 (2.8)
2.100	32.6 (1.4)	36.2 (1.5)	30.7 (1.7)	30.7 (1.8)	34.3 (1.9)	45.7 (2.0)	30.2 (2.5)
2.296	30.7 (1.4)	29.1 (1.4)	25.9 (1.7)	27.6 (1.9)	27.4 (1.7)	34.2 (1.7)	20.8 (1.9)
2.499	26.1 (1.2)	24.2 (1.2)	22.7 (1.5)	23.3 (1.6)	28.6 (1.7)	23.4 (1.5)	16.5 (1.8)
2.703	20.8 (1.2)	21.3 (1.2)	20.4 (1.2)	20.4 (1.2)	20.4 (1.2)	20.4 (1.2)	15.0 (1.8)
2.898	18.6 (1.2)	17.8 (1.2)	14.7 (1.6)	19.6 (1.7)	20.6 (1.5)	33.6 (1.3)	35.9 (1.4)
3.097	18.0 (1.3)	14.5 (1.4)	13.0 (1.7)	17.0 (1.9)	19.2 (1.4)	35.4 (1.2)	19.2 (0.9)
3.296	14.6 (1.1)	12.5 (1.2)	13.5 (1.5)	13.0 (1.5)	25.7 (1.3)	27.0 (0.9)	7.6 (0.5)
3.501	9.9 (1.3)	10.3 (1.3)	12.8 (1.7)	16.2 (1.6)	31.7 (1.3)	15.4 (0.6)	2.5 (0.3)
3.707	7.4 (1.1)	0.2 (1.1)	11.9 (1.4)	27.7 (1.3)	28.5 (0.9)	7.0 (0.4)	
3.906	7.0 (1.0)	7.4 (1.1)	15.6 (1.3)	36.4 (1.2)	19.4 (0.7)	2.9 (0.2)	
4.153	7.1 (1.0)	9.2 (1.1)	15.6 (1.3)	36.4 (1.2)	19.4 (0.7)	2.9 (0.2)	
4.454	6.8 (0.9)	16.7 (0.9)	41.1 (1.0)	13.2 (0.7)	2.7 (0.2)	1.1 (0.2)	
4.752	7.4 (1.0)	27.4 (0.9)	32.7 (0.8)	6.9 (0.4)	0.9 (0.1)		
5.042	6.7 (0.9)	34.3 (0.9)	17.0 (0.5)	2.4 (0.2)			
5.346	13.3 (0.9)	32.3 (0.7)	6.4 (0.3)	1.0 (0.2)			
5.645	22.1 (0.9)	22.6 (0.5)	2.2 (0.2)				
5.938	28.8 (0.8)	12.5 (0.4)	1.0 (0.2)				
6.254	27.8 (0.9)	5.2 (0.2)					
6.556	20.7 (0.7)	2.0 (0.2)					
6.839	13.1 (0.5)	0.9 (0.1)					
7.235	5.6 (0.2)						
7.736	1.6 (0.2)						
8.236	0.5 (0.1)						

INCIDENT ENERGY	2.50- 2.75	2.24- 2.50	2.00- 2.24	1.75- 2.00	1.50- 1.75	1.25- 1.50	1.00- 1.25
SECONDARY ENERGY (MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)
0.907	111.6 (5.4)	105.7 (5.2)	110.0 (5.7)	173.2 (7.5)	73.6 (4.2)	7.1 (1.5)	213.3 (17.5)
1.007	105.2 (4.8)	92.1 (4.4)	134.9 (6.1)	132.3 (5.9)	36.4 (2.5)	34.0 (2.9)	340.3 (25.9)
1.107	50.5 (4.2)	67.3 (3.9)	131.3 (5.5)	81.4 (3.9)	17.3 (1.7)	100.4 (7.4)	314.7 (23.1)
1.205	78.0 (3.5)	85.7 (3.8)	105.7 (4.2)	41.3 (2.4)	25.4 (1.8)	181.5 (11.6)	199.4 (14.5)
1.310	73.7 (3.3)	100.1 (3.6)	72.8 (2.5)	18.7 (1.6)	69.6 (3.6)	203.9 (12.2)	89.9 (6.5)
1.412	77.8 (3.2)	99.9 (3.1)	44.4 (2.0)	21.2 (1.6)	125.9 (6.0)	157.1 (9.1)	31.7 (2.3)
1.511	80.4 (3.0)	69.5 (2.5)	24.5 (1.6)	47.2 (2.3)	150.7 (6.9)	92.1 (5.3)	9.6 (0.7)
1.610	74.4 (2.9)	46.6 (2.0)	14.9 (1.4)	84.8 (3.4)	132.4 (6.0)	43.2 (2.5)	2.7 (0.1)
1.709	60.5 (2.4)	27.7 (1.6)	31.7 (1.6)	111.2 (4.2)	91.2 (4.1)	16.6 (1.0)	0.4 (0.1)
1.813	42.9 (2.2)	18.4 (1.5)	58.9 (2.3)	110.1 (3.9)	50.3 (2.3)	5.0 (0.3)	
1.932	26.0 (2.0)	26.9 (1.6)	84.5 (3.1)	79.4 (2.8)	20.5 (0.9)	1.2 (0.1)	
2.100	21.7 (1.5)	57.9 (2.2)	76.6 (2.7)	33.5 (1.2)	4.5 (0.3)	0.6 (0.1)	
2.296	44.5 (1.9)	65.5 (2.4)	16.1 (1.3)	8.7 (0.4)	1.1 (0.1)		
2.499	44.8 (2.0)	37.5 (1.4)	11.2 (0.5)	2.7 (0.2)			
2.703	34.7 (1.3)	13.5 (0.6)	3.6 (0.2)	1.6 (0.1)			
2.898	15.1 (0.7)	4.7 (0.3)	1.7 (0.1)				
3.097	5.1 (0.1)						
3.296	1.6 (0.2)						

TABLE 2. NR(N,IV) DIFFERENTIAL CROSS SECTIONS AT 129 DEGREES. THE NUMBERS IN PARENTHESES ARE STATISTICAL UNCERTAINTIES (IN THE SAME UNITS AS THE DATA). THE OVERALL UNCERTAINTY IN NORMALIZATION IS ABOUT 10%.

