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AN EVALUATION OF THE Nb–93(n,n′)Nb–93m DOSIMETER REACTION FOR ENDF/B–VI*

by

Donald L. Smith and Luiz P. Geraldo**

November 1990

EVALUATION OF Nb–93(n,n′)Nb–93m. Experimental data. Nuclear model calculations. Cross section errors. Least–squares adjustment.

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TABLE OF CONTENTS

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AN EVALUATION OF THE
Nb–93(n,n′)Nb–93m DOSIMETER REACTION
FOR ENDF/B–VI

Donald L. Smith and Luiz P. Geraldo

ABSTRACT

A differential cross section evaluation for the Nb–93(n,n′)Nb–93m dosimetry reaction from threshold to 20 MeV has been prepared for ENDF/B–VI. This evaluation was obtained by adjusting the results of nuclear model calculations using published differential cross section information. Uncertainties for the measured differential cross sections were derived from information provided in the literature. The uncertainties associated with the model–calculated curve were estimated indirectly through consideration of the total cross section and all other prominent reaction channels. Experimental and model–calculated information was then merged by means of the least–squares method to provide a final evaluation and its covariance matrix.

I. INTRODUCTION

The Nb–93(n,n′)Nb–93m reaction plays an important role in nuclear energy applications. Because of its low threshold energy and relatively long half-life, it is a desirable reaction for long–term neutron fluence dosimetry in nuclear fission reactors. An evaluation of the differential cross section for this reaction was completed in 1985 by this laboratory as part of a comprehensive effort involving all neutron cross sections for niobium. The objective was to provide input for ENDF/B–VI [S+85].

It was difficult to produce a reliable evaluation for this reaction in 1985 because the information available then was sparse and quite uncertain. In fact, that evaluation was based entirely on nuclear model calculations. The evaluated cross sections below 0.7 MeV were derived from calculations carried out in this laboratory by Smith et al. [S+85], while the higher energy values were obtained from the work of Strohmaier and co-workers [STV80,Str82]. In 1985 there was only one published experimental differential cross section value to consider for this reaction (near 14 MeV). Even the half-life of Nb–93m was in serious doubt [S+85]. During the five years between the completion of the earlier evaluation and the finalization of ENDF/B–VI there have been some significant improvements and additions to the experimental database for this reaction. Also, new model calculations have been performed. Therefore, it was considered worthwhile to produce a new evaluation of Nb–93(n,n′)Nb–93m for ENDF/B–VI which would supplant the one that had been completed in 1985.

Details of the present evaluation are discussed in Sections I and II. The basic strategy is outlined below. It was recognized that there have been notable improvements in our knowledge of the decay half-life of Nb–93m (see Section II). This new understanding was taken into consideration in preparing the present evaluation. Although new experimental differential data were available to be considered in this work, this database is still not adequate alone to define the differential cross section from threshold to 20 MeV. Therefore, strong reliance has again been placed on nuclear modeling. Below 0.7 MeV, the earlier calculations of Smith et al. [S+85] were retained. In this threshold region, only three reaction channels must be considered, namely, elastic scattering, inelastic scattering, and neutron capture. These calculations were based on the optical/statistical model. The parameterization of the model that was used by Smith et al. [S+85], as reported in the documentation for their earlier comprehensive evaluation effort [S+85], continues to provide a very adequate representation of what is known about this aspect of the physics of neutron interactions with niobium at these low energies. Above 0.7 MeV, the situation has changed considerably from 1985. Two recent theoretical studies, i.e., by Odano et al. [OIS89] and Strohmaier [Str89], produced values which differ significantly from results of the earlier investigation by Strohmaier and co-workers [STV80,Str82]. The recent nuclear-model results from Strohmaier appear to be in rather good agreement with experimental data, while the predictions from Odano et al. are significantly lower (around 30% in the 2–11 MeV energy range). Furthermore, at 0.7 MeV, the values of Strohmaier are in excellent agreement with calculations by Smith et al. For these reasons, it was decided that these latest values from Strohmaier ought to be utilized in the evaluation process above 0.7 MeV. However, on this occasion it was decided that the newly available differential data should also be incorporated. Therefore, the procedure adopted was to generate an evaluation which merges the results from nuclear modeling with contemporary experimental differential cross–section data. The method of least–squares, as embodied in the code GMA [Poe81], has been employed for this purpose.

II. INFORMATION USED IN THE EVALUATION

Values of the fundamental nuclear constants which were employed in the present evaluation are given in Table 1. The origins of these values are documented in the references of that table. The most dramatic improvement in the knowledge of these constants during the last several years was found to be in the Nb–93m decay half-life. Knowledge of this half-life is critical because it impacts on experimental determinations of the reaction cross section and on the use of Nb–93(n,n′)Nb–93m as a dosimeter.

As pointed out in Section I, the model calculations of Smith et al. [S+85] were incorporated in the present evaluation for neutron energies below 0.7 MeV, while those of Strohmaier [Str89] were included for the energy range 0.7–20 MeV. Since the results from these two sources are in excellent agreement at 0.7 MeV, no ambiguity arises. Cross section values were deduced from these works at thirty–three grid–point neutron energies. These particular energies were selected because they provide an adequate representation of the excitation function with a relatively small number of points. These values are presented in Table 2. Linear interpolation was employed in this analysis since the energies chosen for the present evaluation usually did not coincide with those energies at which the authors of the model calculations reported their respective results.

The generation of plausible uncertainties for these cross sections was a far more subjective enterprise. The procedure employed in the present evaluation is discussed below.

