BgNS TRANSACTIONS volume **25** number 1 (2020/21) pp. 3–5

^{93m}Nb Activation Cross Section Libraries Investigation

M. Mitev, D. Dimitrov, M. Manolova

Institute for Nuclear Research and Nuclear Energy, Bulgarian Academy of Sciences, 72 Tsarigradsko shose Blvd., 1784 Sofia, Bulgaria

Abstract. The isotope ^{93m}Nb is suitable for a test element in the development of principally new approaches in the radiochemistry. In order to correctly evaluate the measurements a suitable experimental setup has to be assured through preliminary calculation of the expected induced activity. These calculations also provide ground for evaluation of the results and adjustment of the approaches used during the experiment. In order to correctly perform the calculations, a reliable activation cross section data is necessary. Two editions of the ENDF format International Reactor Dosimetry File (IRDF) nuclear data library are evaluated in this study: IRDF2002, IRDF2014. The activation cross sections for the ^{93m}Nb based on these libraries are compared directly and by means of neutronic calculations. The results show that the use of activation cross sections based on IRDF2014 give better agreement with the experimental data and should be used as a base for further evaluation of the 93mNb activation measurements.

Keywords: neutron activation, cross section libraries, numerical modeling, experimental verification.

1 Introduction

The prediction of integral nuclear parameters (flunce, activities, doses, etc.) requires a reliable nuclear database such as microscopic nuclear parameters, cross sections, covariance matrices, etc. Not many works are focused on the ^{93m}Nb isotope since its limited application in the industry and medicine. This isotope, however, has important properties in the state-of-the-art activity measurement techniques investigations, due to its low-energy gamma lines and its high level of masking due to other elements, and especially the tantalum. Niobium and tantalum are chemical “twins” of the vanadium triad of the periodic table and are notoriously difficult to separate from one another and from their naturally occurring ores, due to their near-identical physical and chemical properties [1]. That is why ^{93m}Nb is suitable for a test element in the development of principally new approaches in the radiochemistry. In order to correctly evaluate the measurements a suitable experimental setup has to be assured through preliminary calculation of the expected induced activity. These calculations also provide ground for evaluation of the results and adjustment of the approaches used during the experiment. In order to correctly perform the calculations, a reliable activation cross section data is necessary. Two editions of the ENDF format International Reactor Dosimetry File (IRDF) nuclear data library are evaluated in this study: IRDF2002 [2], IRDF2014 [3]. In order to select the better cross sections for evaluating the approaches in ^{93m}Nb measurements the obtained cross sections are transformed to fewer group structure from the original ENDF energy group structure format.

The ENDF format [4] presents the radionuclide of interest in specific way. The information is presented in the form of data blocks called “files” (MF), with a special identifier for each reaction type (MT). The most significant information for radionuclide production can be found in the ENDF files labelled MF2, MF3, MF9 and MF10. MF2 provides information in the resonance domain, where cross sections are characterized by peak values corresponding to the quantum excitation states of the nuclei. MF3 gives the cross sections as a function of energy. In this file, one can also find descriptive data or resonance parameters which are not of interest for this work.

A nuclear reaction can produce a radionuclide in the ground or, as in the case of ^{93m}Nb, in an excited nuclear state (isomer). If isomers can be produced, the associated information can be found in the files MF9 or MF10. MF9 tabulates the multiplicities for production of radionuclide. Multiplicities represent the fraction of the reaction cross section specified in MF3, which produces the Level Final State (LFS) of the residual. These values are given as a function of energy but do not have the same mesh as in MF3. A linear interpolation (as recommended by the interpolation scheme for multiplicities) is then performed to get the multiplicity at the same energy as in the reaction cross-section. Finally, the reaction cross section $\sigma(E)$ from MF3 is multiplied by the multiplicity of the LSF, $\mu_{\mathrm{LFS}}(E)$ from MF9 to get the specific cross section of LFS:

$$\sigma_{\mathrm{LFS}}(E) = \sigma(E)\mu_{\mathrm{LFS}}(E)\,.$$

MF10 contains directly the specific cross section of the LFS, $\sigma_{\mathrm{LFS}}(E)$. In principle, the information in MF10 would be redundant with respect to the one in MF9 (once complemented with MF3). However, in practice the information is either provided in MF9 or in MF10, but never in both files at the same time.