INCIDENT ENERGY	17.46-19.92	15.01-17.46	12.52-15.01	9.98-12.52	9.02- 9.94	7.99- 9.02	6.99- 7.99
GAMMA RAY ENERGY (MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)	CROSS SECTION (MB/SR/MEV)
0.760	205.0 (37.5)	185.3 (29.7)	178.0 (24.2)	244.8 (25.0)	390.9 (40.7)	434.1 (43.0)	445.1 (43.9)
0.800	223.7 (13.9)	199.6 (10.7)	190.6 (4.5)	243.5 (4.3)	397.0 (13.7)	436.8 (14.1)	429.4 (13.2)
0.840	240.8 (8.7)	220.1 (6.5)	204.3 (4.8)	252.5 (4.2)	393.6 (7.2)	449.6 (7.0)	435.0 (6.5)
0.880	263.4 (7.9)	241.6 (5.8)	221.2 (4.3)	267.7 (3.7)	425.5 (6.2)	487.9 (6.0)	469.9 (5.6)
0.920	280.2 (7.4)	251.3 (5.9)	233.7 (4.3)	304.5 (3.7)	455.5 (6.3)	520.5 (6.0)	504.6 (5.6)
0.960	278.1 (7.6)	243.0 (5.6)	232.9 (4.1)	293.8 (3.5)	445.4 (6.1)	508.9 (5.8)	499.9 (5.4)
1.000	254.5 (7.5)	223.7 (5.5)	216.6 (4.0)	258.7 (3.3)	391.4 (5.7)	450.2 (5.4)	440.1 (5.0)
1.040	234.5 (7.2)	203.4 (5.3)	189.9 (3.8)	213.9 (3.1)	320.1 (5.4)	370.6 (5.1)	356.0 (4.7)
1.080	209.2 (7.2)	184.1 (5.2)	161.4 (3.7)	172.5 (3.0)	256.3 (5.2)	294.8 (4.9)	281.2 (4.5)
1.125	179.6 (6.9)	160.7 (5.0)	135.1 (3.5)	134.8 (2.7)	209.0 (5.0)	223.0 (4.8)	225.4 (4.4)
1.175	147.0 (6.7)	136.5 (4.9)	115.5 (3.5)	116.9 (2.8)	167.7 (4.9)	183.2 (4.6)	188.9 (4.2)
1.225	125.4 (6.9)	121.6 (5.0)	102.7 (3.5)	102.6 (2.9)	146.1 (5.0)	160.9 (4.7)	166.2 (4.4)
1.275	121.9 (6.2)	114.6 (5.0)	94.3 (3.6)	91.0 (2.9)	132.4 (5.0)	152.3 (4.8)	152.5 (4.5)
1.325	131.2 (6.4)	111.8 (4.6)	90.8 (3.1)	84.2 (2.6)	126.2 (4.6)	140.8 (4.4)	148.2 (4.0)
1.375	138.3 (6.2)	111.1 (4.4)	90.7 (3.1)	82.8 (2.5)	127.4 (4.3)	148.8 (4.1)	137.1 (3.7)
1.425	128.9 (5.5)	111.0 (4.6)	91.5 (3.2)	83.4 (2.6)	131.7 (4.5)	147.4 (4.3)	137.8 (3.9)
1.475	136.3 (6.4)	111.4 (4.6)	91.2 (3.2)	82.4 (2.5)	133.4 (4.5)	144.4 (4.1)	132.3 (3.4)
1.525	134.5 (6.1)	108.3 (4.3)	98.5 (3.0)	79.1 (2.4)	128.9 (4.3)	134.0 (4.0)	134.7 (3.6)
1.575	130.0 (5.3)	107.7 (4.1)	83.3 (2.8)	74.5 (2.3)	119.5 (4.1)	131.6 (3.8)	134.6 (3.5)
1.640	118.4 (5.7)	92.4 (4.0)	74.0 (2.8)	68.5 (2.3)	107.4 (4.0)	121.9 (3.8)	126.2 (3.4)
1.720	106.7 (5.0)	74.9 (3.9)	69.8 (2.7)	62.9 (2.2)	98.2 (3.9)	110.6 (3.6)	109.4 (3.3)
1.800	107.0 (5.4)	75.5 (4.0)	61.2 (2.7)	59.9 (2.2)	91.7 (3.4)	100.8 (3.6)	99.4 (3.2)
1.880	111.4 (5.5)	85.2 (4.0)	61.1 (2.6)	58.2 (2.2)	97.4 (3.4)	106.0 (3.6)	99.4 (3.2)
1.960	116.7 (5.7)	97.5 (4.0)	67.1 (2.7)	55.0 (2.3)	84.6 (3.3)	96.2 (3.7)	91.3 (3.3)
2.040	122.6 (5.3)	102.3 (3.8)	72.9 (2.5)	50.6 (2.1)	79.8 (3.7)	96.1 (3.4)	86.3 (3.1)
2.120	122.6 (5.3)	99.2 (3.7)	72.5 (2.5)	47.9 (2.1)	73.0 (3.6)	92.4 (3.4)	84.4 (3.0)
2.200	113.3 (5.0)	92.4 (3.5)	64.6 (2.4)	47.1 (2.0)	70.5 (3.5)	86.4 (3.3)	84.3 (2.9)
2.280	60.4 (4.4)	83.7 (3.4)	51.9 (2.3)	45.9 (1.9)	72.9 (3.4)	79.9 (3.2)	81.6 (2.8)
2.360	87.6 (4.7)	73.6 (3.3)	44.7 (2.2)	43.4 (1.9)	74.0 (3.4)	73.5 (3.1)	74.9 (2.8)
2.450	76.1 (4.4)	61.9 (3.0)	37.1 (2.0)	39.4 (1.9)	69.6 (3.3)	67.5 (3.0)	65.6 (2.7)
2.550	68.3 (4.1)	51.1 (2.8)	31.5 (1.9)	35.6 (1.7)	61.4 (3.1)	64.2 (2.9)	58.8 (2.5)
2.650	57.4 (4.1)	43.2 (2.4)	28.4 (1.9)	32.0 (1.7)	56.9 (3.1)	64.9 (2.9)	56.6 (2.5)
2.750	48.5 (3.9)	37.9 (2.6)	25.9 (1.8)	29.6 (1.7)	55.1 (3.0)	64.8 (2.8)	56.1 (2.4)
2.850	44.3 (3.8)	35.2 (2.5)	22.4 (1.7)	27.7 (1.7)	52.5 (3.0)	60.1 (2.7)	55.3 (2.3)
2.950	43.5 (3.7)	34.3 (2.5)	18.6 (1.7)	25.7 (1.7)	48.8 (2.9)	53.7 (2.7)	51.3 (2.3)
3.050	43.8 (3.7)	32.6 (2.4)	16.4 (1.7)	24.1 (1.6)	46.9 (2.9)	51.1 (2.6)	51.3 (2.3)
3.150	43.6 (3.6)	28.5 (2.4)	16.0 (1.6)	23.6 (1.7)	46.6 (2.9)	52.2 (2.7)	48.0 (2.3)
3.250	42.6 (3.6)	23.6 (2.4)	16.1 (1.6)	21.7 (1.7)	44.9 (3.0)	52.7 (2.7)	44.2 (2.2)
3.350	40.2 (3.6)	20.5 (2.3)	15.5 (1.6)	23.2 (1.7)	40.3 (2.9)	49.0 (2.7)	40.2 (2.2)
3.450	36.5 (3.5)	19.6 (2.3)	14.2 (1.6)	22.1 (1.7)	34.8 (2.9)	42.5 (2.6)	37.1 (2.2)
3.550	32.2 (3.5)	19.2 (2.3)	12.5 (1.6)	21.1 (1.7)	31.2 (2.9)	37.2 (2.6)	35.4 (2.2)