In this evaluation it is assumed that the overall uncertainty of the model–calculated Nb–93(n,n′)Nb–93m reaction cross section below ≈ 0.7 MeV is ≈ 30% (which is a larger error than the 25% attributed to this energy range in an earlier evaluation from this laboratory [S+85]). The rationale is as follows: Below ≈ 0.7 MeV, the Nb–93(n,n′)Nb–93m reaction is entirely dominated by neutron–inelastic–scattering excitation of the 1/2– first–excited state of Nb–93 at 0.03082 MeV. This state is, in fact, the isomer. Since the ground state of Nb–93 is 9/2+, and a large orbital angular momentum change is involved in the excitation process, the cross section is quite small and difficult to calculate reliably. This cross section is also extremely hard to determine experimentally by conventional neutron detection methods (e.g., time–of–flight spectroscopy). The key point is that the small component of inelastically scattered neutrons is virtually impossible to resolve from the much larger elastic component at such a low excitation energy. Consequently, this cross section is best determined from optical/statistical model calculations. The uncertainty should be consistent with what one might expect for nuclear model results under such conditions. In this case, it was conservatively taken to be ≈ 30%. This error ought to be treated as only partially correlated in this energy region. For a reason to be discussed below, a correlated error component of ≈ 20% was assumed.

Additional neutron–interaction processes become involved at higher energies. For Nb–93, the dominant processes contributing to the neutron total cross section below 20 MeV are: elastic scattering, inelastic scattering, and the (n,2n) process. The contributions from neutron capture and other reaction channels, denoted collectively by (n,X), are relatively small and, therefore, they can be safely neglected for the sake of the following discussion. For neutron energies > 0.7 MeV, the production of Nb–93m by neutron inelastic scattering is dominated by feeding from the higher levels of Nb–93 rather than by direct excitation of the isomeric state. These higher levels de–excite electromagnetically in one of two ways: (1) through transitions which ultimately feed the isomeric state, or (2) through transitions which ultimately feed the ground state. For energies > 1.5 MeV, so many levels of Nb–93 are involved that a statistical approach can be used to treat the process. In consequence, it is found that the ratio of the cross section for Nb–93(n,n′)Nb–93m, as calculated by Strohmaier [Str89], to the total inelastic scattering cross section, as determined by a completely independent method described in the earlier evaluation from this laboratory [S+85], is constant (to within ≈ 20%) from ≈ 1.5 MeV to 20 MeV. This important result suggests adapting the following plausible strategy for estimating the error in the Nb–93(n,n′)Nb–93m cross section over most of the energy range of this evaluation: For 0.7–20 MeV, we assume that the total inelastic cross section is approximately equivalent to the difference between the total cross section and the sum of the angle–integrated elastic scattering cross section and the (n,2n) cross section. Rational estimates for the uncertainties of these quantities are available from earlier work in this laboratory. Thus, the uncertainty in the the total inelastic scattering cross section can be readily derived from error propagation. It is then assumed that this source of error can be considered as random (uncorrelated) from 0.7–20 MeV. However, to determine the total error in the Nb–93(n,n′)Nb–93m cross section, it is necessary to also fold in a systematic error due to the above–mentioned isomer factor (i.e.,≈ 20%). Finally, we arbitrarily treat this error as fully correlated over the entire energy range of this evaluation, not just for energies above 0.7 MeV. This is the origin of the ≈ 20% component mentioned in the preceding paragraph.

The random and systematic errors which emerged from this analysis appear in Table 3. The corresponding correlation matrix is given in Table 4. Although subjectively generated, this information provides a quite reasonable representation of the cross section uncertainties, and one with acceptable properties from the point of view of data adjustment [Smi87].

Next we turn to an examination of the experimental database. In the present evaluation, consideration was restricted to well–documented, energy–differential measurements which explicitly represented determinations of the activation cross section for production of Nb–93m by the Nb–93(n,n′)Nb–93m reaction. Consequently, only three data sets appeared to qualify for inclusion in the present evaluation: Ryves and Kolkowski [RK81], Gayther et al. [G+87], and Wagner et al. [W+88a,b]. Each of these data sets has been documented in the literature, with detailed information provided on the nuclear constants employed in the data analysis and on the experimental uncertainties. The reader is referred to the original papers for such details, since they are not reproduced in this report. Prior to the evaluation, the quoted values were adjusted as required to account for any known revisions in these nuclear constants, consistent with the information in Table 1. The value from Ryves and Kolkowski had to be increased by 4.6% to account for changes in the Nb–93m half-life, the neutron fluence standard and the K_{alpha} X–ray relative intensity (see Table 1). The data of Gayther et al. needed to be reduced by only 0.2%, a negligible amount. Finally, the results of Wagner et al. needed to be reduced by 2.9%, a relatively modest amount which could be attributed to a change in the mass scale for the calibrated uranium deposit which was used in the neutron fluence measurement for this experiment [M+89]. The uncertainties were generally deduced from information provided in the original papers. The exception was for the fundamental constants, where relevant errors suggested by the authors were superseded by correspondingly more recent values from Table 1. Error correlations were determined for the data in each set, but cross correlations between sets were neglected. The reason is that the half-life is the main source of error common to each set, and its uncertainty is relatively small compared to most of the other error sources. Tables 5 and 6 summarize the experimental input to the present evaluation process.

