



Original Article

Neutron activation analysis: Modelling studies to improve the neutron flux of Americium–Beryllium source



Abdessamad Didi ^{a,*}, Ahmed Dadouch ^a, Otman Jaï ^b, Jaouad Tajmouati ^a, Hassane El Bekkouri ^a

^a Advanced Technology and Integration System, Department of Physics, Faculty of Science Dhar Mehraz, University Sidi Mohamed Ben Abdellah, B.P. 1796, Fez, Morocco

^b Laboratory of Radiation and Nuclear Systems, Department of Physics, Faculty of Sciences, POB 2121, Tetouan, Morocco

ARTICLE INFO

Article history:

Received 15 September 2016

Received in revised form

29 December 2016

Accepted 2 February 2017

Available online 22 February 2017

Keywords:

Americium–Beryllium Source

Evaluated Nuclear Data File

Monte Carlo N-Particle Transport Code-6

Neutron Activation Analysis

Paraffin

ABSTRACT

Americium–beryllium (Am-Be; n, γ) is a neutron emitting source used in various research fields such as chemistry, physics, geology, archaeology, medicine, and environmental monitoring, as well as in the forensic sciences. It is a mobile source of neutron activity (20 Ci), yielding a small thermal neutron flux that is water moderated. The aim of this study is to develop a model to increase the neutron thermal flux of a source such as Am-Be. This study achieved multiple advantageous results: primarily, it will help us perform neutron activation analysis. Next, it will give us the opportunity to produce radio-elements with short half-lives. Am-Be single and multisource (5 sources) experiments were performed within an irradiation facility with a paraffin moderator. The resulting models mainly increase the thermal neutron flux compared to the traditional method with water moderator.

© 2017 Korean Nuclear Society, Published by Elsevier Korea LLC. This is an open access article under the CC BY-NC-ND license (<http://creativecommons.org/licenses/by-nc-nd/4.0/>).

1. Introduction

Americium–beryllium (Am-Be) is a neutron-emitting source used in many domains. It provides many advantages such as high stability of the produced flux [1]. Furthermore, it is used for source calibration via neutron instrumentation, which can be employed in several areas [2] such as neutron activation analysis (NAA).

Due to the high flux, the NAA method is frequently used with reactors and accelerators for the generation of radionuclides used in many fields such as Medicine, Industry, and Agriculture [3–5]. However, this technology is only available in developed countries due to its high cost [3,6,7]. In this work, using an Am-Be neutron

source, we attempt to develop a new model for nuclear instrumentation. The main focus is to perform neutron activation analysis using thermal and fast neutrons in order to study samples outside reactors and accelerators [2,8–11].

2. Methods and materials

2.1. Am-Be source

Am-Be is a mixture of heavy isotopes. It is a compact mixture of finely powdered AmO₂ [8]. The mixture is compressed into a cylindrical capsule with a density of 1.3 g/cm³, as shown in Fig. 1



* Corresponding author.

E-mail address: abdessamad.didi1@usmba.ac.ma (A. Didi).

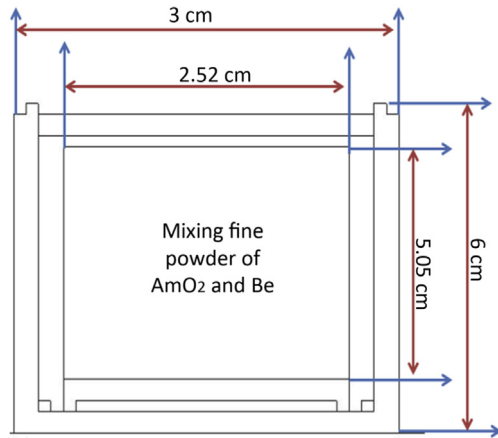


Fig. 1. Cross-sectional view of ²⁴¹Am-Be neutron source assembly.

[12,13]. Therefore, the created source produces a large number of neutrons [14,15]:

Americium-241 has two important irradiations. The first is called α radiation, and reaches 85% with a spectrum energy of 5.486 MeV. The second is called γ whit, and reaches 36% with an energy of 60 KeV.