3.655	27.8 (3.5)	18.1 (2.2)	11.0 (1.6)	20.2 (1.7)	30.5 (2.9)	35.6 (2.6)	34.1 (2.2)
3.795	24.7 (3.4)	16.4 (2.2)	9.8 (1.6)	18.2 (1.7)	31.4 (3.0)	37.0 (2.6)	32.1 (2.1)
3.925	23.1 (3.4)	15.6 (2.2)	8.7 (1.6)	15.5 (1.8)	21.3 (3.0)	37.0 (2.6)	25.8 (2.1)
4.055	21.7 (3.5)	14.7 (2.3)	7.3 (1.7)	14.3 (1.9)	30.4 (3.1)	34.2 (2.6)	24.1 (2.1)
4.185	19.8 (3.5)	12.8 (2.3)	5.8 (1.7)	15.6 (1.3)	29.2 (3.2)	30.3 (2.7)	26.3 (2.2)
4.315	18.0 (3.7)	10.2 (2.4)	5.0 (1.8)	17.7 (2.0)	27.6 (3.4)	26.7 (2.9)	23.6 (2.2)
4.445	15.9 (3.9)	8.0 (2.5)	5.6 (1.9)	18.4 (2.2)	25.1 (3.7)	23.5 (3.1)	20.5 (2.3)
4.575	13.5 (3.8)	6.9 (2.4)	7.0 (1.9)	17.1 (2.3)	22.3 (3.8)	21.5 (3.2)	17.6 (2.3)
4.705	11.4 (3.9)	6.5 (2.4)	7.8 (1.9)	16.2 (2.1)	21.0 (3.5)	21.4 (3.2)	15.2 (2.3)
4.835	10.2 (3.9)	6.6 (2.4)	7.0 (1.9)	10.7 (2.3)	21.9 (4.0)	22.4 (3.2)	13.3 (2.3)
4.985	10.5 (3.3)	6.7 (2.4)	4.4 (1.9)	7.7 (2.5)	23.5 (4.2)	22.7 (3.3)	12.3 (2.3)
5.155	11.3 (3.9)	6.7 (2.3)	1.6 (1.8)	6.7 (2.5)	22.2 (4.3)	20.1 (3.4)	12.6 (2.3)
5.325	11.2 (3.9)	6.4 (2.2)	0.9 (1.8)	7.4 (2.6)	16.4 (4.3)	15.5 (3.4)	12.7 (2.2)
5.495	10.4 (3.7)	5.7 (2.1)	1.8 (1.8)	8.1 (2.6)	10.1 (4.3)	11.3 (3.3)	11.2 (2.0)
5.665	9.3 (3.4)	4.5 (1.9)	1.2 (1.7)	7.6 (2.6)	7.6 (4.1)	4.3 (3.2)	4.6 (1.8)
5.835	7.7 (2.3)	2.2 (1.6)	3.0 (1.4)	6.1 (2.8)	0.7 (4.0)	8.1 (2.9)	6.7 (1.6)
6.005	5.8 (2.9)	2.2 (1.4)	3.5 (1.4)	5.0 (2.2)	13.1 (3.7)	7.5 (2.6)	4.5 (1.4)
6.175	3.9 (2.6)	1.4 (1.3)	2.5 (1.2)	4.4 (2.0)	14.4 (3.3)	9.4 (2.2)	3.5 (1.2)
6.345	2.9 (2.3)	0.9 (1.1)	1.4 (1.1)	4.4 (1.4)	12.5 (2.4)	8.4 (2.0)	2.4 (0.4)
6.515	2.6 (2.0)	0.7 (0.9)	0.6 (0.9)	4.4 (1.5)	0.8 (2.4)	6.7 (1.6)	1.7 (0.8)
6.700	2.8 (1.6)	0.3 (0.7)	0.2 (0.8)	4.1 (1.8)	5.0 (1.7)	4.6 (1.2)	1.5 (0.6)
6.900	2.0 (1.8)	0.4 (0.6)	0.4 (0.9)	3.9 (1.8)	2.0 (1.5)	2.9 (0.9)	1.2 (0.9)
7.100	2.5 (1.0)	0.3 (0.5)	0.7 (0.6)	3.3 (0.7)	3.1 (1.2)	1.8 (0.7)	1.0 (0.3)
7.300	1.9 (0.9)	0.3 (0.4)	0.4 (0.5)	2.8 (0.6)	3.9 (1.0)	1.3 (0.6)	0.5 (0.2)
7.500	1.4 (0.7)	0.3 (0.3)	1.0 (0.4)	2.3 (0.5)	4.1 (0.8)	1.1 (0.5)	0.6 (0.2)
7.700	0.9 (0.5)	0.3 (0.2)	0.8 (0.4)	1.7 (0.4)	3.5 (0.6)	1.0 (0.4)	0.4 (0.1)
7.900	0.6 (0.4)	0.4 (0.2)	0.5 (0.3)	1.1 (0.3)	2.4 (0.5)	0.9 (0.2)	0.2 (0.1)
8.100	0.4 (0.3)	0.4 (0.2)	0.4 (0.3)	0.7 (0.2)	1.4 (0.4)	0.7 (0.2)	0.1 (0.1)
8.300	0.3 (0.2)	0.4 (0.2)	0.2 (0.2)	0.4 (0.2)	0.6 (0.3)	0.5 (0.1)	0.0 (0.1)
8.500	0.2 (0.2)	0.4 (0.1)	0.2 (0.2)	0.3 (0.2)	0.3 (0.2)	0.3 (0.1)	0.0 (0.1)
8.725	0.2 (0.2)	0.3 (0.1)	0.2 (0.1)	0.2 (0.1)	0.1 (0.2)	0.2 (0.1)	0.0 (0.1)
8.975	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.0 (0.1)
9.225	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.0 (0.1)	0.0 (0.1)
9.475	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.0 (0.1)	0.0 (0.1)
9.725	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	-0.0 (0.1)
9.975	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	-0.0 (0.1)

INCIDENT ENERGY*	5.99- 6.03	5.01- 5.99	4.50- 5.01	3.99- 4.50	3.51- 3.99	3.00- 3.51	2.75- 3.00
GAMMA RAY ENERGY (MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)	CROSS SECTION (NB/SR/MEV)
0.760	454.4 (43.4)	452.5 (40.2)	459.1 (39.3)	437.8 (36.9)	455.7 (34.2)	505.4 (36.9)	444.6 (39.5)
0.800	448.6 (12.9)	436.5 (12.6)	438.1 (11.1)	426.7 (12.2)	435.0 (11.4)	446.9 (11.8)	435.2 (13.5)
0.840	453.3 (6.1)	432.6 (5.6)	431.9 (6.8)	420.9 (6.2)	422.0 (5.9)	421.5 (6.1)	416.3 (6.5)
0.880	450.5 (5.2)	460.5 (5.7)	455.7 (5.6)	437.5 (5.3)	443.5 (5.0)	432.1 (4.8)	425.1 (6.5)
0.920	512.1 (5.1)	407.6 (4.9)	479.6 (5.8)	442.6 (5.3)	440.3 (5.0)	445.4 (4.9)	438.4 (6.4)
0.960	500.7 (5.0)	473.8 (4.5)	462.7 (5.5)	423.9 (5.0)	420.8 (4.7)	425.6 (4.5)	418.8 (6.1)
1.000	442.6 (4.7)	416.1 (4.2)	401.5 (5.1)	375.3 (4.6)	368.0 (4.4)	369.6 (4.1)	359.8 (5.6)
1.040	361.9 (4.4)	341.6 (3.9)	325.1 (4.8)	311.7 (4.4)	310.4 (4.1)	299.3 (3.8)	284.8 (5.1)
1.080	226.5 (4.2)	274.3 (3.7)	259.3 (4.5)	255.0 (4.1)	247.1 (3.9)	234.9 (3.5)	218.6 (4.1)