III. LEAST–SQUARES EVALUATION PROCEDURE AND RESULTS

The information described in Section II formed the basis for a least–squares evaluation procedure which employed the computer code GMA [Poe81]. The actual version of GMA that was used in this analysis is one which had been modified by S. Chiba [Chi90]. It is designed to run on a personal computer and incorporates a chi–square test for checking the consistency of all input parameters, a valuable feature which is not available in the original version of GMA.

The results of this evaluation appear in Table 7. The least–squares adjustment procedure yielded a chi–square per degree of freedom of 1.01, indicating very acceptable consistency for the input information. The estimated errors in the evaluation are also presented in Table 7, and the corresponding correlations are given in Table 8. Since the errors in the measured data are considerably smaller than those for the a priori cross section based on nuclear–model calculations, the impact of these experimental data on the evaluation is substantial. Since these data and the results from model calculations are reasonably consistent, the obvious changes in the evaluated cross section relative to the nuclear model results are relatively modest. Therefore, the greatest impact of the experimental information is to reduce the uncertainties in the evaluated cross section relative to the a priori. The situation is summarized in Fig. 1. The evaluated excitation function is not as "smooth" as the a priori cross section curve, which is based entirely on nuclear modeling results. This structure cannot be attributed to any known physical phenomena. It is an artifact resulting from the numerical procedure used in this evaluation. Nevertheless, the present evaluation represents the best possible knowledge of the Nb–93(n,n′)Nb–93m reaction differential cross section that the available differential information (experimental and theoretical) can provide at this time. One could smooth the evaluated curve by using a spline–fitting technique. However, this subjective action would be a cosmetic one which would serve no useful purpose. In fact it might actually be misleading. For this reason, we chose not to pursue this matter. It is clear that it would require many more experimental data points, or a much more definitive and reliable nuclear modeling procedure, in order to significantly improve our knowledge of the shape of the excitation function.

The results of the present evaluation are documented in ENDF/B–VI format in the Appendix. This information has also been transmitted to the National Nuclear Data Center at Brookhaven National Laboratory for inclusion in ENDF/B–VI.

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Table 1: Fundamental constants for niobium and Nb-93(n,n')Nb-93m

Isotopic abundance of Nb-93:	100% (mono isotopic)
Mass of Nb-93:	92.906376 a.m.u. [WPF89]
Reaction Q-value:	−0.03082 ± 0.00017 MeV [Sie88]
Reaction threshold:	0.03115 MeV (based on Q-value)
Half life of Nb-93m:	16.1 ± 0.2 years [Llo81,Van83,SS86]
Internal conversion factor:	$I_{electron}/I_{gamma}$ = 181800 [Sie88]
X-ray energies:	K_{alpha} = 16.6 keV, K_{beta} = 18.6 keV [Sie88]
X-ray emission probability:	p_k = 0.1114 ± 0.0020 [ASS89]
X-ray intensities:	K_{alpha} = 0.09238, K_{beta} = 0.01802 [Pes82]
Ratio of K_{beta} to K_{alpha} intensities:	0.195 ± 0.001 [Pes82]

Table 2: Model–calculated cross sections for Nb–93(n,n′)Nb–93m which are incorporated in the present evaluation[a]

Grid Point	Energy (MeV)	Cross Section (mb)
NA[b]	0.0	0.0
NA[b]	0.03115	0.0
1	0.04	0.06
2	0.05	0.15
3	0.06	0.25
4	0.08	0.52
5	0.1	0.86
6	0.15	1.97
7	0.2	3.43
8	0.3	7.19
9	0.4	11.88
10	0.5	17.27
11	0.6	23.17
12	0.7	28.95
13	0.7625	40.0
14	1.0	89.80
15	1.288	111.2
16	1.75	209.2
17	2.063	210.6
18	2.563	257.1
19	4.05	260.2
20	4.588	258.2
21	6.156	245.5
22	8.344	225.5
23	9.375	207.1
24	10.375	133.7
25	11.394	83.67
26	12.394	57.14
27	13.406	43.47
28	14.356	35.71
29	15.52	30.61
30	17.0	26.94
31	20.0	22.45

[a] The values at neutron energies below 0.7 MeV are from the work of Smith et al. [S+85] while those at higher energies are from Strohmaier [Str89], as discussed in Section II.

[b] Energies at or below threshold are not included in the evaluation grid structure for uncertainty purposes.

Table 3: Estimated errors for the model-calculated cross sections of Nb-93(n,n')Nb-93m

Grid Point	Energy (MeV)	Error Components (%) Random*	Systematic**	Total Error (%)
1	0.04	≈ 22	20	≈ 30
2	0.05	≈ 22	20	≈ 30
3	0.06	≈ 22	20	≈ 30
4	0.08	≈ 22	20	≈ 30
5	0.1	≈ 22	20	≈ 30
6	0.15	≈ 22	20	≈ 30
7	0.2	≈ 22	20	≈ 30
8	0.3	≈ 22	20	≈ 30
9	0.4	≈ 22	20	≈ 30
10	0.5	≈ 22	20	≈ 30
11	0.6	≈ 22	20	≈ 30
12	0.7	≈ 22	20	≈ 30
13	0.7625	15	20	25
14	1.0	10	20	22.4
15	1.288	10	20	22.4
16	1.75	8.1	20	21.6
17	2.063	5	20	20.6
18	2.563	5	20	20.6
19	4.05	5	20	20.6
20	4.588	5	20	20.6
21	6.156	6	20	20.9
22	8.344	6	20	20.9
23	9.375	6	20	20.9
24	10.375	6	20	20.9
25	11.394	15	20	25
26	12.394	22	20	29.7
27	13.406	25	20	32
28	14.356	30	20	36.1
29	15.52	35	20	40.3
30	17.0	38	20	42.9
31	20.0	36	20	41.2

* Errors are completely uncorrelated from one grid-point energy to another.