The ²⁴¹Am source is contained in a standard capsule called Amersham X.14 (QSA Global, USA), and is referenced with the AMN241 code [13]. The capsule has an activity of 20 Ci and an emission of 2.2×10^6 n/cm²s [14,16,17]. It consists of double layers made of stainless steel (Figs. 1 and 2), with the following composition: C (0.004%); Mn (1.59%); P (0.011%); S (0.008%); Si (0.37%); Cr (16.96%); Ni (3.61%); Mo (2.29%); Fe (65.16%) [14].

2.2. Monte Carlo N-particle code

The Monte Carlo N-particle transport (MCNP) code simulates individual particles and their interactions to record the average properties of the total system. The code can be used to model neutron, photon, and electron fields as well as to model coupled neutron–photon–electron fields in which secondary effects can be considered. Each stage in the history of a particle is determined by random sampling of the probability distributions that govern the interaction of the particle with the material [18–20]; MCNP simulates every single particle and its track through the geometry of the problem. The probability of a particle traveling a certain distance ℓ before collision is given by Eq. (1):

$$P(\ell) = \text{Exp}\left(-\sum_t \ell\right) \tag{1}$$

The probability of a particle undergoing a collision between ℓ and $\ell + d\ell$ is therefore Eq. (2):

$$P(\ell)d\ell = \text{Exp}\left(-\sum_t \ell\right) * \sum_t d\ell \tag{2}$$

Integrating Eq. (2) for the path length of the particle and setting it equal to some random number, the result is Eq. (3):

$$\xi = \int_0^1 \text{Exp}\left(-\sum_t s\right) * \sum_t ds = 1 - \text{Exp}\left(-\sum_t \ell\right) \tag{3}$$

Solving for ℓ gives Eq. (4):

$$\ell = -\frac{1}{\sum_t} \ln(1 - \xi) \tag{4}$$

The random numbers ξ are evenly distributed between 0 and 1 [20–22]. It follows that $1 - \xi$ follows the same distribution. Thus, the distance to collision is randomly sampled according to Eq. (5):

$$\ell = -\frac{1}{\sum_t} \ln(\xi) \tag{5}$$

If there are more possibilities of the behavior, MCNP makes a decision in such a way that the probability of each behavior corresponds to the real statistical probability distribution [23]. By tracking all the particles, it is possible to calculate so-called tallies. A tally is output information a user can receive from MCNP [18,20,23].

In this study, to calculate the flux in each cylinder, we use the card F4 tally [18,20,24], presented in Eq. (6):

$$F4 = \frac{\int \phi(E)dE}{N} \tag{6}$$

where: ϕ represents the neutron flux (n/cm²s); N represents the number of source neutrons/s; and E represents the energy.

Experimental simulation was performed using the MCNP-6 code through the cross section and nuclear data from the ENDF/B-VII library. In addition, the thermal neutrons are described by the free gas $S(\alpha,\beta)$ [22,24]. In this work, we use the thermal flux with a range of energy between 0 and 6.25×10^{-7} MeV.

2.3. The proposed models

Our proposal consists of four models, described as follows. The first model is based on using a single source and replacing water,

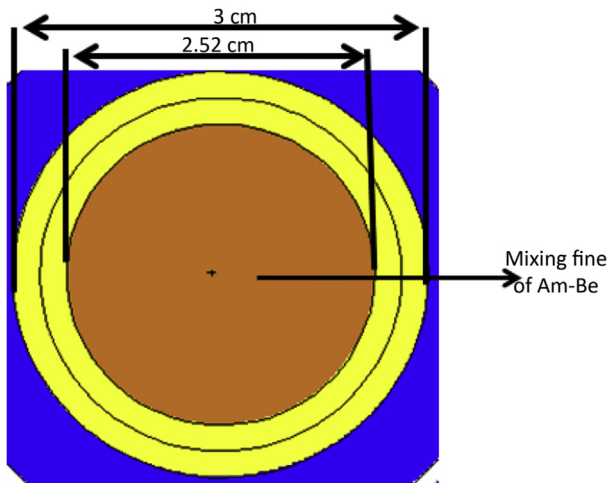


Fig. 2. Plan view of ²⁴¹Am-Be neutron source.