1.125	224.8 (4.0)	217.4 (3.4)	207.6 (4.1)	202.0 (3.9)	191.0 (3.6)	180.9 (3.2)	166.0 (4.3)
1.175	184.1 (3.9)	177.4 (3.4)	172.1 (4.1)	162.7 (3.7)	155.6 (3.4)	148.5 (3.0)	125.2 (3.7)
1.225	161.2 (4.0)	154.1 (3.5)	150.1 (4.1)	140.1 (3.7)	133.5 (3.4)	124.4 (3.0)	104.1 (3.8)
1.275	148.8 (3.9)	141.4 (3.4)	133.3 (4.1)	124.3 (3.7)	117.9 (3.4)	110.4 (3.0)	86.2 (3.6)
1.325	143.1 (3.7)	135.0 (3.2)	123.2 (3.8)	120.8 (3.4)	110.1 (3.1)	100.9 (2.7)	78.4 (3.3)
1.375	143.2 (3.4)	133.0 (2.9)	121.4 (3.5)	116.7 (3.1)	105.4 (2.8)	97.2 (2.5)	79.9 (3.2)
1.425	145.2 (3.6)	133.0 (3.1)	125.4 (3.6)	115.3 (3.2)	105.0 (2.5)	97.2 (2.5)	84.7 (3.2)
1.475	145.4 (3.4)	132.0 (3.0)	128.3 (3.7)	114.3 (3.2)	108.0 (2.3)	96.0 (2.5)	86.1 (3.2)
1.525	141.6 (3.3)	128.4 (2.8)	125.1 (3.4)	110.8 (3.0)	106.7 (2.7)	91.4 (2.3)	81.8 (3.0)
1.575	133.9 (3.1)	122.3 (2.7)	115.5 (3.2)	103.1 (2.8)	100.4 (2.5)	84.4 (2.3)	73.2 (2.7)
1.625	119.5 (3.1)	112.9 (2.5)	98.8 (3.0)	90.9 (2.7)	86.7 (2.4)	74.1 (2.0)	60.8 (2.9)
1.720	108.1 (2.3)	102.6 (2.5)	82.4 (2.9)	81.1 (2.6)	74.4 (2.2)	62.4 (1.5)	49.1 (2.1)
1.800	100.9 (2.3)	96.1 (2.5)	75.2 (2.9)	73.7 (2.5)	64.2 (2.2)	55.7 (1.9)	40.4 (2.1)
1.880	98.3 (2.4)	91.0 (2.4)	73.9 (2.8)	69.4 (2.4)	62.4 (2.1)	52.5 (1.7)	35.5 (2.0)
1.960	53.2 (2.9)	85.0 (2.5)	73.3 (2.8)	65.9 (2.4)	60.4 (2.1)	47.4 (1.6)	30.5 (1.9)
2.040	85.7 (2.7)	78.1 (2.3)	71.2 (2.6)	61.6 (2.2)	55.6 (1.9)	39.3 (1.5)	24.8 (1.7)
2.120	79.7 (2.6)	71.0 (2.2)	67.0 (2.6)	56.1 (2.2)	47.4 (1.9)	32.7 (1.4)	19.1 (1.5)
2.200	78.9 (2.6)	66.3 (2.1)	61.0 (2.4)	50.0 (2.0)	38.7 (1.6)	26.6 (1.3)	15.2 (1.5)
2.280	78.0 (2.5)	59.5 (2.0)	54.4 (2.3)	44.2 (1.9)	34.6 (1.5)	21.7 (1.2)	12.5 (1.2)
2.360	72.3 (2.4)	57.5 (2.0)	49.1 (2.2)	39.7 (1.9)	30.7 (1.5)	17.5 (1.1)	11.2 (1.1)
2.450	62.1 (2.3)	56.3 (1.9)	44.5 (2.1)	34.4 (1.7)	25.9 (1.4)	14.5 (1.0)	10.4 (1.1)
2.550	55.4 (2.1)	52.8 (1.7)	40.4 (1.9)	34.4 (1.6)	21.2 (1.2)	12.5 (0.9)	8.9 (0.9)
2.650	56.0 (2.1)	47.6 (1.7)	38.0 (1.9)	33.1 (1.5)	19.7 (1.2)	10.9 (0.8)	6.7 (0.8)
2.750	56.9 (2.0)	43.5 (1.6)	37.1 (1.8)	30.8 (1.4)	17.2 (1.1)	9.7 (0.8)	4.7 (0.7)
2.850	52.3 (1.9)	40.7 (1.5)	34.6 (1.7)	26.4 (1.3)	15.0 (1.0)	9.0 (0.7)	3.1 (0.6)
2.950	45.3 (1.9)	37.7 (1.4)	29.5 (1.6)	22.3 (1.2)	13.1 (0.9)	8.4 (0.6)	2.7 (0.5)
3.050	40.8 (1.8)	35.4 (1.3)	24.8 (1.5)	19.1 (1.2)	11.1 (0.9)	7.1 (0.6)	1.3 (0.5)
3.150	39.2 (1.8)	34.0 (1.4)	22.2 (1.5)	16.8 (1.1)	10.5 (0.9)	5.6 (0.5)	0.6 (0.4)
3.250	37.6 (1.8)	32.3 (1.4)	20.7 (1.4)	14.8 (1.1)	10.1 (0.9)	3.8 (0.5)	0.3 (0.4)
3.350	34.7 (1.8)	29.3 (1.4)	14.7 (1.3)	12.5 (1.0)	9.1 (0.7)	2.3 (0.4)	0.4 (0.4)
3.450	31.6 (1.7)	25.6 (1.3)	16.5 (1.3)	10.5 (0.9)	7.4 (0.6)	1.3 (0.3)	0.7 (0.4)
3.550	30.0 (1.7)	22.6 (1.2)	14.9 (1.2)	9.1 (0.9)	5.5 (0.6)	0.9 (0.3)	0.4 (0.4)
3.665	25.4 (1.7)	20.8 (1.2)	14.1 (1.1)	9.0 (0.9)	3.4 (0.5)	0.3 (0.3)	0.4 (0.4)
3.761	26.1 (1.6)	19.3 (1.1)	12.7 (1.0)	7.1 (0.7)	3.7 (0.4)	0.6 (0.3)	0.7 (0.4)
3.925	25.5 (1.6)	17.3 (1.0)	11.3 (0.9)	5.6 (0.6)	0.4 (0.4)	0.4 (0.3)	0.5 (0.4)
4.055	22.5 (1.6)	14.9 (1.0)	8.4 (0.8)	4.2 (0.5)	0.5 (0.3)	0.1 (0.3)	0.4 (0.4)
4.185	20.9 (1.5)	12.4 (1.0)	5.6 (0.7)	2.9 (0.5)	0.5 (0.3)	-0.0 (0.3)	0.4 (0.4)
4.315	19.0 (1.5)	10.6 (0.9)	3.7 (0.7)	1.7 (0.4)	0.6 (0.3)	0.0 (0.3)	0.4 (0.4)
4.445	16.6 (1.6)	8.3 (0.9)	2.7 (0.6)	1.0 (0.3)	0.7 (0.3)	0.2 (0.3)	0.4 (0.4)
4.575	13.5 (1.5)	6.2 (0.9)	2.1 (0.5)	0.5 (0.3)	0.6 (0.3)	0.4 (0.3)	0.3 (0.4)
4.705	10.8 (1.4)	4.7 (0.7)	1.6 (0.4)	0.2 (0.3)	0.4 (0.3)	0.3 (0.3)	0.3 (0.4)
4.835	8.6 (1.3)	3.7 (0.6)	1.2 (0.3)	0.1 (0.3)	0.2 (0.3)	0.3 (0.3)	0.3 (0.4)
4.965	7.7 (1.2)	3.0 (0.5)	0.7 (0.3)	-0.0 (0.2)	0.1 (0.2)	0.5 (0.3)	0.3 (0.3)
5.155	7.0 (1.1)	2.2 (0.4)	0.3 (0.2)	-0.0 (0.2)	0.0 (0.2)	0.4 (0.3)	0.3 (0.3)
5.325	5.9 (1.0)	1.3 (0.4)	0.0 (0.2)	0.1 (0.2)	0.1 (0.2)	0.3 (0.3)	0.4 (0.3)
5.495	4.3 (0.7)	0.6 (0.3)	-0.1 (0.2)	0.2 (0.2)	0.2 (0.2)	0.2 (0.2)	0.3 (0.2)