** Errors are 100% correlated between all grid-point energies.

Table 4: Correlations of the estimated errors for the calculated cross sections of Nb-93(n,n')Nb-93m*

	1	2	3	4	5	6	7	8	9	10	11	12
	13	14	15	16	17	18	19	20	21	22	23	24
	25	26	27	28	29	30	31					
1	1000											
2	452	1000										
3	452	452	1000									
4	452	452	452	1000								
5	452	452	452	452	1000							
6	452	452	452	452	452	1000						
7	452	452	452	452	452	452	1000					
8	452	452	452	452	452	452	452	1000				
9	452	452	452	452	452	452	452	452	1000			
10	452	452	452	452	452	452	452	452	452	1000		
11	452	452	452	452	452	452	452	452	452	452	1000	
12	452	452	452	452	452	452	452	452	452	452	452	1000
13	538	538	538	538	538	538	538	538	538	538	538	538
	1000											
14	602	602	602	602	602	602	602	602	602	602	602	602
	716	1000										
15	602	602	602	602	602	602	602	602	602	602	602	602
	716	800	1000									
16	623	623	623	623	623	623	623	623	623	623	623	623
	741	829	829	1000								
17	653	653	653	653	653	653	653	653	653	653	653	653
	776	868	868	899	1000							
18	653	653	653	653	653	653	653	653	653	653	653	653
	776	868	868	899	941	1000						
19	653	653	653	653	653	653	653	653	653	653	653	653
	776	868	868	899	941	941	1000					
20	653	653	653	653	653	653	653	653	653	653	653	653
	776	868	868	899	941	941	941	1000				
21	644	644	644	644	644	644	644	644	644	644	644	644
	766	857	857	888	929	929	929	929	1000			
22	644	644	644	644	644	644	644	644	644	644	644	644
	766	857	857	888	929	929	929	929	917	1000		
23	644	644	644	644	644	644	644	644	644	644	644	644
	766	857	857	888	929	929	929	929	917	917	1000	
24	644	644	644	644	644	644	644	644	644	644	644	644
	766	857	857	888	929	929	929	929	917	917	917	1000

Table 4: (continued)

	1	2	3	4	5	6	7	8	9	10	11	12
	13	14	15	16	17	18	19	20	21	22	23	24
	25	26	27	28	29	30	31					
25	538	538	538	538	538	538	538	538	538	538	538	538
	640	716	716	741	776	776	776	776	766	766	766	766
	1000											
26	452	452	452	452	452	452	452	452	452	452	452	452
	538	602	602	623	653	653	653	653	644	644	644	644
	538	1000										
27	420	420	420	420	420	420	420	420	420	420	420	420
	500	559	559	579	606	606	606	606	598	598	598	598
	500	420	1000									
28	373	373	373	373	373	373	373	373	373	373	373	373
	444	496	496	514	538	538	538	538	531	531	531	531
	444	373	347	1000								
29	334	334	334	334	334	334	334	334	334	334	334	334
	397	444	444	460	481	481	481	481	475	475	475	475
	397	334	310	275	1000							
30	313	313	313	313	313	313	313	313	313	313	313	313
	373	417	417	432	452	452	452	452	446	446	446	446
	373	313	291	258	231	1000						
31	327	327	327	327	327	327	327	327	327	327	327	327
	389	434	434	450	471	471	471	471	465	465	465	465
	389	327	303	269	241	226	1000					

* Indices for the correlation matrices correspond in order to the cross section values listed in Tables 2 and 3. Correlations are scaled so 1000 is equivalent to full correlation.

Table 5: Experimental differential cross sections of Nb-93(n,n')Nb-93m which are included in the present evaluation

Authors	Energy (MeV)	Adjusted Cross Section (mb)*	Errors (%) Random	Errors (%) Systematic	Errors (%) Total
Ryves and Kolkowski [RK81]	14.3	38.168	8.2	NA**	8.2
Gayther et al. [G+87]	1.09	78.0	4.3	4.6	6.3
	2.10	206.0	3.1	4.6	5.5
	2.70	246.0	1.5	4.6	4.8
	3.29	260.0	4.1	4.6	6.2
	3.73	280.0	2.2	4.6	5.1
	4.03	273.0	7.3	4.6	8.6
	4.45	255.0	2.8	4.6	5.4
	5.05	270.0	3.5	4.6	5.8
	5.53	254.0	3.3	4.6	5.7
	5.76	240.0	3.9	4.6	6.0
Wagner et al. [W+88a,b]	2.83	245.0	4.2	2.2	4.7
	7.91	258.2	3.6	2.2	4.2

* See the discussion in Section II of the text.
** Irrelevant since there is only a single data point.

Table 6: Uncertainty correlations of the experimental differential cross sections of Nb-93(n,n')Nb-93m which are included in the present evaluation*

Gayther et al. [G+87]:

	1	2	3	4	5	6	7	8	9	10
1	1000									
2	611	1000								
3	700	802	1000							
4	542	621	711	1000						
5	659	754	864	669	1000					
6	391	447	513	397	482	1000				
7	622	712	816	632	768	456	1000			
8	579	663	760	588	715	424	676	1000		
9	589	675	773	599	728	432	687	640	1000	
10	560	641	735	569	691	410	653	608	619	1000

Wagner et al. [W+88a,b]:

	1	2
1	1000	
2	245	1000

* Indices for the correlation matrices correspond in order to the cross section values listed in Table 5. Correlations are scaled so 1000 is equivalent to full correlation.