Table 1. Reactions types and their MT label

MT number	Reaction type	MT number	Reaction type	MT number	Reaction type
4	(n, n')	111	$(n, 2p)$	174	$(n, 5n\,t)$
11	$(n, 2n\,d)$	112	$(n, p\,\alpha)$	175	$(n, 6n\,t)$
16	$(n, 2n)$	113	$(n, t\,2\alpha)$	176	$(n, 2n\,\mathrm{He}_3)$
17	$(n, 3n)$	114	$(n, d\,2\alpha)$	177	$(n, 3n\,\mathrm{He}_3)$
22	$(n, n\,\alpha)$	115	$(n, p\,d)$	178	$(n, 4n\,\mathrm{He}_3)$
23	$(n, n'\,3\alpha)$	116	$(n, p\,t)$	179	$(n, 3n\,2p)$
24	$(n, 2n\,\alpha)$	117	$(n, d\,\alpha)$	180	$(n, 3n\,2\alpha)$
25	$(n, 3n\,\alpha)$	152	$(n, 5n)$	181	$(n, 3n\,p\,\alpha)$
28	$(n, n\,p)$	153	$(n, 6n)$	182	$(n, d\,t)$
29	$(n, n\,2\alpha)$	154	$(n, 2n\,t)$	183	$(n, n'\,p\,d)$
30	$(n, 2n\,2\alpha)$	155	$(n, t\,\alpha)$	184	$(n, n'\,p\,t)$
32	$(n, n\,d)$	156	$(n, 4n\,p)$	185	$(n, n'\,d\,t)$
33	$(n, n\,t)$	157	$(n, 3r\,d)$	186	$(n, n'\,p\,\mathrm{He}_3)$
34	$(n, n\,\mathrm{He}_3)$	158	$(n, n'\,d\,\alpha)$	187	$(n, n'\,d\,\mathrm{He}_3)$
35	$(n, n'\,d\,2\alpha)$	159	$(n, 2n\,p\,\alpha)$	188	$(n, n'\,t\,\mathrm{He}_3)$
36	$(n, n'\,t\,2\alpha)$	160	$(n, 7n)$	189	$(n, n'\,t\,\alpha)$
37	$(n, 4n)$	161	$(n, 8n)$	190	$(n, 2n\,2p)$
41	$(n, 2n\,p)$	162	$(n, 5n\,p)$	191	$(n, p\,\mathrm{He}_3)$
42	$(n, 3n\,p)$	163	$(n, 6n\,p)$	192	$(n, d\,\mathrm{He}_3)$
44	$(n, n'\,2p)$	164	$(n, 7n\,p)$	193	$(n, \alpha\,\mathrm{He}_3)$
45	$(n, n'\,p\,\alpha)$	165	$(n, 4n\,\alpha)$	194	$(n, 4n\,2p)$
102	(n, g)	166	$(n, 5n\,\alpha)$	195	$(n, 4n\,2\alpha)$
103	(n, p)	167	$(n, 6n\,\alpha)$	196	$(n, 4n\,p\,\alpha)$
104	(n, d)	168	$(n, 7n\,\alpha)$	197	$(n, 3p)$
105	(n, t)	169	$(n, 4t\,h)$	198	$(n, n'\,3p)$
106	$(n, n\,\mathrm{He}_3)$	170	$(n, 5t\,h)$	199	$(n, 3n\,2p\,\alpha)$
107	(n, α)	171	$(n, 6t\,h)$	200	$(n, 5n\,2p)$
108	$(n, 2\alpha)$	172	$(n, 3n\,t)$		
109	$(n, 3\alpha)$	173	$(n, 4n\,t)$		

The cross sections are used to compute the reaction rate $\theta_{k\leftarrow m}$ of a formation of product nuclide k from a target nuclide m as follows:

$$\theta_{k\leftarrow m} = \sum \int \sigma_{q,\,k\leftarrow m}(E)\Phi(E)dE,$$

where $\Phi(E)$ is the neutron flux at energy E, and $\sigma_{q,\,k\leftarrow m(E)}$ is the microscopic cross section of the reaction q at energy E producing the nuclide k from target nuclide m.

