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Evaluations of the fast neutron cross sections of ^{60}Ni including complete covariance information

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Kernphysik der Universität Wien, Austria

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1. Abstract

A new evaluation of all important neutron cross sections of ^{60}Ni was performed in the neutron energy range 0.450 – 20 MeV, that is for the whole energy range above the resonance region. The evaluation combines the results of nuclear model calculations and the complete existing experimental data base in order to obtain the most accurate description of the cross sections possible within our present knowledge. The evaluation was performed in the following way: The cross sections from the EFF – 2 file (results of model calculations) and their estimated covariances are used as prior information which is successively improved by adding experimental data and by applying Bayes' theorem to obtain the posterior information. For this process the code GLUCS was used. For some cross sections, not covered in EFF – 2, priors were taken from ENDF/B-VI and JENDL – 3. For the (n,p) cross – section, where we had already done such evaluations previously, we used that result directly. As results we obtained evaluated cross sections and their covariances for a chosen set of 15 independent cross sections. A final coupled set of evaluated cross sections and covariances was obtained by a last GLUCS run including the experimental data for "redundant" cross sections that is all cross sections which can be expressed as sum or differences of the basic cross sections chosen for the evaluation. The results of our new evaluation agree with ENDF/B–VI and EFF – 2 within the uncertainties of these evaluations. Most of the uncertainties of our evaluated cross sections, however are considerably smaller than those of ENDF/B–VI and EFF – 2.

2. General evaluation procedure

The general principle of our evaluation is essentially the same as used in (*Vonach 92*). For a better understanding of this report we will give a short description of this procedure; it is shown schematically in Fig. 1. First we choose a set of non-redundant cross sections which give a complete and sufficiently detailed description of the interaction of fast neutrons with ^{60}Ni as the subject of the evaluation. As the starting point we use the EFF – 2 evaluation (*Uhl 91*) and its covariances (*Tagesen 91*) with some modifications which are going to be discussed later in section 3. This constitutes our prior knowledge of the neutron cross sections of ^{60}Ni. Each type of cross section is represented by a cross section vector T and its covariance matrix M. For some rare reactions, not contained in EFF – 2, we have used ENDF/B–VI and JENDL – 3 (see section 3). Then Bayes' theorem was used to add successively the experimental data for the various ^{60}Ni cross sections to the prior. This is done in the following way: If the data are described by a vector *R* with the covariance matrix *V*, application of Bayes' theorem results in the following relations for the improved cross sections *T'* and the covariances *M'*

$$T' = T + MG^{+}(GMG^{+} + V)^{-1}(R - R_T) \qquad (1)$$

$$M' = M - MG^{+}(GMG^{+} + V)^{-1}GM \qquad (2)$$

where R_T presents the prior value interpolated at the point where *R* is given, *G* is the sensitivity matrix of the new experimental data relative to the prior data with the matrix elements $g_{ij} = \delta R_i/\delta T_j$, and the up scripts (+) and (–1) mean transpose and inverse operation respectively. One of the most important conditions for obtaining these formulae is an absence of correlations between the data vectors *T* and *R*. This condition is fulfilled as *T* was derived from nuclear model calculations and *R* are results of measurements.

From this procedure (depicted at the left side of Figure 1) we get a set of improved cross sections with much reduced uncertainties compared to the prior EFF – 2 values. Cross sections for which no experimental data exist (e.g. $\sigma_{n,np}$, $\sigma_{n,n\alpha}$) remain unchanged at this step. This procedure however does not use the complete experimental data base. In addition to cross section measurements for our basic cross sections there exist always additional measurements on so-called redundant cross sections, which are sums or differences of our basic, linearly independent cross sections (see. Fig. 1 and Section 5.2). In order to also use this information in a final evaluation step (see right side of Figure 1) the results of our evaluations for the basic cross sections are used as a new

improved prior and the data for the redundant cross sections are added as data for the corresponding sums or differences of basic cross sections in a final evaluation step again using eq. 1 and 2.

Thus the evaluations proceed in the following steps:

1) Establishment of the prior data for all cross sections of interest.
2) Establishment of the experimental data base.
3) Calculation of the improved cross sections T' and covariances M' for all important cross sections for which data are available.
4) Further improvement of the evaluation by adding the information from all redundant cross sections in a final evaluation step applied to the joint cross section vector for all reaction types.

This leads to a final result of the evaluation in form of a cross section vector T' containing a complete set of independent cross sections and one large covariance matrix M' which can be subdivided into covariance matrices for the individual cross sections and covariance matrices between different cross section types (interreaction covariance matrices).

Technically this procedure is performed by means of the code GLUCS (*Hetrick 80*) which implements Equ. (1) and (2) and provides output on T' and M' directly in ENDF/B format. As modified recently (*Tagesen 94*) it can also be used for the constrained least squares adjustment of step 4 of our evaluation procedure.

3. Establishment of the prior information for all cross sections of interest

For this evaluation we chose the following cross sections as our complete basic set of non-redundant cross sections, from which all other cross sections of interest can be derived as linear functions: σ_{tot} (MT1), $\sigma_{n,2n}$ (MT16), $\sigma_{n,n\alpha}$ (MT22), $\sigma_{n,np}$ (MT28), $\sigma_{n,n1}$ (MT51), $\sigma_{n2\text{-}3}$ (MT52), $\sigma_{n4\text{-}5}$ (MT851), $\sigma_{n6\text{-}11}$ (MT852), $\sigma_{n,n\ cont}$ (MT91), $\sigma_{n,\gamma}$ (MT102), $\sigma_{n,p}$ (MT103), $\sigma_{n,d}$ (MT104), $\sigma_{n,t}$ (MT105), $\sigma_{n,^3He}$ (MT106) and $\sigma_{n,\alpha}$ (MT107).
Most of the choices are obvious, the only important point is the amount of detail used to describe inelastic neutron scattering to discrete levels. Our starting point EFF-2 contains cross sections for the excitation of 26 levels of ^{60}Ni; thus in principle excitation of all those levels could have been included separately in our evaluation. It is however intended to combine the file 3 and 33 data of this evaluation with the file 6 of ENDF/B-VI for the energy/angle distributions of the emitted particles. In this evaluation only cross sections for the excitation of the first 11 levels of ^{60}Ni are given separately, whereas inelastic scattering to all higher levels is included in the continuum cross section (MT91). Thus in order to be consistent with this choice, we decided to add the cross sections for all levels above No. 11 (that is MT62-76) to $\sigma_{n,n\ cont}$ (MT91) and to retain only MT51-61. In addition the excitation energies of levels 2 and 3, 4 and 5 and 6 to 11 respectively, are so close together that it is reasonable to treat them together and only evaluate the sum of their cross sections. From these considerations we finally decided to describe inelastic scattering by 5 cross sections (n,n_1), (n,$n_{2\text{-}3}$), (n,$n_{4\text{-}5}$), (n,$n_{6\text{-}11}$) and (n,n_{cont}) as listed before.
We decided to use the EFF – 2 evaluation as far as possible as the basis for the prior vector and covariance matrix respectively (*T*,*M*) in this evaluation because it provides an already good description of the ^{60}Ni cross sections, has sufficiently detailed covariance information and is essentially uncorrelated with the experimental data to be added (see Table 1). In detail, however, some modifications had to be made. Therefore, in the following a brief description of the prior quantities actually used is given:

A) Cross section

1) For the cross sections σ_{tot}, $\sigma_{n,n1}$, $\sigma_{n,n2\text{-}3}$, $\sigma_{n,n4\text{-}5}$, $\sigma_{n,n6\text{-}11}$, $\sigma_{n,\gamma}$, $\sigma_{n,\alpha}$, $\sigma_{n,np}$ and $\sigma_{n,n\ cont}$ the cross sections from EFF-2 were used as prior values without any change.
2) For the (n,p) cross section we made use of recent accurate evaluations of our group (*Badikov 96*) which already include all experimental data. Therefore step 1 of our evaluation (see Fig. 1) for this cross section was replaced by the use of this existing accurate evaluation.

3) For the small cross sections $\sigma_{n,d}$ and $\sigma_{n,n\alpha}$ not given in EFF-2 cross sections from ENDF/B-VI were used as prior, for $\sigma_{n,t}$ and $\sigma_{n,^3He}$ the only existing evaluation JENDL-3 was used.

B) Covariances

1. EFF-2 cross sections:
   Relative uncertainties as a function of neutron energy were taken from the EFF-2 covariance estimates (*Tagesen 91*) for $\sigma_{n,2n}$, $\sigma_{n,np}$, $\sigma_{n,\alpha}$ and $\sigma_{n,\gamma}$. For inelastic neutron scattering to discrete levels EFF-2 only gives covariances to the sum of all discrete cross sections. Therefore uncertainties for $\sigma_{n,n1}$, $\sigma_{n,n2-3}$, $\sigma_{n,n4-5}$, $\sigma_{n,n6-11}$ and $\sigma_{n,n\ cont}$ were estimated from the differences of those cross sections between the evaluations EFF-2, ENDF/B-VI, JENDL-3 and BROND using the procedures developed in (*Tagesen 91)*.
2. For σ_{tot}, where good experimental data are available for the whole energy range of the evaluation uncertainties for the prior were set to 10% about an order of magnitude larger than the experimental uncertainties in order to obtain a so-called uninformative prior which does not influence the evaluation result.
3. For $\sigma_{n,p}$ the covariance matrix given in (*Badikov 96*) was used.
4. For the cross sections ($\sigma_{n,d}$ and $\sigma_{n,n\alpha}$) taken from ENDF/B-VI also the corresponding covariance matrices from ENDF/B-VI were used. For $\sigma_{n,t}$ and $\sigma_{n,^3He}$ taken from JENDL-3 approximate covariance data were estimated based on our experience about the reliability of the calculations of such cross sections.

In EFF – 2 the energy range of the evaluations has been divided into intervals of 0.5 MeV and 1 MeV for the representation of the covariance matrices within which cross sections are fully correlated. In the lower energy range of our evaluation these intervals appeared to be too large for a detailed description of the excitation function. Therefore a finer energy grid (33 intervals) was adopted for this evaluation. Energy bins of 0.2 MeV were chosen in the energy range up to 3.0 MeV, 0.5 MeV in the energy range 3.0 – 6.0 MeV and 14.0 – 15.0 MeV and 1.0 MeV for the rest of the energy range (see e.g. Table 10). This structure of the covariance matrices was used for all cross sections.

For EFF – 2 a triangular decrease of correlation with increasing distance in energy was assumed for all cross sections in order to describe the (positive) correlations between the cross section uncertainties at different neutron energies E_1 and E_2 (see discussion on

page 6 in *Pavlik 91*). To prevent high rigidity of shape of the excitation functions at low energies a variable FWHM of the triangular decrease was calculated as

$$FWHM = (E_1 + E_2)/10. + 2.e+5 \text{ (eV)}$$

which results in FWHM ranging from ≈ 400 keV at 1 MeV up to ≈ 4 MeV at 20 MeV incident neutron energy for generating the off – diagonal elements of the covariances of our priors. Correlation coefficients between cross section uncertainties at the energies E_1 and E_2 were calculated according to the relation

$$\mathrm{cov}(\sigma_1\sigma_2) = \sqrt{Var(\sigma_1)\cdot Var(\sigma_2)}\cdot\frac{(FWHM - ABS(E_1 - E_2)}{FWHM} \qquad (3).$$

in case of ABS (E_1 - E_2) ≤ FWHM and zero otherwise

4. Establishment of the experimental data base including construction of covariance matrices for all data sets

We used the experimental data compiled in EXFOR (*Lemmel 86*, *McLane 88*) and supplemented them by very recent ones which were mostly obtained directly from the authors. In addition to measurements on ^{60}Ni we also used measurements on natural nickel for such cross sections for which the difference between ^{60}Ni and ^{nat}Ni is known to be small as is the case for σ_{tot}, σ_{non} and σ_{el} above 4 MeV. Additionally, in order to widen our data base also some more complex cross sections like the γ – production cross section for the first 2^+ level were included in our data base, if good measurements have existed and accurate conversion procedures to basic cross sections, e.g. σ_{inel}, could be developed. Differential elastic and inelastic scattering cross sections measured over a sufficient angular range were used to derive the total elastic and inelastic scattering cross sections by means of fits with Legendre polynomials in those cases where the integrations had not been performed by the authors.

All data sets were critically reviewed, obviously wrong data were rejected. The accepted data were renormalized if necessary with regard to the standard cross sections or decay data used. In some cases renormalizations were also applied if comparisons of a data set with other data consistently have indicated the need for such renormalizations.

For the construction of the covariance matrices of the experimental data sets it is necessary to have detailed information on all uncertainty components of the measurements and the correlation of each component within the data set. As this information is not given for most of the experiments various approximations had to be used.

We assumed that the covariance matrix of total uncertainties can be split into two matrices of partial uncertainties:

1) a diagonal covariance matrix of partial uncertainties describing short – energy – range (SER) correlation properties such as statistical uncertainties due to a finite number of counts per channel;

2) a constant covariance matrix of partial uncertainties connected with properties which induce large – energy – range (LER) correlations, such as systematic uncertainties due to any normalization of the cross sections in order to get absolute values, to the determination of the number of nuclei in a sample, to geometrical sizes and distances and to sample self – absorption properties for the non – resonance energy region. This means we assume complete correlation over all energy groups for these long – range uncertainty components.

The magnitudes of the described two components were chosen according to the uncertainty information given by the authors, or in case of missing information they were estimated by the authors according to their experience about typical uncertainties at the time of the respective experiments.
Only for the total cross sections it was necessary to use a third partial uncertainty matrix, describing so-called medium-energy range correlations (MER). For this covariance matrix the correlations between the uncertainties for different energy groups are described by a linear model of correlation propagation with a certain energy E_c (typical 2 MeV) within which the correlation decreases linearly from 100% to zero.

A summary on the data base prepared in this way is given in Tables 2 – 9. For each data set, accepted for the evaluation, the tables give the energy range of the experiment, the number of data points, the short and long-range uncertainties assigned to the respective data set and in some cases information about special treatment of the data prior to the use in the evaluation. The numerical values of the cross sections and the covariances cannot be given in this report, they are however available on request at our institute.
While most of the cross section values could be directly put into the evaluation some pre-treatment was necessary for a part of the cross section types and some special data sets. In detail the following procedures were used .

1. Total cross sections: A number of total cross section measurements have been performed with high-resolution using white neutron sources resulting in a very large number of data points (see Table 2). These data sets have been pre-averaged into bins corresponding to the group structure of the evaluation and only these averaged values were used as evaluation input. Thus also our evaluated total cross sections are in principle group cross sections (averages over the 0.2 – 1.0 MeV intervals chosen for the evaluation). This fact is important for the lower energy range up to about 4 MeV, where the total cross sections exhibit considerable fluctuations (which have to be re-introduced in a final step following this evaluation). Above 4 MeV the fluctuations are small and the general energy dependence is rather weak, therefore above 4 MeV these group cross sections can also be considered as point cross sections at the respective group centers. Total cross section data for natural nickel have been included for neutron energies above 4 MeV, as in this energy region the cross section difference between ^{60}Ni and ^{nat}Ni is expected to be smaller than the uncertainty of the measurements. In addition these cross sections of ^{nat}Ni have already been used for the evaluation of ^{58}Ni; thus small errors will be canceled if these two evaluations are combined to describe ^{nat}Ni. All measurements of σ_{tot}

below 4 MeV were checked for self-shielding effects and only the measurements done with sufficiently thin targets were used. Self shielding effects were estimated according to *Fröhner 00*. Based on this work the data of *Harvey 82* were rejected completely and the data of *Stoler 71* were only used in the energy region above 2 MeV. For the accepted data self-shielding effects are expected to be below 1%.

2. Inelastic scattering cross sections for excitation of the first level (E = 1.33 MeV): As noted in Table 3 a number of data sets reporting only differential cross sections were integrated over angle using the code GPOLFIT (*Pavlik 90*).
3. Inelastic cross sections for excitation of higher levels (Table 4): In addition to the direct neutron scattering measurements, also the γ-production measurements of Traiforos et al. (*Traiforos 79*) were used. These data which are not available in numerical form were read from the graphs in the original paper and inelastic neutron scattering cross sections for the various levels were derived by adding the γ-production cross sections for all γ-transitions deexciting the particular level and subtracting (if necessary) γ-transitions feeding the level from higher levels.
4. Total inelastic cross sections (see Table 6) were derived from the measurements of the γ-production cross sections for the $^{1}2^{+} \rightarrow$ g.s. transition (see Figure 2) using an energy-dependent correction factor for the small fraction of direct γ-transitions from higher levels to the ground-state. An uncertainty of 5% was assumed for this correction factor for neutron energies above 3 MeV. Below this energy there are no direct ground state transitions and the correction factor is exactly unity.
5. Elastic cross sections for ^{nat}Ni were included for E_n > 4 MeV for the reasons discussed before.
6. The results of *Pavlik 91* at E_n = 14 MeV in Tables 6 and 9 are taken from the evaluation of Pavlik et al. (*Pavlik 91*) which give the weighted average of all cross section measurements at 14 MeV. Details about the original data can be found in (*Pavlik 91*).
7. Finally it has to be mentioned, that the experimental database for the ^{60}Ni(n,p) reaction is not given here, as a good recent evaluation of this reaction is available (*Badikov 96*), which has been used by us in order to avoid unnecessary duplication of work as discussed in Section 3.
8. In addition to the data listed in Tables 2-9 the following single data points in the 14-MeV region were used:

$\sigma_{n,t}$:	61 ± 20 μb	at	14.6	MeV	(*Qaim 76*),
$\sigma_{n,p\text{-}prod}$:	325 ± 40 mb	at	14.8	MeV	(*Grimes 79*),
$\sigma_{n,2n}$:	410 ± 120 mb	at	14.8	MeV	(*Weselka 91*).

The cross section value for $\sigma_{n,\ n\ cont}$ (MT 91) at E_n = 14.1 MeV derived from the secondary neutron production cross section at 14.1 MeV of ^{nat}Ni integrated over the secondary neutron energy range 5-11 MeV (*Pavlik 93*) was also used in the evaluation. The conversion of this cross section integral into values for $\sigma_{n,\ n\ cont}$ for both ^{58}Ni and ^{60}Ni was done by comparing the measured cross section integral to that calculated with ENDF/B-VI and scaling the ENDF/B-VI cross sections for $\sigma_{n,\ n\ cont}$ with the ratio of these two integrals. The uncertainty of the experimental cross section integral was also assigned to the values of $\sigma_{n,\ n\ cont}$ derived in this way.

There is consistency both within the database and between the priors and the data as can be seen from Fig. 3 - 20 discussed in section 6.

5. Cross section evaluation

5.1. Evaluation of the cross sections for individual reactions

In the first evaluation step the experimental data for the basic cross sections were combined with the respective priors by means of the code GLUCS according to the general procedure described in Section two (see Fig. 1). This was done for the cross sections σ_{tot}, $\sigma_{n,n1}$, $\sigma_{n,n2\text{-}3}$, $\sigma_{n,n4\text{-}5}$, $\sigma_{n,n6\text{-}11}$, $\sigma_{n,n\ cont}$, $\sigma_{n,2n}$, $\sigma_{n,\alpha,}$ $\sigma_{n,d}$ and $\sigma_{n,t}$. For the n,p cross section this work has already been done in the evaluation (*Badikov 94*) used as prior, thus this prior could be used directly as input for the second evaluation step. For the cross sections $\sigma_{n,\gamma}$, $\sigma_{n,^3He}$, $\sigma_{n,np}$, and $\sigma_{n,n\alpha}$ there are no experimental data; accordingly no improvement of the priors was possible in the first evaluation step.

5.2. Consistent joint evaluation of all cross sections

The cross sections σ_{inel}, σ_{non}, σ_{el} and $\sigma_{\alpha\text{-prod}}$ summarized in Table 8-12 are “redundant” cross sections that is they are sums or differences of the basic cross sections chosen for our evaluation (see Table 1). They are connected to the basic cross sections by means of the relations

$$\sigma_{inel} = \sigma_{n,n_1} + \sigma_{n,n_{2-3}} + \sigma_{n,n_{4-5}} + \sigma_{n,n_{6-11}} + \sigma_{n,n_{cont}} \tag{4}$$

$$\sigma_{non} = \sigma_{inel} + \sigma_{n,2n} + \sigma_{n,p} + \sigma_{n,np} + \sigma_{n,\alpha} + \sigma_{n,n\alpha} + \sigma_{n,d} + \sigma_{n,t} + \sigma_{n,\gamma} + \sigma_{n,^3He} \tag{5}$$

$$\sigma_{el} = \sigma_{tot} - \sigma_{non} \tag{6}$$

$$\sigma_{\alpha-prod} = \sigma_{n,\alpha} + \sigma_{n,n\alpha} \tag{7}$$

$$\sigma_{p-em} = \sigma_{n,p} + \sigma_{n,np} \tag{8}$$

As the final step of the evaluation (see right side of Figure 1), an improved evaluation using the information contained in both our basic and redundant cross sections, was obtained in the following way: The redundant cross sections were added as "data" of sums or differences of basic cross sections, according to Equation 4 – 8, by using again the code GLUCS, based on Equation 1 and 2 (see Section 2). The posterior data derived in this way not only strictly fulfill the consistency relations (Equation 4 – 8) but are also considerably improved in quality as many of the redundant cross sections (e.g. σ_{non} or σ_{inel}) are known rather accurately and this accuracy is in part transferred to the basic cross sections by means of the applied constrained least squares fit. Technically the

accepted redundant cross sections (see Tables 6 -9) of all types were added as one large data vector to the prior consisting of the coupled set of all basic cross sections in one GLUCS run.

Because of the conditions (4 – 8) and the consideration of all basic cross sections as one coupled set, the resulting correlation matrix now includes parts, which describe correlations between different energy intervals of different cross sections. In most cases these correlations are small (< 10%), in some cases however, e.g. between different partial inelastic cross sections, they are important and have to be taken into account.

6. Results of the evaluation

The main result of this evaluation is a complete but non – redundant set of cross sections (σ_{tot}, $\sigma_{n,n1}$, $\sigma_{n,n2-3}$, $\sigma_{n,n4-5}$, $\sigma_{n,n6-11}$, $\sigma_{n,n\ cont}$, $\sigma_{n,p}$, $\sigma_{n,d}$, $\sigma_{n,t}$, $\sigma_{n,\alpha}$, $\sigma_{n,n\alpha}$, $\sigma_{n,^3He}$, $\sigma_{n,2n}$, $\sigma_{n,np}$ and $\sigma_{n,\gamma}$) and their covariances in the fast neutron energy range 0.450 – 20 MeV. In addition to this, cross sections and covariances for σ_{el}, σ_{non} and σ_{inel} were obtained by expressing these cross sections as linear functions of the basic cross sections (see Equations 4 – 8). In Table 10 the final results of this evaluation, i.e., the cross sections and their uncertainties, are listed. There is, however, some difference in the meaning of the listed cross section values between σ_{tot} and σ_{el} on the one hand and all the other cross sections on the other hand. Due to the special evaluation procedure used for σ_{tot} (see Section 4) the evaluated cross sections are group cross sections averaged over the bins of our 33 group bin structure. As σ_{el} was derived essentially as difference between σ_{tot} and all other cross sections, the listed σ_{el} values are group – averaged cross sections, too. All other cross sections, however, are point cross sections, as their priors are the point cross sections from EFF – 2. Also the added data are approximately point cross sections. This difference however is only important in the energy range below 4 MeV. Therefore in Table 10 σ_{tot} and σ_{el} are given as group cross sections in the energy range below 4 MeV, above this energy all cross sections are given as point cross sections. For higher energies both σ_{tot} and σ_{el} are smooth functions of energy and the listed values can also be considered as point cross sections at the appropriate energy bin centers. In the energy region below 4 MeV, the known fine structure of the total (and elastic) cross sections will have to be superimposed on our evaluated group cross sections for an accurate description of σ_{tot} in file three of our evaluated data file, while retaining our course group structure in the description of the covariances in file 33.

These results are also presented in Figures 3 - 20 and compared to the EFF-2 and ENDF/B-VI evaluations and to the experimental data used as input to the evaluation. Two figures are given for each cross section. The first one compares our new evaluation to the prior (EFF-2) and to the experimental data; the second one compares our new evaluation to ENDF/B-VI. (For those cross sections where ENDF/B-VI had to be used as prior, because no EFF-2 evaluation exists, of course only one figure is given.) For each evaluation both the evaluated cross sections and the “error bands” calculated from the respective covariance are shown.

In these figures the progress, achieved in this evaluation, is immediately obvious. Our main conclusions are rather similar to those obtained in our previous evaluations for ^{56}Fe and ^{52}Cr (*Vonach 92, Pronyaev 95*):

1. Except for the total cross sections in the fluctuation region below 4 MeV, the results of this evaluation remain within the uncertainty limits of EFF – 2 for all reaction types and energies. Thus our latest evaluation, which is more accurate, confirms the validity of the estimations, concerning the uncertainties for the EFF – 2 cross sections which were derived from the dispersion of recent evaluations (*Tagesen 91*).

2. The largest improvement compared to EFF-2 in the evaluated cross sections was obtained for σ_{tot} and σ_{el} in the energy region below 3 MeV. At these low energies, the theoretical description of the cross section, by means of the optical model, becomes rather poor, so that rather large uncertainties are to be assigned to any calculated cross sections (*Tagesen 91*) and experimental data are more accurate.

3. Our new evaluation agrees with ENDF/B-VI within the uncertainties of both evaluations except for a small discrepancy for $\sigma_{n,\alpha}$ in the energy range 6 - 11 MeV. This discrepancy, also existing for ^{58}Ni, is due to the fact that the only data existing at the time of ENDF/B-VI (for ^{58}Ni) was in error as found by later experiments and is thus clarified.

4. The most important improvement of our new evaluation compared to both EFF-2 and ENDF/B–VI is certainly the considerable reduction of the cross section uncertainties in all energy ranges where accurate measurements of the respective cross section exist. For the most important cross section σ_{tot} the uncertainties in all energy ranges could be reduced by about a factor of five compared to both EFF-2 and ENDF/B–VI over the whole energy range. Considerable improvement in uncertainties compared to the two previous evaluations could also be achieved for σ_{el}, $\sigma_{n,n1}$, $\sigma_{n,n\ cont}$, and $\sigma_{n,\alpha}$. Due to the large number of measurements around E_n = 14 MeV our new evaluation gives especially low uncertainties in this energy region, which is especially important for fusion neutronics.

   Reduction of uncertainties could not only be obtained for those cross sections where accurate measurements of these cross sections exist, but also for a number of cross sections, where no direct measurements are available as $\sigma_{n,n\ cont}$, by the use of redundant cross sections as described in Section 5.2 (for example accurate values of σ_{ncont} could be derived from measurements of σ_{inel}, $\sigma_{n,n1}$ and $\sigma_{n,n2}$). For the reactions $\sigma_{n,n2}$, $\sigma_{n,nd}$, $\sigma_{n,\gamma}$, $\sigma_{n,t}$ and $\sigma_{n,^3He}$ and inelastic scattering to high discrete states above 10 MeV no improvement has been possible due to a complete lack of experimental data.

One question may be the rather small uncertainties, resulting from our evaluation because of possible correlations between our prior data and the added data sets. However, this objection is not valid, because the statistical weight of the priors becomes negligible, if the added data are much more accurate than the prior data, and this is exactly the situation which exists in those parts of our evaluation, where the uncertainties are very low.

Of course, as is the case with any evaluation of experimental data, the uncertainties of our results could be too small because of unrealistically low uncertainty estimates, given for the data or because of neglecting correlations between different data sets. As it has been discussed in the previous chapters, we accounted for such effects by increasing the uncertainty components, estimated by the authors in all cases, which seemed to be doubtful. The correlations between different data sets were checked and generally seemed to be small. Finally our uncertainty estimates are confirmed by the fact that in all evaluations of individual cross sections and in the final joint evaluation χ^2 values of unity were obtained. According to our judgement, which is also based on our previous experience with evaluations of experimental data (*Pavlik 88, Wagner 90, Pavlik 91, Vonach 92*), the final uncertainties of the present evaluation are realistic effective standard deviations at the 1σ confidence level.

All correlation matrices for the uncertainties of the different reactions are positive definite. There are strong positive correlations between cross sections for neighboring energies, which decrease strongly with the energy difference between the considered points (see Fig. 21 - 35). The detailed structure of these matrices is determined by the strength of the mainly positive correlations, which are presented in the various data sets, used in the evaluation and varies considerably between different cross section types. The cross correlations between cross sections for different reactions are small relatively to most of the reaction pairs.

For most applications concerning neutron transport and activation in fusion devices, the accuracy of the present evaluation will probably be sufficient. The weakest point is to our judgement the capture cross sections in the lower MeV range where data are almost completely missing and thus very large uncertainties have to be assigned to the calculated cross sections. A new accurate measurement of $\sigma_{n,\gamma}$ in the MeV range therefore seems to us the most urgent neutron data requirement for ^{60}Ni.