Table 7: Evaluated cross sections for Nb-93(n,n')Nb-93m which result from the least-squares procedure*

Grid Point	Energy (MeV)	Cross Section (mb)	Total Error (%)
NA*	0.0	0.0	---
NA*	0.03115	0.0	---
1	0.04	0.06053	22.3
2	0.05	0.1513	22.3
3	0.06	0.2522	22.3
4	0.08	0.5246	22.3
5	0.1	0.8676	22.3
6	0.15	1.987	22.3
7	0.2	3.460	22.3
8	0.3	7.253	22.3
9	0.4	11.98	22.3
10	0.5	17.42	22.3
11	0.6	23.37	22.3
12	0.7	29.21	22.3
13	0.7625	40.35	15.5
14	1.0	74.84	5.1
15	1.288	112.2	10.7
16	1.75	211.0	8.9
17	2.063	206.0	4.1
18	2.563	248.6	3.2
19	4.05	278.2	3.7
20	4.588	262.9	3.7
21	6.156	245.5	3.9
22	8.344	244.6	3.8
23	9.375	208.9	7.0
24	10.375	134.9	7.0
25	11.394	84.41	15.5
26	12.394	57.64	22.3
27	13.406	43.85	25.3
28	14.356	37.56	8.0
29	15.52	30.88	35.2
30	17.0	27.18	38.2
31	20.0	22.65	36.2

* Energies at or below threshold are not included in the evaluation grid structure for uncertainty purposes.

Table 8: Correlations of the evaluated cross sections for Nb-93(n,n')Nb-93m*

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31
1	1000																														
2	26	1000																													
3	26	26	1000																												
4	26	26	26	1000																											
5	26	26	26	26	1000																										
6	26	26	26	26	26	1000																									
7	26	26	26	26	26	26	1000																								
8	26	26	26	26	26	26	26	1000																							
9	26	26	26	26	26	26	26	26	1000																						
10	26	26	26	26	26	26	26	26	26	1000																					
11	26	26	26	26	26	26	26	26	26	26	1000																				
12	26	26	26	26	26	26	26	26	26	26	26	1000																			
13	38	38	38	38	38	38	38	38	38	38	38	38	1000																		
14	83	83	83	83	83	83	83	83	83	83	83	83	120	1000																	
15	55	55	55	55	55	55	55	55	55	55	55	55	80	174	1000																
16	66	66	66	66	66	66	66	66	66	66	66	66	96	208	139	1000															
17	109	109	109	109	109	109	109	109	109	109	109	109	158	468	229	274	1000														
18	119	119	119	119	119	119	119	119	119	119	119	119	172	562	250	299	688	1000													
19	116	116	116	116	116	116	116	116	116	116	116	116	167	523	242	290	644	771	1000												
20	115	115	115	115	115	115	115	115	115	115	115	115	166	518	240	288	638	764	712	1000											
21	108	108	108	108	108	108	108	108	108	108	108	108	157	494	227	273	608	729	679	672	1000										
22	68	68	68	68	68	68	68	68	68	68	68	68	99	237	143	172	306	364	330	327	310	1000									
23	84	84	84	84	84	84	84	84	84	84	84	84	121	263	176	211	347	379	367	365	345	217	1000								
24	84	84	84	84	84	84	84	84	84	84	84	84	121	263	176	211	347	379	367	365	345	217	267	1000							

Table 8: (continued)

	1	2	3	4	5	6	7	8	9	10	11	12
	13	14	15	16	17	18	19	20	21	22	23	24
	25	26	27	28	29	30	31					
25	38	38	38	38	38	38	38	38	38	38	38	38
	55	120	80	96	158	172	167	166	157	99	121	121
	1000											
26	26	26	26	26	26	26	26	26	26	26	26	26
	38	83	55	66	109	119	116	115	108	6[illegible]	84	84
	38	1000										
27	23	23	23	23	23	23	23	23	23	23	23	23
	34	73	49	58	96	105	102	101	96	60	74	74
	34	23	1000									
28	5	5	5	5	5	5	5	5	5	5	5	5
	8	17	11	14	22	25	24	24	22	14	17	17
	8	5	5	1000								
29	17	17	17	17	17	17	17	17	17	17	17	17
	24	52	35	42	69	75	73	73	69	43	53	53
	24	17	15	3	1000							
30	15	15	15	15	15	15	15	15	15	15	15	15
	22	48	32	39	64	70	67	67	63	40	49	49
	22	15	14	3	10	1000						
31	16	16	16	16	16	16	16	16	16	16	16	16
	23	51	34	41	67	73	71	71	67	42	52	52
	23	16	14	3	10	9	1000					

* Indices for the correlation matrices correspond in order to the cross section values listed in Table 7. Correlations are scaled so 1000 is equivalent to full correlation.