5.665	2.6 (0.8)	0.2 (0.2)	-0.0 (0.1)	0.3 (0.2)	0.3 (0.2)	0.1 (0.2)	0.3 (0.2)
5.835	1.4 (0.6)	-0.0 (0.2)	0.0 (0.1)	0.3 (0.2)	0.3 (0.2)	0.1 (0.2)	0.3 (0.2)
6.005	0.8 (0.6)	-0.0 (0.2)	0.0 (0.1)	0.2 (0.2)	0.2 (0.2)	0.1 (0.2)	0.0 (0.2)
6.175	0.0 (0.5)	0.0 (0.1)	0.2 (0.1)	0.1 (0.2)	0.2 (0.2)	0.2 (0.2)	0.0 (0.2)
6.345	0.9 (0.4)	0.1 (0.1)	0.2 (0.1)	0.0 (0.2)	0.1 (0.1)	0.2 (0.2)	0.0 (0.1)
6.515	0.3 (0.3)	0.2 (0.1)	0.3 (0.1)	0.0 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)
6.700	0.7 (0.2)	0.2 (0.1)	0.3 (0.1)	0.0 (0.1)	0.0 (0.1)	0.2 (0.1)	0.1 (0.1)
6.900	0.5 (0.2)	0.2 (0.1)	0.2 (0.1)	0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)
7.100	0.3 (0.1)	0.2 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)
7.300	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)
7.500	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)
7.700	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)
7.900	0.0 (0.1)	0.1 (0.1)	-0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)
8.100	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)
8.300	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
8.500	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
8.725	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
8.975	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.225	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.475	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	-0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.725	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	-0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.975	0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)

TABLE 2. $W(\pi, \pi)$ DIFFERENTIAL CROSS SECTIONS AT 120° DEGREES. THE NUMBERS IN PARENTHESES ARE STATISTICAL UNCERTAINTIES (IN THE SAME UNITS AS THE DATA). THE OVERALL UNCERTAINTY IN NORMALIZATION IS ABOUT 10%.

INCIDENT ENERGY GAMMA RAY ENERGY (MEV)	2.50- 2.75 CROSS SECTION (MB/SR/MEV)	2.24- 2.50 CROSS SECTION (MB/SR/MEV)	2.00- 2.24 CROSS SECTION (MB/SR/MEV)	1.75- 2.00 CROSS SECTION (MB/SR/MEV)	1.50- 1.75 CROSS SECTION (MB/SR/MEV)	1.25- 1.50 CROSS SECTION (MB/SR/MEV)	1.00- 1.25 CROSS SECTION (MB/SR/MEV)
0.760	466.5 (30.2)	449.5 (32.0)	437.9 (28.4)	187.4 (25.0)	161.4 (22.0)	128.6 (20.6)	285.5 (16.8)
0.900	428.6 (11.6)	444.0 (10.3)	390.6 (9.2)	352.3 (7.9)	316.1 (7.0)	295.1 (6.5)	289.4 (5.4)
0.940	403.6 (6.2)	391.0 (4.9)	374.5 (6.2)	328.7 (3.6)	286.8 (7.2)	272.8 (2.9)	216.2 (2.5)
0.980	414.6 (5.2)	403.4 (4.1)	383.7 (3.5)	140.2 (2.9)	294.0 (2.5)	261.1 (2.4)	211.6 (2.0)
0.920	430.4 (5.2)	419.5 (4.1)	397.1 (3.4)	357.9 (2.9)	311.0 (2.5)	299.0 (2.3)	215.3 (2.0)
0.960	409.5 (4.8)	399.8 (3.8)	376.3 (3.2)	142.3 (2.6)	306.3 (2.3)	291.4 (2.2)	205.8 (1.3)
1.000	347.3 (4.0)	338.9 (3.4)	318.9 (2.8)	285.7 (2.3)	262.7 (2.1)	251.2 (1.5)	173.4 (1.6)
1.040	267.5 (4.0)	261.1 (3.1)	236.4 (2.5)	210.4 (2.0)	199.3 (1.9)	192.9 (1.6)	129.2 (1.3)
1.080	200.7 (3.7)	192.0 (2.7)	165.3 (2.2)	141.2 (1.8)	135.0 (1.6)	132.0 (1.3)	85.0 (1.1)
1.125	146.5 (3.3)	137.8 (2.4)	107.0 (1.9)	86.8 (1.5)	87.1 (1.1)	81.2 (1.1)	68.8 (0.8)
1.175	113.0 (3.0)	103.0 (2.2)	69.0 (1.7)	56.3 (1.4)	56.1 (1.1)	46.6 (0.8)	25.0 (0.7)
1.225	96.9 (2.9)	82.6 (2.1)	53.8 (1.6)	46.4 (1.3)	44.7 (1.1)	29.4 (0.7)	13.1 (0.6)
1.275	87.2 (2.8)	67.9 (2.1)	47.6 (1.6)	44.6 (1.1)	41.3 (1.0)	20.6 (0.6)	7.5 (0.5)
1.325	80.0 (2.6)	58.5 (1.9)	44.9 (1.5)	44.7 (1.2)	39.0 (0.9)	15.1 (0.6)	4.7 (0.5)
1.375	75.5 (2.4)	55.7 (1.7)	44.4 (1.4)	45.7 (1.1)	35.4 (0.9)	11.1 (0.5)	3.4 (0.5)
1.425	73.0 (2.4)	57.5 (1.7)	46.2 (1.4)	46.9 (1.1)	32.5 (0.8)	8.2 (0.5)	3.1 (0.5)
1.475	70.3 (2.3)	59.1 (1.8)	48.3 (1.4)	47.0 (1.1)	29.5 (0.7)	6.2 (0.5)	3.3 (0.5)
1.525	65.8 (2.2)	57.1 (1.6)	48.1 (1.3)	44.7 (1.0)	23.6 (0.7)	4.7 (0.4)	1.6 (0.5)
1.575	59.0 (2.0)	51.1 (1.5)	44.7 (1.2)	40.0 (0.9)	18.0 (0.6)	3.4 (0.4)	3.6 (0.5)