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Table 1: Choice of Priors for the ^{60}Ni evaluation

Reaction	Cross Section	Covariance Matrix
σ_{tot}	EFF–2	uninformative
$\sigma_{n,n1}$	EFF–2	new estimate
$\sigma_{n,n2-3}$	EFF–2	new estimate
$\sigma_{n,n4-5}$	EFF–2	new estimate
$\sigma_{n,n6-11}$	EFF–2	new estimate
$\sigma_{n,\ n\ cont}$	EFF–2	EFF–2
$\sigma_{n,p}$	Badikov 96	Badikov 96
$\sigma_{n,d}$	ENDF/B-VI	ENDF/B-VI
$\sigma_{n,t}$	JENDL/3	new estimate
$\sigma_{n,^3He}$	JENDL/3	new estimate
$\sigma_{n,\alpha}$	EFF–2	EFF–2
$\sigma_{n,np}$	EFF–2	EFF–2
$\sigma_{n,n\alpha}$	ENDF/B-VI	ENDF/B-VI
$\sigma_{n,\gamma}$	EFF–2	EFF–2
$\sigma_{n,2n}$	EFF–2	EFF–2

Table 2: Experimental database for the total cross sections

Part one: Total cross section measurements for ^{60}Ni

EXFOR ENTRY	Reference	Type of experiment	Target thickness at./barn)	No. of Datapoints original	binned	E_n-range [MeV]	SER [%]	MER [%]	LER [%]
10124	Stoler 71	white source	0.27	374	7	2 – 4**	1	0.3	1
10879	Smith 79	mono n*	0.18	21	7	2 – 4	1	0.6	1
11601	Farell 66	mono n	0.06	280	1	0.53	2	–	–
22314	Brusegan 91	white source	0.075	25599	15	0.5 – 4.0	1 – 2	1.2	1
–	Larson 99	white source	0.075	1453	15	0.46 – 4.0	1 – 2	1.1	1
40614	Fedorov 80	white source	–	53	13	0.46 – 3.5	1 – 2	1.5	2

* monoenergetic neutrons

**The energy range of the experiment extends down to 0.45 MeV, because of the relatively thick target cross sections at low energy are distorted by self-shielding, therefore only data above 2 MeV were used for the evaluation

Part two: Total cross section measurements for ^{nat}Ni

10047	Foster 71	145	15	4.0 – 14.9	1	0.2	1
10225	Green 71 a	53	7	4.0 – 8.55	1	0.4	1
10225	Green 71 b	72	7	4.0 – 8.67	1	0.6	1
10342	Perey 73 a	469	20	4.0 – 20.0	1	0.4	1
10342	Perey 73 b	449	20	4.0 – 20.0	1.5	0.1	1.5
10416	Schwartz 74	698	16	4.0 – 15.13	1	0.2	1
10593	Guenther 76	35/20	3	4.0 – 5.0	1	1.7	1
10823	Smith 81	5	1	4.0 – 4.23	1	–	–
11056	Coon 52	1	1	14.12	1	–	–
11057	Goodman 52	1	1	14.0	1	–	–
11108	Peterson 60	2	2	17.5 – 18.4	1	0.2	1
11155	Bratenahl 58	5	5	7.0 – 14.48	1	0.7	1
12882	Larson 81	317	20	4.0 – 20.0	1	0.2	1
20012	Cierjacks 68	1137	20	4.0 – 20.0	1.5	0.2	1.5
20480	Cabe 73	54	5	4.0 – 6.19	1	2.5	1
21122	McCallum 60	13	8	12.37 – 20.0	1	0.6	1
30037	Mazari 55	7	6	13.0 – 16.2	1.5	0.9	1.5
30141	Angeli 71	1	1	14.7	1	–	–
31468	Polycroniades 94	3	2	4.04 – 6.71	1	1.4	1
40559	Tutubalin 77	1	1	14.7	1	–	–
68023	Tsukada 66	1	1	4.7	1.5	–	–

Table 3: Experimental database for the inelastic scattering cross sections with excitation of the first level (MT51)

EXFOR ENTRY	Reference	Meth. used	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
10879	Smith 79A	n-det.	34	2.0 – 3.9	3.8 – 6	10
10879	Smith 79B	γ-det.	14	1.39 – 2.04	7	10
10146	Rodgers 71	γ-det.	4	1.42 – 1.79	13	10
10487	Perey 70	n-det.	5	6.4 – 8.6	5 – 9.0	7
10113	Kinney 74	n-det.	2	4.34 – 4.92	5	7
10037	Boschung 71	n-det.	2	5.05 – 5.58	5	5
12930	Guss 85	n-det.	4	7.9 – 13.94	5 – 6.5	5
22128	Olsson 89	n-det.	1	21.6	5	–
40531	Korzh 77	n-det.	3	2.0 – 3.0	2.5 – 6	5

Table 4: Experimental database for the inelastic scattering cross sections for excitation of ^{60}Ni levels (second and higher)

EXFOR ENTRY	Reference	Type of cross section	Method	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
10487	Perey 70	(n,n_{2-3})	n-TOF	5	6.4 – 8.6	19 – 130	7
		(n,n_{4-5})	n-TOF	5	6.4 – 8.6	10 – 33	7
		(n,n_{6-11})	n-TOF	5	6.4 – 8.6	5 – 16	7
10879	Smith 79A	(n,n_{2-3})	n-TOF	17	2.5 – 3.9	10	10
		(n,n_{4-5})	n-TOF	10	2.5 – 3.9	7.8 – 8.6	10
10852	Traiforos 79	(n,n_{2-3})	γ-prod.	2	2.45 – 4.0	10 – 15	7
		(n,n_{4-5})	γ-prod.	2	2.45 – 4.0	14 – 21	7
10037	Boschung 71	(n,n_{2-3})	n-TOF	2	5.05 – 5.58	7	7
		(n,n_{4-5})	n-TOF	2	5.05 – 5.58	4.6 – 7	7
10113	Kinney 74	(n,n_{2-3})	n-TOF	6	4.1 – 5.5	5.4 – 15	7
		(n,n_{4-5})	n-TOF	6	4.1 – 5.5	10 – 13	7

Table 5: Experimental database for the (n,d) cross sections (MT104)

EXFOR ENTRY	Reference	Type of experiment	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
10827	Grimes 79	d-detection	1	14.8	36 total	
12999	Graham 87	d-detection	1	11.0	62 total	

Table 6: Experimental database for the total inelastic cross sections (MT 4)

EXFOR ENTRY	Reference	Type of experiment	corr. appl.	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
10852	Traiforos 79	$\sigma(2^+ \rightarrow \text{g.s.})$	–	23	1.42 – 3.95	8 – 9	8
10439	Tessler 75	$\sigma(2^+ \rightarrow \text{g.s.})$	–	4	3.43 – 5.42	12.5	10
20744	Voss 75	$\sigma(2^+ \rightarrow \text{g.s.})$	renormalized	30	1.4 – 13.0	7	7
20905	Towle 67	n-spect. int.	–	1	7.0	10	–
	Larson 85	$\sigma(2^+ \rightarrow \text{g.s.})$	–	21			7
	Pavlik 91	Evaluation	–	1	14.0	7.5	–

Table 7: Experimental database for the elastic scattering cross sections (MT 2)

Part 1: Data for ^{60}Ni

EXFOR ENTRY	Reference	No. of datapoints	corr.appl.	Energy range [MeV]	SER [%]	LER [%]
110879	Smith 79	9	int.of diff. c.s	1.675 – 3.7	3	5
10037	Boschung 71	2	–	5.05 – 5.58	4	4
12930	Guss 85	4	–	7.9 – 13.94	5 – 9	5
	Part 2: Data for ^{nat}Ni					
10487	Perey 70	3	–	6.4 – 8.6	5	7
10113	Kinney 74	5	–	4.34 – 4.92	5	7
22048	Olsson 87	1	–	21.6	5	–
20019	Holmquist 69	5	–	4.0 – 8.05	5	5

Table 8: Experimental database for the (n,α-em) cross sections

EXFOR ENTRY	Reference	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
10827	Grimes 79	1	14.8	7	–
12999	Graham 87	2	9.4 – 11	15	7
21920	Fischer 84	1	14.1	5	–
10933	Kneff 86	1	15.0	7	–
13598	Haight 96	1	9.85	5	–
	Haight 97	29	3.25 – 20	≥ 5	7

Table 9: Experimental database for the total non-elastic cross sections of ^{nat}Ni (MT 3)

EXFOR ENTRY	Reference	Type of experiment	No. of datapoints	Energy range [MeV]	SER [%]	LER [%]
11217	Taylor 55	sphere transm.	3	4.7 – 12.7	3	3
11216	Beyster 55	sphere transm.	2	4.0 – 4.5	3	3
11220	Beyster 56	sphere transm.	1	7.0	4	–
	Pavlik 91	sphere transm.	1	14.0	1	–

Table 10: Evaluated cross sections for neutron induced reactions on ^{60}Ni

MT = 1 total cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.0000E-11 - 4.5000E-01	0.0000E+00	0.000E+00	0.00
4.5000E-01 - 6.0000E-01	3.4703E+00	6.354E-02	1.83
6.0000E-01 - 8.0000E-01	3.2452E+00	6.035E-02	1.86
8.0000E-01 - 1.0000E+00	3.3563E+00	4.457E-02	1.33
1.0000E+00 - 1.2000E+00	3.0853E+00	4.479E-02	1.45
1.2000E+00 - 1.4000E+00	3.0899E+00	4.250E-02	1.38
1.4000E+00 - 1.6000E+00	3.1745E+00	3.306E-02	1.04
1.6000E+00 - 1.8000E+00	3.0454E+00	3.179E-02	1.04
1.8000E+00 - 2.0000E+00	3.0252E+00	3.394E-02	1.12
2.0000E+00 - 2.2000E+00	3.0985E+00	2.798E-02	0.90
2.2000E+00 - 2.4000E+00	2.9845E+00	2.861E-02	0.96
2.4000E+00 - 2.6000E+00	3.1585E+00	3.318E-02	1.05
2.6000E+00 - 2.8000E+00	3.0943E+00	3.207E-02	1.04
2.8000E+00 - 3.0000E+00	3.1937E+00	2.972E-02	0.93
3.0000E+00 - 3.5000E+00	3.3221E+00	3.207E-02	0.97
3.5000E+00 - 4.0000E+00	3.3972E+00	3.523E-02	1.04
4.0000E+00	3.4673E+00	3.313E-02	0.96
4.2500E+00	3.5219E+00	1.274E-02	0.36
4.7500E+00	3.5881E+00	1.254E-02	0.35
5.2500E+00	3.6834E+00	1.488E-02	0.40
5.7500E+00	3.6913E+00	1.484E-02	0.40
6.5000E+00	3.6634E+00	1.417E-02	0.39
7.5000E+00	3.5638E+00	1.259E-02	0.35
8.5000E+00	3.4320E+00	1.212E-02	0.35
9.5000E+00	3.2972E+00	1.219E-02	0.37
1.0500E+01	3.1833E+00	1.160E-02	0.36
1.1500E+01	3.0438E+00	1.174E-02	0.39
1.2500E+01	2.9122E+00	1.222E-02	0.42
1.3250E+01	2.8261E+00	1.359E-02	0.48
1.3750E+01	2.7516E+00	1.404E-02	0.51
1.4250E+01	2.6962E+00	1.347E-02	0.50
1.4750E+01	2.6297E+00	1.357E-02	0.52
1.5500E+01	2.5721E+00	1.247E-02	0.48
1.6500E+01	2.4899E+00	1.259E-02	0.51
1.7500E+01	2.4377E+00	1.198E-02	0.49
1.8500E+01	2.3943E+00	1.218E-02	0.51
1.9500E+01	2.3616E+00	1.897E-02	0.80
2.0000E+01	2.3471E+00	3.076E-02	1.31

MT = 2 elastic cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.0000E-11 - 4.5000E-01	0.0000E+00	0.0000E+00	0.00
4.5000E-01 - 6.0000E-01	3.4629E+00	6.3509E-02	1.83
6.0000E-01 - 8.0000E-01	3.2374E+00	6.0399E-02	1.87
8.0000E-01 - 1.0000E+00	3.3482E+00	4.4637E-02	1.33
1.0000E+00 - 1.2000E+00	3.0766E+00	4.4882E-02	1.46
1.2000E+00 - 1.4000E+00	3.0805E+00	4.2637E-02	1.38
1.4000E+00 - 1.6000E+00	2.7156E+00	4.2294E-02	1.56
1.6000E+00 - 1.8000E+00	2.5452E+00	3.7176E-02	1.46
1.8000E+00 - 2.0000E+00	2.4500E+00	4.1018E-02	1.67
2.0000E+00 - 2.2000E+00	2.4605E+00	4.0170E-02	1.63
2.2000E+00 - 2.4000E+00	2.2125E+00	4.6900E-02	2.12
2.4000E+00 - 2.6000E+00	2.2666E+00	4.4589E-02	1.97
2.6000E+00 - 2.8000E+00	2.1561E+00	4.3132E-02	2.00
2.8000E+00 - 3.0000E+00	2.1588E+00	4.3455E-02	2.01
3.0000E+00 - 3.5000E+00	2.1270E+00	4.5773E-02	2.15
3.5000E+00 - 4.0000E+00	2.1232E+00	5.2122E-02	2.45
4.0000E+06	2.1561E+00	5.0283E-02	2.33
4.2500E+06	2.1440E+00	6.3607E-02	2.97
4.7500E+06	2.0324E+00	5.3806E-02	2.65
5.2500E+06	2.1746E+00	6.5397E-02	3.01
5.7500E+06	2.1426E+00	6.4466E-02	3.01
6.5000E+06	2.1717E+00	5.8469E-02	2.69
7.5000E+06	2.1462E+00	5.4547E-02	2.54
8.5000E+06	2.0521E+00	5.5577E-02	2.71
9.5000E+06	1.9210E+00	5.9501E-02	3.10
1.0500E+07	1.8247E+00	5.4036E-02	2.96
1.1500E+07	1.6731E+00	6.4398E-02	3.85
1.2500E+07	1.5301E+00	6.1350E-02	4.01
1.3250E+07	1.3454E+00	6.5221E-02	4.85
1.3750E+07	1.2817E+00	7.9965E-02	6.24
1.4250E+07	1.2522E+00	4.3030E-02	3.44
1.4750E+07	1.1940E+00	4.9005E-02	4.10
1.5500E+07	1.1335E+00	1.1121E-01	9.81
1.6500E+07	1.0447E+00	1.4771E-01	14.14
1.7500E+07	9.8164E-01	1.6067E-01	16.37
1.8500E+07	9.6188E-01	1.4788E-01	15.37
1.9500E+07	9.7057E-01	9.7953E-02	10.09
2.0000E+07	9.7152E-01	5.3055E-02	5.46

MT = 3 nonelastic cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
5.2500E+05	7.3856E-03	1.4771E-03	20.00
7.0000E+05	7.8229E-03	2.3560E-03	30.12
9.0000E+05	8.1379E-03	2.4574E-03	30.20
1.1000E+06	8.6866E-03	2.8884E-03	33.25
1.3000E+06	9.4366E-03	3.5118E-03	37.22
1.5000E+06	4.5887E-01	2.6607E-02	5.80
1.7000E+06	5.0016E-01	2.0515E-02	4.10
1.9000E+06	5.7516E-01	2.3559E-02	4.10
2.1000E+06	6.3802E-01	2.9683E-02	4.65
2.3000E+06	7.7201E-01	3.8223E-02	4.95
2.5000E+06	8.9187E-01	3.2926E-02	3.69
2.7000E+06	9.3822E-01	3.2505E-02	3.46
2.9000E+06	1.0349E+00	3.5062E-02	3.39
3.2500E+06	1.1951E+00	3.7545E-02	3.14
3.7500E+06	1.2740E+00	4.6336E-02	3.64
4.0000E+06	1.3112E+00	4.1295E-02	3.15
4.2500E+06	1.3779E+00	6.2687E-02	4.55
4.7500E+06	1.5557E+00	5.2658E-02	3.38
5.2500E+06	1.5088E+00	6.4774E-02	4.29
5.7500E+06	1.5487E+00	6.4008E-02	4.13
6.5000E+06	1.4917E+00	5.7368E-02	3.85
7.5000E+06	1.4176E+00	5.3608E-02	3.78
8.5000E+06	1.3799E+00	5.4729E-02	3.97
9.5000E+06	1.3762E+00	5.8606E-02	4.26
1.0500E+07	1.3586E+00	5.3134E-02	3.91
1.1500E+07	1.3707E+00	6.3632E-02	4.64
1.2500E+07	1.3821E+00	6.0293E-02	4.36
1.3250E+07	1.4807E+00	6.4031E-02	4.32
1.3750E+07	1.4699E+00	7.9208E-02	5.39
1.4250E+07	1.4440E+00	4.1561E-02	2.88
1.4750E+07	1.4357E+00	4.6956E-02	3.27
1.5500E+07	1.4386E+00	1.1050E-01	7.68
1.6500E+07	1.4452E+00	1.4716E-01	10.18
1.7500E+07	1.4561E+00	1.6027E-01	11.01
1.8500E+07	1.4324E+00	1.4764E-01	10.31
1.9500E+07	1.3910E+00	9.8809E-02	7.10
2.0000E+07	1.3756E+00	5.8574E-02	4.26

MT = 4 inelastic cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.5000E+06	4.5268E-01	2.6485E-02	5.85
1.7000E+06	4.9493E-01	2.0380E-02	4.12
1.9000E+06	5.7020E-01	2.3433E-02	4.11
2.1000E+06	6.3302E-01	2.9578E-02	4.67
2.3000E+06	7.6728E-01	3.8151E-02	4.97
2.5000E+06	8.8734E-01	3.2854E-02	3.70
2.7000E+06	9.3445E-01	3.2459E-02	3.47
2.9000E+06	1.0314E+00	3.5028E-02	3.40
3.2500E+06	1.1917E+00	3.7521E-02	3.15
3.7500E+06	1.2707E+00	4.6328E-02	3.65
4.0000E+06	1.3076E+00	4.1301E-02	3.16
4.2500E+06	1.3724E+00	6.2675E-02	4.57
4.7500E+06	1.5438E+00	5.2648E-02	3.41
5.2500E+06	1.4874E+00	6.4737E-02	4.35
5.7500E+06	1.5138E+00	6.3963E-02	4.23
6.5000E+06	1.4295E+00	5.7292E-02	4.01
7.5000E+06	1.3220E+00	5.3417E-02	4.04
8.5000E+06	1.2623E+00	5.4360E-02	4.31
9.5000E+06	1.2402E+00	5.8106E-02	4.69
1.0500E+07	1.2005E+00	5.2150E-02	4.34
1.1500E+07	1.1850E+00	6.2559E-02	5.28
1.2500E+07	1.0425E+00	6.2957E-02	6.04
1.3250E+07	9.4738E-01	5.2346E-02	5.53
1.3750E+07	8.0413E-01	6.6156E-02	8.23
1.4250E+07	6.6699E-01	2.1643E-02	3.24
1.4750E+07	5.7206E-01	5.4660E-02	9.55
1.5500E+07	4.7328E-01	5.4734E-02	11.56
1.6500E+07	3.9985E-01	5.6852E-02	14.22
1.7500E+07	3.5205E-01	5.1234E-02	14.55
1.8500E+07	3.0528E-01	4.8593E-02	15.92
1.9500E+07	2.6456E-01	4.6959E-02	17.75
2.0000E+07	2.4708E-01	4.4816E-02	18.14

MT = 16 (n,2n) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.1580E+01	0.0000E+00	0.000E+00	0.00
1.1696E+01	0.0000E+00	0.000E+00	0.00
1.1696E+01	2.6981E-03	5.456E-04	20.22
1.2000E+01	2.4788E-02	4.860E-03	19.61
1.2500E+01	8.7906E-02	1.626E-02	18.50
1.3000E+01	1.7847E-01	3.019E-02	16.92
1.3500E+01	2.7594E-01	4.009E-02	14.53
1.4000E+01	3.7196E-01	4.145E-02	11.14
1.4500E+01	4.4015E-01	4.876E-02	11.08
1.5000E+01	4.9880E-01	6.860E-02	13.75
1.5500E+01	5.5082E-01	9.072E-02	16.47
1.6000E+01	5.8541E-01	1.077E-01	18.40
1.6500E+01	6.0770E-01	1.196E-01	19.68
1.7000E+01	6.2940E-01	1.271E-01	20.20
1.7500E+01	6.4257E-01	1.302E-01	20.27
1.8000E+01	6.4355E-01	1.289E-01	20.03
1.8500E+01	6.3602E-01	1.240E-01	19.50
1.9000E+01	6.2465E-01	1.157E-01	18.52
1.9500E+01	6.0973E-01	1.036E-01	16.99
2.0000E+01	5.9712E-01	9.404E-02	15.75

MT = 22 (n,na) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
6.3969E+00	0.0000E+00	0.000E+00	0.00
1.0000E+01	5.1708E-07	5.171E-07	100.00
1.1000E+01	1.4667E-04	4.258E-05	29.03
1.2000E+01	2.2347E-03	6.575E-04	29.42
1.3000E+01	3.9640E-03	1.110E-03	28.01
1.4500E+01	1.1446E-02	2.999E-03	26.21
1.6000E+01	2.6254E-02	6.145E-03	23.41
1.7500E+01	4.5233E-02	9.767E-03	21.59
2.0000E+01	7.6578E-02	1.175E-02	15.35

MT = 28 (n,np) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
9.6935E+00	0.0000E+00	0.000E+00	0.00
1.1000E+01	0.0000E+00	0.000E+00	0.00
1.1000E+01	7.5940E-05	3.973E-05	52.31
1.1500E+01	2.9597E-03	1.574E-03	53.17
1.1696E+01	8.2996E-03	4.402E-03	53.04
1.2000E+01	1.7166E-02	8.933E-03	52.04
1.2500E+01	3.6196E-02	1.789E-02	49.42
1.3000E+01	6.2243E-02	2.225E-02	35.74
1.3500E+01	9.5693E-02	2.303E-02	24.07
1.4000E+01	1.3040E-01	2.693E-02	20.65
1.4500E+01	1.6144E-01	2.884E-02	17.87
1.5000E+01	1.8923E-01	3.312E-02	17.50
1.5500E+01	2.0920E-01	4.466E-02	21.35
1.6000E+01	2.2655E-01	5.548E-02	24.49
1.6500E+01	2.4125E-01	6.530E-02	27.07
1.7000E+01	2.5555E-01	7.394E-02	28.93
1.7500E+01	2.6716E-01	8.065E-02	30.19
1.8000E+01	2.7833E-01	8.524E-02	30.62
1.8500E+01	2.8970E-01	8.819E-02	30.44
1.9000E+01	2.9984E-01	8.954E-02	29.86
1.9500E+01	3.0720E-01	8.856E-02	28.83
2.0000E+01	3.1647E-01	8.835E-02	27.92

MT = 51 (n,n‘1) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.3524E+00	0.0000E+00	0.000E+00	0.00
1.3525E+00	5.0123E-02	6.386E-03	12.74
1.4150E+00	2.3161E-01	1.361E-02	5.88
1.5200E+00	5.0470E-01	2.613E-02	5.18
1.6512E+00	4.7656E-01	1.956E-02	4.11
2.0004E+00	6.0799E-01	2.842E-02	4.67
2.1966E+00	6.5730E-01	3.494E-02	5.32
2.3288E+00	6.5011E-01	3.263E-02	5.02
2.5525E+00	6.2570E-01	2.990E-02	4.78
2.6745E+00	5.9428E-01	2.909E-02	4.90
2.8798E+00	5.9730E-01	3.037E-02	5.08
2.9463E+00	5.8905E-01	2.924E-02	4.96
3.1728E+00	5.5610E-01	2.808E-02	5.05
3.2400E+00	5.3871E-01	2.852E-02	5.29
3.3254E+00	5.2932E-01	2.849E-02	5.38
3.3955E+00	5.1209E-01	2.767E-02	5.40
3.4383E+00	4.9976E-01	2.602E-02	5.21
3.6305E+00	4.2526E-01	2.160E-02	5.08
3.7321E+00	4.0223E-01	2.089E-02	5.19
3.7932E+00	3.9278E-01	2.095E-02	5.33
3.8267E+00	3.8075E-01	2.050E-02	5.38
3.9386E+00	3.6355E-01	2.119E-02	5.83
3.9904E+00	3.5452E-01	2.633E-02	7.43
4.0759E+00	3.4330E-01	2.817E-02	8.21
4.1074E+00	3.3698E-01	2.817E-02	8.36
4.1491E+00	3.1528E-01	2.834E-02	8.99
4.5000E+00	2.8151E-01	2.709E-02	9.62
5.0000E+00	2.3245E-01	1.318E-02	5.67
5.5000E+00	2.1101E-01	1.510E-02	7.16
6.0000E+00	1.6522E-01	2.265E-02	13.71
6.5000E+00	1.3913E-01	1.098E-02	7.89
7.0000E+00	1.1051E-01	9.275E-03	8.39
7.5000E+00	1.1140E-01	9.001E-03	8.08
8.0000E+00	9.5152E-02	5.946E-03	6.25
8.5000E+00	8.1169E-02	8.456E-03	10.42
9.0000E+00	7.5533E-02	1.554E-02	20.57
9.4890E+00	6.9076E-02	1.391E-02	20.14
1.0000E+01	7.0174E-02	5.180E-03	7.38
1.0500E+01	6.5763E-02	1.351E-02	20.54
1.0683E+01	6.5505E-02	1.486E-02	22.69
1.1000E+01	6.3147E-02	1.532E-02	24.27
1.1500E+01	6.0003E-02	1.346E-02	22.43
1.1696E+01	6.0060E-02	1.052E-02	17.52
1.1994E+01	6.0468E-02	4.284E-03	7.09
1.2500E+01	5.9579E-02	8.823E-03	14.81
1.2961E+01	6.0626E-02	9.731E-03	16.05
1.3500E+01	6.0486E-02	8.020E-03	13.26
1.4000E+01	5.9751E-02	4.710E-03	7.88
1.4500E+01	5.5416E-02	8.242E-03	14.87
1.5000E+01	5.1295E-02	9.716E-03	18.94
1.5500E+01	4.6041E-02	1.040E-02	22.58
1.6000E+01	4.2831E-02	1.083E-02	25.29
1.6500E+01	4.2180E-02	1.126E-02	26.69
1.7000E+01	4.1142E-02	1.102E-02	26.78

1.7500E+01	4.0560E-02	1.040E-02	25.64
1.8000E+01	4.0883E-02	9.537E-03	23.33
1.8500E+01	4.1111E-02	8.443E-03	20.54
1.9020E+01	4.1278E-02	7.064E-03	17.11
1.9500E+01	4.1457E-02	5.241E-03	12.64
2.0000E+01	4.1552E-02	2.165E-03	5.21

MT = 851 (n,n‘2-3) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
2.1964E+00	0.0000E+00	0.000E+00	0.00
2.2500E+00	8.0522E-02	1.250E-02	15.52
2.5000E+00	2.5591E-01	2.034E-02	7.95
2.7500E+00	2.9784E-01	2.176E-02	7.31
3.0000E+00	3.2138E-01	2.557E-02	7.96
3.2500E+00	3.5250E-01	2.503E-02	7.10
3.5000E+00	3.4415E-01	2.370E-02	6.89
3.7500E+00	3.0647E-01	2.114E-02	6.90
4.0000E+00	2.4524E-01	1.604E-02	6.54
4.2500E+00	1.9360E-01	1.435E-02	7.41
4.5000E+00	1.4918E-01	1.362E-02	9.13
4.7500E+00	1.3123E-01	1.243E-02	9.47
5.0000E+00	1.2638E-01	8.428E-03	6.67
5.5000E+00	9.6227E-02	7.014E-03	7.29
6.0000E+00	6.5301E-02	7.400E-03	11.33
6.5000E+00	4.3404E-02	6.332E-03	14.59
7.0000E+00	2.8557E-02	5.868E-03	20.55
7.5000E+00	1.9764E-02	5.054E-03	25.57
8.0000E+00	1.3401E-02	4.352E-03	32.48
8.5000E+00	8.9855E-03	3.482E-03	38.75
9.0000E+00	5.9418E-03	2.725E-03	45.86
9.5000E+00	3.8953E-03	2.057E-03	52.80
1.0000E+01	2.5890E-03	1.527E-03	58.97
1.0500E+01	1.7524E-03	1.136E-03	64.80
1.1000E+01	1.2038E-03	8.420E-04	69.95
1.1500E+01	8.3369E-04	6.251E-04	74.98
1.2000E+01	5.7832E-04	4.623E-04	79.94
1.2500E+01	4.0565E-04	3.449E-04	85.04
1.3000E+01	2.8714E-04	2.579E-04	89.81
1.3500E+01	2.0359E-04	1.827E-04	89.72
1.4000E+01	1.4522E-04	1.302E-04	89.66
1.4500E+01	1.0367E-04	9.300E-05	89.70
1.5000E+01	7.4225E-05	6.666E-05	89.81
1.6000E+01	3.8657E-05	3.476E-05	89.91
1.7000E+01	2.0321E-05	1.829E-05	89.99
1.8000E+01	1.0805E-05	9.725E-06	90.00
1.9000E+01	5.7961E-06	5.216E-06	90.00
2.0000E+01	3.1623E-06	2.944E-06	93.10