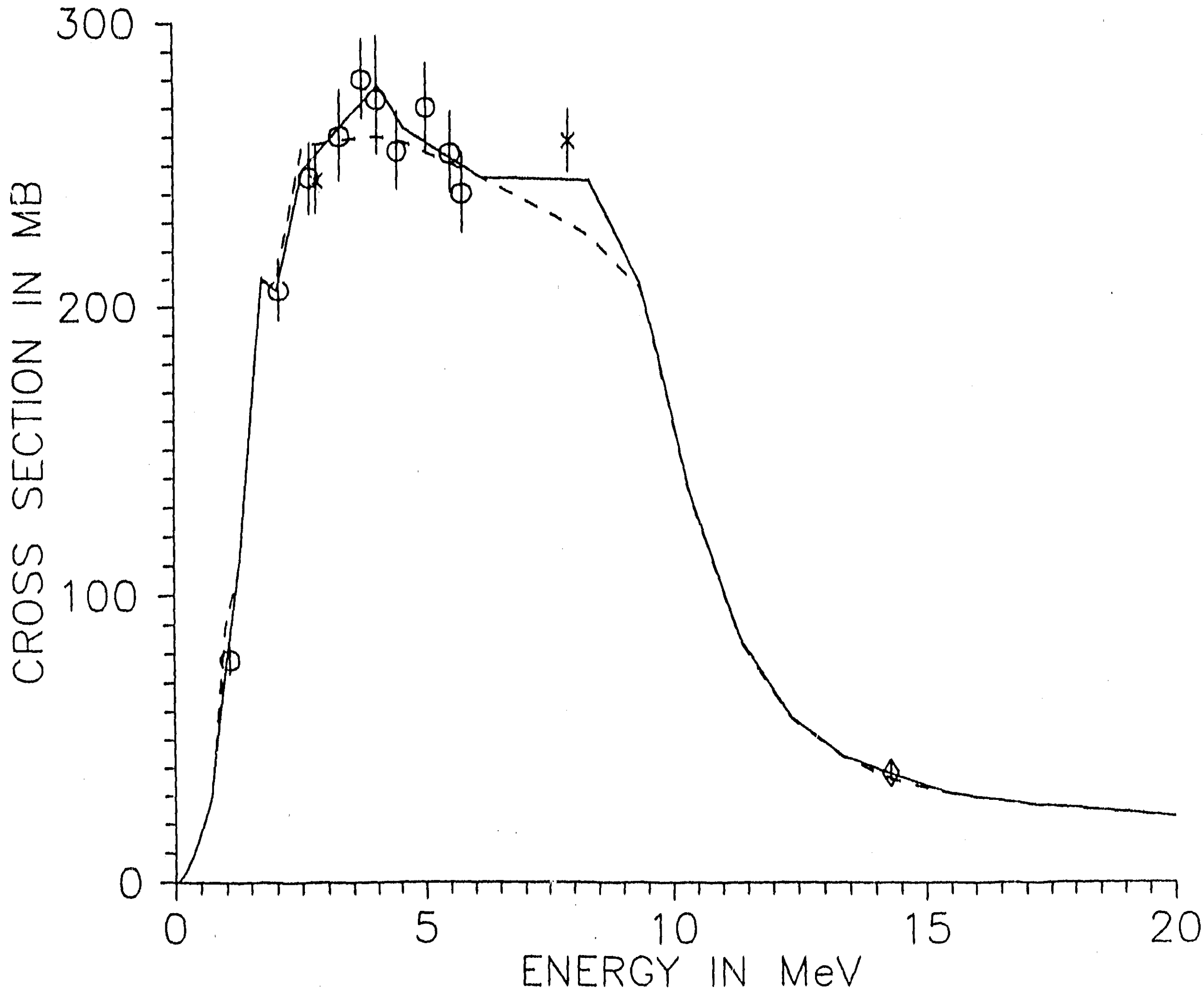

Fig. 1: Differential cross sections for Nb–93(n,n′)Nb–93m: Gayther et al. [G+87], symbol "O"; Ryves and Kolkowski [RK81], symbol "<>"; Wagner et al. [W+88a,b], symbol "X"; a priori based on nuclear model calculations (see Section II), dashed curve; present evaluation, solid curve. Error bars are shown for the experimental data points only.