1.600	48.4 (1.9)	41.0 (1.3)	37.8 (1.1)	31.4 (0.8)	11.3 (0.5)	2.3 (0.4)	3.2 (0.4)
1.720	36.2 (1.7)	31.5 (1.2)	30.1 (1.0)	21.2 (0.6)	5.8 (0.4)	1.8 (0.4)	3.2 (0.4)
1.800	30.3 (1.5)	28.4 (1.2)	25.0 (0.9)	12.5 (0.5)	3.2 (0.4)	2.1 (0.4)	3.4 (0.4)
1.880	30.1 (1.5)	28.7 (1.1)	22.0 (0.9)	6.8 (0.4)	2.4 (0.3)	2.6 (0.4)	3.0 (0.4)
1.960	29.7 (1.4)	26.8 (1.0)	17.2 (0.7)	3.7 (0.4)	2.4 (0.3)	2.0 (0.4)	2.2 (0.4)
2.000	25.7 (1.3)	21.5 (0.9)	11.6 (0.6)	2.4 (0.3)	2.4 (0.3)	2.6 (0.4)	1.9 (0.4)
2.120	19.7 (1.1)	15.1 (0.7)	6.7 (0.5)	1.0 (0.3)	1.4 (0.3)	2.1 (0.4)	2.1 (0.4)
2.200	14.0 (0.9)	9.8 (0.6)	1.8 (0.4)	1.1 (0.3)	1.3 (0.3)	1.8 (0.3)	2.1 (0.4)
2.280	4.7 (0.8)	0.0 (0.7)	2.3 (0.4)	1.0 (0.3)	0.4 (0.3)	1.7 (0.3)	2.1 (0.4)
2.360	6.8 (0.7)	3.6 (0.4)	1.5 (0.3)	1.2 (0.3)	1.0 (0.3)	1.6 (0.3)	1.9 (0.4)
2.450	4.7 (0.6)	2.1 (0.4)	1.0 (0.3)	1.5 (0.3)	1.1 (0.3)	1.6 (0.3)	1.6 (0.4)
2.550	3.3 (0.5)	1.1 (0.4)	0.7 (0.3)	1.3 (0.3)	1.0 (0.3)	1.1 (0.3)	1.4 (0.3)
2.650	2.2 (0.5)	0.4 (0.3)	0.8 (0.3)	0.8 (0.3)	0.8 (0.3)	0.8 (0.3)	1.6 (0.3)
2.750	1.4 (0.4)	0.4 (0.3)	0.9 (0.3)	0.4 (0.3)	0.9 (0.3)	0.6 (0.3)	1.9 (0.3)
2.850	1.1 (0.4)	0.4 (0.3)	0.8 (0.3)	0.5 (0.3)	1.0 (0.3)	0.7 (0.3)	1.9 (0.3)
2.950	1.1 (0.4)	0.6 (0.3)	0.6 (0.3)	0.8 (0.3)	1.0 (0.3)	0.9 (0.3)	1.5 (0.3)
3.050	1.0 (0.4)	0.7 (0.3)	0.5 (0.3)	1.1 (0.3)	0.8 (0.3)	1.1 (0.3)	1.0 (0.3)
3.150	0.6 (0.3)	0.7 (0.3)	0.5 (0.3)	1.2 (0.3)	0.7 (0.3)	1.2 (0.3)	0.8 (0.3)
3.250	0.2 (0.3)	0.6 (0.3)	0.5 (0.3)	1.0 (0.3)	0.6 (0.3)	1.1 (0.3)	0.9 (0.3)
3.350	-0.2 (0.3)	0.6 (0.3)	0.5 (0.3)	0.8 (0.3)	0.8 (0.3)	1.5 (0.3)	1.1 (0.3)
3.450	-0.2 (0.3)	0.7 (0.3)	0.7 (0.3)	0.6 (0.2)	1.0 (0.3)	1.5 (0.3)	1.4 (0.3)
3.550	0.1 (0.3)	0.8 (0.3)	0.9 (0.3)	0.5 (0.2)	1.0 (0.3)	1.3 (0.3)	1.6 (0.3)
3.650	0.5 (0.3)	0.8 (0.3)	1.0 (0.3)	0.6 (0.2)	0.9 (0.3)	0.9 (0.3)	1.7 (0.3)
3.795	0.7 (0.3)	0.7 (0.3)	0.8 (0.3)	0.6 (0.3)	0.5 (0.3)	0.6 (0.3)	1.7 (0.3)
3.925	0.7 (0.4)	0.6 (0.3)	0.4 (0.3)	0.6 (0.3)	0.4 (0.3)	0.7 (0.3)	1.2 (0.3)
4.055	0.6 (0.4)	0.6 (0.3)	0.1 (0.3)	0.5 (0.3)	0.5 (0.3)	0.8 (0.3)	0.7 (0.3)
4.185	0.3 (0.4)	0.6 (0.3)	0.1 (0.3)	0.5 (0.3)	0.7 (0.3)	0.8 (0.3)	0.3 (0.3)
4.315	0.1 (0.4)	0.4 (0.3)	0.3 (0.3)	0.5 (0.3)	0.7 (0.3)	0.7 (0.3)	0.4 (0.4)
4.445	-0.1 (0.4)	0.1 (0.3)	0.6 (0.3)	0.6 (0.3)	0.7 (0.3)	0.7 (0.3)	0.7 (0.4)
4.575	-0.1 (0.4)	-0.1 (0.3)	0.8 (0.3)	0.6 (0.3)	0.7 (0.3)	0.6 (0.3)	0.9 (0.4)
4.705	0.0 (0.3)	-0.2 (0.3)	0.6 (0.3)	0.5 (0.3)	0.7 (0.3)	0.6 (0.3)	0.9 (0.4)
4.835	0.3 (0.3)	-0.1 (0.3)	0.4 (0.3)	0.4 (0.3)	0.6 (0.3)	0.5 (0.3)	0.9 (0.4)
4.965	0.6 (0.3)	0.2 (0.3)	0.2 (0.3)	0.3 (0.3)	0.5 (0.3)	0.4 (0.3)	0.8 (0.4)
5.155	0.8 (0.3)	0.4 (0.3)	0.1 (0.3)	0.3 (0.3)	0.3 (0.3)	0.3 (0.3)	0.7 (0.4)
5.325	0.7 (0.3)	0.6 (0.3)	0.3 (0.3)	0.3 (0.3)	0.3 (0.3)	0.3 (0.3)	0.6 (0.4)
5.495	0.5 (0.3)	0.6 (0.2)	0.5 (0.2)	0.3 (0.3)	0.3 (0.3)	0.4 (0.3)	0.5 (0.4)
5.665	0.2 (0.3)	0.6 (0.2)	0.5 (0.2)	0.1 (0.2)	0.4 (0.3)	0.5 (0.3)	0.3 (0.4)
5.835	0.1 (0.3)	0.5 (0.2)	0.5 (0.2)	0.3 (0.2)	0.5 (0.2)	0.5 (0.3)	0.2 (0.4)
6.005	0.2 (0.2)	0.3 (0.2)	0.4 (0.2)	0.3 (0.2)	0.4 (0.2)	0.5 (0.3)	0.3 (0.3)
6.175	0.3 (0.2)	0.2 (0.2)	0.3 (0.2)	0.2 (0.2)	0.3 (0.2)	0.5 (0.2)	0.4 (0.3)
6.345	0.3 (0.1)	0.0 (0.1)	0.2 (0.2)	0.2 (0.2)	0.1 (0.2)	0.4 (0.2)	0.5 (0.3)
6.515	0.4 (0.1)	-0.0 (0.1)	0.2 (0.1)	0.2 (0.2)	0.0 (0.1)	0.3 (0.2)	0.5 (0.2)
6.700	0.3 (0.1)	0.0 (0.1)	0.2 (0.1)	0.2 (0.1)	0.0 (0.1)	0.2 (0.2)	0.5 (0.2)
6.880	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.4 (0.2)
7.100	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.2 (0.1)	0.1 (0.1)	0.2 (0.1)
7.300	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.2 (0.1)
7.500	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)
7.700	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.1 (0.1)	0.2 (0.1)	0.1 (0.1)	0.1 (0.1)
7.900	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)