MT = 852 (n,n‘4-5) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
2.5522E+00	0.0000E+00	0.000E+00	0.00
2.7500E+00	6.7376E-02	8.577E-03	12.73
3.0000E+00	1.6290E-01	1.560E-02	9.58
3.2500E+00	2.2597E-01	2.017E-02	8.92
3.5000E+00	2.6746E-01	2.284E-02	8.54
3.7500E+00	2.8239E-01	2.091E-02	7.40
4.0000E+00	2.7390E-01	2.067E-02	7.55
4.2500E+00	2.3340E-01	1.932E-02	8.28
4.5000E+00	2.0257E-01	1.831E-02	9.04
4.7500E+00	1.8373E-01	1.659E-02	9.03
5.0000E+00	1.6509E-01	1.076E-02	6.52
5.5000E+00	1.2779E-01	9.100E-03	7.12
6.0000E+00	9.4193E-02	9.497E-03	10.08
6.5000E+00	7.3154E-02	7.895E-03	10.79
7.0000E+00	5.7963E-02	7.329E-03	12.64
7.5000E+00	4.3699E-02	6.721E-03	15.38
8.0000E+00	3.5043E-02	6.411E-03	18.29
8.5000E+00	2.5096E-02	6.118E-03	24.38
9.0000E+00	1.7930E-02	6.847E-03	38.19
9.5000E+00	1.3366E-02	6.187E-03	46.29
1.0000E+01	9.9240E-03	5.464E-03	55.06
1.0500E+01	8.0565E-03	4.782E-03	59.35
1.1000E+01	7.0145E-03	4.206E-03	59.96
1.1500E+01	6.3174E-03	3.780E-03	59.84
1.2000E+01	5.8551E-03	3.464E-03	59.17
1.2500E+01	5.4820E-03	3.228E-03	58.89
1.3000E+01	5.1489E-03	2.990E-03	58.08
1.3500E+01	4.8439E-03	2.787E-03	57.53
1.4000E+01	4.5731E-03	2.616E-03	57.21
1.4500E+01	4.2860E-03	2.471E-03	57.66
1.5000E+01	4.0204E-03	2.346E-03	58.34
1.6000E+01	3.5231E-03	2.589E-03	73.48
1.7000E+01	3.2244E-03	2.866E-03	88.89
1.8000E+01	3.1802E-03	2.820E-03	88.68
1.9000E+01	3.0979E-03	2.747E-03	88.69
2.0000E+01	3.0283E-03	2.691E-03	88.85

MT = 853 (n,n‘6-11) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
3.1725E+00	0.0000E+00	0.000E+00	0.00
3.2500E+00	7.5640E-02	2.059E-02	27.22
3.5000E+00	2.1552E-01	5.223E-02	24.23
3.7500E+00	1.9448E-01	5.987E-02	30.78
4.0000E+00	2.1353E-01	7.091E-02	33.21
4.2500E+00	2.5288E-01	8.183E-02	32.36
4.5000E+00	2.7811E-01	8.507E-02	30.59
4.7500E+00	2.9936E-01	8.309E-02	27.76
5.0000E+00	2.6648E-01	7.956E-02	29.86
5.5000E+00	2.3827E-01	5.758E-02	24.17
6.0000E+00	1.9584E-01	3.447E-02	17.60
6.5000E+00	1.6335E-01	1.498E-02	9.17
7.0000E+00	1.1502E-01	1.180E-02	10.26
7.5000E+00	8.8840E-02	1.052E-02	11.84
8.0000E+00	7.7517E-02	1.018E-02	13.14
8.5000E+00	5.3282E-02	9.033E-03	16.95
9.0000E+00	3.6220E-02	1.533E-02	42.33
9.5000E+00	2.3221E-02	1.403E-02	60.43
1.0000E+01	1.4651E-02	1.178E-02	80.40
1.0500E+01	1.0658E-02	9.664E-03	90.67
1.1000E+01	8.4473E-03	7.590E-03	89.85
1.1500E+01	6.8062E-03	6.083E-03	89.38
1.2000E+01	5.6930E-03	5.003E-03	87.88
1.2500E+01	4.8164E-03	4.239E-03	88.02
1.3000E+01	4.3076E-03	3.681E-03	85.44
1.3500E+01	3.8654E-03	3.248E-03	84.02
1.4000E+01	3.5097E-03	2.921E-03	83.22
1.4500E+01	3.1667E-03	2.667E-03	84.23
1.5000E+01	2.8668E-03	2.466E-03	86.01
1.6000E+01	2.4110E-03	2.118E-03	87.84
1.7000E+01	2.1298E-03	1.902E-03	89.30
1.8000E+01	1.9464E-03	1.736E-03	89.21
1.9000E+01	1.8781E-03	1.675E-03	89.19
2.0000E+01	1.7873E-03	1.597E-03	89.36

MT = 91 (n,n'cont) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
3.3952E+00	0.0000E+00	0.000E+00	0.00
3.5000E+00	2.4591E-02	1.594E-02	64.81
3.7500E+00	8.7919E-02	4.652E-02	52.92
4.0000E+00	2.2168E-01	6.524E-02	29.43
4.2500E+00	3.8766E-01	8.748E-02	22.57
4.5000E+00	5.3668E-01	9.139E-02	17.03
4.7500E+00	6.7430E-01	9.363E-02	13.88
5.0000E+00	6.5273E-01	9.198E-02	14.09
5.5000E+00	8.5943E-01	7.974E-02	9.28
6.0000E+00	9.7802E-01	7.100E-02	7.26
6.5000E+00	1.0105E+00	5.958E-02	5.90
7.0000E+00	1.0474E+00	3.967E-02	3.79
7.5000E+00	1.0583E+00	5.501E-02	5.20
8.0000E+00	1.1419E+00	6.097E-02	5.34
8.5000E+00	1.0938E+00	5.560E-02	5.08
9.0000E+00	1.1016E+00	7.786E-02	7.07
9.5000E+00	1.1306E+00	6.042E-02	5.34
1.0000E+01	1.1985E+00	6.292E-02	5.25
1.0500E+01	1.1143E+00	5.421E-02	4.86
1.1000E+01	1.1633E+00	7.722E-02	6.64
1.1500E+01	1.1110E+00	6.360E-02	5.72
1.2000E+01	1.1198E+00	6.524E-02	5.83
1.2500E+01	9.7225E-01	6.357E-02	6.54
1.3000E+01	9.3174E-01	5.627E-02	6.04
1.3500E+01	8.2326E-01	7.394E-02	8.98
1.4000E+01	6.4763E-01	2.290E-02	3.54
1.4500E+01	5.5539E-01	5.962E-02	10.73
1.5000E+01	4.6750E-01	6.118E-02	13.09
1.6000E+01	3.7404E-01	6.089E-02	16.28
1.7000E+01	3.2995E-01	5.528E-02	16.75
1.8000E+01	2.8252E-01	5.274E-02	18.67
1.9000E+01	2.3569E-01	5.019E-02	21.29
2.0000E+01	2.0071E-01	4.447E-02	22.16

MT = 102 (n,cap) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.0000E-11	0.0000E+00	0.000E+00	0.00
1.0000E-02	0.0000E+00	0.000E+00	0.00
1.0000E-02	0.0000E+00	0.000E+00	0.00
1.0000E-02	1.0000E-04	2.000E-05	20.00
1.4000E-02	1.0700E-04	2.140E-05	20.00
2.0000E-02	1.1500E-04	2.300E-05	20.00
2.4000E-02	1.2200E-04	2.440E-05	20.00
3.0000E-02	1.3000E-04	2.600E-05	20.00
3.4000E-02	1.3500E-04	2.700E-05	20.00
4.0000E-02	1.4300E-04	2.860E-05	20.00
4.4000E-02	1.4900E-04	2.980E-05	20.00
5.0000E-02	1.5600E-04	3.120E-05	20.00
6.0000E-02	1.7373E-04	3.475E-05	20.00
7.0000E-02	1.7945E-04	3.589E-05	20.00
8.0000E-02	1.9376E-04	3.875E-05	20.00
9.0000E-02	2.0906E-04	4.181E-05	20.00
1.0000E-01	2.2520E-04	4.504E-05	20.00
1.2000E-01	2.7471E-04	5.494E-05	20.00
1.4000E-01	3.4218E-04	6.844E-05	20.00
1.6000E-01	4.1990E-04	8.398E-05	20.00
1.8000E-01	5.1572E-04	1.031E-04	20.00
2.0000E-01	6.2166E-04	1.243E-04	20.00
2.2000E-01	7.2901E-04	1.458E-04	20.00
2.4000E-01	8.3714E-04	1.674E-04	20.00
2.6000E-01	9.5386E-04	1.908E-04	20.00
2.8000E-01	1.0617E-03	2.123E-04	20.00
3.0000E-01	1.1602E-03	2.320E-04	20.00
3.2000E-01	1.2573E-03	2.515E-04	20.00
3.4000E-01	1.3444E-03	2.689E-04	20.00
3.6000E-01	1.4214E-03	2.843E-04	20.00
3.8000E-01	1.4884E-03	2.977E-04	20.00
4.0000E-01	1.5541E-03	3.108E-04	20.00
4.2000E-01	1.6188E-03	3.238E-04	20.00
4.4000E-01	1.6657E-03	3.331E-04	20.00
4.6000E-01	1.7204E-03	3.441E-04	20.00
4.8000E-01	1.7665E-03	3.533E-04	20.00
5.0000E-01	1.8124E-03	3.625E-04	20.00
5.2000E-01	1.8500E-03	3.700E-04	20.00
5.3000E-01	1.8731E-03	3.746E-04	20.00
5.4000E-01	1.8961E-03	3.792E-04	20.00
5.5000E-01	1.9152E-03	5.748E-04	30.01
5.5000E-01	7.8511E-03	2.368E-03	30.16
6.0000E-01	7.8092E-03	2.357E-03	30.19
6.5000E-01	7.8017E-03	2.357E-03	30.21
7.0000E-01	7.8229E-03	2.365E-03	30.23
7.5000E-01	7.8692E-03	2.380E-03	30.25
8.0000E-01	7.9371E-03	2.403E-03	30.27
8.5000E-01	8.0246E-03	2.431E-03	30.29
9.0000E-01	8.1298E-03	2.464E-03	30.31
9.5000E-01	8.2511E-03	2.503E-03	30.33
1.0000E+00	8.3816E-03	2.626E-03	31.33
1.1000E+00	8.6866E-03	2.896E-03	33.33
1.2000E+00	9.0414E-03	3.195E-03	35.33
1.2500E+00	9.2334E-03	3.356E-03	36.34

1.3000E+00	9.4366E-03	3.525E-03	37.35
1.3549E+00	9.6714E-03	3.719E-03	38.46
1.4000E+00	7.2963E-03	2.872E-03	39.37
1.4500E+00	6.6291E-03	2.677E-03	40.38
1.5000E+00	6.1858E-03	2.561E-03	41.40
1.5500E+00	5.8435E-03	2.479E-03	42.42
1.6000E+00	5.5788E-03	2.424E-03	43.46
1.7000E+00	5.2328E-03	2.381E-03	45.50
1.8000E+00	5.0460E-03	2.398E-03	47.53
1.9000E+00	4.9612E-03	2.458E-03	49.54
2.0000E+00	4.9499E-03	2.551E-03	51.53
2.1000E+00	4.9973E-03	2.573E-03	51.49
2.1951E+00	5.0825E-03	2.615E-03	51.45
2.3000E+00	4.7370E-03	2.435E-03	51.40
2.4000E+00	4.5786E-03	2.339E-03	51.08
2.5480E+00	4.5095E-03	2.293E-03	50.84
2.6000E+00	4.0706E-03	2.064E-03	50.71
2.6702E+00	3.8946E-03	1.971E-03	50.60
2.7000E+00	3.7499E-03	1.893E-03	50.47
2.8000E+00	3.5528E-03	1.786E-03	50.28
2.9000E+00	3.4423E-03	1.724E-03	50.09
3.0000E+00	3.3969E-03	1.693E-03	49.84
3.1722E+00	3.3841E-03	1.679E-03	49.63
3.2397E+00	3.2665E-03	1.617E-03	49.50
3.3246E+00	3.1026E-03	1.534E-03	49.43
3.3738E+00	3.0155E-03	1.489E-03	49.37
3.4379E+00	2.9064E-03	1.433E-03	49.30
3.5000E+00	2.7950E-03	1.375E-03	49.20
3.6301E+00	2.6895E-03	1.319E-03	49.03
3.7990E+00	2.4983E-03	1.222E-03	48.91
4.0000E+00	2.3358E-03	1.142E-03	48.88
4.0070E+00	2.3349E-03	1.140E-03	48.83
4.5000E+00	1.9027E-03	9.321E-04	48.99
5.0000E+00	1.6037E-03	7.900E-04	49.26
5.5000E+00	1.3400E-03	6.625E-04	49.44
6.0000E+00	1.1169E-03	5.534E-04	49.54
6.5000E+00	1.0208E-03	5.069E-04	49.66
7.0000E+00	9.3194E-04	4.638E-04	49.77
7.5000E+00	8.6335E-04	4.303E-04	49.84
8.0000E+00	8.0231E-04	3.999E-04	49.85
8.5000E+00	7.4629E-04	3.721E-04	49.86
9.0000E+00	6.9480E-04	3.464E-04	49.86
9.5000E+00	6.4750E-04	3.227E-04	49.84
1.0000E+01	6.0390E-04	3.010E-04	49.84
1.0500E+01	5.6499E-04	2.820E-04	49.91
1.1000E+01	5.3477E-04	2.672E-04	49.96
1.1500E+01	5.1648E-04	2.582E-04	49.98
1.1584E+01	5.1494E-04	2.575E-04	50.00
1.2000E+01	5.1634E-04	2.583E-04	50.02
1.2500E+01	5.4022E-04	2.703E-04	50.04
1.3000E+01	5.9291E-04	3.102E-04	52.32
1.3500E+01	6.7063E-04	3.660E-04	54.57
1.4000E+01	7.5960E-04	4.316E-04	56.82
1.4500E+01	8.3788E-04	4.954E-04	59.12
1.5000E+01	8.8811E-04	5.456E-04	61.43
1.5500E+01	9.0515E-04	5.771E-04	63.75
1.6000E+01	8.9448E-04	5.912E-04	66.09
1.6500E+01	8.6635E-04	5.929E-04	68.44
1.7000E+01	8.3007E-04	5.874E-04	70.77
1.7250E+01	8.1108E-04	5.834E-04	71.93

1.7500E+01	7.9198E-04	5.789E-04	73.10
1.8000E+01	7.5651E-04	5.706E-04	75.42
1.8500E+01	7.2631E-04	5.647E-04	77.75
1.9000E+01	7.0252E-04	5.626E-04	80.09
1.9500E+01	6.7397E-04	5.399E-04	80.11
2.0000E+01	6.7330E-04	5.395E-04	80.12

MT = 103 (n,p) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
2.2000E+00	0.0000E+00	0.000E+00	0.00
2.5000E+00	1.4736E-12	9.574E-13	64.97
2.7500E+00	4.3855E-09	2.846E-09	64.90
3.0000E+00	3.5435E-07	2.294E-07	64.75
3.5000E+00	3.6663E-05	2.361E-05	64.39
4.0000E+00	4.4344E-04	2.836E-04	63.96
4.5000E+00	3.7908E-03	1.763E-03	46.51
5.0000E+00	9.4715E-03	2.772E-03	29.27
5.5000E+00	1.7134E-02	2.040E-03	11.91
6.0000E+00	2.6792E-02	2.659E-03	9.93
6.5000E+00	3.7863E-02	3.010E-03	7.95
7.0000E+00	4.9466E-02	3.932E-03	7.95
7.5000E+00	6.0794E-02	4.831E-03	7.95
8.0000E+00	7.1283E-02	5.649E-03	7.92
8.5000E+00	8.0481E-02	6.369E-03	7.91
9.0000E+00	8.8642E-02	7.005E-03	7.90
9.5000E+00	9.6366E-02	7.601E-03	7.89
1.0000E+01	1.0417E-01	8.709E-03	8.36
1.0500E+01	1.1201E-01	9.972E-03	8.90
1.1000E+01	1.2095E-01	1.082E-02	8.94
1.1500E+01	1.3108E-01	1.175E-02	8.96
1.2000E+01	1.4177E-01	1.271E-02	8.97
1.2250E+01	1.4663E-01	1.317E-02	8.98
1.2500E+01	1.5102E-01	1.236E-02	8.18
1.2750E+01	1.5452E-01	1.140E-02	7.38
1.3000E+01	1.5696E-01	1.032E-02	6.57
1.3250E+01	1.5775E-01	9.107E-03	5.77
1.3500E+01	1.5674E-01	7.794E-03	4.97
1.3750E+01	1.5412E-01	7.274E-03	4.72
1.4000E+01	1.5020E-01	6.707E-03	4.47
1.4500E+01	1.3938E-01	5.530E-03	3.97
1.5000E+01	1.2770E-01	6.744E-03	5.28
1.5500E+01	1.1616E-01	7.679E-03	6.61
1.6000E+01	1.0574E-01	8.409E-03	7.95
1.6500E+01	9.6917E-02	8.372E-03	8.64
1.7000E+01	8.9567E-02	8.349E-03	9.32
1.7500E+01	8.3485E-02	8.350E-03	10.00
1.8000E+01	7.8388E-02	8.633E-03	11.01
1.8500E+01	7.4166E-02	8.916E-03	12.02
1.9000E+01	7.0559E-02	9.196E-03	13.03
1.9500E+01	6.7457E-02	9.475E-03	14.05
2.0000E+01	6.4763E-02	9.747E-03	15.05

MT = 104 (n,d) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
7.4313E+00	0.0000E+00	0.000E+00	0.00
9.0413E+00	1.7540E-07	1.595E-07	90.93
9.5000E+00	4.2337E-04	1.260E-04	29.76
1.0000E+01	1.9555E-03	5.108E-04	26.12
1.0500E+01	2.7406E-03	6.703E-04	24.46
1.1000E+01	3.3110E-03	7.814E-04	23.60
1.1500E+01	3.8402E-03	9.048E-04	23.56
1.2000E+01	4.4287E-03	9.197E-04	20.77
1.2500E+01	5.3504E-03	1.079E-03	20.16
1.3000E+01	6.4766E-03	1.277E-03	19.72
1.3500E+01	7.6170E-03	1.482E-03	19.46
1.4000E+01	8.9470E-03	2.164E-03	24.19
1.4500E+01	1.0201E-02	2.312E-03	22.66
1.5000E+01	1.1737E-02	2.545E-03	21.68
1.5500E+01	1.2635E-02	2.820E-03	22.32
1.6000E+01	1.4601E-02	4.467E-03	30.59
1.6500E+01	1.6002E-02	4.736E-03	29.60
1.7000E+01	1.7380E-02	5.008E-03	28.82
1.7500E+01	1.8935E-02	5.324E-03	28.12
1.8000E+01	2.0268E-02	5.600E-03	27.63
1.8500E+01	2.1382E-02	5.834E-03	27.28
1.9000E+01	2.2096E-02	5.986E-03	27.09
1.9500E+01	2.2613E-02	6.097E-03	26.96
2.0000E+01	2.2825E-02	6.144E-03	26.92

MT = 105 (n,t) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.1694E+01	0.0000E+00	0.000E+00	0.00
1.4000E+01	1.1823E-07	4.126E-08	34.90
1.4500E+01	3.4371E-05	9.927E-06	28.88
1.5000E+01	2.4934E-04	8.550E-05	34.29
1.6000E+01	1.5615E-03	7.725E-04	49.47
1.7000E+01	4.0234E-03	2.611E-03	64.90
1.8000E+01	9.2037E-03	6.963E-03	75.66
1.9000E+01	1.6803E-02	1.445E-02	86.01
2.0000E+01	2.6296E-02	2.526E-02	96.08

MT = 106 (n,3He) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
9.3286E+00	0.0000E+00	0.000E+00	0.00
9.6980E+00	9.8934E-27	8.904E-27	90.00
1.0000E+01	4.0584E-25	3.653E-25	90.00
1.0494E+01	1.0560E-22	9.504E-23	90.00
1.1000E+01	2.4833E-20	2.235E-20	90.00
1.1585E+01	1.2075E-17	1.087E-17	90.00
1.1694E+01	3.5965E-17	3.237E-17	90.00
1.2000E+01	9.4006E-16	8.461E-16	90.00
1.3000E+01	3.0144E-11	2.713E-11	90.00
1.3500E+01	3.8242E-09	3.442E-09	90.00
1.4000E+01	3.0000E-07	2.700E-07	90.00
1.4500E+01	1.9331E-06	1.740E-06	90.00
1.5000E+01	6.0002E-06	5.400E-06	90.00
1.5500E+01	1.0360E-05	9.324E-06	90.00
1.6000E+01	1.6001E-05	1.440E-05	90.00
1.6500E+01	2.3821E-05	2.144E-05	90.00
1.7000E+01	3.5031E-05	3.153E-05	90.00
1.7500E+01	4.8957E-05	4.406E-05	90.00
1.8000E+01	6.2993E-05	5.670E-05	90.01
1.8500E+01	7.4298E-05	6.688E-05	90.01
1.9000E+01	8.2393E-05	7.417E-05	90.02
1.9500E+01	8.7139E-05	7.844E-05	90.02
2.0000E+01	8.8974E-05	8.008E-05	90.01

MT = 107 (n,a) cross section

Neutron energy (in MeV)	cross section (in barn)	std.dev. of sigma(E) (in barn)	std.dev. (in %)
1.0000E-11	0.0000E+00	0.000E+00	0.00
2.5525E+00	0.0000E+00	0.000E+00	0.00
2.5525E+00	3.0424E-06	1.081E-06	35.53
2.5553E+00	3.0889E-06	1.071E-06	34.68
2.6435E+00	9.2899E-06	3.107E-06	33.44
2.6745E+00	1.0899E-05	3.397E-06	31.17
2.8798E+00	2.6301E-05	7.486E-06	28.46
2.9128E+00	3.0357E-05	8.386E-06	27.63
2.9463E+00	3.7685E-05	9.967E-06	26.45
3.0014E+00	5.0110E-05	1.248E-05	24.90
3.0575E+00	6.5835E-05	1.468E-05	22.31
3.1728E+00	1.0478E-04	1.810E-05	17.27
3.3254E+00	1.7564E-04	2.646E-05	15.07
3.4494E+00	2.4686E-04	3.940E-05	15.96
3.7321E+00	5.0494E-04	4.994E-05	9.89
3.9610E+00	8.0619E-04	1.237E-04	15.34
4.1491E+00	1.0097E-03	1.036E-04	10.26
4.5000E+00	2.2670E-03	2.692E-04	11.87
5.0000E+00	5.3152E-03	6.108E-04	11.49
5.2440E+00	6.5461E-03	3.642E-04	5.56
5.5000E+00	8.0482E-03	1.535E-03	19.07
6.0000E+00	1.5466E-02	1.624E-03	10.50
6.5000E+00	2.3319E-02	2.092E-03	8.97
7.0000E+00	2.9017E-02	2.595E-03	8.94
7.5000E+00	3.3890E-02	3.147E-03	9.29
8.0000E+00	3.5129E-02	3.388E-03	9.64
8.5000E+00	3.6344E-02	3.655E-03	10.06
9.0000E+00	3.8652E-02	3.839E-03	9.93
9.4890E+00	3.8333E-02	3.704E-03	9.66
1.0000E+01	4.6976E-02	3.063E-03	6.52
1.0500E+01	4.2685E-02	4.403E-03	10.31
1.1000E+01	5.1019E-02	3.849E-03	7.54
1.1500E+01	4.6173E-02	4.726E-03	10.24
1.1994E+01	5.5111E-02	4.556E-03	8.27
1.2500E+01	5.5450E-02	4.401E-03	7.94
1.2961E+01	5.6566E-02	3.617E-03	6.39
1.3500E+01	5.6505E-02	3.635E-03	6.43
1.4000E+01	5.9333E-02	3.307E-03	5.57
1.4500E+01	6.0023E-02	4.287E-03	7.14
1.5000E+01	5.8753E-02	4.783E-03	8.14
1.5500E+01	5.3655E-02	5.398E-03	10.06
1.6000E+01	5.2743E-02	8.192E-03	15.53
1.6500E+01	4.8036E-02	7.792E-03	16.22
1.7000E+01	4.3725E-02	1.016E-02	23.25
1.7500E+01	3.9924E-02	9.726E-03	24.36
1.8000E+01	3.7170E-02	1.000E-02	26.91
1.8500E+01	3.4867E-02	8.672E-03	24.87
1.9020E+01	3.0876E-02	9.392E-03	30.42
1.9500E+01	2.7387E-02	9.018E-03	32.93
2.0000E+01	2.3686E-02	8.818E-03	37.23

Fig. 1 ^{60}Ni evaluation-flow chart

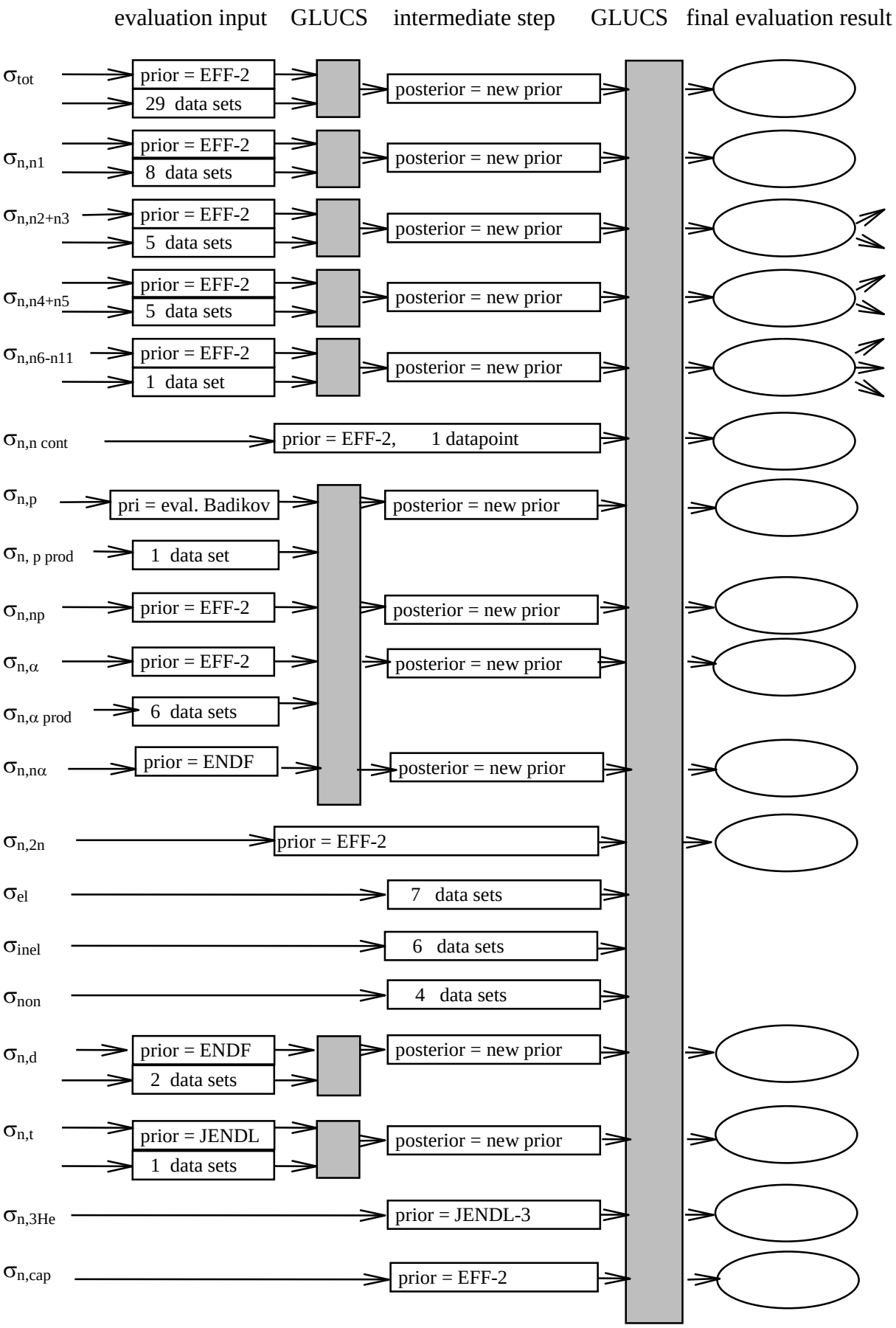


Fig. 2: Correction factor for conversion of $\sigma(^{1}2^{+} \rightarrow$ g.s.) to σ_{inel}