APPENDIX

Results of the Evaluation in ENDF/B Format

```
 4.10930+ 1 9.21051+ 1          -1          0          0          1MAT# 1451    1
 0.00000+ 0 0.00000+ 0           0          0          0          6MAT# 1451    2
 0.10000+ 1 0.00000+ 0           0          0         10          6MAT# 1451    3
 0.00000+ 0 0.00000+ 0           0          0         42          4MAT# 1451    4
 41-NB-093 ANL        EVAL-MAR90 D.L. SMITH AND L.P. GERALDO       MAT# 1451    5
 REPORT ANL/NDM-117   DIST-DEC90 REVO-DEC90            901231      MAT# 1451    6
----ENDF/B-VI         MAT#                 REVISION 0              MAT# 1451    7
-----INCIDENT NEUTRON DATA                                         MAT# 1451    8
------ENDF-6                                                       MAT# 1451    9
                                                                   MAT# 1451   10
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *  MAT# 1451   11
                                                                   MAT# 1451   12
NB-93(N,N')NB-93M DOSIMETRY REACTION                               MAT# 1451   13
                                                                   MAT# 1451   14
EVALUATED BY D.L. SMITH AND L.P. GERALDO, ANL, MARCH 1990.         MAT# 1451   15
                                                                   MAT# 1451   16
PRODUCTION OF THE ISOMER NB-93M BY THE (N,N') PROCESS IS           MAT# 1451   17
ROUTINELY EMPLOYED FOR NEUTRON DOSIMETRY APPLICATIONS. ISOMER      MAT# 1451   18
IS THE FIRST-EXCITED STATE OF THE ISOTOPE NB-93 (30.82 KEV         MAT# 1451   19
EXCITATION ENERGY). THE REACTION THRESHOLD ENERGY IS 31.15 KEV.    MAT# 1451   20
THE ISOTOPIC ABUNDANCE OF NB-93 IN NATURAL NIOBIUM IS 100          MAT# 1451   21
PERCENT. THE HALF LIFE OF NB-93M IS 16.1 YEARS. THE DECAY          MAT# 1451   22
IS ENTIRELY BY ISOMERIC TRANSITION WITH NEARLY 100 PERCENT         MAT# 1451   23
INTERNAL CONVERSION. ACTIVITY MEASUREMENT IS BY OBSERVATION OF     MAT# 1451   24
X-RAYS. X-RAY YIELDS: 16.6 KEV K-ALPHA (0. 09238 PER               MAT# 1451   25
DISINTEGRATION), 18.6 KEV K-BETA (0.01802 PER DISINTEGRATION).     MAT# 1451   26
                                                                   MAT# 1451   27
EVALUATION IS BASED ON A LEAST-SQUARES ADJUSTMENT PROCEDURE.       MAT# 1451   28
INPUT INFORMATION INCLUDES RESULTS FROM NUCLEAR MODEL              MAT# 1451   29
CALCULATIONS AND RECENT DIFFERENTIAL ACTIVATION CROSS SECTION      MAT# 1451   30
DATA FROM THE LITERATURE. UNCERTAINTIES ARE DERIVED FROM           MAT# 1451   31
EXPERIMENTAL ERRORS AND THE CONSIDERATION OF SYSTEMATICS.          MAT# 1451   32
                                                                   MAT# 1451   33
EVALUATION DOCUMENTATION: D.L. SMITH AND L.P. GERALDO,             MAT# 1451   34
REPORT ANL/NDM-117, ARGONNE NATIONAL LABORATORY (1990).            MAT# 1451   35
```

```
                                                                  MAT# 1451   36
THE FOLLOWING FILE SECTIONS ARE INCLUDED -                        MAT# 1451   37
    MF= 1    GENERAL INFORMATION (MT=451)                         MAT# 1451   38
    MF= 8    RADIOACTIVITY DATA (MT=51)                           MAT# 1451   39
    MF=10    CROSS SECTION FOR RADIOACTIVE NUCLIDE                MAT# 1451   40
               PRODUCTION (MT=51)                                 MAT# 1451   41
    MF=40    DATA COVARIANCES FOR RADIOACTIVE NUCLIDE             MAT# 1451   42
               PRODUCTION (MT=51)                                 MAT# 1451   43
                                                                  MAT# 1451   44
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * MAT# 1451   45
                                                                  MAT# 1451   46
                                1          451         50        1MAT# 1451   47
                                8           51          4        1MAT# 1451   48
                               10           51         14        1MAT# 1451   49
                               40           51         92        1MAT# 1451   50
                                                                  MAT# 1  0   51
                                                                  MAT# 0  0   52
 4.10930+ 4 9.21051+ 1          0          0          1          0MAT# 8 51   53
 4.10930+ 4 3.08200+ 4         10          1          6       MAT#MAT# 8 51   54
 5.08118+ 8 3.00000+ 0 4.10930+ 4 1.00000+ 0 3.08200+ 4 1.00000+ 0MAT# 8 51   55
                                                                 0MAT# 8  0   56
                                                                  MAT# 0  0   57
 4.10930+ 4 9.21051+ 1          0          0          1          0MAT#10 51   58
 0.00000+ 0-3.08200+ 4          0          1          1         32MAT#10 51   59
        32          2                                             MAT#10 51   60
 3.11500+ 4 0.00000+ 0 4.00000+ 4 6.05300- 5 5.00000+ 4 1.51300- 4MAT#10 51   61
 6.00000+ 4 2.52200- 4 8.00000+ 4 5.24600- 4 1.00000+ 5 8.67600- 4MAT#10 51   62
 1.50000+ 5 1.98700- 3 2.00000+ 5 3.46000- 3 3.00000+ 5 7.25300- 3MAT#10 51   63
 4.00000+ 5 1.19800- 2 5.00000+ 5 1.74200- 2 6.00000+ 5 2.33700- 2MAT#10 51   64
 7.00000+ 5 2.92100- 2 7.62500+ 5 4.03500- 2 1.00000+ 6 7.48400- 2MAT#10 51   65
 1.28800+ 6 1.12200- 1 1.75000+ 6 2.11000- 1 2.06300+ 6 2.06000- 1MAT#10 51   66
 2.56300+ 6 2.48600- 1 4.05000+ 6 2.78200- 1 4.58800+ 6 2.62900- 1MAT#10 51   67
 6.15600+ 6 2.45500- 1 8.34400+ 6 2.44600- 1 9.37500+ 6 2.08900- 1MAT#10 51   68
 1.03750+ 7 1.34900- 1 1.13940+ 7 8.44100- 2 1.23940+ 7 5.76400- 2MAT#10 51   69
 1.34060+ 7 4.38500- 2 1.43560+ 7 3.75600- 2 1.55200+ 7 3.08800- 2MAT#10 51   70
 1.70000+ 7 2.71800- 2 2.00000+ 7 2.26500- 2                      MAT#10 51   71
                                                                  MAT#10  0   72
                                                                  MAT# 0  0   73
 4.10930+ 1 9.21051+ 1          0          0          1          0MAT#40 51   74
 .00000E+00-3.08200+ 4          0          1          0          1MAT#40 51   75
 1.00000+ 1 1.00000+ 0          0         51          0          1MAT#40 51   76
 .00000E+00 .00000E+00          1          5        528         32MAT#40 51   77
 .10000E-04 .40000E+05 .50000E+05 .60000E+05 .80000E+05 .10000E+06MAT#40 51   78
 .15000E+06 .20000E+06 .30000E+06 .40000E+06 .50000E+06 .60000E+06MAT#40 51   79
```