8.100	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.1 (0.1)	0.1 (0.1)
8.300	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)	0.0 (0.1)	0.1 (0.1)
8.500	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.1 (0.1)
8.725	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
8.975	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.225	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.475	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.725	0.0 (0.1)	-0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)	0.0 (0.1)
9.975	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	-0.0 (0.1)	0.0 (0.1)	-0.0 (0.1)	0.0 (0.1)

TABLE 3. TOTAL SECONDARY YIELDS AND AVERAGE SECONDARY ENERGIES FOR NEUTRONS AND GAMMA RAYS DUE TO NEUTRON INTERACTIONS WITH NIOBIUM. THESE RESULTS ARE FOR SECONDARY NEUTRONS WITH ENERGIES GREATER THAN 0.764 MEV AND GAMMA RAYS WITH ENERGIES GREATER THAN 0.700 MEV. UNCERTAINTIES ARE GIVEN IN PARENTHESES IN THE SAME UNITS AS THE DATA. UNCERTAINTIES IN TOTAL YIELDS DO NOT INCLUDE A 10% ERROR IN ABSOLUTE NORMALIZATION. THE ANGLE IS 120 DEGREES.

INCIDENT NEUTRON ENERGY (MEV)	ENERGY SPREAD	SECONDARY NEUTRON YIELD (NB/SR)	AVERAGE NEUTRON ENERGY (MEV)	SECONDARY PHOTON YIELD (NB/SR)	AVERAGE PHOTON ENERGY (MEV)
1.026	0.049	159.07 (5.91)	0.935 (.019)	69.19 (1.05)	1.109 (.023)
1.076	0.052	139.98 (5.00)	0.927 (.019)	44.65 (1.10)	1.063 (.018)
1.127	0.050	120.73 (4.32)	1.102 (.023)	37.96 (1.24)	1.037 (.017)
1.175	0.047	110.34 (3.92)	1.151 (.025)	106.24 (1.32)	1.034 (.017)
1.224	0.050	102.53 (3.40)	1.135 (.024)	117.13 (1.31)	1.018 (.016)
1.276	0.053	95.61 (3.01)	1.154 (.025)	116.15 (1.30)	1.005 (.015)
1.327	0.050	87.05 (2.93)	1.242 (.031)	120.36 (1.38)	0.992 (.015)
1.375	0.045	86.05 (2.69)	1.317 (.033)	124.31 (1.47)	1.002 (.016)
1.425	0.055	82.06 (2.27)	1.300 (.030)	125.42 (1.33)	1.001 (.014)
1.477	0.050	84.19 (2.13)	1.404 (.030)	132.87 (1.51)	0.997 (.014)
1.524	0.044	85.78 (2.08)	1.362 (.030)	112.75 (1.52)	1.003 (.015)
1.573	0.055	93.41 (1.88)	1.337 (.025)	138.20 (1.38)	1.004 (.013)
1.625	0.048	100.81 (1.92)	1.231 (.025)	139.90 (1.49)	1.016 (.014)
1.674	0.050	101.95 (1.79)	1.315 (.025)	143.16 (1.48)	1.019 (.014)
1.725	0.053	108.56 (1.76)	1.309 (.023)	149.95 (1.52)	1.021 (.013)
1.773	0.044	117.00 (2.00)	1.295 (.025)	155.56 (1.74)	1.020 (.015)
1.826	0.057	127.00 (1.70)	1.265 (.020)	159.93 (1.54)	1.024 (.013)
1.876	0.048	130.40 (1.56)	1.300 (.022)	167.09 (1.74)	1.019 (.014)
1.925	0.050	130.29 (2.01)	1.345 (.023)	163.03 (1.76)	1.015 (.014)
1.975	0.052	130.39 (1.94)	1.346 (.023)	170.04 (1.76)	1.029 (.014)
2.045	0.095	121.46 (1.45)	1.411 (.019)	193.16 (1.40)	1.029 (.010)
2.107	0.102	124.40 (1.41)	1.416 (.019)	192.44 (1.45)	1.041 (.010)
2.245	0.094	124.44 (1.57)	1.442 (.021)	200.05 (1.62)	1.054 (.011)
2.342	0.100	126.65 (1.62)	1.449 (.021)	210.74 (1.70)	1.061 (.011)
2.404	0.104	127.45 (1.70)	1.533 (.022)	216.52 (1.76)	1.066 (.011)
2.545	0.094	135.39 (2.00)	1.515 (.024)	224.61 (2.07)	1.068 (.013)
2.642	0.100	128.12 (2.09)	1.591 (.028)	228.96 (2.24)	1.090 (.014)
2.745	0.105	142.15 (2.47)	1.548 (.029)	227.78 (2.59)	1.105 (.016)
2.853	0.112	144.89 (2.50)	1.605 (.029)	244.49 (2.60)	1.102 (.015)
2.956	0.094	146.34 (2.70)	1.674 (.032)	251.77 (2.84)	1.119 (.016)
3.053	0.099	143.46 (2.59)	1.645 (.032)	254.98 (2.73)	1.141 (.015)
3.148	0.091	135.61 (2.60)	1.765 (.036)	268.92 (2.75)	1.178 (.016)
3.246	0.109	134.12 (2.40)	1.603 (.034)	268.63 (2.64)	1.178 (.015)
3.352	0.100	128.76 (2.34)	1.893 (.037)	277.90 (2.78)	1.199 (.015)
3.454	0.104	121.74 (2.17)	1.995 (.039)	280.31 (2.73)	1.221 (.015)
3.553	0.093	122.29 (2.25)	1.970 (.041)	264.92 (2.91)	1.233 (.016)
3.648	0.097	122.66 (2.14)	2.027 (.041)	292.63 (2.93)	1.253 (.016)
3.747	0.101	120.24 (2.08)	2.079 (.043)	295.01 (2.83)	1.258 (.015)
3.850	0.105	130.70 (2.05)	2.060 (.039)	300.00 (2.80)	1.282 (.015)
