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# Determination of Production Cross-Section of Zn-65 Using NIRR-1

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**ABSTRACT**
The unique decay modes of $^{65}$Zn, including both its positron emission and electron capture, places it as an important theragnostic radioisotope with emerging application in nuclear medicine, cancer therapy and other similar applications. Comprehensive study of its cross section is therefore of essence in its production technology design and clinical application, especially as it relates with the analysis of its purity and yield. Hence, for this and other emerging radioisotope of interest, a clear protocol needs to be established as part of expanding the overall utilization of Research Reactors to cover this important area of need. In this work, Nigerian Research Reactor 1 (NIRR-1), a Miniature Neutron Source Reactor (MNSR), has been utilized in the determination of the neutron induced cross section of $^{64}$Zn at thermal energy region. The measurement of the cross section for the production of $^{65}$Zn radioisotope via $^{64}$Zn(n,γ)$^{65}$Zn nuclear reaction was carried out at the recator facility at Ahmadu Bello University (ABU), Zaria and shown to be suitable in establishing the protocol for the possibility of expanding utilization of NIRR-1 to medical radioisotope production with the right technical efforts. The thermal neutron cross section obtained for the $^{64}$Zn(n,γ)$^{65}$Zn reaction, given as 1.98E+03 mb, is more than all available retrieved data but within the same order of magnitude and with a standard deviation uncertainty of 98.99495 to 893.0759 when compared with extant studies retrieved from the IAEA standard measured cross section data library. The result indicates promising prospect for the exploration of NIRR-1 for medical radioisotope production.

**Keywords**: Cross-section, Radioisotopes, Thermal neutron, Production Cross-section, NIRR-1

## 1.0 INTRODUCTION

The Nigeria Research Reactor-1 (NIRR-1) is a 30 kW Miniature Neutron Source Reactor (MNSR) designed mainly for Neutron Activation Analysis (NAA), training and limited radioisotope production. These radioisotopes have many important applications across different fields of interest. However, the limit of NIRR-1's radioisotope production capacity has not been explored exhaustively for different types of radioisotopes, especially for those finding application in nuclear medicine.

Some of the radioisotopes of interest include Cu-60, 61,62,64,67, Zn-65, Au-198 and Mg-27. Several of these find production possibility via neutron capture (n,$\gamma$), neutron-induced fission (n,f) and neutron-proton reactions (n,p). These radioisotopes have numerous benefits, including improved healthcare, increased food production and advanced scientific research (Zahra et al, 2018).

Zinc-65 is a one of the most prominent radioisotopes of Zinc with half-life of 243.93 days. It decays via electron capture (EC 98.3%) and positron emission ($B_+$ 1.7%). Its feature includes Mean Positron Energy, $E_{\beta+} = 142.5KeV$, End-point Positron Energy $E_{\beta+} = 330.1KeV$; Gamma Energy $E_\gamma = 1115.55\,KeV$ and Gamm Intensity $I_\gamma = 50.04$ (Browne and Tuli, 2010). It can be produced through cyclotron production technology via the $^{65}$Cu (p,n)$^{65}$Zn reaction production route. It can also be produced via a neutron induced process via $^{64}$Zn (n, γ) $^{65}$Zn reaction production route. The latter is associated with high activity, achieved without carrier-addition (NCA) and with high radionuclidic purity of almost 99.9% (Alafeshat et al., 2026). These provide incentive for deeper interest in the neutron induced production reaction of Zn-65.

Applications of Zn-65 cut across medical research, where it is used to study Zinc metabolism, diagnose Zinc deficiency and monitor Zinc levels in the body. It is also useful in agricultural research for the study of Zinc uptake and utilization in plants and in material sciences where it is used to study corrosion and surface properties of Zinc alloys (Brambillaet al, 2002). Zn-65 is a valuable tool for studying Zinc biology and material sciences (Rowshanfarzad et al, 2005).