.70000E+06	.76250E+06	.10000E+07	.12880E+07	.17500E+07	.20630E+07	MAT#40	51	80
.25630E+07	.40500E+07	.45880E+07	.61560E+07	.83440E+07	.93750E+07	MAT#40	51	81
.10375E+08	.11394E+08	.12394E+08	.13406E+08	.14356E+08	.15520E+08	MAT#40	51	82
.17000E+08	.20000E+08	.49818E-01	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	83
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	84
.12953E-02	.12953E-02	.13104E-02	.93665E-03	.13074E-02	.13084E-02	MAT#40	51	85
.99480E-03	.84782E-03	.95176E-03	.95203E-03	.93867E-03	.57159E-03	MAT#40	51	86
.13154E-02	.13154E-02	.13104E-02	.12953E-02	.12983E-02	.89180E-04	MAT#40	51	87
.13368E-03	.12793E-03	.12935E-03	.49818E-01	.12953E-02	.12953E-02	MAT#40	51	88
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	89
.12953E-02	.12953E-02	.13104E-02	.93665E-03	.13074E-02	.13084E-02	MAT#40	51	90
.99480E-03	.84782E-03	.95176E-03	.95203E-03	.93867E-03	.57159E-03	MAT#40	51	91
.13154E-02	.13154E-02	.13104E-02	.12953E-02	.12983E-02	.89180E-04	MAT#40	51	92
.13368E-03	.12793E-03	.12935E-03	.49818E-01	.12953E-02	.12953E-02	MAT#40	51	93
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	94
.12953E-02	.13104E-02	.93665E-03	.13074E-02	.13084E-02	.99480E-03	MAT#40	51	95
.84782E-03	.95176E-03	.95203E-03	.93867E-03	.57159E-03	.13154E-02	MAT#40	51	96
.13154E-02	.13104E-02	.12953E-02	.12983E-02	.89180E-04	.13368E-03	MAT#40	51	97
.12793E-03	.12935E-03	.49818E-01	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	98
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.13104E-02	MAT#40	51	99
.93665E-03	.13074E-02	.13084E-02	.99480E-03	.84782E-03	.95176E-03	MAT#40	51	100
.95203E-03	.93867E-03	.57159E-03	.13154E-02	.13154E-02	.13104E-02	MAT#40	51	101
.12953E-02	.12983E-02	.89180E-04	.13368E-03	.12793E-03	.12935E-03	MAT#40	51	102
.49818E-01	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	MAT#40	51	103
.12953E-02	.12953E-02	.13104E-02	.93665E-03	.13074E-02	.13084E-02	MAT#40	51	104
.99480E-03	.84782E-03	.95176E-03	.95203E-03	.93867E-03	.57159E-03	MAT#40	51	105
.13154E-02	.13154E-02	.13104E-02	.12953E-02	.12983E-02	.89180E-04	MAT#40	51	106
.13368E-03	.12793E-03	.12935E-03	.49818E-01	.12953E-02	.12953E-02	MAT#40	51	107
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.13104E-02	.93665E-03	MAT#40	51	108
.13074E-02	.13084E-02	.99480E-03	.84782E-03	.95176E-03	.95203E-03	MAT#40	51	109
.93867E-03	.57159E-03	.13154E-02	.13154E-02	.13104E-02	.12953E-02	MAT#40	51	110
.12983E-02	.89180E-04	.13368E-03	.12793E-03	.12935E-03	.49818E-01	MAT#40	51	111
.12953E-02	.12953E-02	.12953E-02	.12953E-02	.12953E-02	.13104E-02	MAT#40	51	112
.93665E-03	.13074E-02	.13084E-02	.99480E-03	.84782E-03	.95176E-03	MAT#40	51	113
.95203E-03	.93867E-03	.57159E-03	.13154E-02	.13154E-02	.13104E-02	MAT#40	51	114
.12953E-02	.12983E-02	.89180E-04	.13368E-03	.12793E-03	.12935E-03	MAT#40	51	115
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.93665E-03	.13074E-02	.13084E-02	.99480E-03	.84782E-03	.95176E-03	MAT#40	51	117
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.93665E-03	.13074E-02	.13084E-02	.99480E-03	.84782E-03	.95176E-03MAT#40	51	128
.95203E-03	.93867E-03	.57159E-03	.13154E-02	.13154E-02	.13104E-02MAT#40	51	129
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.13154E-02	.13104E-02	.12953E-02	.12983E-02	.89180E-04	.13368E-03MAT#40	51	133
.12793E-03	.12935E-03	.23870E-01	.93738E-03	.13163E-02	.13174E-02MAT#40	51	134
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.13529E-03	.12824E-03	.63856E-02	.84457E-05	.91601E-05	.86830E-05MAT#40	51	164
.12412E-02	.13461E-04	.12760E-04	.14600E-02	.12456E-04	.13119E-02MAT#40	51	165
					MAT#40	0	166
					MAT# 0	0	167

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