3.948	0.091	131.57 (2.23)	2.101 (.043)	302.84 (3.03)	1.300 (.016)
4.051	0.114	135.13 (2.02)	2.141 (.040)	303.17 (2.71)	1.325 (.015)
4.157	0.098	134.02 (2.17)	2.201 (.045)	316.78 (2.98)	1.341 (.016)
4.257	0.102	130.45 (2.14)	2.283 (.047)	314.81 (2.99)	1.361 (.016)
4.350	0.094	134.65 (2.48)	2.313 (.053)	324.38 (3.45)	1.370 (.018)
4.447	0.109	137.35 (2.16)	2.338 (.047)	326.59 (3.02)	1.399 (.016)
4.546	0.090	139.77 (2.38)	2.441 (.051)	329.64 (3.16)	1.401 (.015)
4.656	0.116	149.01 (2.18)	2.417 (.045)	341.21 (3.05)	1.420 (.016)
4.756	0.096	149.17 (2.40)	2.489 (.049)	344.06 (3.34)	1.445 (.017)
4.854	0.099	146.03 (2.36)	2.575 (.051)	347.76 (3.32)	1.451 (.017)
4.955	0.103	146.89 (2.33)	2.605 (.051)	363.94 (3.39)	1.448 (.017)
5.100	0.137	150.08 (1.55)	2.644 (.042)	360.48 (2.54)	1.490 (.013)
5.293	0.198	151.82 (1.84)	2.714 (.043)	368.19 (2.52)	1.532 (.013)
5.497	0.210	149.04 (1.81)	2.795 (.044)	366.98 (2.55)	1.539 (.013)
5.690	0.190	151.18 (1.87)	2.848 (.048)	384.98 (2.70)	1.578 (.014)
5.891	0.200	154.37 (1.83)	2.952 (.049)	401.38 (2.74)	1.620 (.014)
6.096	0.210	154.63 (1.76)	3.015 (.050)	410.49 (2.72)	1.623 (.013)
6.293	0.184	151.08 (1.85)	3.052 (.055)	408.08 (2.90)	1.657 (.015)
6.500	0.231	151.57 (1.67)	3.158 (.054)	424.39 (2.68)	1.693 (.013)
6.717	0.203	154.07 (1.82)	3.144 (.056)	436.18 (2.93)	1.714 (.014)
6.903	0.169	156.43 (2.06)	3.209 (.060)	446.43 (3.29)	1.727 (.016)
7.098	0.220	154.30 (1.90)	3.232 (.056)	449.99 (2.94)	1.773 (.015)
7.299	0.184	149.33 (2.14)	3.298 (.064)	451.23 (3.27)	1.803 (.017)
7.487	0.191	145.61 (2.11)	3.315 (.064)	437.65 (3.15)	1.817 (.017)
7.681	0.198	139.19 (1.96)	3.229 (.062)	426.84 (3.02)	1.851 (.017)
7.884	0.206	135.03 (1.88)	3.209 (.063)	427.91 (3.00)	1.882 (.017)
8.094	0.215	137.50 (1.88)	3.169 (.063)	443.52 (3.07)	1.905 (.017)
8.313	0.223	142.66 (1.90)	3.108 (.061)	470.23 (3.19)	1.420 (.017)
8.512	0.174	140.79 (2.17)	3.168 (.071)	481.64 (3.76)	1.948 (.019)
8.688	0.179	140.93 (2.13)	3.145 (.070)	484.33 (3.73)	1.979 (.019)
8.902	0.248	134.78 (1.78)	3.204 (.068)	470.92 (3.13)	2.001 (.017)
9.122	0.193	126.73 (1.94)	3.237 (.075)	456.63 (3.44)	2.024 (.020)
9.318	0.199	124.03 (1.90)	3.259 (.075)	432.58 (3.36)	2.040 (.020)
9.520	0.206	124.22 (1.94)	3.161 (.076)	429.46 (3.40)	2.048 (.021)
9.727	0.212	121.62 (2.07)	3.097 (.075)	406.60 (3.16)	2.071 (.022)
9.906	0.146	132.93 (2.58)	3.176 (.093)	381.17 (4.05)	2.094 (.029)
10.249	0.536	144.31 (1.71)	3.029 (.052)	399.66 (2.06)	2.067 (.016)
10.764	0.495	155.77 (2.20)	2.893 (.052)	275.67 (2.00)	2.021 (.019)
11.277	0.531	163.10 (2.67)	2.941 (.053)	243.46 (1.92)	1.939 (.020)
11.778	0.472	176.41 (3.12)	2.925 (.058)	217.38 (2.04)	1.891 (.023)
12.266	0.502	174.05 (3.24)	2.822 (.060)	209.43 (1.99)	1.793 (.023)
12.784	0.534	165.17 (3.64)	2.921 (.062)	207.74 (2.08)	1.748 (.023)
13.277	0.453	190.78 (4.08)	2.975 (.072)	218.19 (2.41)	1.704 (.025)
13.742	0.477	200.40 (4.44)	2.930 (.073)	225.53 (2.47)	1.728 (.025)
14.232	0.503	194.74 (4.32)	3.004 (.076)	240.08 (2.52)	1.690 (.023)
14.749	0.531	196.40 (4.31)	3.174 (.080)	249.95 (2.57)	1.751 (.024)
15.223	0.416	197.08 (4.63)	3.206 (.099)	263.22 (3.15)	1.766 (.028)
15.724	0.585	198.90 (4.39)	3.201 (.089)	268.55 (2.79)	1.809 (.024)
16.247	0.461	204.44 (4.75)	3.123 (.105)	285.30 (3.15)	1.948 (.028)
16.719	0.481	204.55 (4.81)	3.405 (.112)	300.81 (3.48)	1.892 (.028)
17.210	0.503	214.39 (5.01)	3.293 (.112)	317.22 (3.65)	1.940 (.029)
17.726	0.526	222.19 (5.25)	3.407 (.119)	346.67 (3.87)	1.966 (.028)
18.262	0.550	228.51 (5.39)	3.398 (.121)	353.33 (4.01)	2.004 (.029)
18.727	0.381	234.32 (6.15)	3.181 (.152)	363.64 (5.05)	2.051 (.036)
19.215	0.594	244.73 (5.82)	3.429 (.129)	375.47 (4.29)	2.036 (.030)
19.710	0.412	246.49 (6.76)	3.517 (.154)	372.46 (5.35)	2.070 (.038)

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