For this radioisotope, insufficient nuclear data, such as cross section, has hindered essential information on nuclear reactions, radiation interactions and material properties. Such data are crucial for medical applications. Zn-65 has been noted to possess anticancer potential of selective target of cancer cells in therapy, altering cell growth and inducing programmed cell death. Thus, it has become imperative to study the production cross section of Zn-65.

The Cross Section, which is the probability that a nuclear reaction will occur, gives the effective area of the target nuclei presented to the bombarding beam of incident particle. In particular, the Production Cross Section is a measure of the probability that a specific nuclear reaction will produce a particular nucleus or reaction product when an incident particle interacts with the target nucleus (Jonah et al, 2008).

This study thus aims to determine the cross section for the production of Zn-65 via the $^{64}$Zn-(n,γ)$^{65}$Zn reaction for nuclear medicine application using measurement of its activity and reaction characteristics from activation in NIRR-1. For this study, the cross sections with respect to the production of $^{65}$Zn has been explored with focus on the capture cross section from thermal activation of $^{64}$Zn.

The determined cross section will provide insight into production yield and purity of the isotope. Generally, higher cross section imply that more Zn-65 would be produced per unit natural Zn irradiation. Increasing yield also imply decreasing production cost. Similarly, a full profile of the nuclear interaction will allow the determination of the Purity Index of the isotope. The Purity Index is very key to radiation safety in production strategy and treatment planning. The cross-section data will further provide comprehensive guidance and establish Standard Operating Procedure (SOP) with respect to irradiation time, neutron energy and flux optimization for efficient production of Zn-65 and other similar radioisotope.

## 2.0 MATERIALS AND METHOD

This study involves four stages namely Zn- Sample Preparation, Irradiation in NIRR-1, Gamma Spectroscopic Counting and the Cross Section Calculation.

**2.1 Material**

A 99.50% purity goodfellow Zn sample was used in the study. NIRR-1 was used for irradiation of the sample. The thermal flux is suitable for the neutron capture (n,γ) study for Zn-64, hence the irradiation was carried out at the Bare Inner Irradiation Channel B2 to provide maximum exposure to thermal neutron during the activation. For the short irradiation process, NIRR-1 was operated at Half-Power 17 kW (5.0 x $10^{11}$ n/cm$^2$s). After irradiation, radiation counting was carried out with ORTEC coaxial Hyper Pure Germanium (HpGe) detector with relative efficiency of 30% and resolution of 1.95 keV FWHM, at 1.33MeV, Co-60. The MAESTRO gamma-ray data acquisition Multi Channel Analyzer (MCA) emulation software card by ORTEC is coupled to interphase with the detector system through electronic module also by ORTEC.

**2.2 Method**

Figure 1 provide a schematic representation of the procedure for the study**.** The study was carried out at the irradiation facility of the Center for Energy Research and Training (CERT), Ahmadu Bello University (ABU) Zaria.

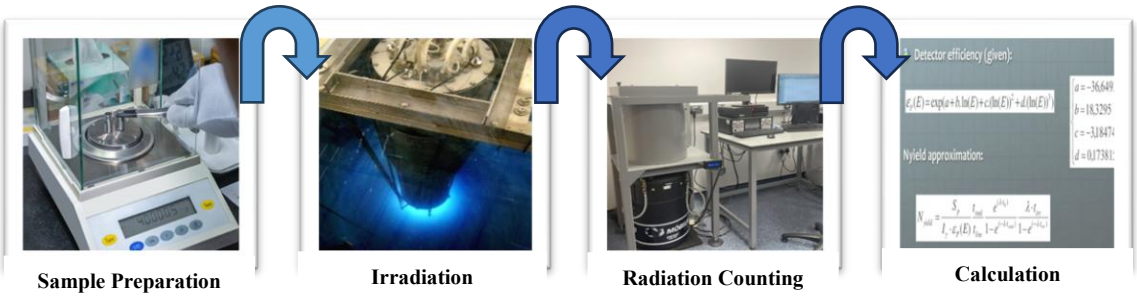


Figure 1: Schematic Represenation of the Study Pocedure

**2.2.1 Sample Preparation**

The preparation of the sample for irradiation involved cutting the foils to suitable sizes influenced by the vials size; cleaning with analytic grade acetone and determination of the different masses of the primary Zn foils (Goodfellow). The masses were determined with Mettler AE240, high precision analytical weighing balance.