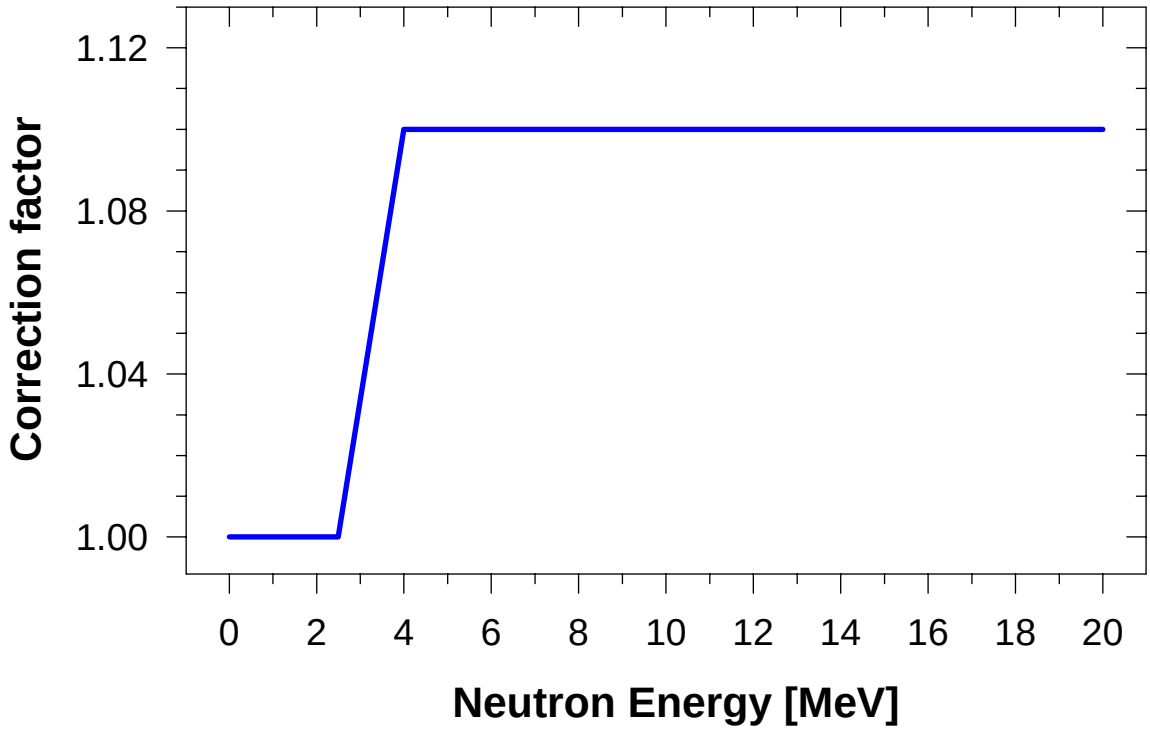
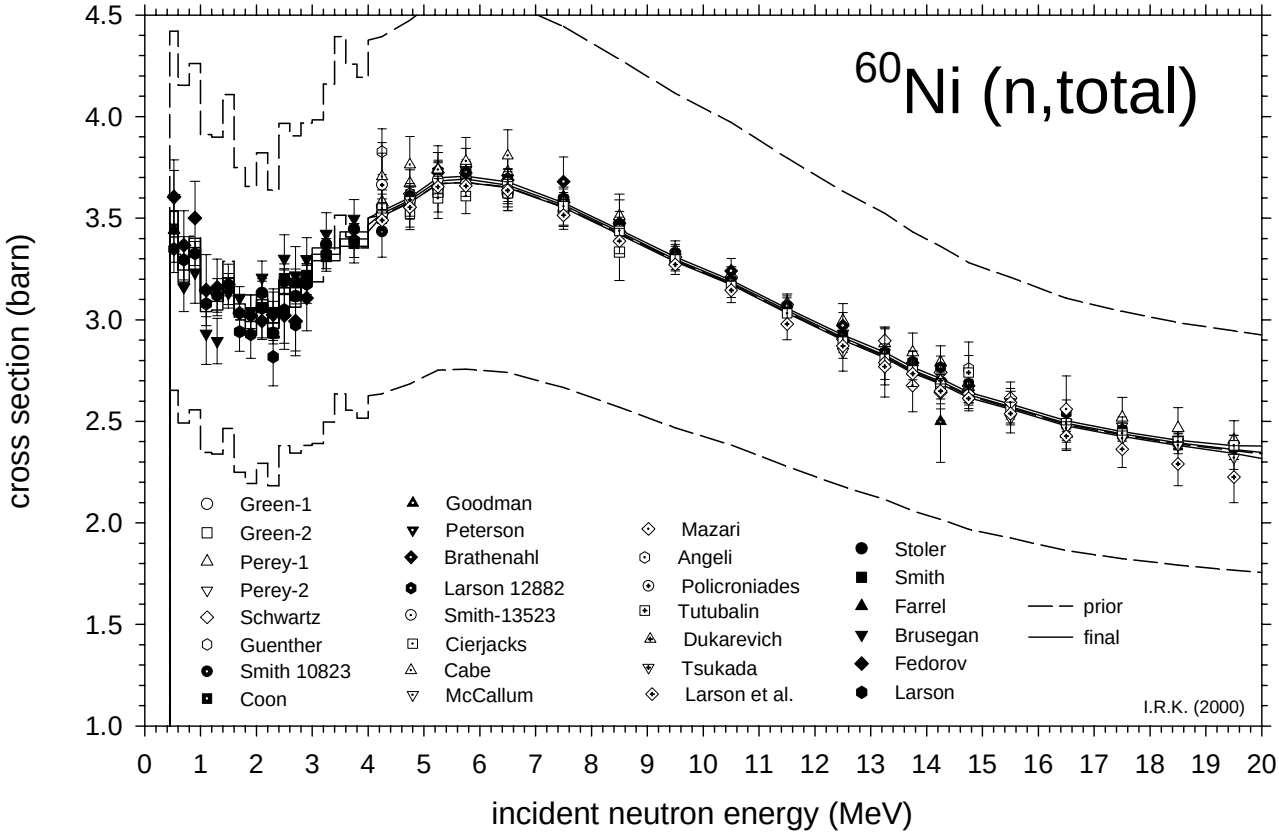


Fig 3: Total cross section and comparison with ENDF/B-VI



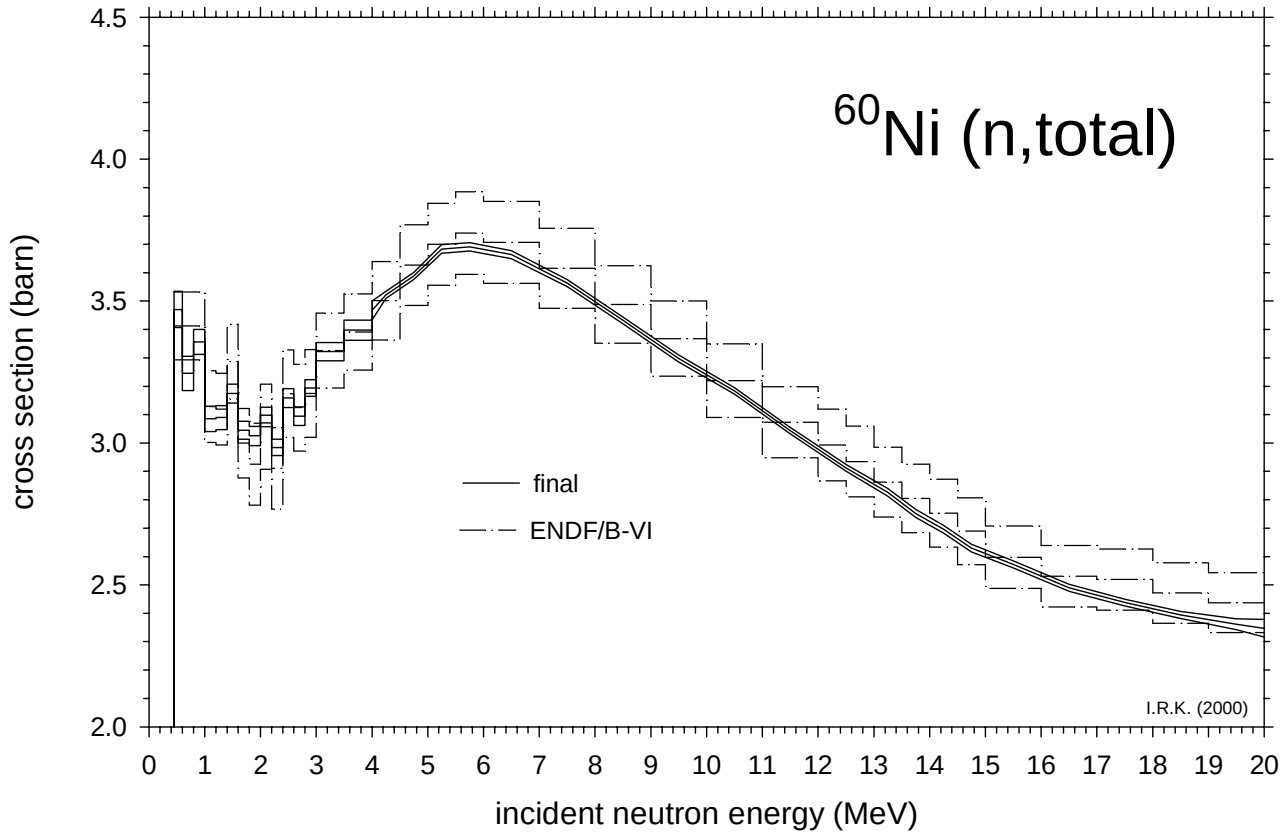
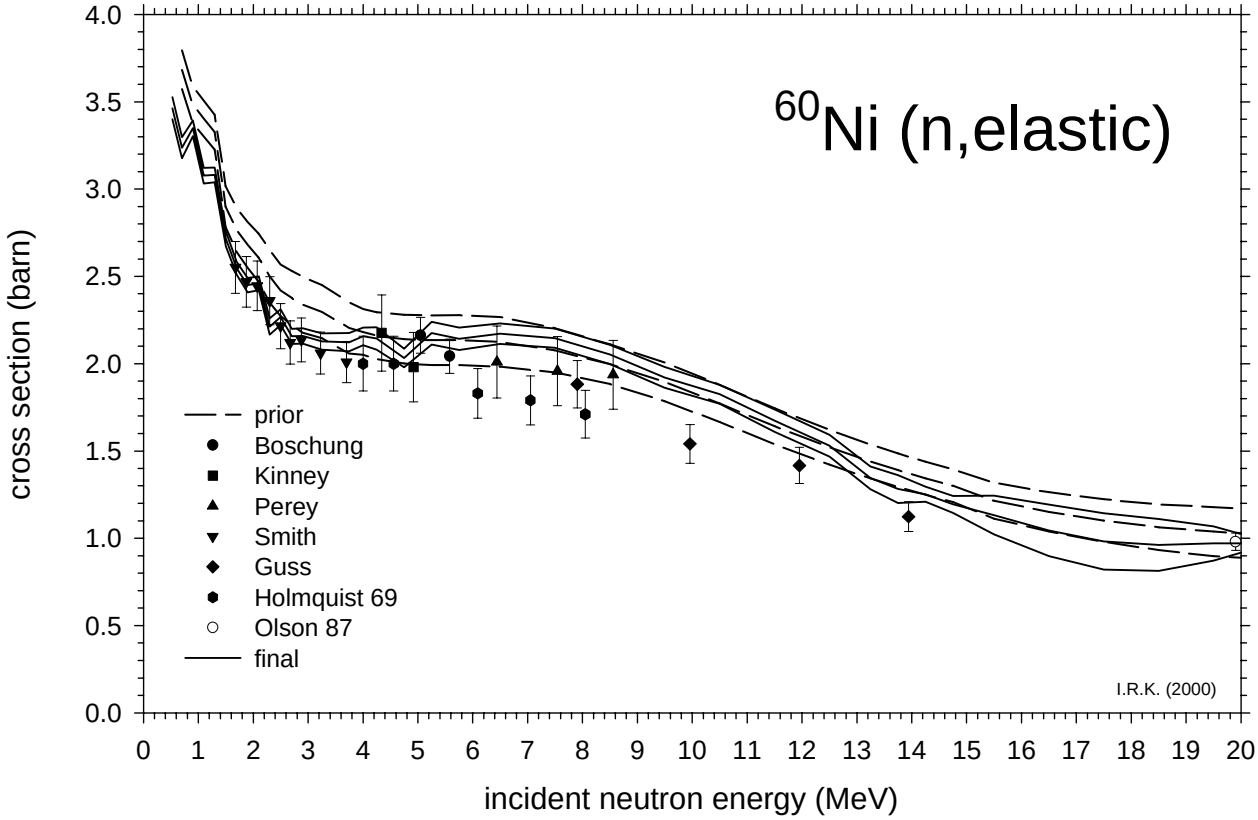


Fig 4: Elastic cross section and comparison with ENDF/B-VI



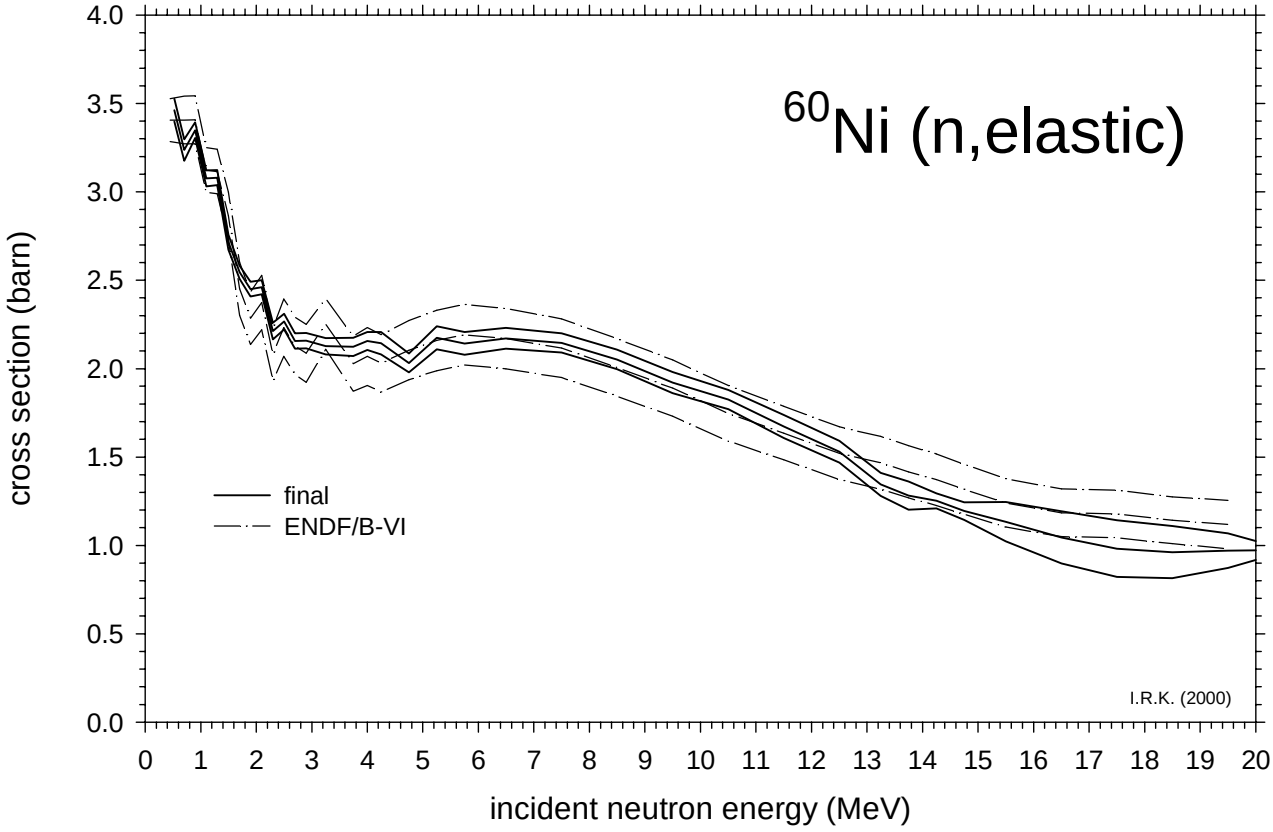
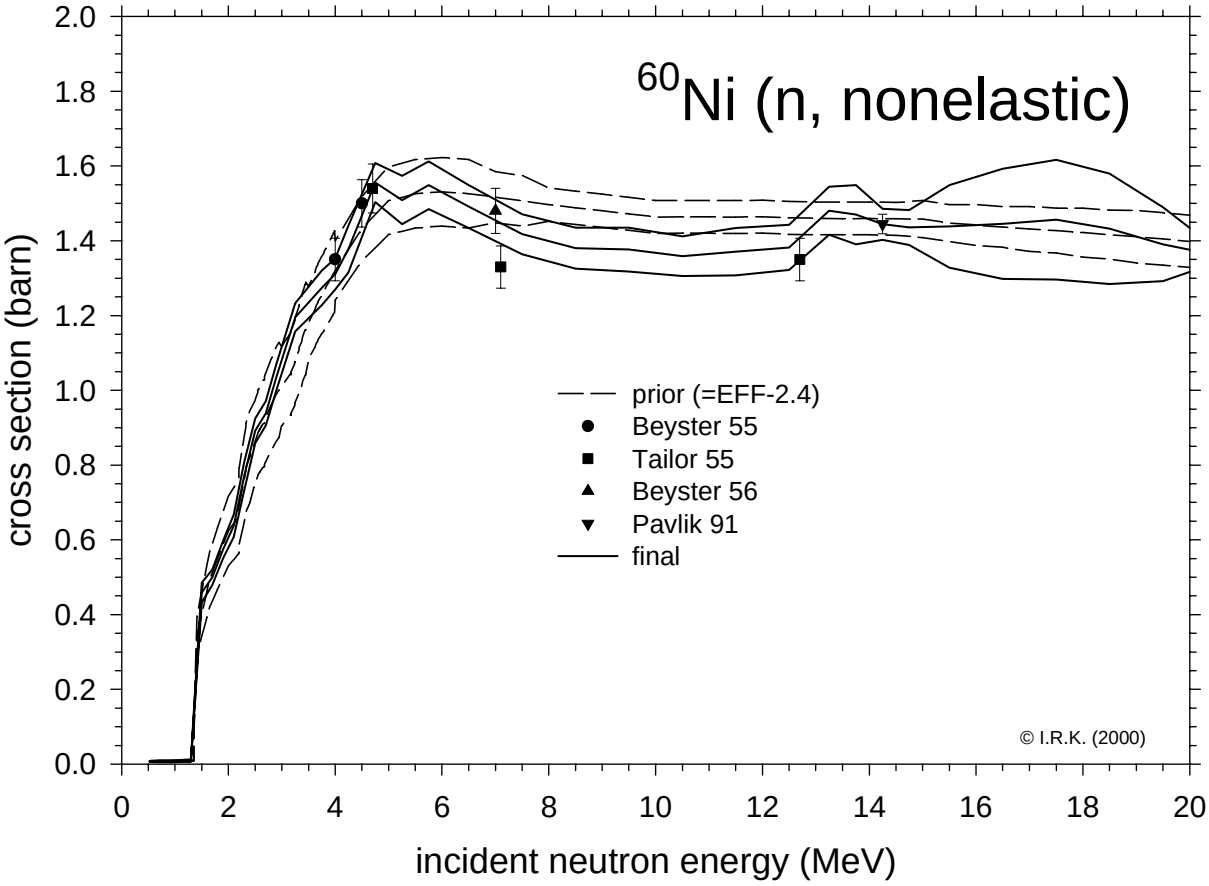


Fig 5: Nonelastic cross section and comparison with ENDF/B-VI



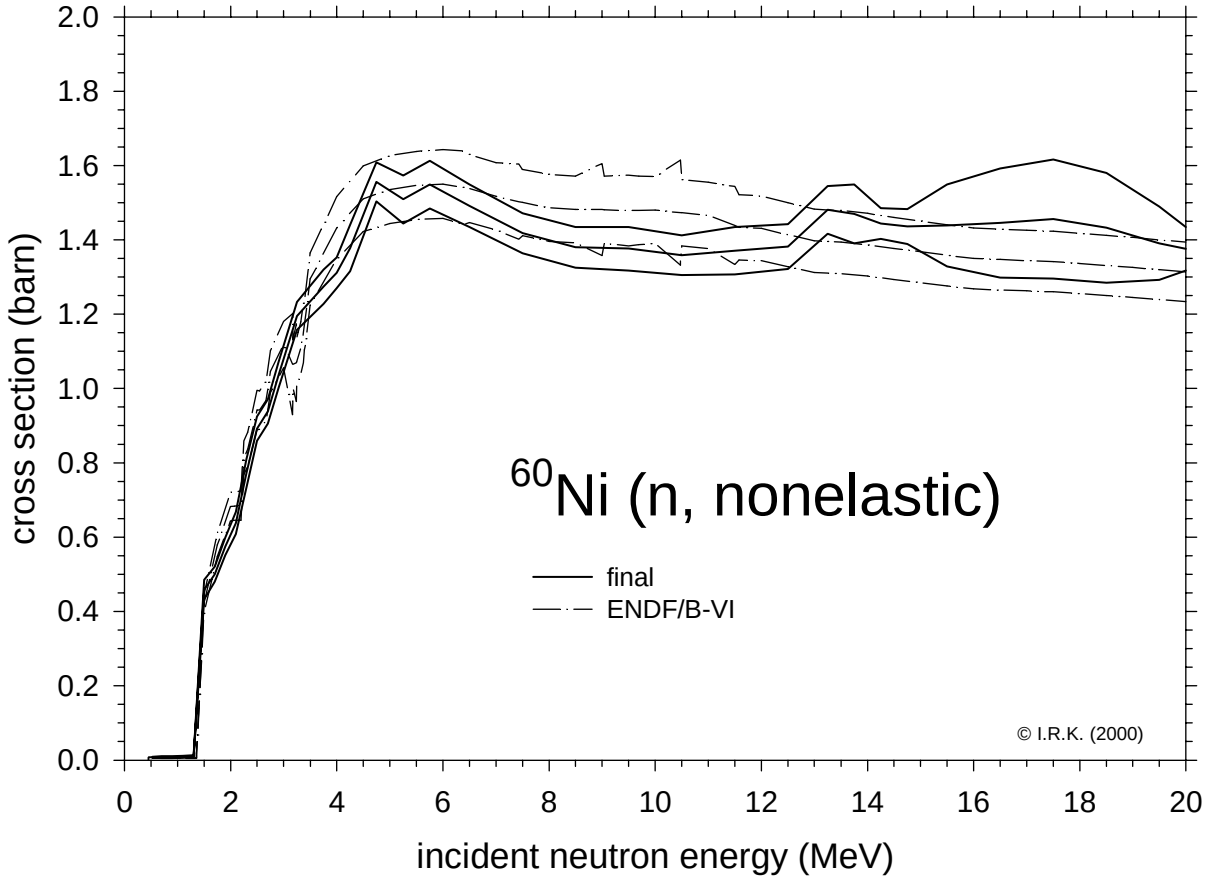
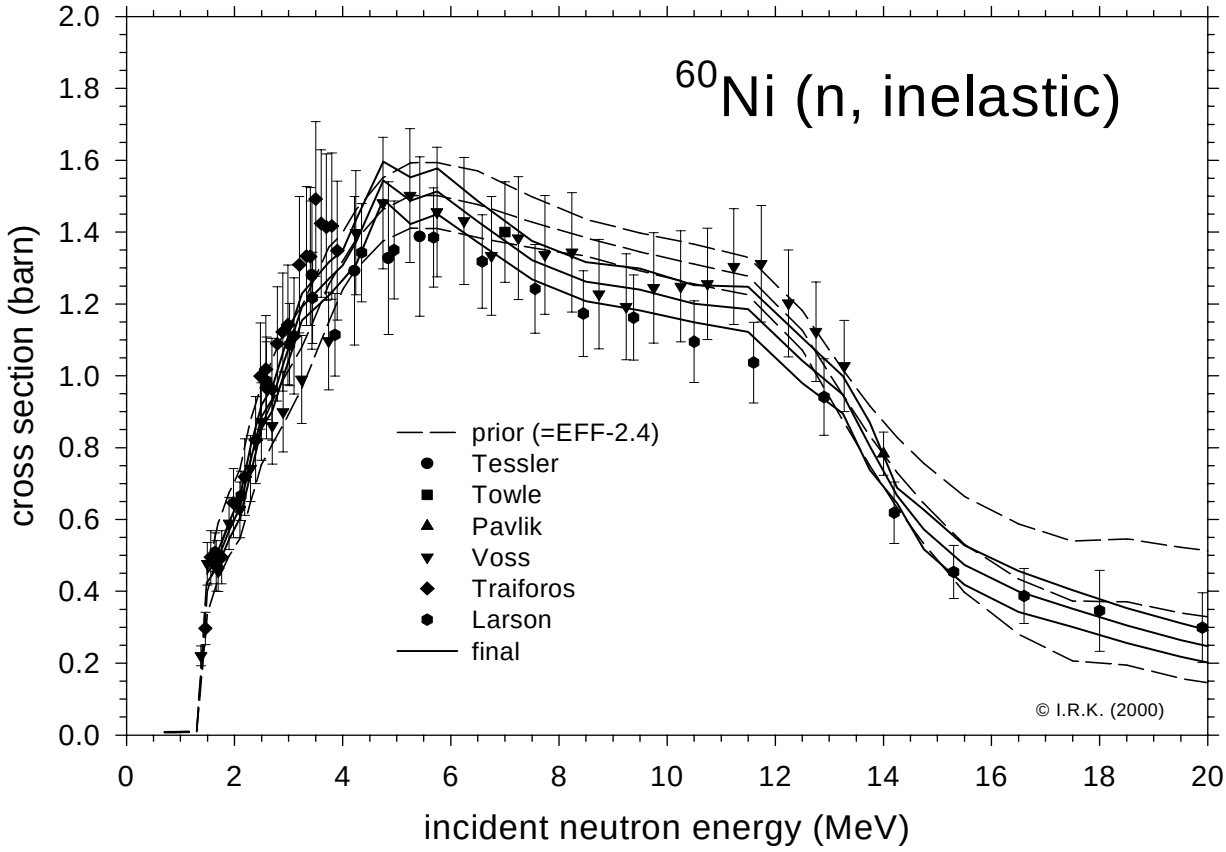