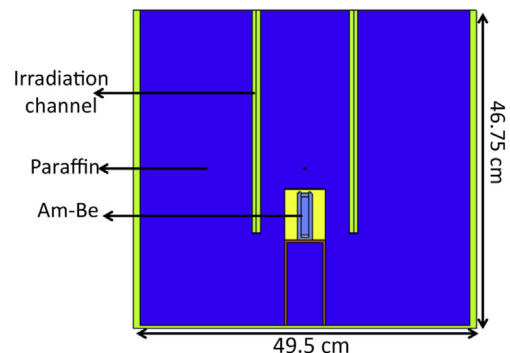


Fig. 3. Vertical view of the ²⁴¹Am-Be neutron source of 20 Ci.

the classic moderator, with paraffin (Fig. 3).

The second model consists of using the first model and adding two parallel barriers of solid beryllium (Fig. 4).

For the third model, we combine five identical sources of Am-Be (20 Ci for each) with paraffin as moderator (Fig. 5).

The last model combines the previous model (Model 3) with two parallel barriers of solid beryllium. The results in the output file are normalized by Equation (7) using Microsoft Excel and then analyzed and interpreted in the form of graphs and tables.

$$\text{Flux} = \text{Tally} * \text{Source Strength} \left(\frac{n}{\text{cm}^2\text{s}} \right) \quad (7)$$

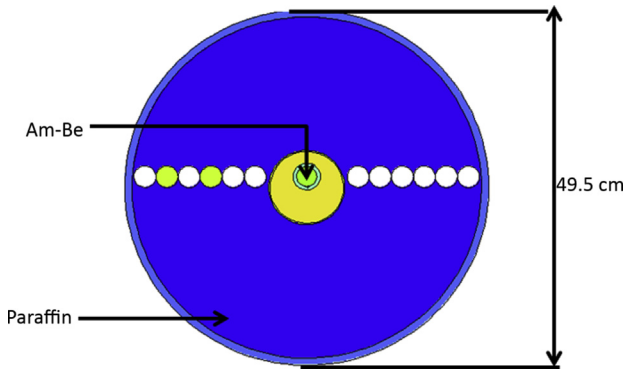


Fig. 4. Horizontal view of the ²⁴¹Am-Be neutron source of 20 Ci.

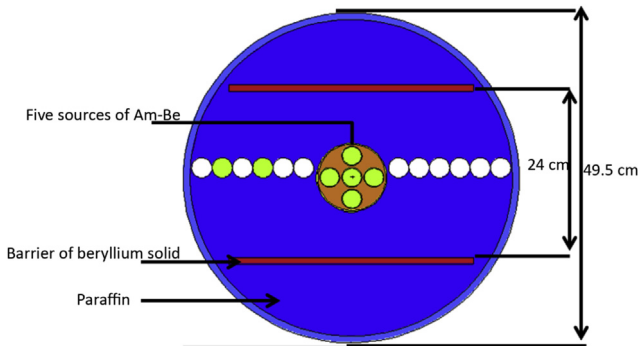


Fig. 5. Horizontal view of five sources of ²⁴¹Am-Be neutron source with Barriers of Be of 20 Ci for each.

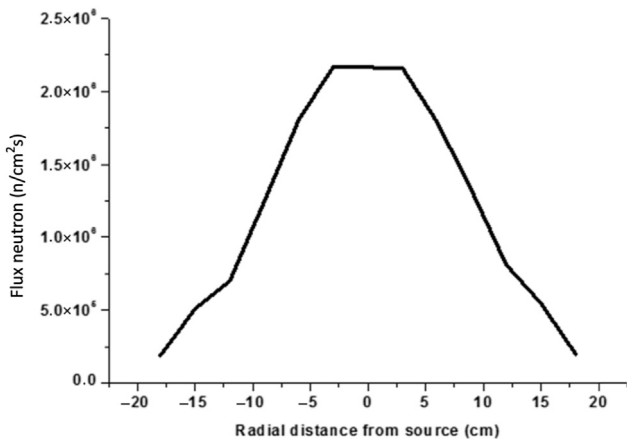


Fig. 6. Radial neutron flux of one source of ²⁴¹Am-Be of 20 Ci.

3. Results

After simulation using the MCNP-6 code of 30×10^9 stories of particles for the three designs, we noticed very quickly that the neutron flux was distributed evenly on both sides of the source.