**2.2.2 Irradiation with NIRR-1**

Activation was carried out using the inner irradiation Bare Channel (B2). Irradiation was carried out with Zn samples inserted into the vial. The heat sealed 7cm$^3$ polyethylene vial was introduced into the channel for the irradiation using the available rabbit system, a pneumatic auxiliary device for introducing and retrieving samples through tubes into the different irradiation sites for NAA. Foil sample. Time of introduction and retrieval of vials were noted and used to ensure irradiation time ($t_{ir}$) of 3600 seconds. The Irradiations was carried out at thermal power level of 17kW (half power), corresponding to a thermal neutron flux value of 5.0 x $10^{11}$ n/cm$^2$s preset on the control console for short irradiation process of 1 hour each for the Zn sample in vial. Figure 2 present the core of NIRR-1 showing the arrangement of the irradiation channels.

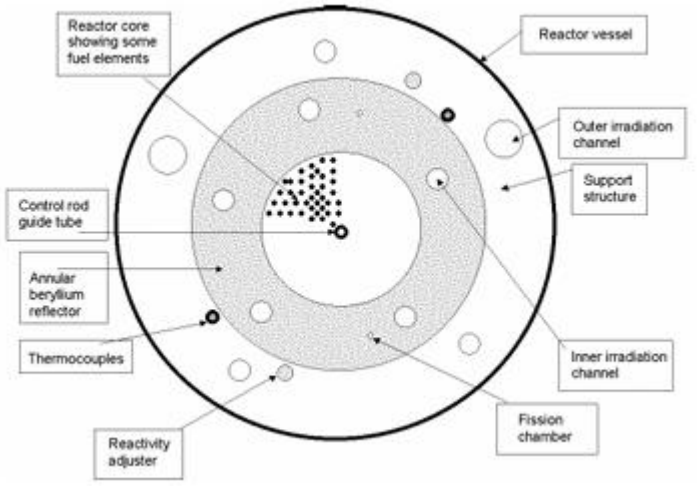


Figure 2: NIRR Core showing Irradiation Channels

### 2.2.3 Gamma Spectroscopy Counting HpGe Detector

Gamma spectrometry counting was preceded by radiation monitoring which was carried out with RDS-120 Universal Survey Meter to estimate the activity of the retrieved sample in the vial after 1 hour irradiation and retrieval through the rabbit system. The time of retrieval is noted. For the retrieved sample, the effective dose was measured repeatedly at close proximity until each was assessed as ready for counting. Also, preliminary assessment was carried out using the counting system to establish dead time < 10% before actual counting was commenced. The time when counting starts is noted and the Deay Time ($t_D$) is determined as the difference between the time counting starts and the time the foil sample is brought back through the rabbit system.

Radiation counting was carried out with the ORTEC coaxial HpGe detector. The GEM 30195 HPGe coaxial, vertical dip-stick detector (ORTEC) is dedicated for Neutron Activation Analysis (NAA) at the Reactor Facility at the CERT, ABU, Zaria. Gamma ray data acquisition was achieved using MAESTRO MCA through the electronic module which are both also from ORTEC. Measurement of the radioactivity in the foil samples can be carried out at two source-detector geometry, namely H1= 2 and H2=15 cm, based on dead time of < 10%. For this study, the H1=2cm was seen to be suitable. The Counting Time ($t_C$) is determined as the difference between the time of placement of the irradiated foil sample on the detector surface at the appropriate geometry and the time counting ends.