Fig 6: Inelastic cross section and comparison with ENDF/B-VI



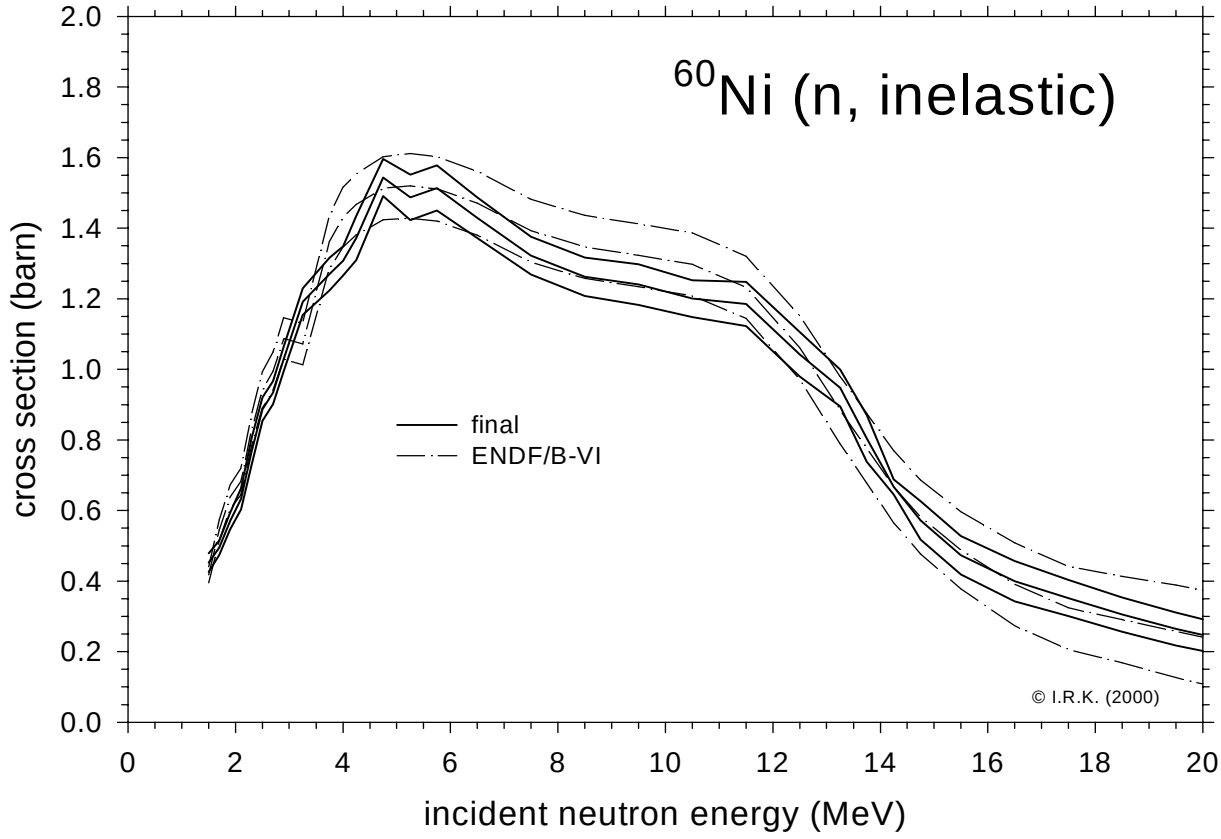
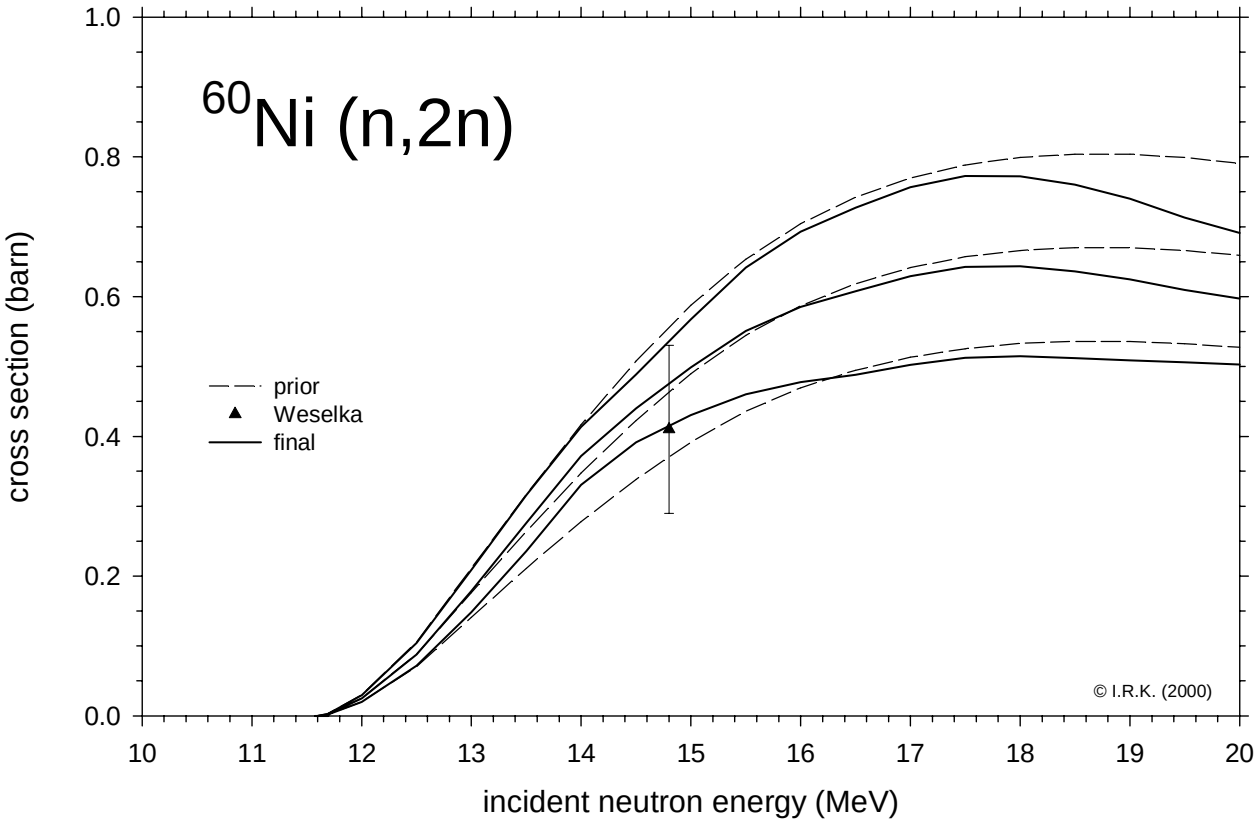


Fig 7: (n, 2n) cross section and comparison with ENDF/B-VI



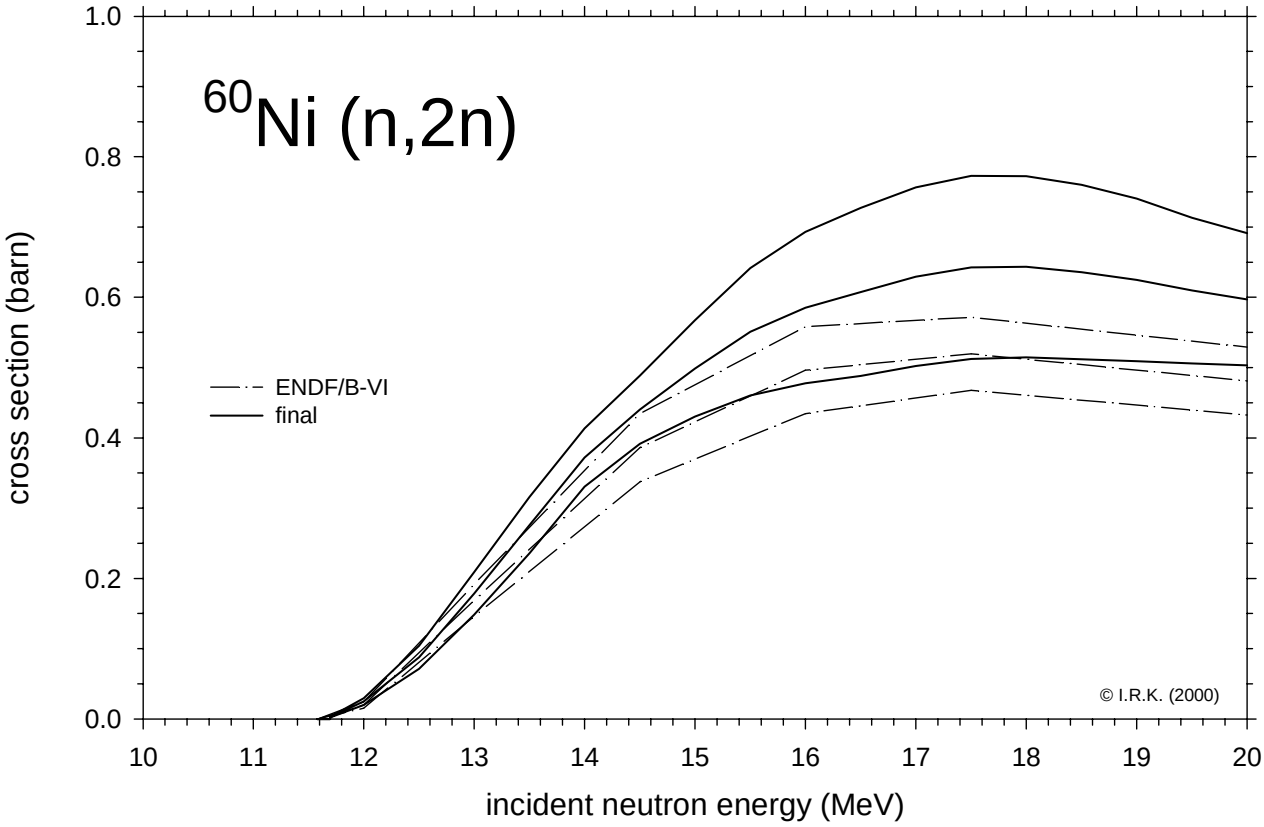


Fig 8: (n, nα) cross section with ENDF/B-VI as "prior"

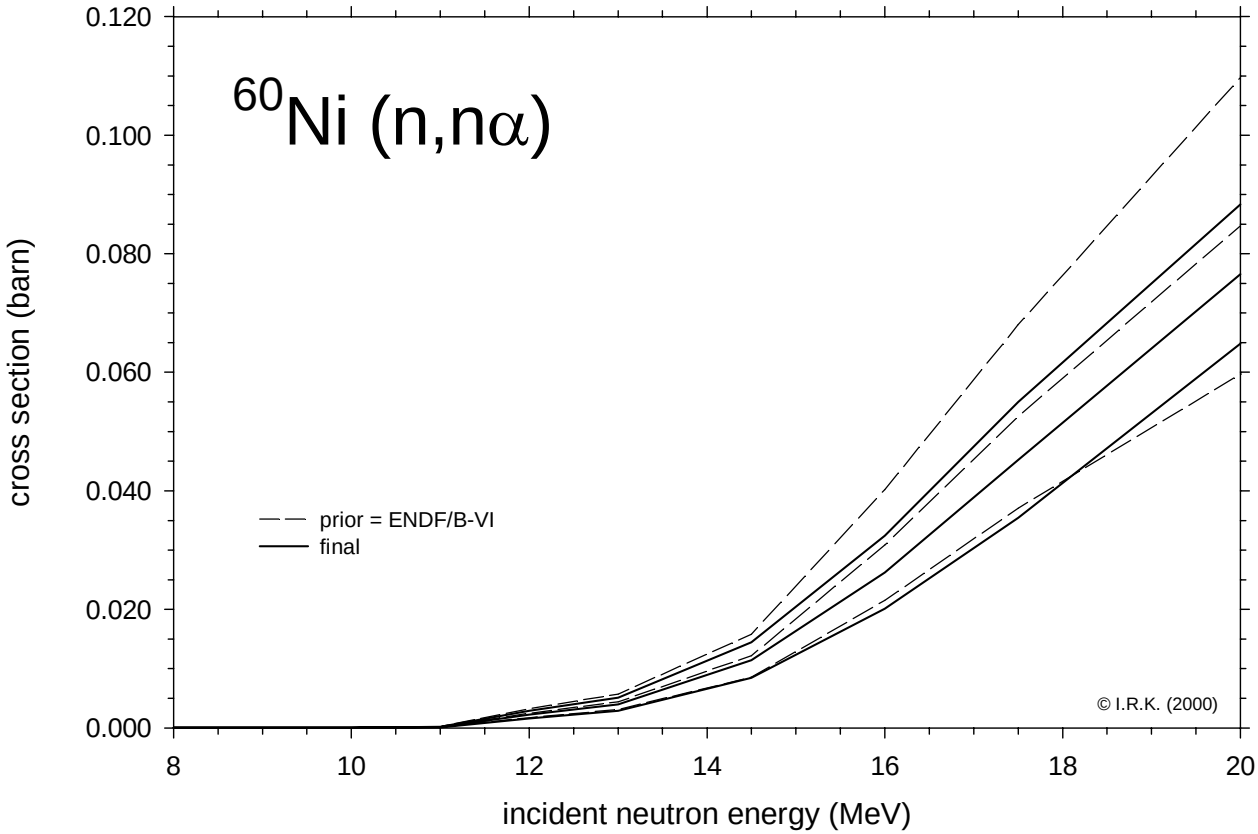


Fig 9: (n, d) cross section with ENDF/B-VI as "prior"

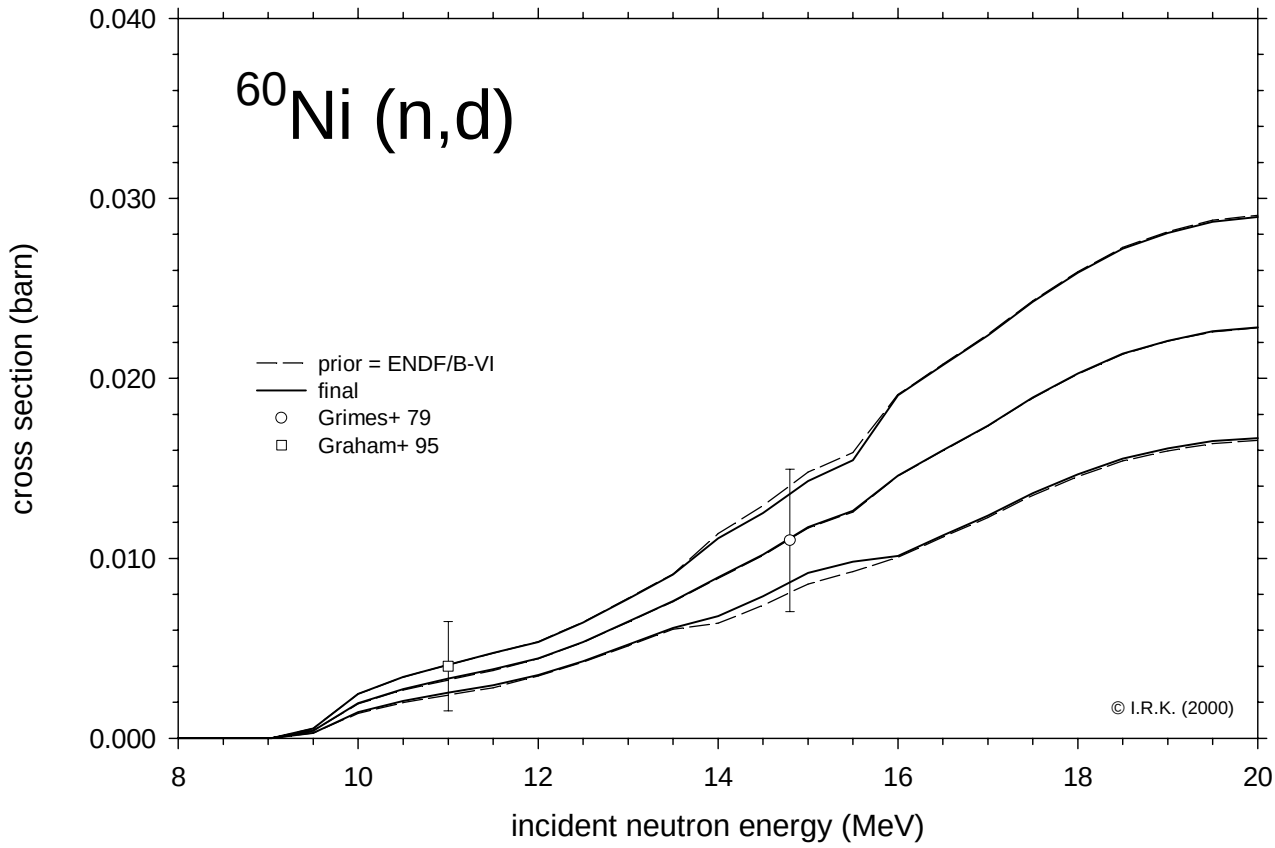
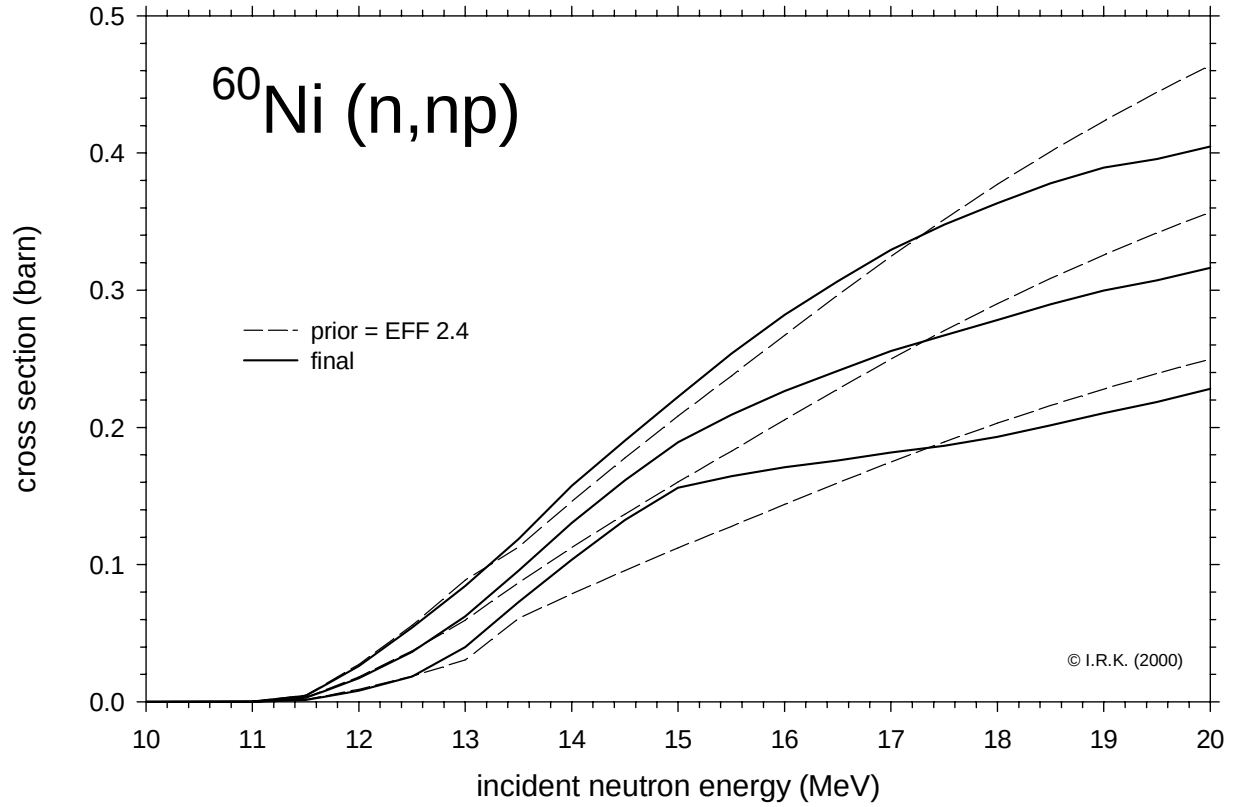


Fig 10: (n, np) cross section and comparison with ENDF/B-VI



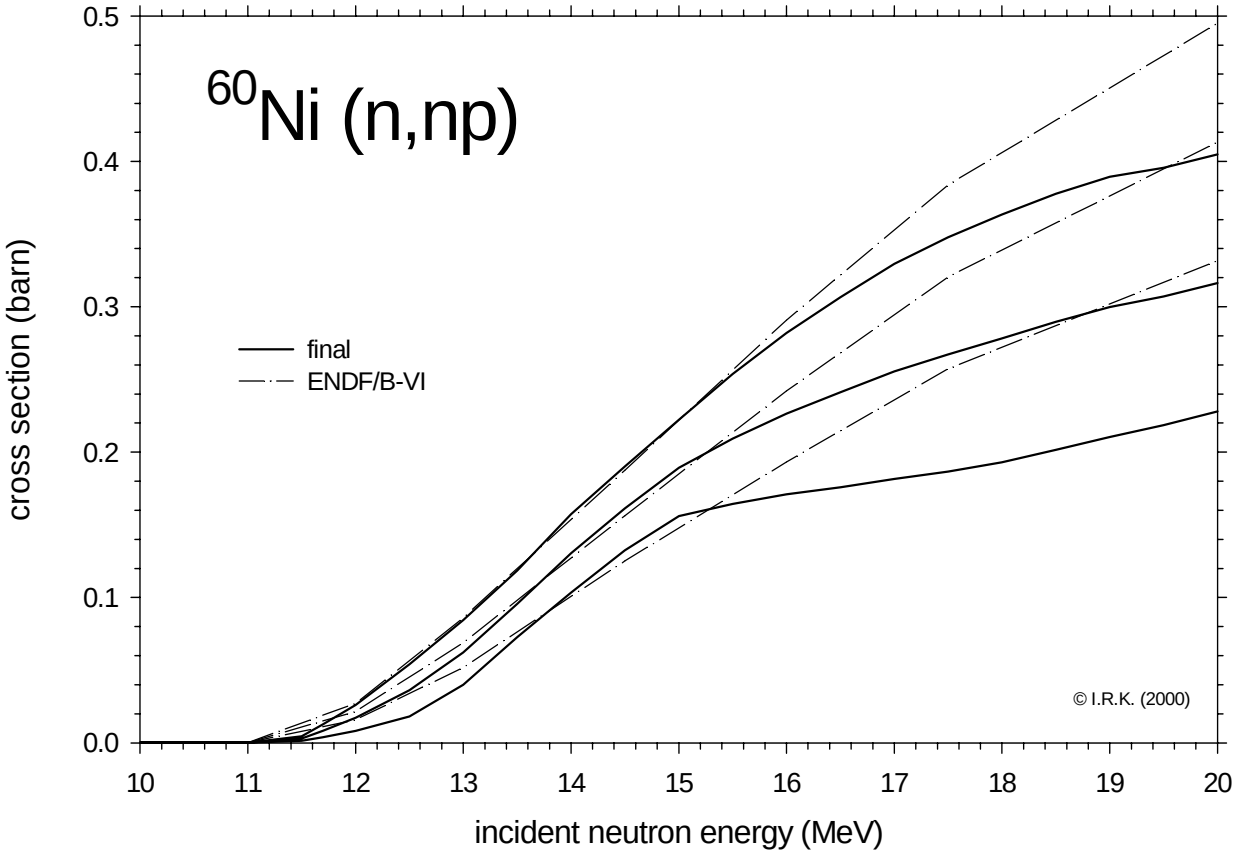
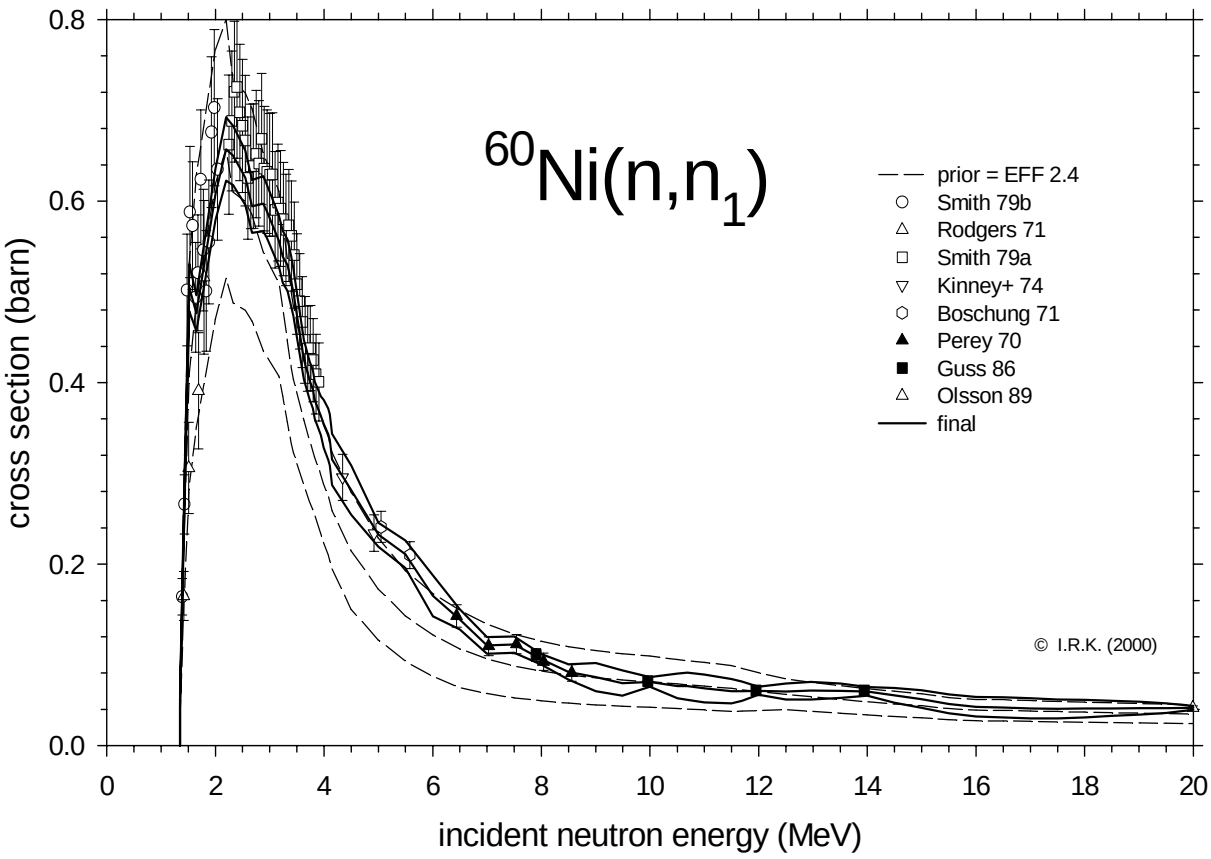


Fig 11: (n, n'$_1$) cross section and comparison with ENDF/B-VI



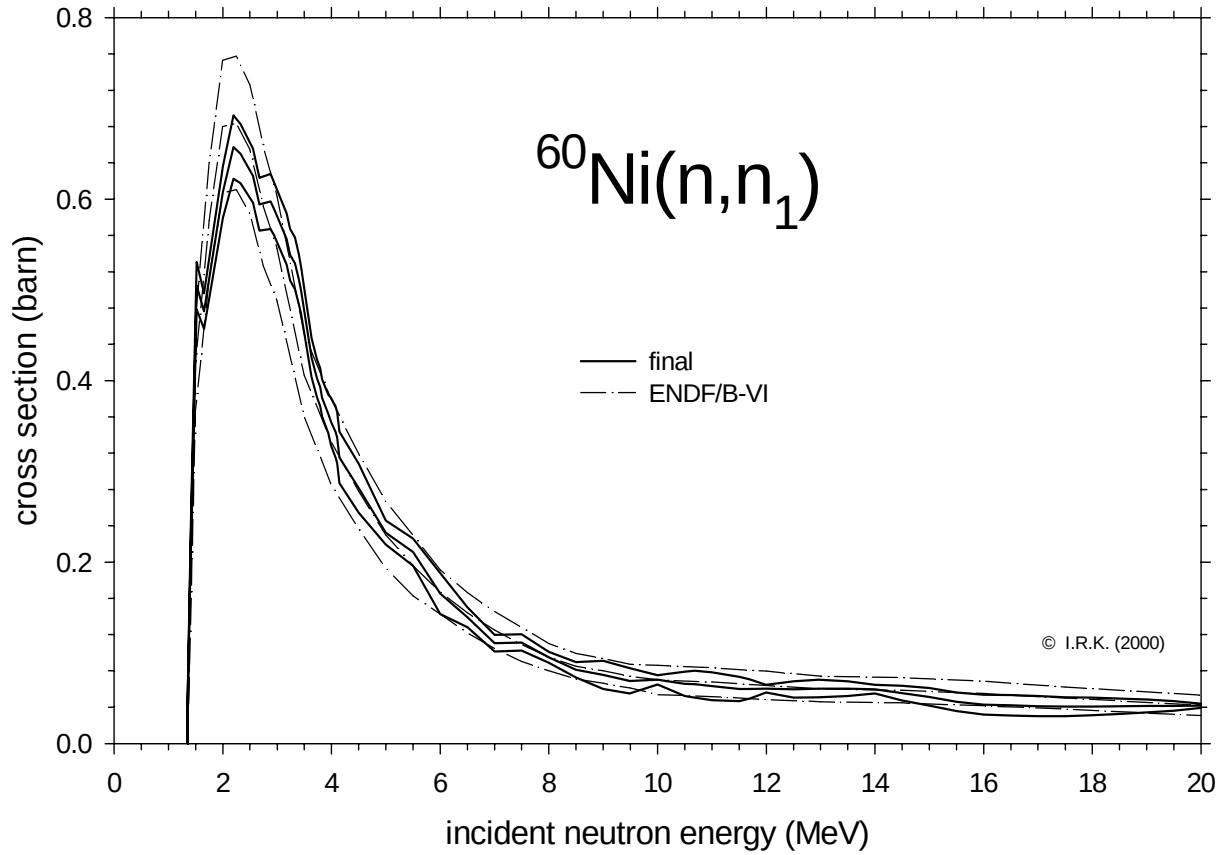
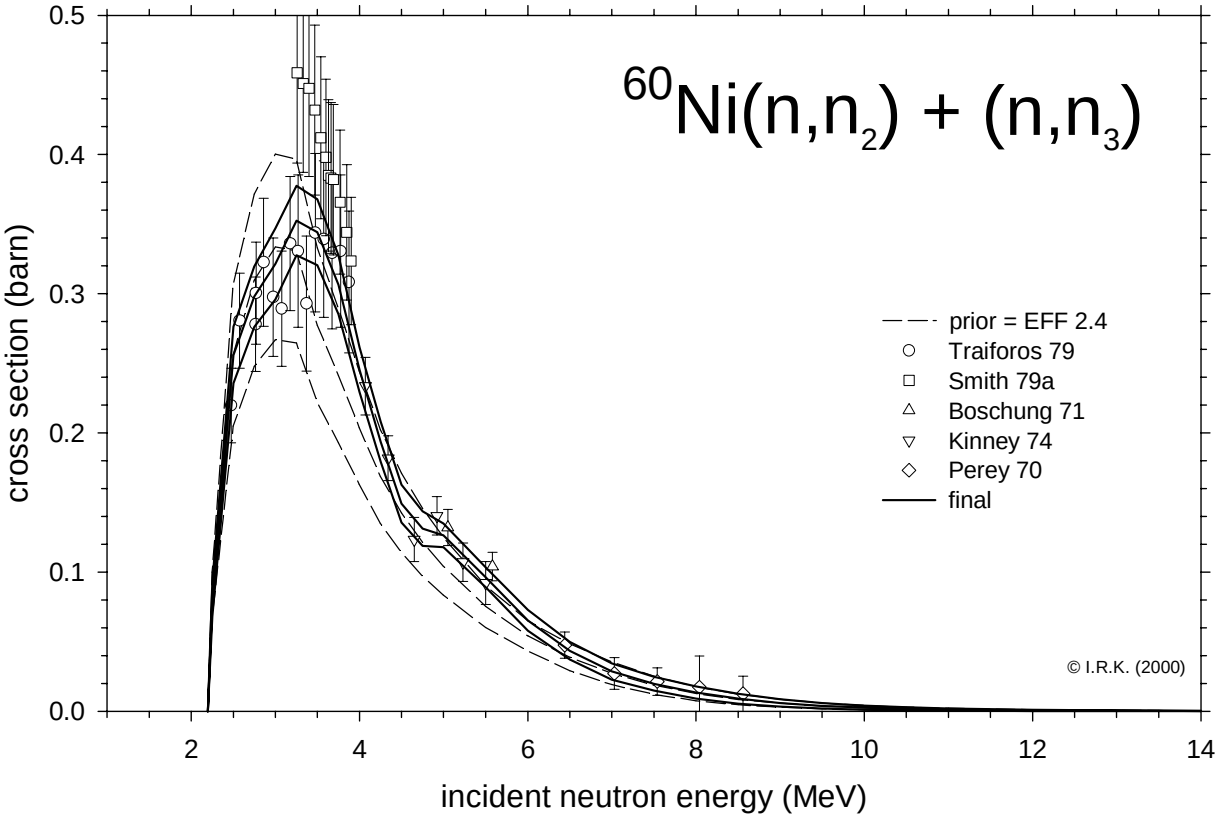