2 ^{93m}Nb Activation Cross Section Comparison

The cross section values for the production of ^{93m}Nb generated from the IRDF2002 and IRDF2014 through the above described approach are fit in the fewer energy group structure BUGLE [5], as the latter is more suitable for use in evaluation of measurements results. It contains 47 neutron energy groups to describe the cross sections information necessary for performing calculations of integral nuclear parameters, among which the neutron activation for production of ^{93m}Nb via the reaction:

$$^{93}\mathrm{Nb}(n, n')^{93m}\mathrm{Nb}\,.$$

The comparison of the cross sections from the two libraries is shown in Figure 1.

It is seen from the figure, that the major differences are in the energy region up to 1 MeV, thus the effect of application of the newer cross sections will strongly depend

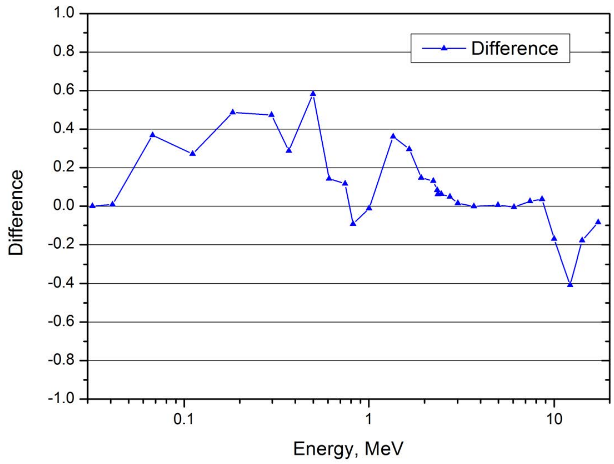

Figure 1. Comparison of cross sections (IRDF2014/IRDF2002-1).

on the experimental setup and the irradiation condition. However, bearing in mind, that the lower energy threshold of the ^{93m}Nb reaction is accepted to be 1 MeV, great effect may be expected from application of the new cross sections in the calculations.

3 ^{93m}Nb Activation Calculations

Calculations for the production of ^{93m}Nb through neutron activation have been carried out, using the activation cross sections from IRDF2014 and IRDF2002 libraries. The results, obtained with the different cross sections were compared. The comparison is presented in Figure 2.

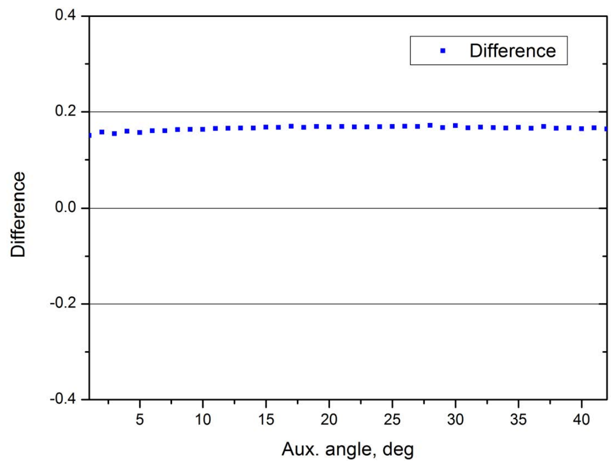

Figure 2. Comparison of calculational results (IRDF2014/IRDF2002-1).

It is seen from the figure, that the numerical results for the production of ^{93m}Nb through neutron activation, obtained using the cross sections derived from IRDF2014 are steadily higher with about 16% from those, obtained using the cross sections derived from IRDF2014.

4 Experimental Verification for the ^{93m}Nb Activation Calculations

The results from the calculations for the production of ^{93m}Nb through neutron activation, using the activation cross sections based on IRDF2014 and IRDF2002 libraries, have been compared with the experimental results for an activated niobium foil. The experimental setup reflects the parameters of the numerical modeling.

The comparison between the numerical results for the production of ^{93m}Nb through neutron activation, obtained using the cross sections derived from IRDF2002 and the experiment are shown in Figure 3.