The yields of neutron flux are at a maximum in the design of a single source of Am-Be in paraffin; yield is $(2.2 \pm 0.0008) \times 10^6$ n/cm²s, as shown in Fig. 6. This result can be compared with the

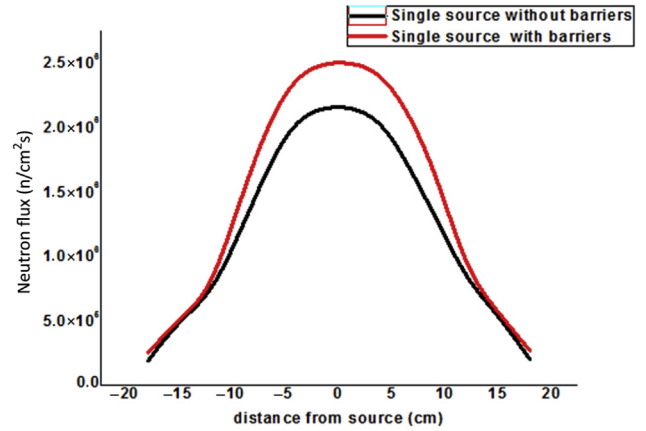


Fig. 7. Radial neutron flux of one source of ²⁴¹Am-Be of 20 Ci with two barriers.

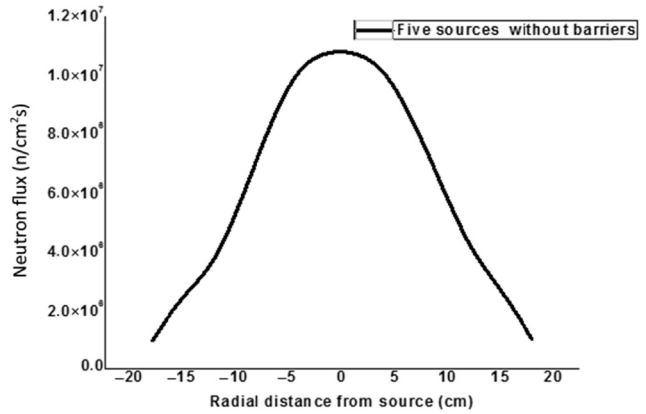


Fig. 8. Radial neutron flux of five sources of ²⁴¹Am-Be of 20 Ci without barriers.

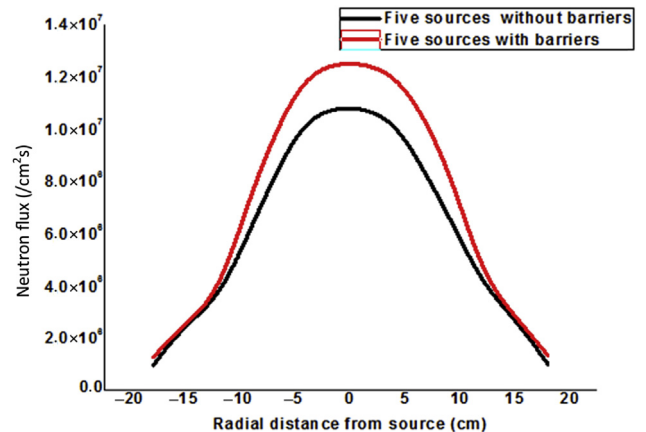


Fig. 9. Radial neutron fluxes of five sources of ²⁴¹Am-Be of 20 Ci with barriers.

Table 1
Performance of neutron fluxes of single and multi-sources of Am-Be.

	Sogbadji et al [21]		Our study with single source		Our study with 5 sources	
	Single source	4 sources	Without barriers	With barriers	Without barriers	With barriers
Neutron Flux n/cm ² s	$(1.8 \pm 0.0007) \times 10^6$	$(5.4 \pm 0.0008) \times 10^6$	$(2.2 \pm 0.0008) \times 10^6$	$(2.5 \pm 0.0008) \times 10^6$	$(1.09 \pm 0.0008) \times 10^7$	$(1.22 \pm 0.0008) \times 10^7$
Increase Factor	1	3.4	1.25	1.4	6.10	6.8

results of experiments using water as moderator; those results yielded a value of $(1.8 \pm 0.0007) \times 10^6$ n/cm²s [21].

In the second design, we have fixed two parallel barriers of solid beryllium at a distance of 24 cm, as shown in Fig. 5; thermal neutron simulation increased to $(2.5 \pm 0.0008) \times 10^6$ n/cm²s. Fig. 7 shows that increasing the flux of neutrons can be explained by the fact that several fast neutrons are reflected by both barriers, which contributes to the increase of the thermal neutrons' flux in the source between the two barriers.