Fig 12: (n, n'$_{2+3}$) cross section and comparison with ENDF/B-VI



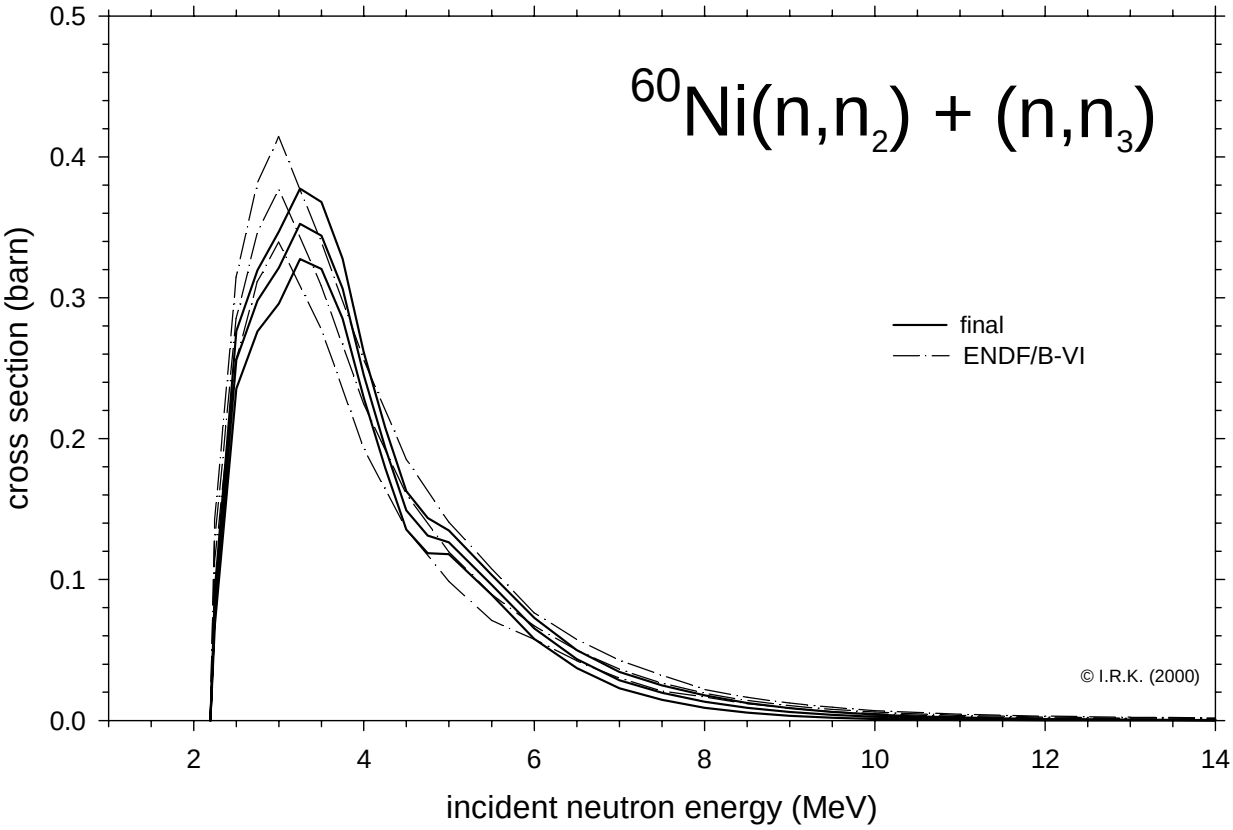
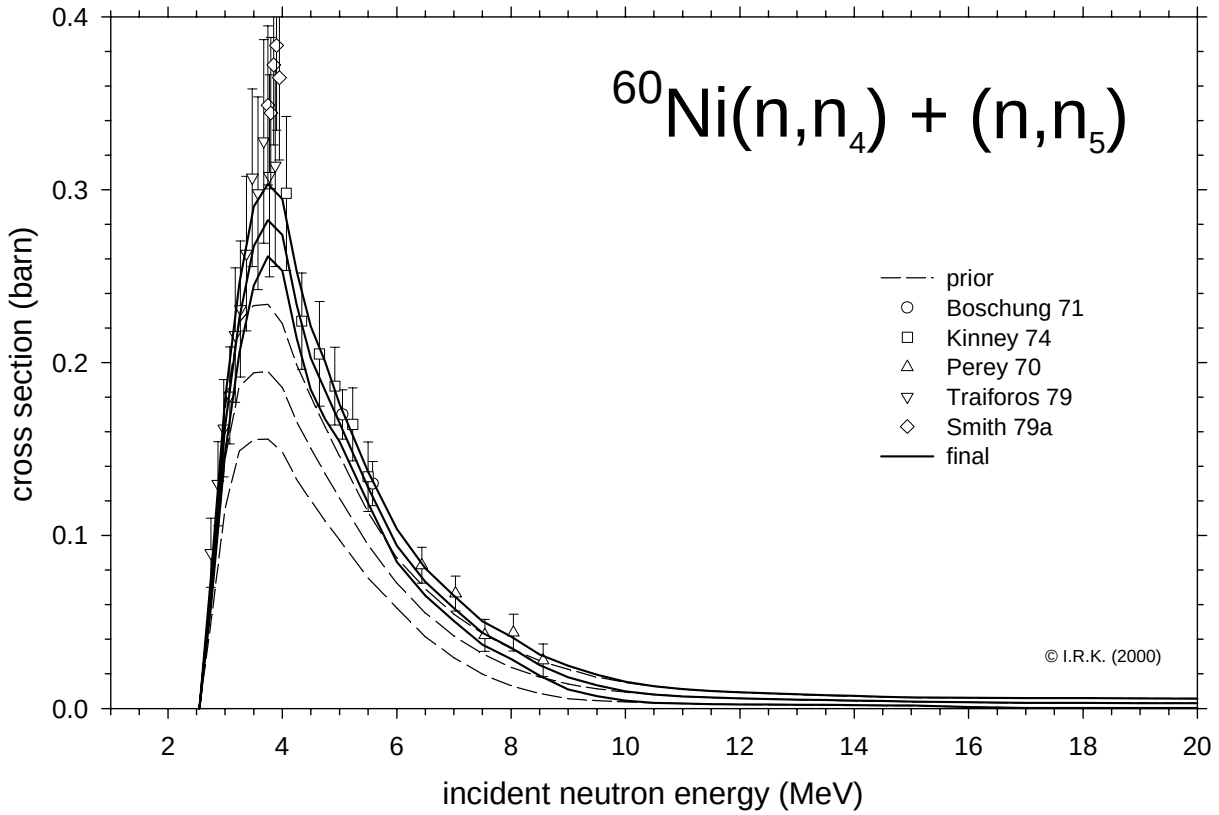


Fig 13: (n, n'$_{4+5}$) cross section and comparison with ENDF/B-VI



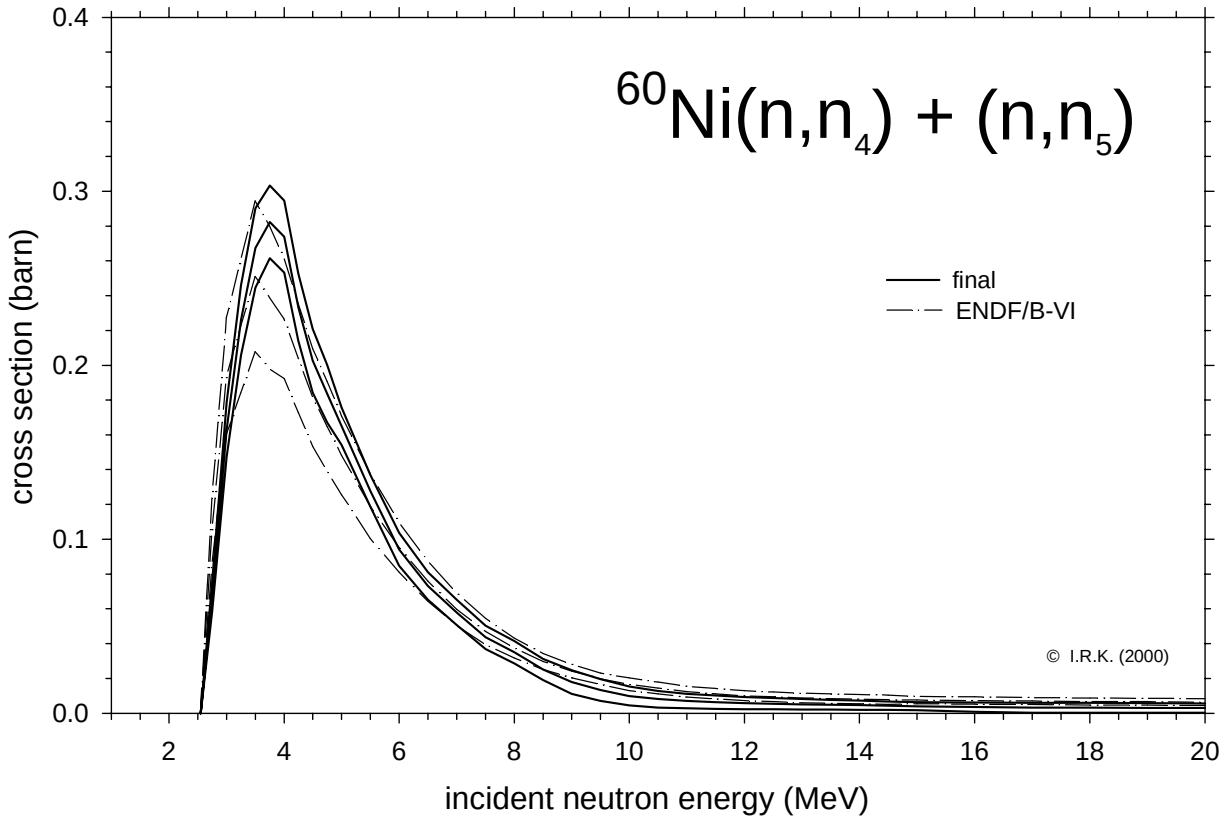
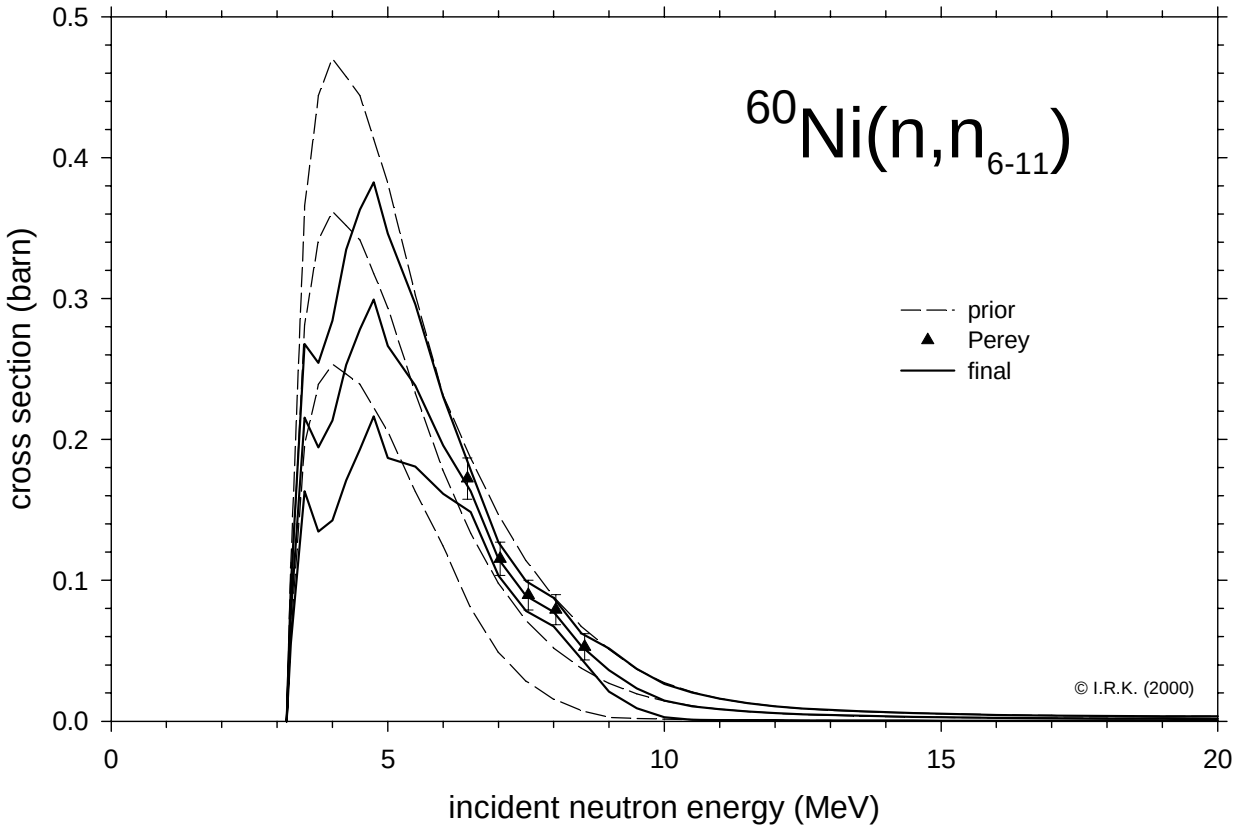


Fig 14: (n, n'$_6$) to (n, n'$_{11}$) cross section and comparison with ENDF/B-VI



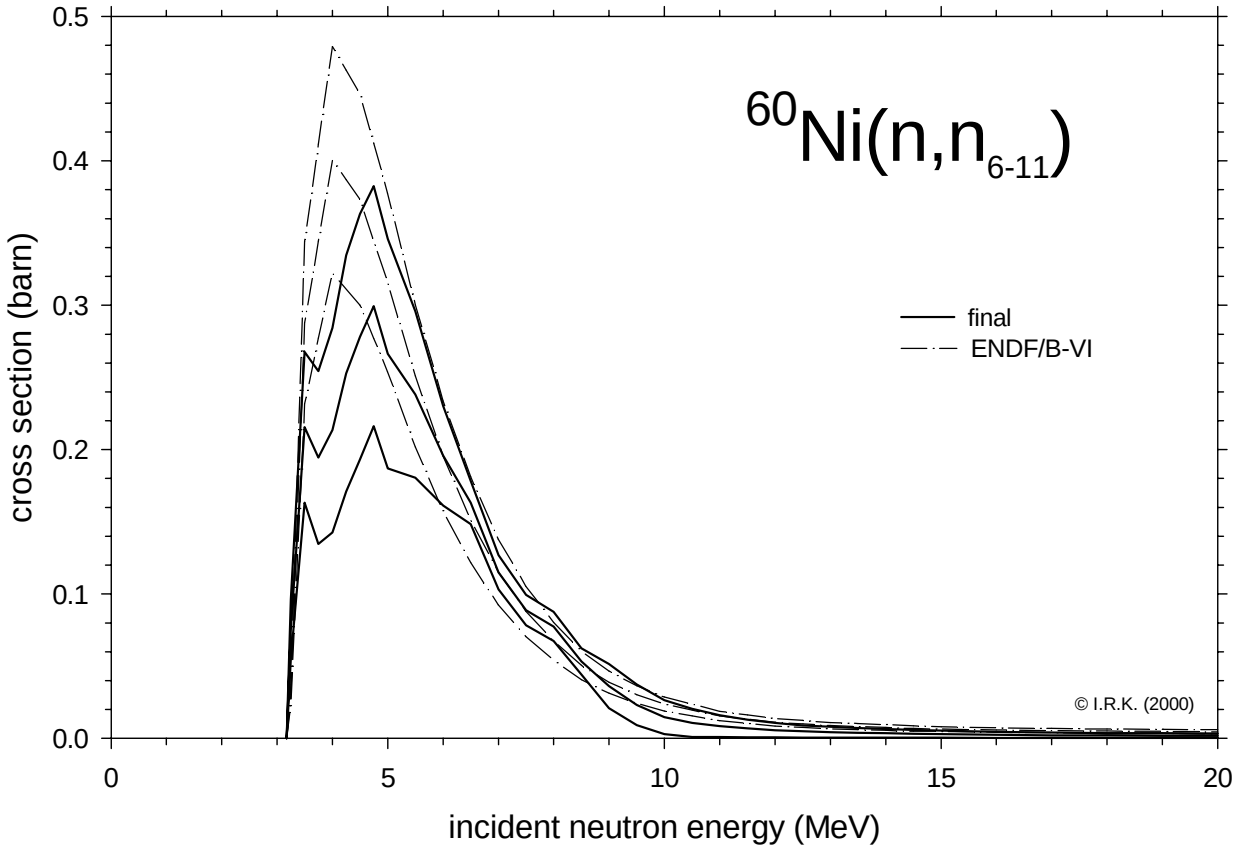
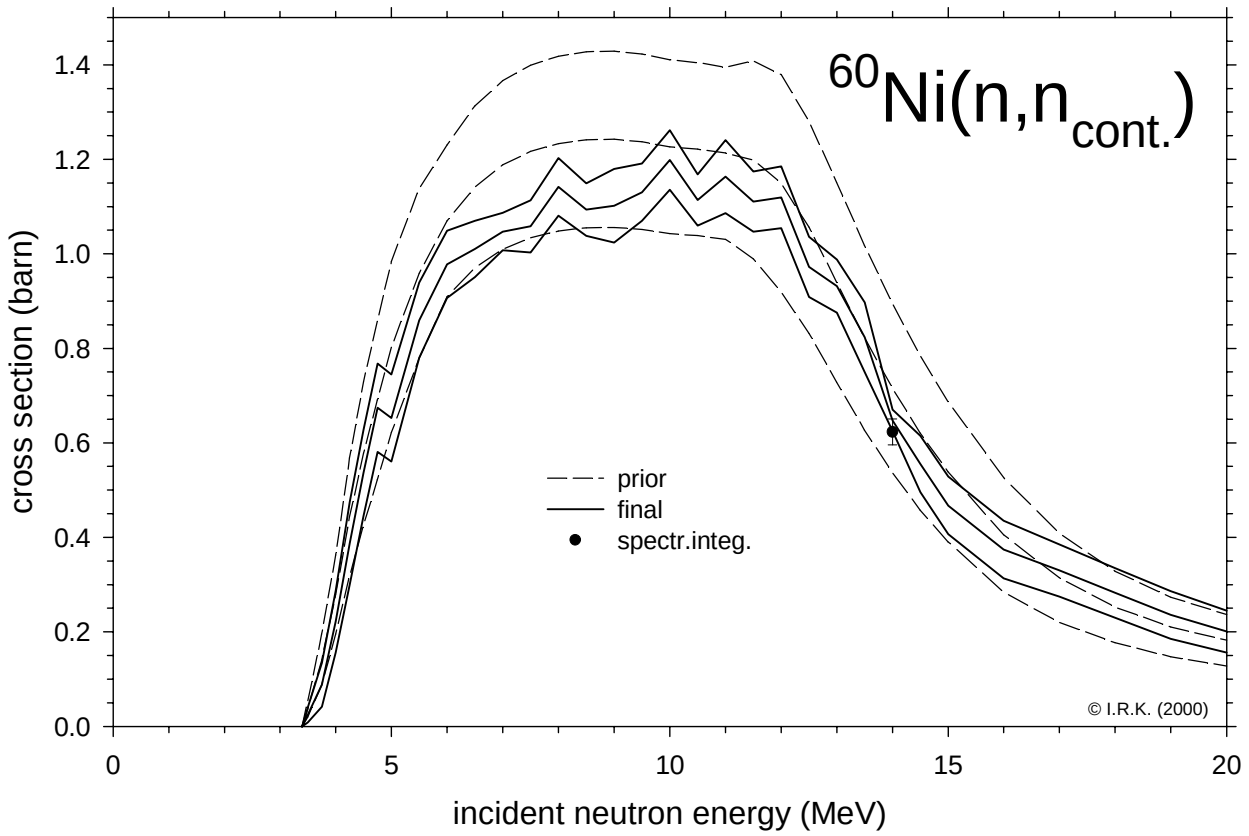


Fig 15: (n, n'$_{cont}$) cross section and comparison with ENDF/B-VI



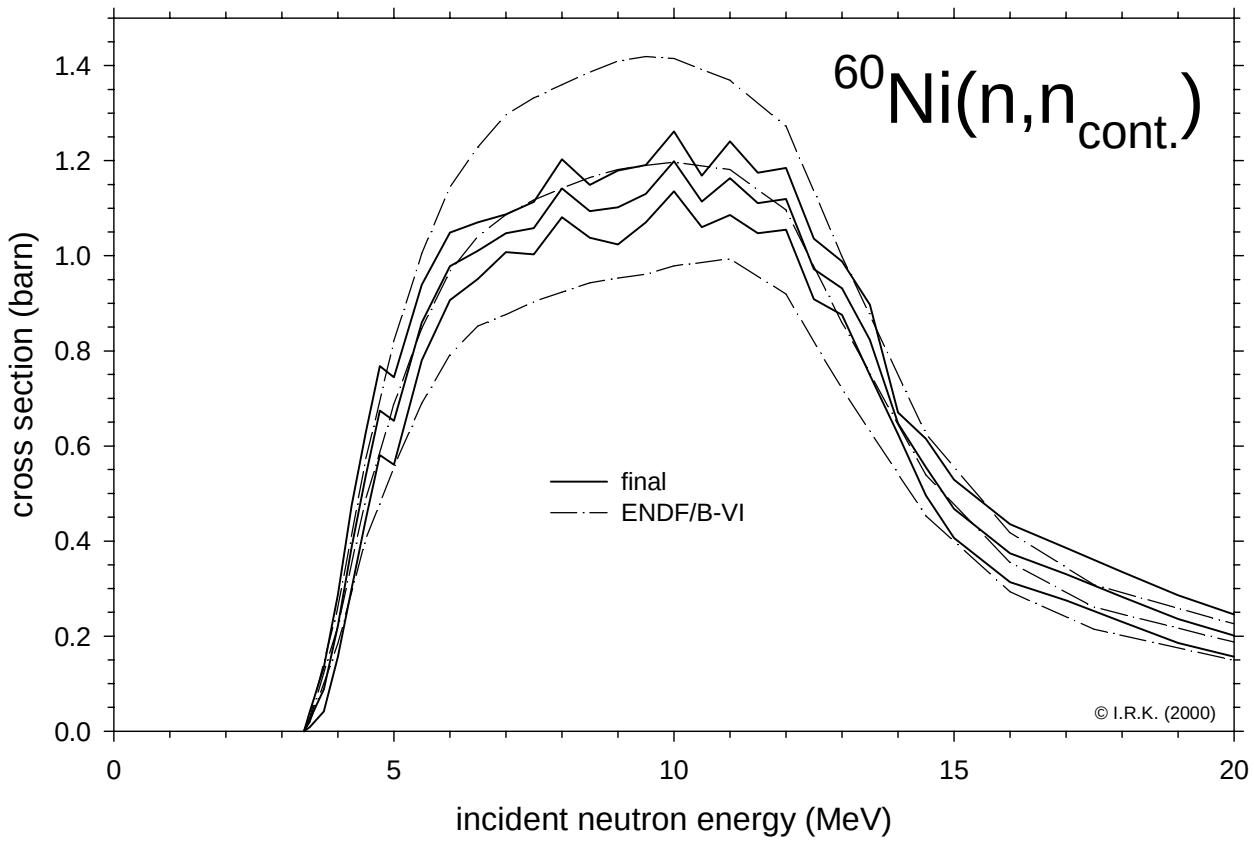
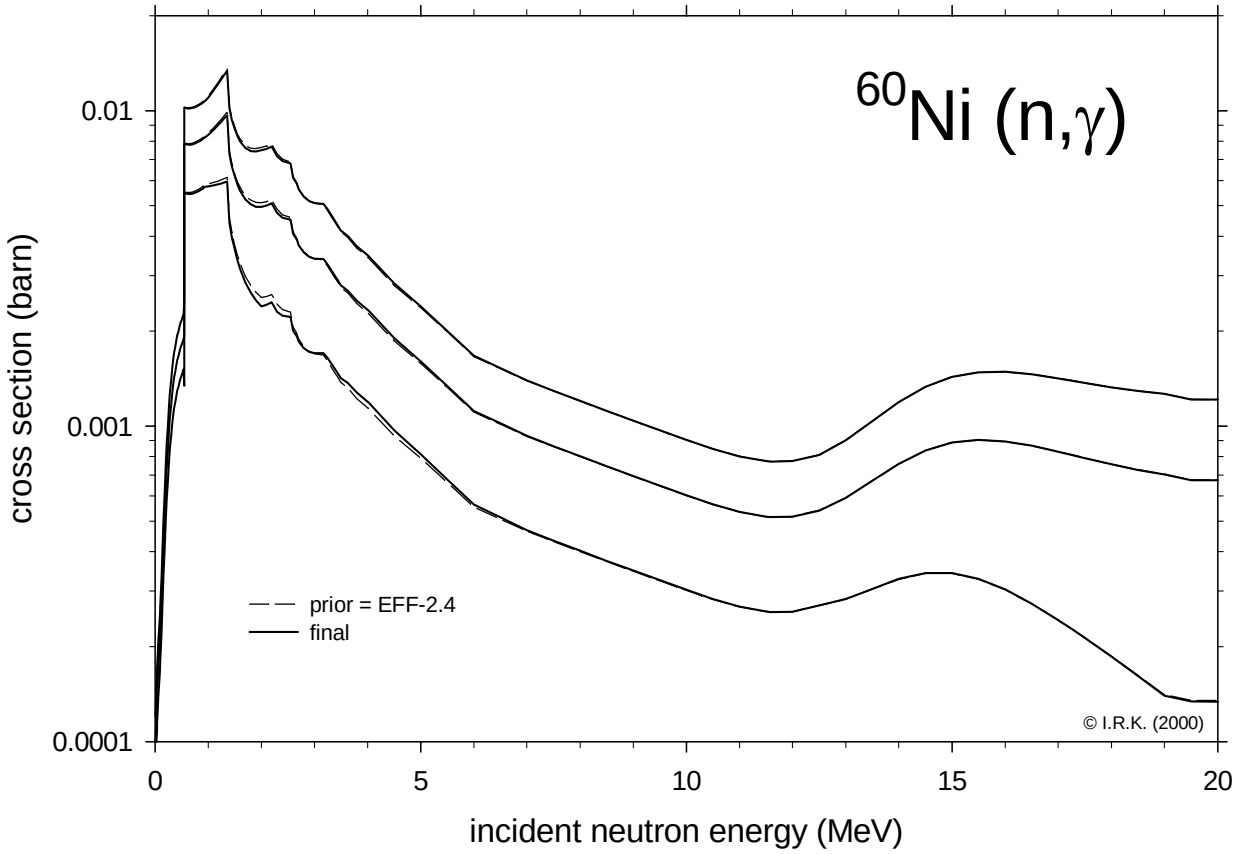