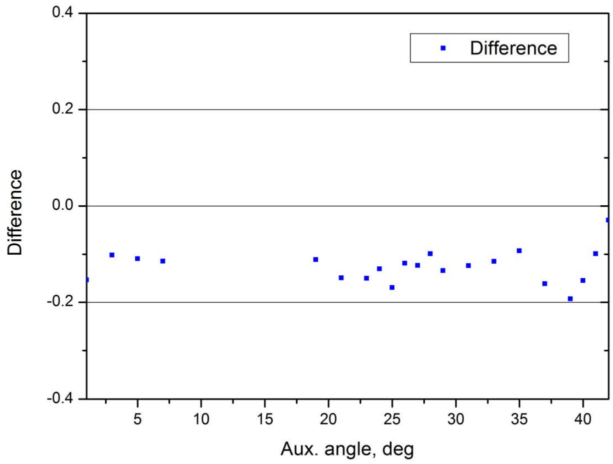

Figure 3. Comparison of calculations results (IRDF2002/experiment-1).

It is seen that the numerical modeling steadily underestimates the experimental values by 10 to 20%.

The comparison between the numerical results for the production of ^{93m}Nb through neutron activation, obtained using the cross sections derived from IRDF2014 and the experiment are shown in Figure 4.

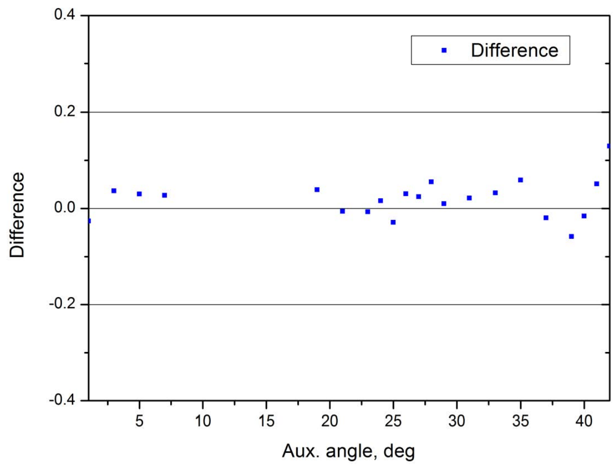

Figure 4. Comparison of calculations results (IRDF2014/experiment-1).

It is seen that the numerical modeling agrees with the experimental values within 5% for the most cases.

5 Conclusions

Two editions of the ENDF format International Reactor Dosimetry File (IRDF) nuclear data library are evaluated in this study: IRDF2002, IRDF2014. The activation cross sections for the ^{93m}Nb derived from both libraries are compared directly and by means of neutronic calculations. The results show that the use of IRDF2014 for activation cross sections give better agreement with the experimental data and should be used as a base for further evaluation of the ^{93m}Nb activation measurements.

Acknowledgments

This work was fully supported by the Bulgarian National Science Fund under contract number КП-06-Н39/14 (12.12.2019) “Research of new approaches in the activation analysis and radiochemistry of technogenic radionuclides important for man and environment.”

References

[1] Schotzig U., Schrader H. (2000) Halbwertszeiten und Photonen-Emissionswahrscheinlichkeiten von haufigverwendeten Radionukliden, 5. erweiterte und korrigierte Auflage, PTB-Bericht PTB-Ra-16/5: Braunschweig, May 2000, ISBN 3-89701-279-0.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY (2006) International Reactor Dosimetry File 2002 (IRDF-2002), Technical Reports Series No. 452, IAEA, Vienna.

[3] IRDF-2014: A. Trkov et al. (2014) International Reactor Dosimetry and Fusion File IRDFF v.1.05, 09 October, 2014.

[4] Herman M., Trkov A. (2010) CSEWG Document ENDF-102 Report BNL-90365-2009 Rev.1 ENDF-6 Formats Manual, BNL, July 2010.

[5] White J., Ingersoll D., Slater C., Roussin R. (1996) BUGLE-96: A Revised Multigroup Cross Section Library for LWR Applications Based on ENDF/B-VI Release 3, DLC-185, Oak Ridge National Laboratory, Oak Ridge, TN.