Then, the neutron flux generated by identical five sources (Am-Be), is more intense than that of a single source; the thermal flux is $(1.15 \pm 0.0007) \times 10^7$ n/cm²s, as shown in Fig. 8; the increase rate is five times that of the flux of a single source. We can conclude that, more than the numbers of sources, the degree of thermal flux is important.

Finally, the performance profile of the thermal flux increased to $(1.22 \pm 0.0008) \times 10^7$ n/cm²s in the case of two barriers, with a rate of increase equal to 6.8, as can be seen in Fig. 9 and in Table 1.

4. Discussion

Due to the presence of the paraffin moderator, and owing to the concept of thermalization, the majority of the fast neutrons emitted by the Am-Be become slow and change to thermal neutrons. The high rate of production of thermal neutrons can be explained by the interaction between the neutrons and the material used.

Furthermore, during elastic collisions between two particles, the transfer of kinetic energy between the neutron-protons is maximized when their masses are equal. The fast neutrons produced by the Am-Be then pass through the paraffin blocks, where they lose energy through their interaction with the paraffin hydrogen atoms, transforming into low energy neutrons [24].

According to the measurements made, the results clearly show that the neutron flux increases when water is replaced as moderator with paraffin and when two parallel barriers are added. By explaining this phenomenon according to the fact that the concentration of paraffin is lower than that of water, we can assert that both barriers play roles as neutron scatterers; when a neutron has just escaped from the source, it undergoes diffusion towards the interior towards the two barriers.

The subsequent increase of thermal neutron flux within the paraffin is the result of the paraffin's density of 0.88 g/cm³. Water has a higher density (1 g/cm³) than that of paraffin. As a result, neutron absorption in water is greater than in paraffin.

5. Conclusion

In this work, from an analysis of experimental and simulated data using the MCNP code, we can infer that paraffin as a moderating material is superior to water, producing a greater thermal neutron flux. We can see that the neutron flux increases from 1.8×10^6 n/cm²s to 2.2×10^6 n/cm²s; this is beneficial for neutron activation analysis. The intense thermal neutron flux recorded in the multi-source design (5 sources) with both levels of barriers flux increased to 1.22×10^7 n/cm²s, which can be useful for advancing research across multiple disciplines, such as the production of radioelements

with small half-lives; the detection and identification of light elements such as carbon and boron; neutron activation analysis by prompt γ method NAA; and conventional instrumental NAA.

We propose a multisource installation of Am-Be. This would solve various national level socio-economic problems.

Conflicts of interest

All authors declare no conflicts of interest.