Fig 16: (n,γ) cross section and comparison with ENDF/B-VI



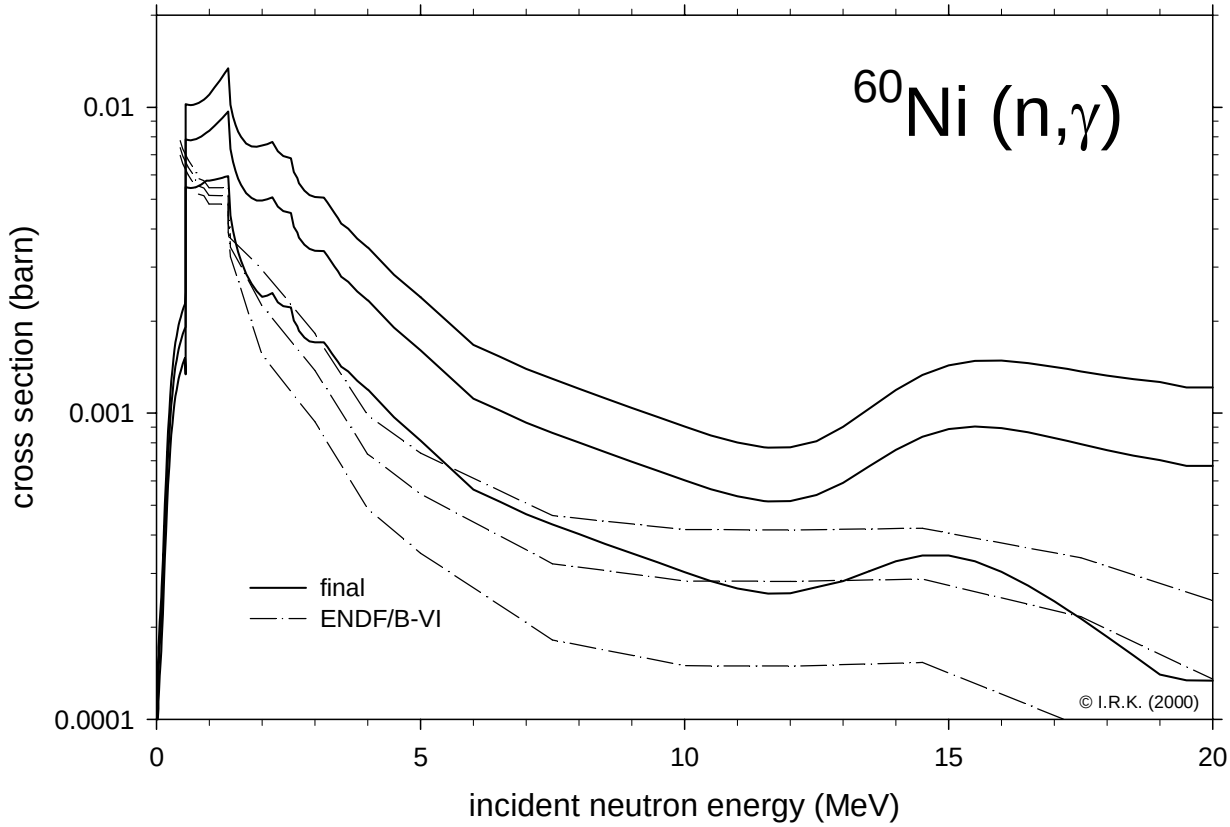
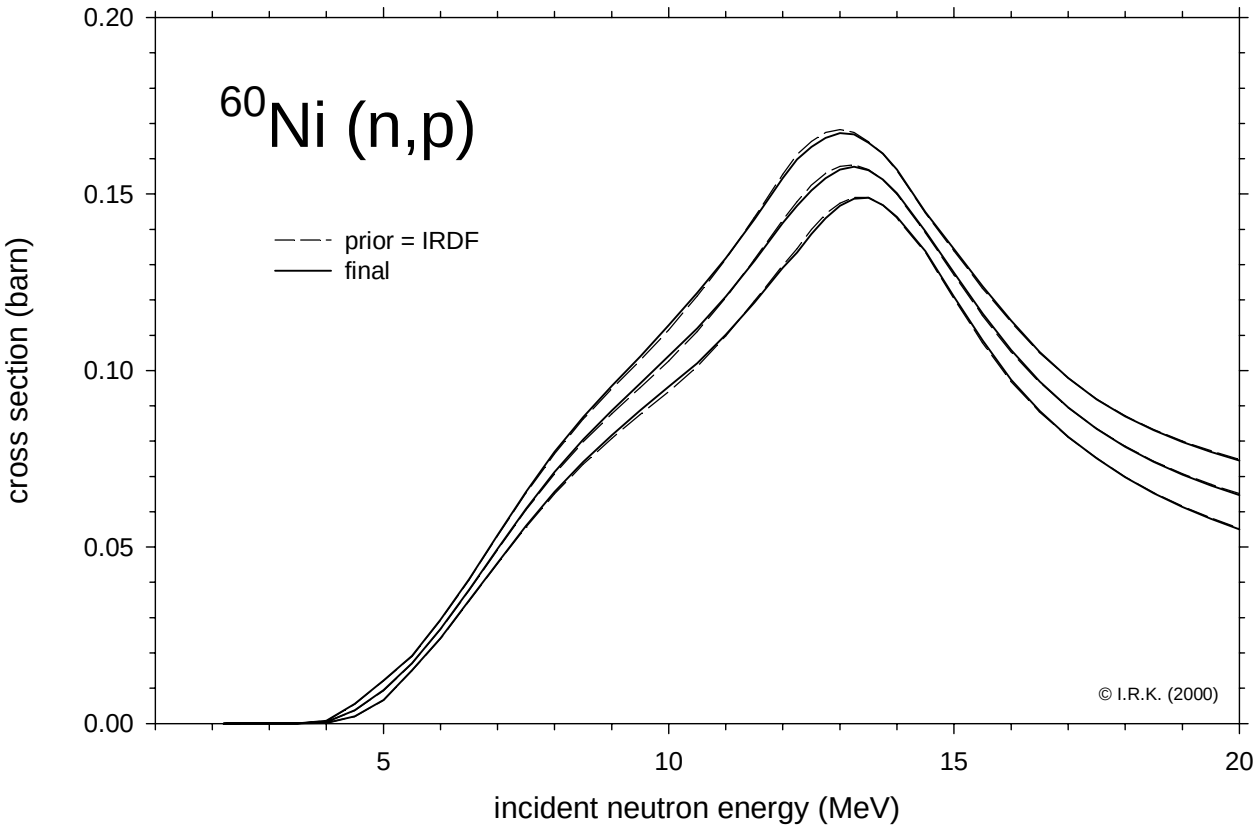


Fig 17: (n,p) cross section and comparison with ENDF/B-VI



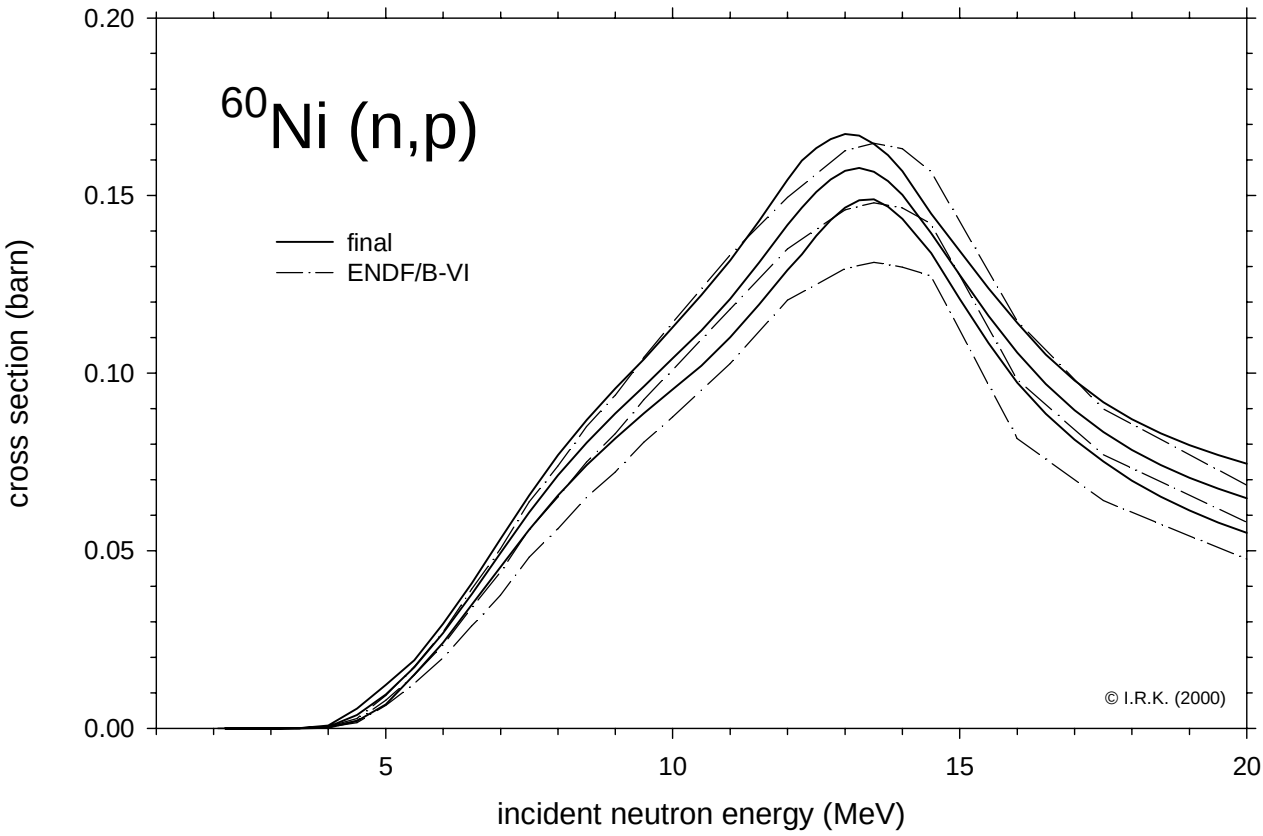


Fig 18: (n,t) cross section

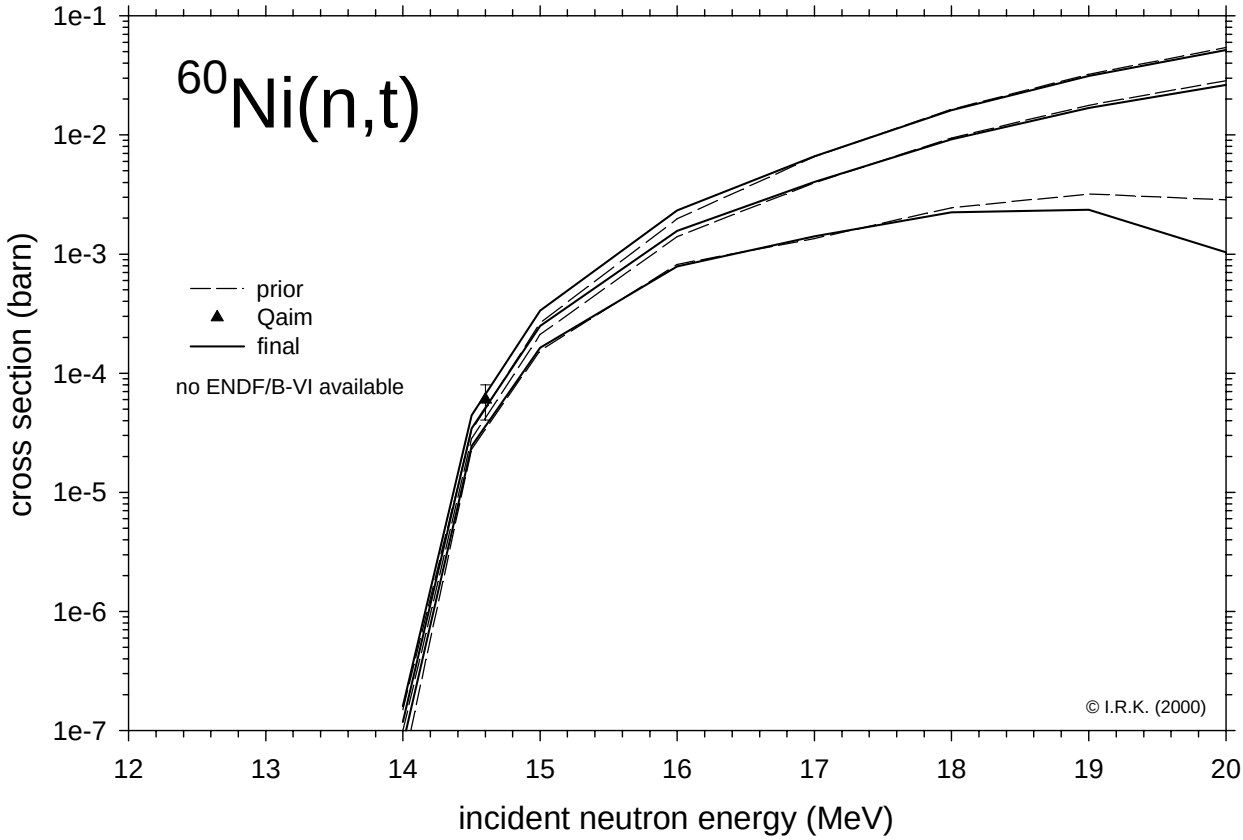


Fig 19: (n,^{3}He) cross section

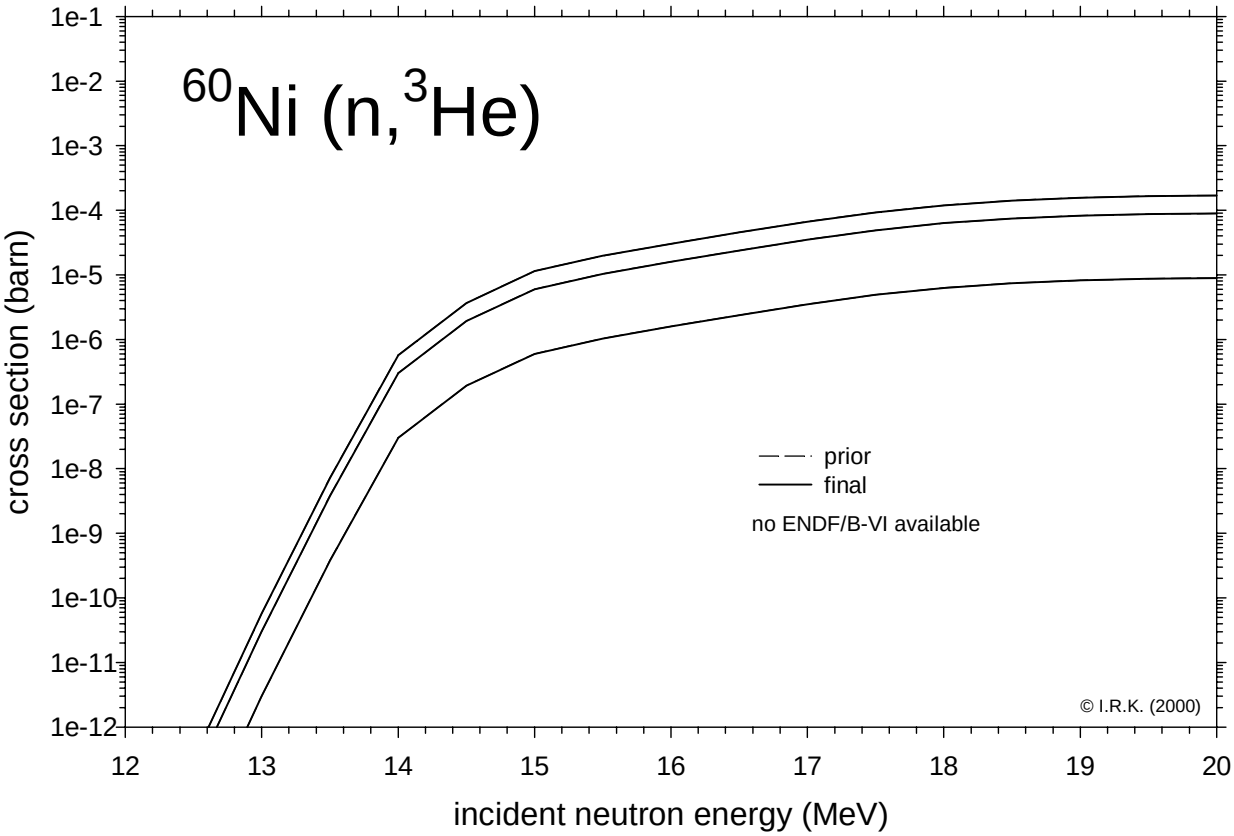
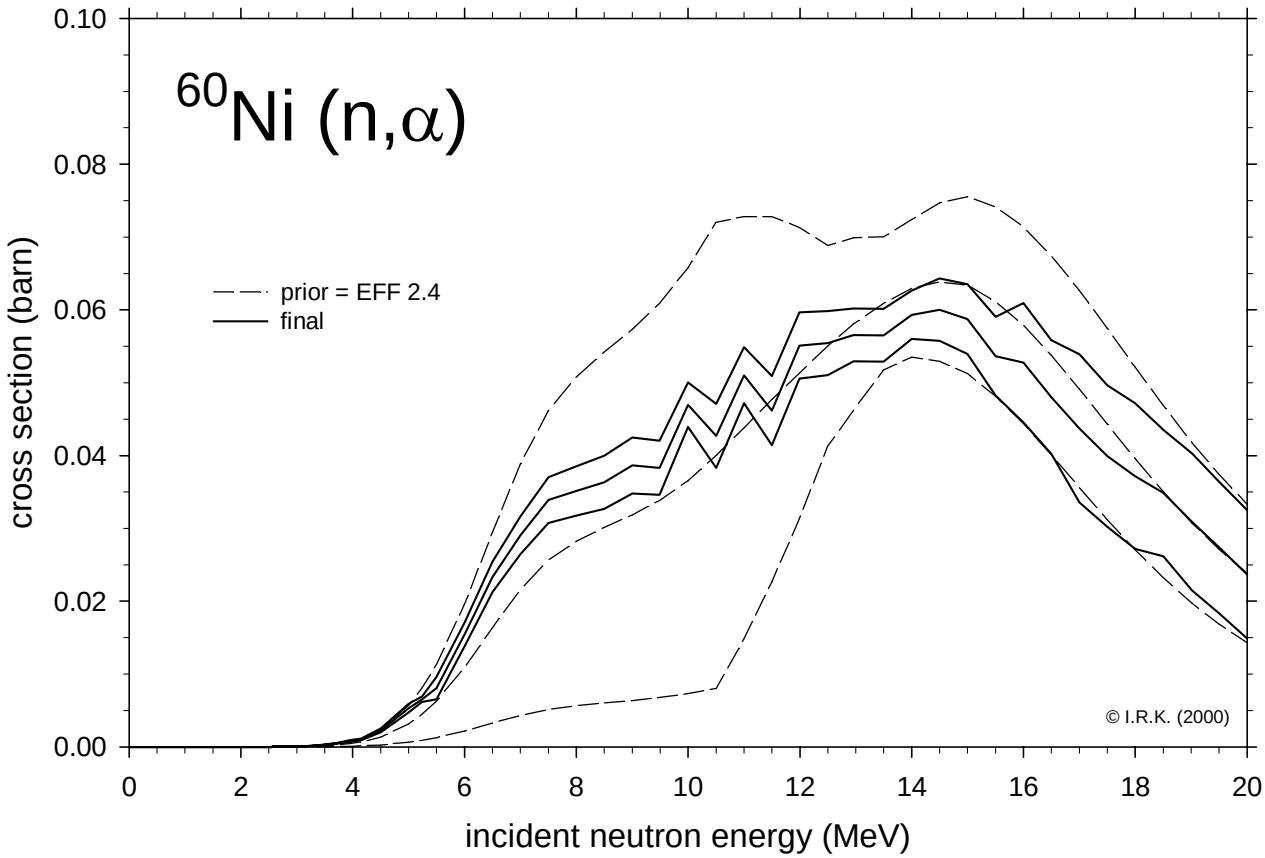


Fig 20: (n,α) cross section and comparison with ENDF/B-VI



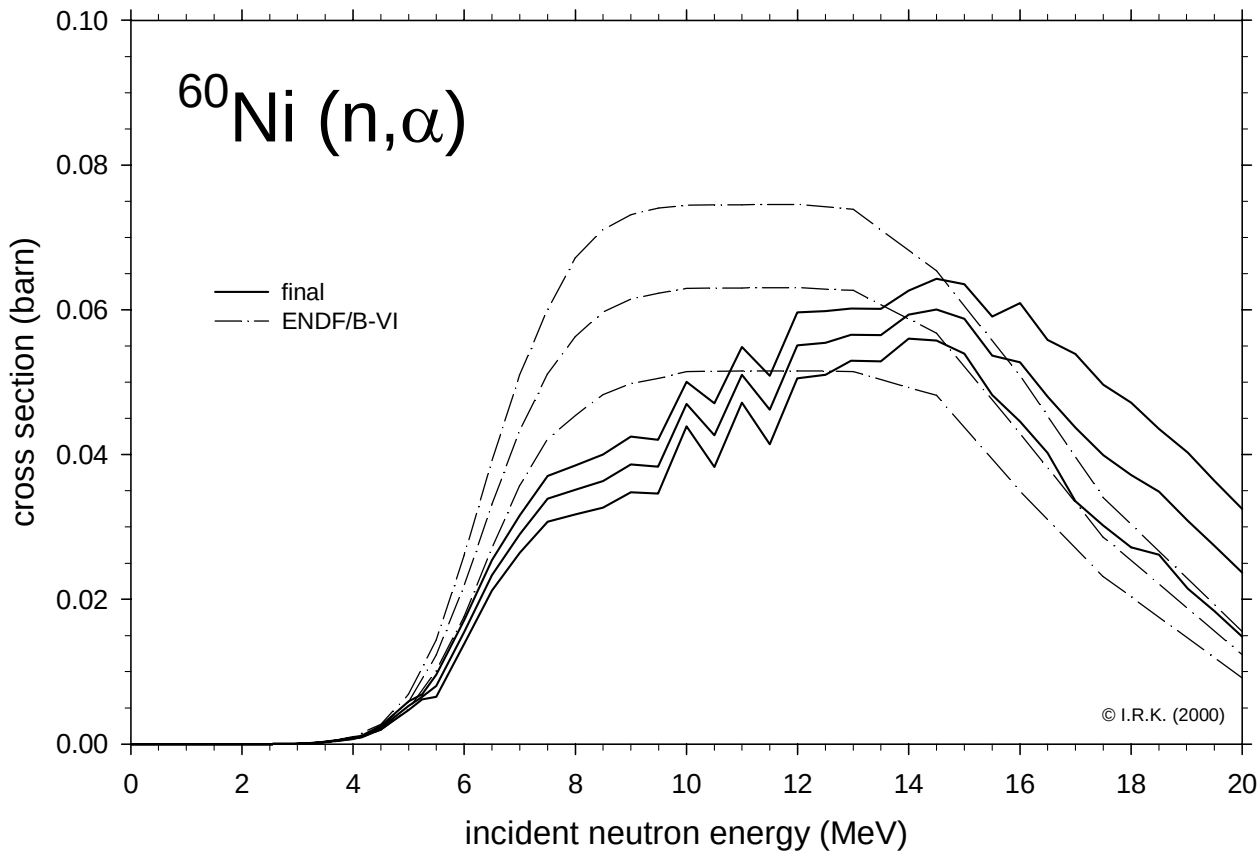


Fig. 21 Correlation matrix for the total cross section

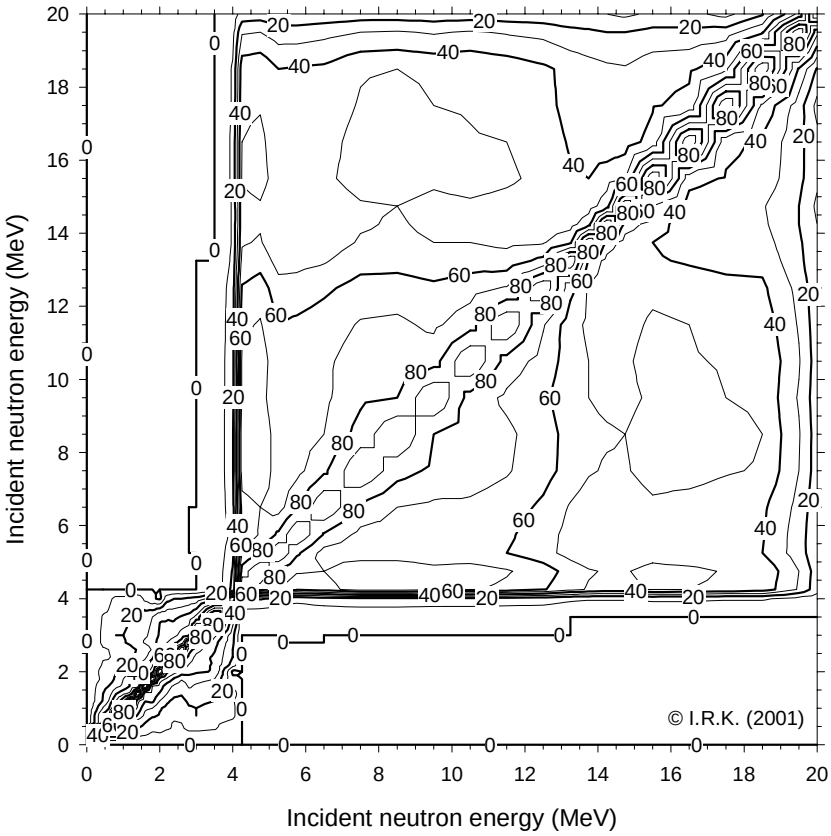


Fig. 22 Correlation matrix for the (n, 2n) cross section

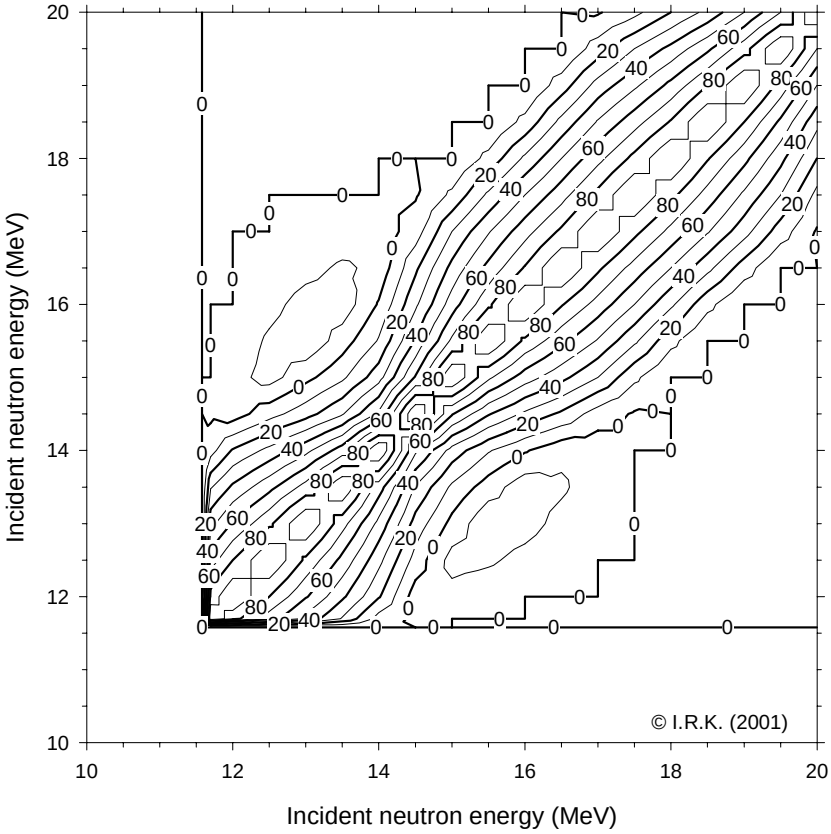


Fig. 23 Correlation matrix for the (n, nα) cross section

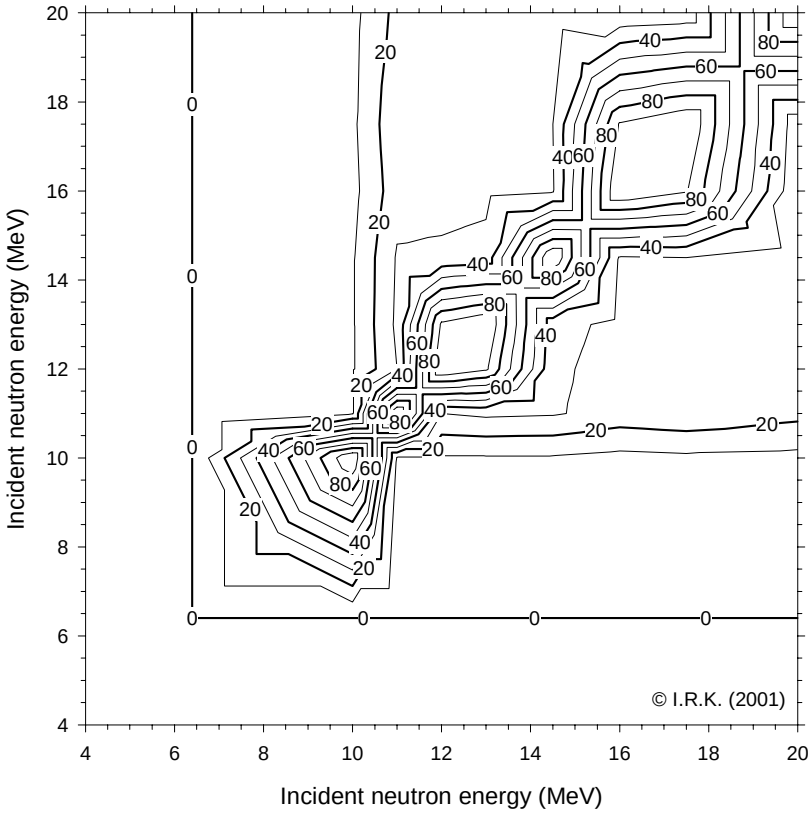


Fig. 24 Correlation matrix for the (n, np) cross section

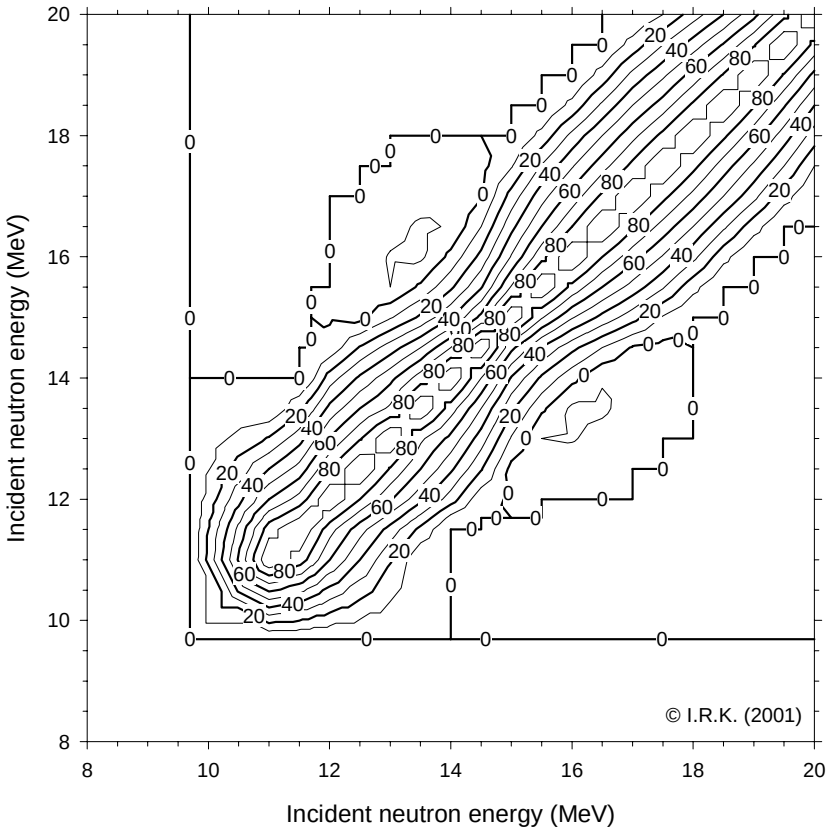


Fig. 25 Correlation matrix for the (n, n'$_1$) cross section

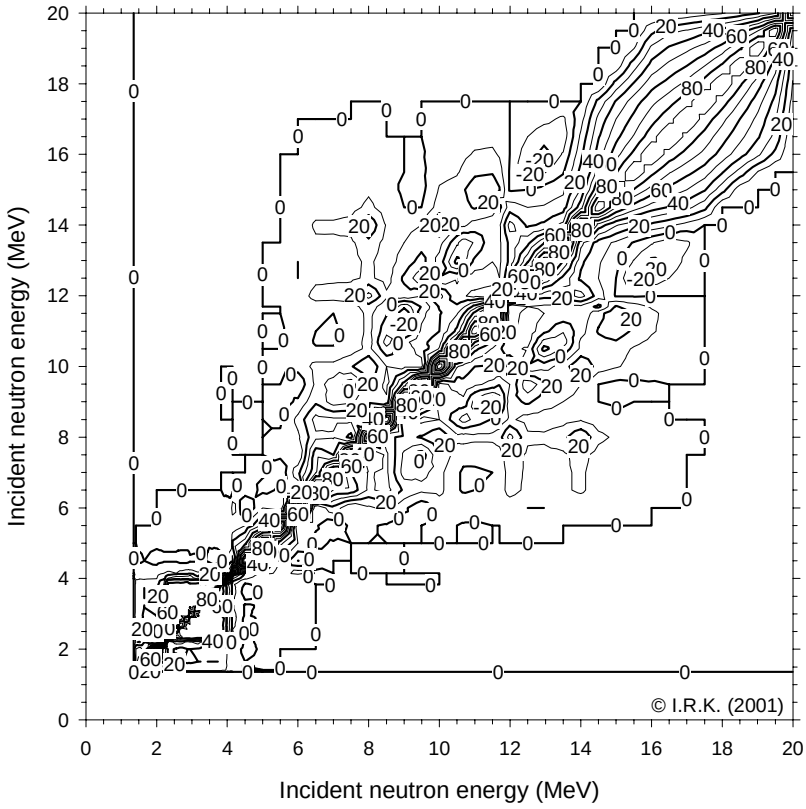


Fig. 26 Correlation matrix for the (n, n'$_2$) + (n, n'$_3$) cross section

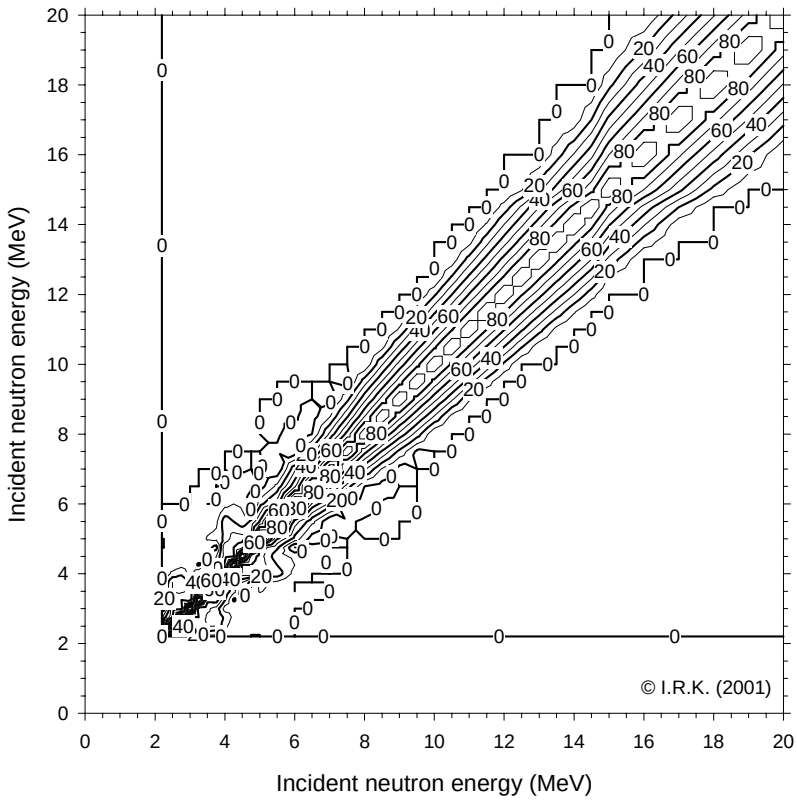


Fig. 27 Correlation matrix for the (n, n'$_4$) + (n, n'$_5$) cross section

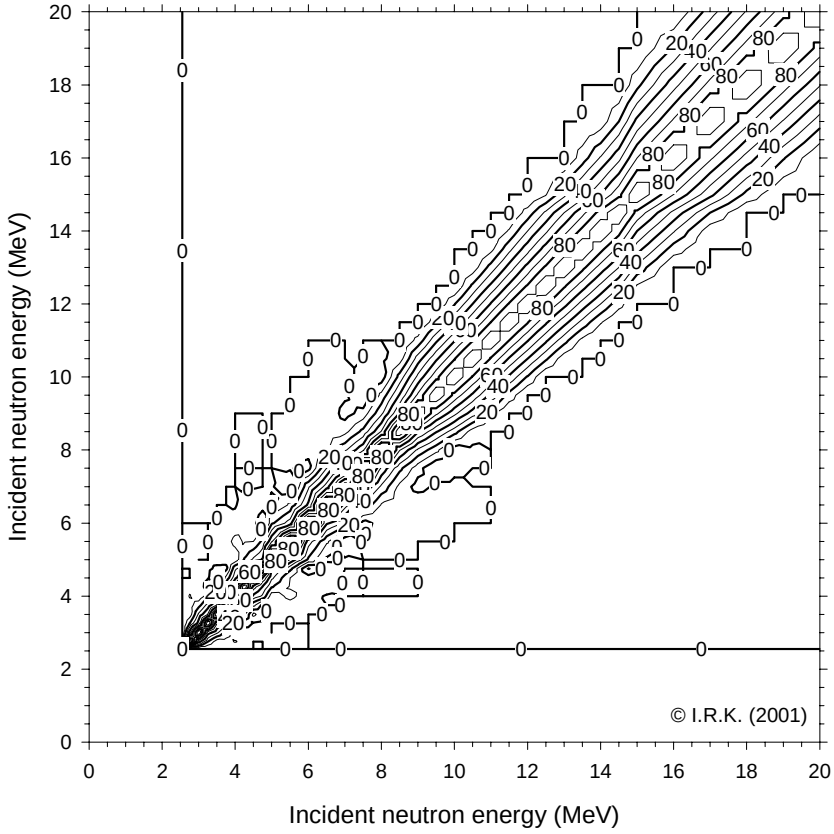


Fig. 28 Correlation matrix for the (n, n'$_6$) + ...+ (n, n'$_{11}$) cross section

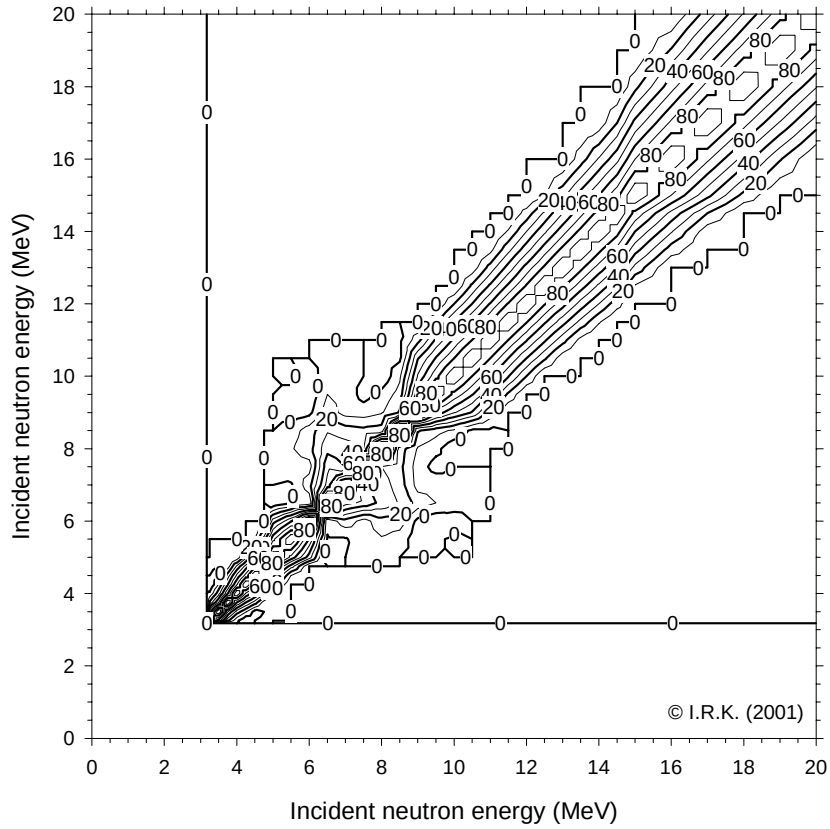


Fig. 29 Correlation matrix for the (n, n'$_{cont}$) cross section

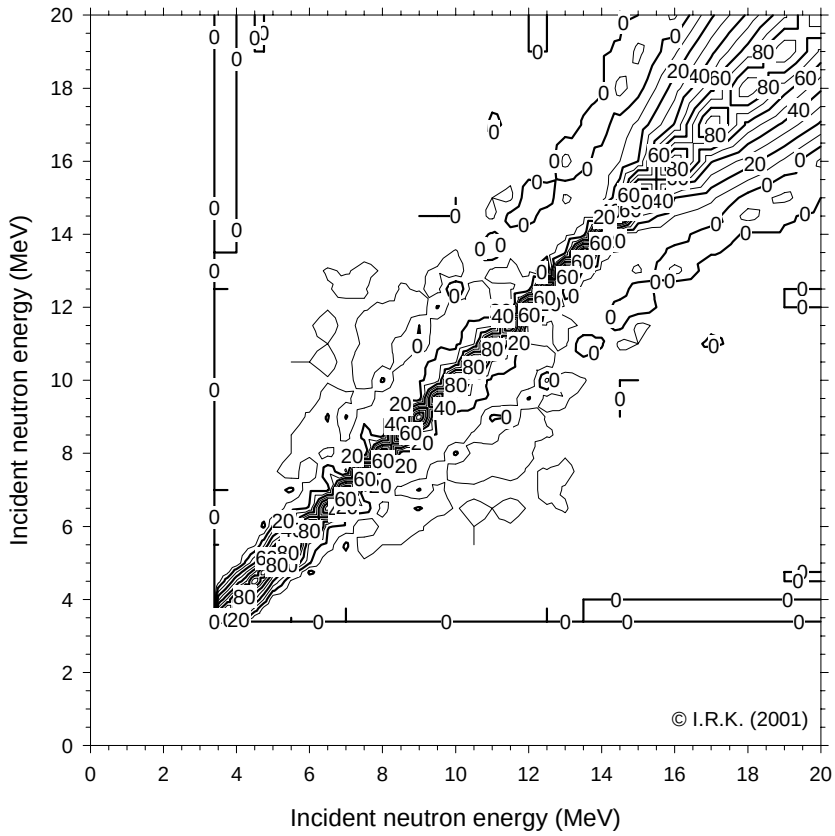


Fig. 30 Correlation matrix for the (n, γ) cross section

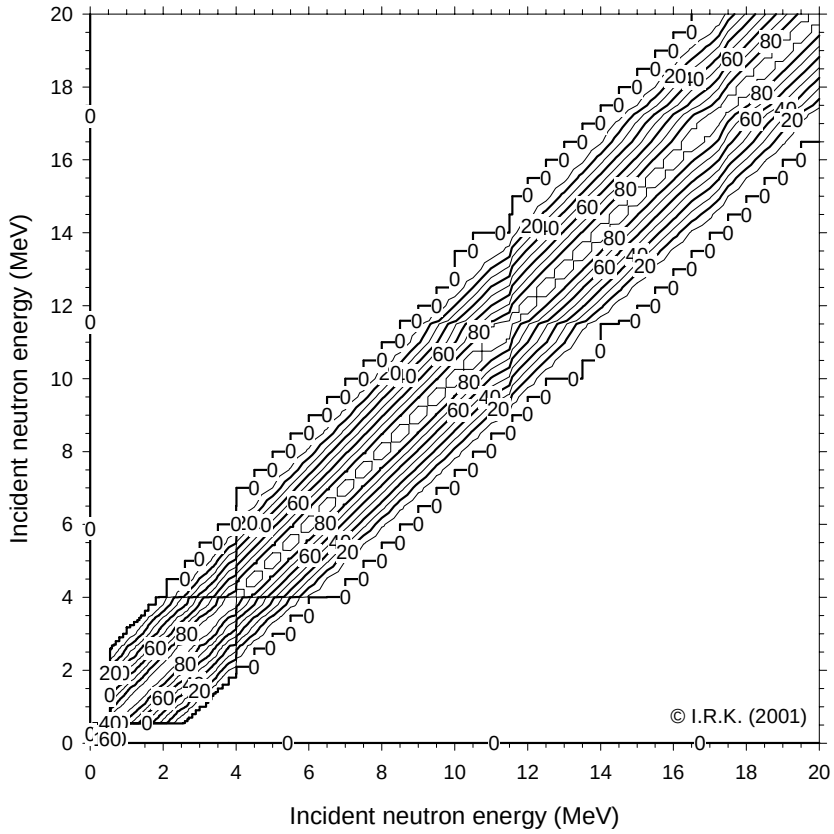


Fig. 31 Correlation matrix for the (n, p) cross section

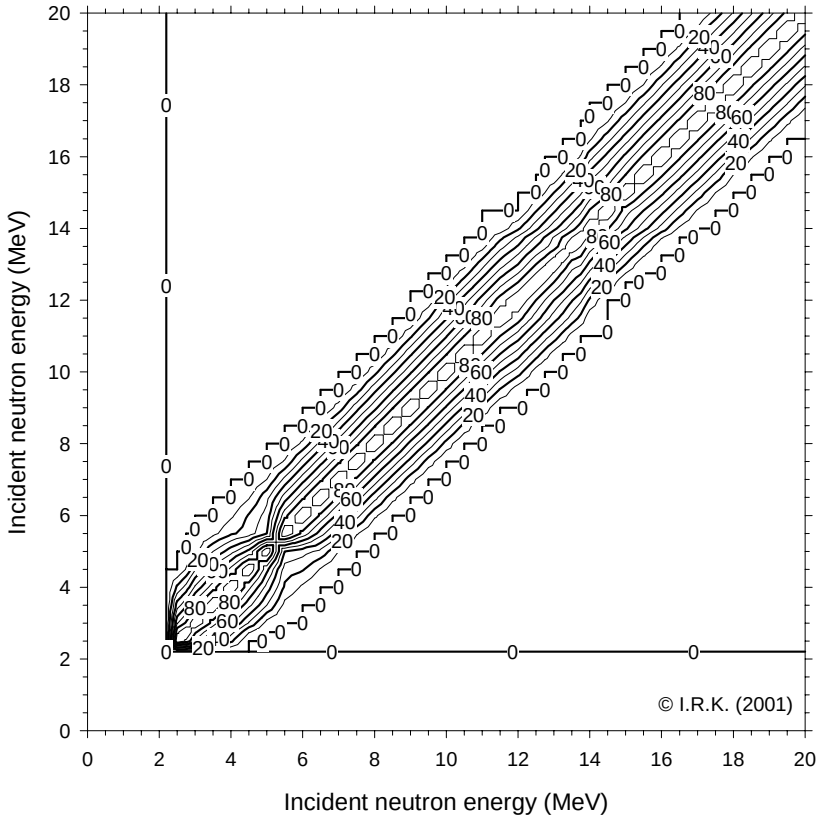


Fig. 32 Correlation matrix for the (n, d) cross section

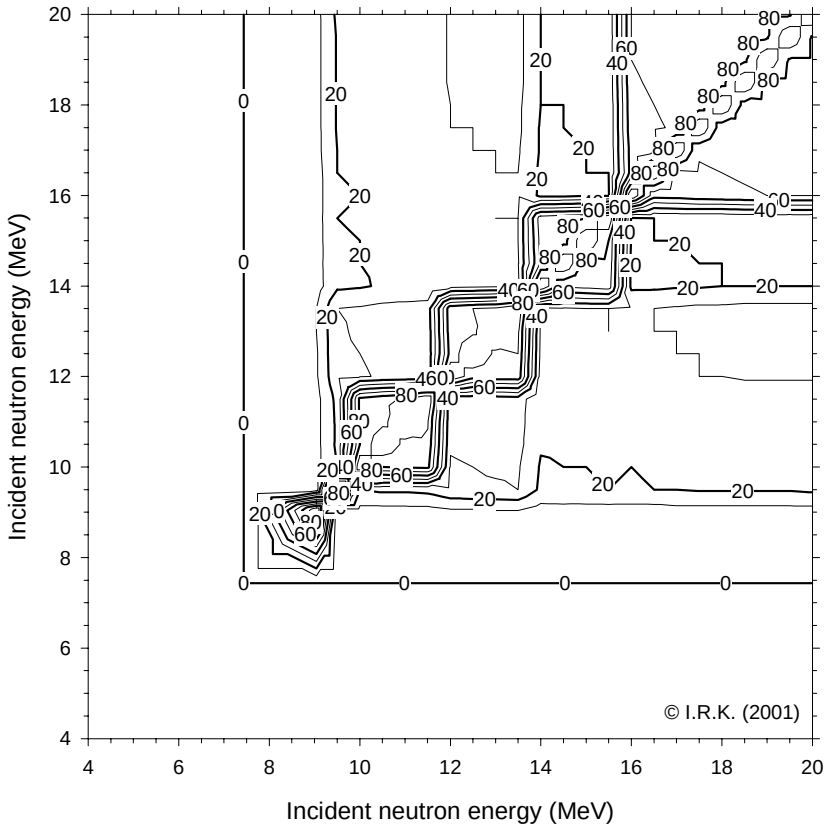


Fig. 33 Correlation matrix for the (n, t) cross section

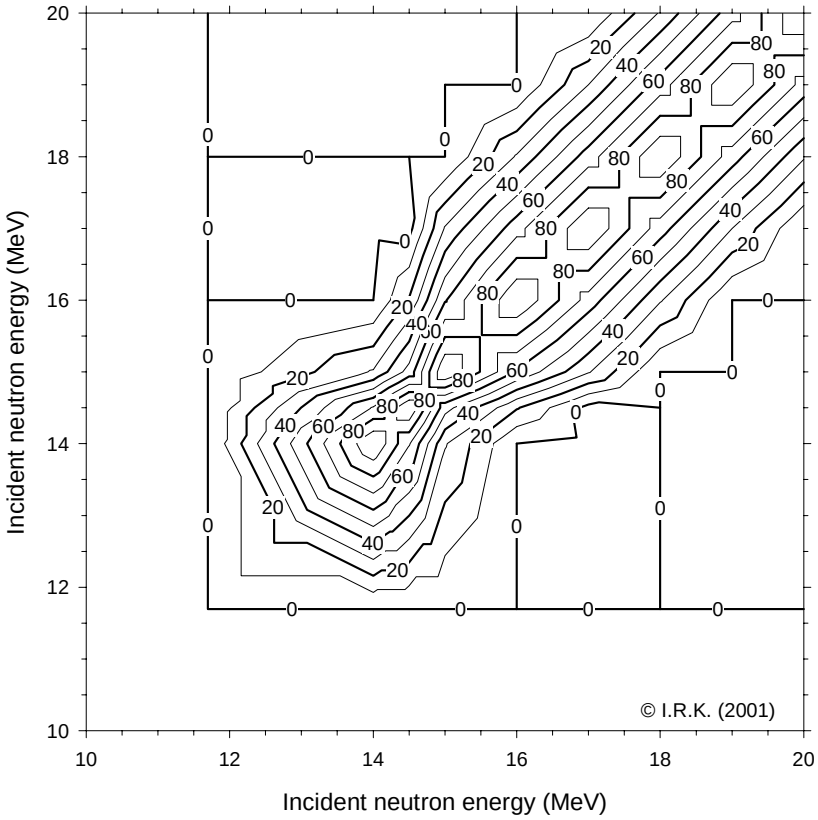


Fig. 34 Correlation matrix for the (n, ^{3}He) cross section

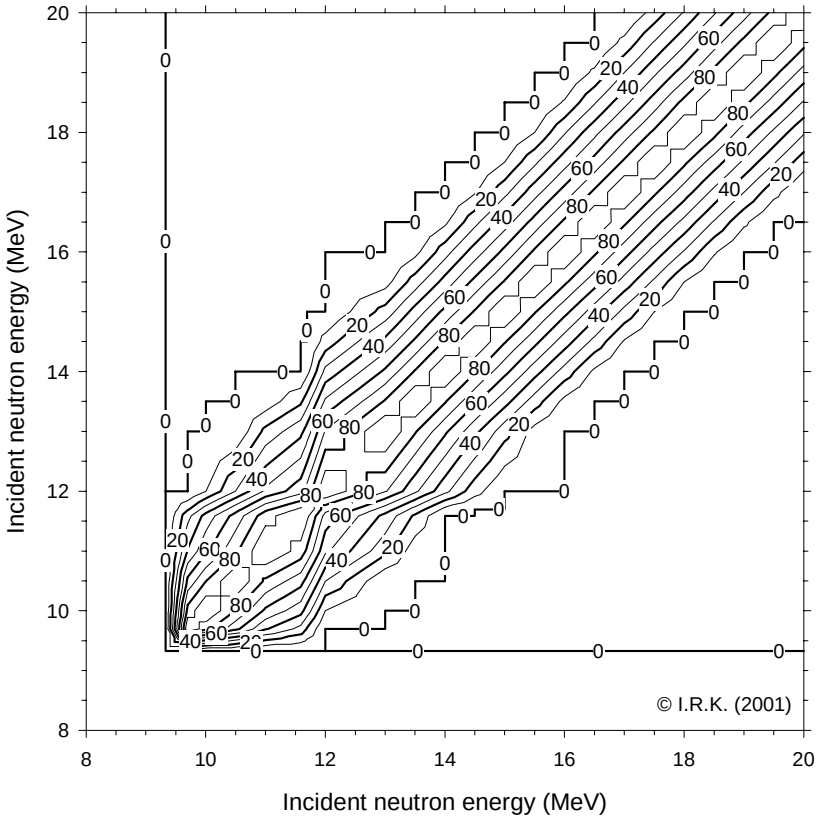
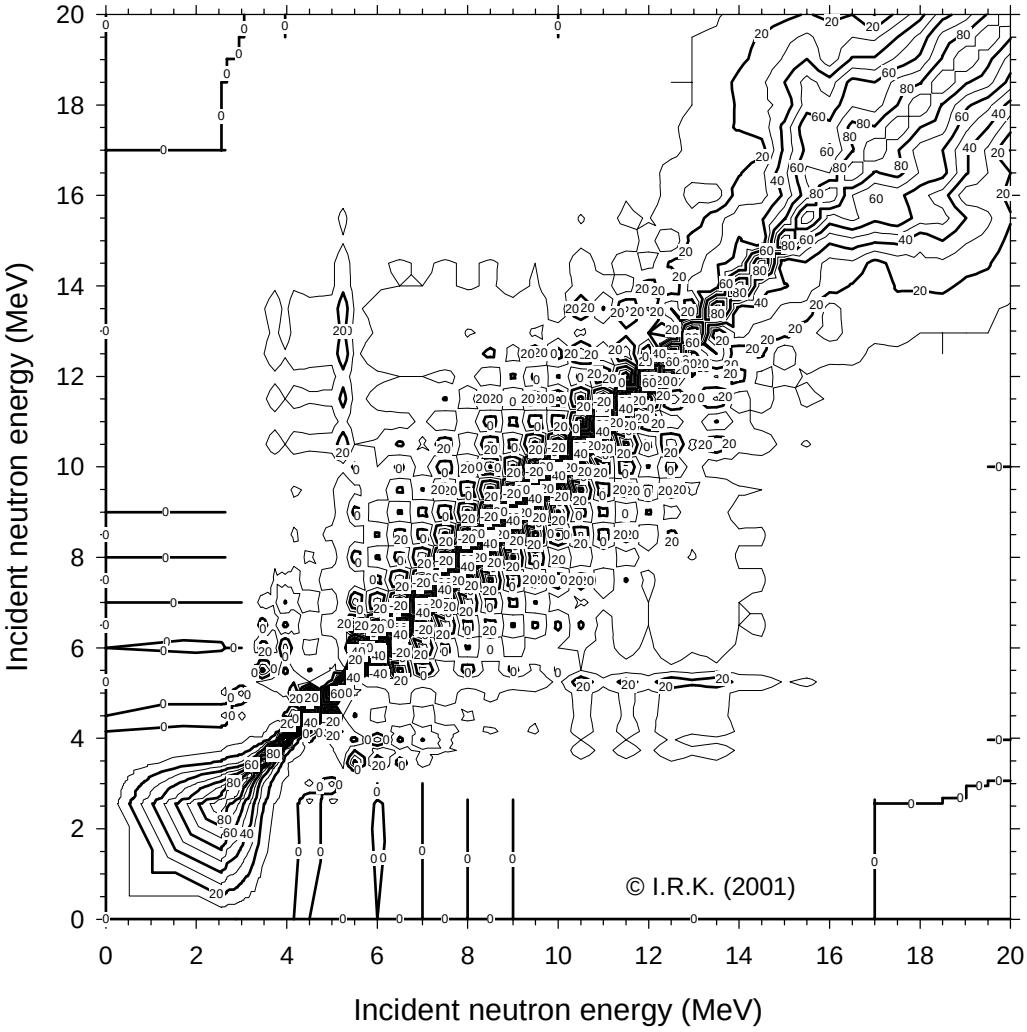


Fig. 35 Correlation matrix for the (n, α) cross section



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		RIPL for FTP file transfer of RIPL;
		NDSONL for FTP access to files sent to NDIS “open” area.
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