References

- [1] J.Y. Zevallos-Chávez, C.B. Zamboni, Evaluation of the neutron flux distribution in an AmBe irradiator using the MCNP-4C code, *Braz. J. Phys.* 35 (2005) 787–800.
- [2] G.F. Knoll, *Radiation Detection and Measurement*, second ed, John Wiley and Sons, New York, 1969, pp. 21–25.
- [3] M. Ali-Abdallah, N.A. Mansour, Neutron activation analysis of cement bulk samples, *Adv. Appl. Sci. Res.* 2 (2011) 613–620.
- [4] A. Didi, A. Dadouch, H. El Bekkouri, Feasibility study for production of iodine-131 using dioxide of tellurium-130, *Int. J. Pharm. Pharmaceut. Sci.* 8 (2016) 327–331.
- [5] A. Didi, A. Dadouch, J. Tajmouati, O. Jai, Calculating concentrations of elements in sample and compare with standard certified results of the International Atomic Energy Agency (IAEA) Soil-7, *Pharm. Chem.* 8 (2016) 250–255.
- [6] J. Yang, Y. Yang, Y. Li, M. Liu, Y. Cheng, Monte-Carlo simulation of cement neutron field distribution characteristics in PGNA, *Nucl. Sci. Tech.* 23 (2012) 337–343.
- [7] K.G. Siddappa, K.M. Balakrishna, Neutron and gamma activation analyses using microtron facility, *Trends NDE Sci. Technol. Proc. 14th World Conf. Non-Destructive Testing*, New Delhi 2 (1996) 389–392.
- [8] P.A. Lessing, I. Glagolenko, High Purity Americium-241 for Fuel Cycle R & D Program, Idaho National Laboratory Idaho Falls, ID 83415–3870 INMM, 2011.
- [9] J.C. Vitorelli, A.X. Silva, V.R. Crispim, E.S. da Fonseca, W.W. Pereira, Monte Carlo simulation of response function for a NaI(Tl) detector for gamma rays from 241 Am/Be source, *Appl. Radiation Isotopes* 62 (2005) 619–622.
- [10] A. Didi, J. Tajmouati, A. Maghnoou, M. Benchiekh, O. Jai, New design of thermal neutron flux distribution of Am–Be neutron source irradiation in the paraffin moderator using MCNP-6, *Mor. J. Chem.* 4 (2) (2016) 285–288.
- [11] T. Miura, H. Matsue, Precise determination of silicon in ceramic reference materials by prompt gamma activation analysis at JRR-3, *Nucl. Eng. Technol.* 48 (2016) 299–303.
- [12] E.K. Osae, G.K. Amoh, Multi-element analysis of Gold Ore mill heads by INAA, *J. Univ. Sci. Technol. Kumasi* 16 (1–2) (1996) 104–107.
- [13] M. Asamoah, B.J.B. Nyarko, J.J. Fletcher, R.B.M. Sogbadji, S. Yamoah, S.E. Agbemava, E. Mensimah, Neutron flux distribution in the irradiation channels of Am–Be neutron source irradiation facility, *Ann. Nucl. Energy* 38 (6) (2011) 1219–1224, <http://dx.doi.org/10.1016/j.anucene.2011.02.014>.
- [14] Z. Liu, J. Chen, P. Zhu, Y. Li, G. Zhang, The 4.438 MeV gamma to neutron ratio for the Am-Be neutron source, *Appl. Radiat. Isot* 65 (12) (2007) 1318–1321.
- [15] C.R. Peeples, *Alternative to Am–Be neutron source for the compensated neutron porosity log*, North Carolina State University, Raleigh (NC), 2007.
- [16] Los Alamos National Laboratory report LA-UR-13-22934, Initial MCNP6 Release Overview MCNP6 Version 1.0, Monte Carlo N–Particle Transport Code System Including MCNP6.1, 2013.
- [17] A.T. Augousti, Neil M. White, *Sensors and their Applications VIII*, Proceedings of the Eighth Conference on Sensors and their Applications, University of Southampton, Institute of Physics Pub, Bristol, Philadelphia, United Kingdom, Glasgow vol. 54, 1997, pp. 109–114.
- [18] T.S. Coelho, F. Massicano, R.G. Possani, G. Fedorenko, R.S. Viana, V.A. Castro, M.A.R. Fernandes, H.A. Miot, H. Yoriyaz, International Nuclear Atlantic Conference - INAC 2011 Belo Horizonte, MG, Brazil, October 24–28, 2011 Associação Brasileira de Energia Nuclear - ABEN, 2011.
- [19] M. Těšínský, Measurement and Monte Carlo Simulation of the Neutron Spectra of the Subcritical Reactor Experiment “Yalina Booster” Master of Science Thesis, Department of Nuclear and Reactor Physics Royal Institute of Technology In collaboration with Department of Nuclear Reactor, Czech Technical University, Stockholm, 2006.
- [20] X-6 Monte Carlo Team, N-Particle Transport Code System Including MCNP6.1, and Data Libraries, MCNP6 Users Manual, Code Version 6.1.1beta, LA-CP-

- 14–00745, Los Alamos National Security, LLC, Los Alamos National Laboratory, Los Alamos (NM), 2014.
- [21] R.B.M. Sogbadji, R.G. Abrefah, B.J.B. Nyarko, E.H.K. Akaho, H.C. Odoi, S. Attakorah-Birinkorang, The design of a multisource americium–beryllium neutron irradiation facility using MCNP for the neutronic performance calculation, *Appl. Radiation Isotopes* 90 (2014) 192–196.
- [22] J.M. Carpenter, C.K. Loong, *Element of slow neutron scattering, Basics, Techniques, and Applications*, M.R.S Materials Research Society, Cambridge, UK, 2015.
- [23] D.P. Gierga, *Electron Photon Calculations using MCNP*, Nuclear Engineering, Rensselaer Polytechnic Institute, Massachusetts Institute of Technology, Boston (MA), 1998.
- [24] R.R. Greenberg, P. Bode, E.A. de Nadai-Fernandes, Neutron activation analysis: a primary method of measurement, *Spectrochim. Acta Part B* 66 (2011) 193–241.