The Full Energy Peak Efficiency (ε) of the detector was determined preceding the activity counting using standard gamma ray sources with well-known features. It is an essential parameter for empirical determination of cross section and necessary for the calibration of the detector system for high quality cross-section measurements. Experimental technique and a semi-empirical method as suggested by Sadiq and Jonah (2007) and Gunnink (1990) has been

reported elsewhere where Eu-152 was used (Ige et al., Unpublished Manuscript). In this case, only value for the near geometry (H1=2 cm) is reported as used in this current study.

In Table 1, reactions and foil sample characteristics are provided. The Table also includes other necessary nuclear data and the irradiation and counting statistics.

**2.2.4 Cross Section Calculation**

After the counting, the cross section was determined from the formalism of Equation (1). The formalism determines the measured cross form measured activity concentration. Equation (1) is suitable specifically for thermal neutron cross section determination.

$$\sigma = {\lambda N_c(t_c)M}\Big/{m\boldsymbol{I_\gamma}\theta N_A\varepsilon\varphi SDCt_c} \qquad (1)$$

where $\mathbf{N_c}$ = net peak counts under gamma line of interest, $\boldsymbol{\lambda}$ = decay constant, **M** = atomic mass of target element, **S** = $\left(1-e^{-\lambda t_c}\right)$ known as saturation factor, **D** = $e^{-\lambda t_D}$ known as decay or cooling factor, **C** = $\left(1-e^{-\lambda t_{ir}}\right)$ known as counting or measuring factor, $\mathbf{N_A}$ = Avogadro's number, $\boldsymbol{\theta}$ = Isotopic abundance of target nucleus, $\boldsymbol{I_\gamma}$ = Gamma ray intensity of the residual radionuclide, $\boldsymbol{\epsilon}$ = Full-energy peak efficiency, $\mathbf{t_{irr}}$ = Irradiation time, $\mathbf{t_d}$ = Cooling/Decay time, $\mathbf{t_m}$ = Measuring time, **m** = Mass of target element in sample

**3.0 Results and Discussion**

Table 1 presents data describing the target material, the Irradiation Counting Data and other Nuclear Data relevant to the study. In Table 2, the measurements data obtained from the gamma spectroscopy counting using the HpGe detector are presented. In Table 3, a comparison of the cross-section data obtained in this work with Literature data is presented.

**Table 1: Description of Target Material, Irradiation, Counting and other Nuclear Data**

| Parameter | Quantity |
|---|---|
| Target | Zn |
| Reaction | $^{64}$Zn(n,γ)$^{64}$Cu |
| Sample ID | S1 |
| $E_{thr}$ (KeV) | 1345.77 |
| Neutron Spectrum | Thermal |
| Mass (g) | 0.1127 |
| Thickness (mm) | 0.25 |
| Size (mm) | 25 x 25 |
| Purity (%) | 99.50 |
| Target Source | Goodfellow |
| Vial No | 1 |
| Eγ (KeV) | 115.5390 |
| $I_\gamma$ (%) | 50.04 |
| Half Life (Days) | 243.93 |
| Atomic Mass (amu) | 64.9292405 |

**Table 2: Gamma Spectrum Measurement Data Using HPGe Detector**

| Parameters | Quantity |
|---|---|
| Target | Zn |
| Reaction | $^{64}Zn(n,\gamma)^{64}Cu$ |
| Irradiation Time $t_{ir}$ (s) | 3600 |
| Decay Time $t_D$ (s) | 4200 |
| Counting Time $t_C$ (s) | 3600 |
| Geometry at H1 (cm) | 2.00 |
| Full Energy Peak Efficiency ε at Geometry H1 | 0.001235 |
| Net Peak Count $N_c$ (c/s) | 28100.00 ± |

**Table 3: A comparison of Cross Section Data Obtained in this work with Literature Data**

| Reaction | Flux Type | Cross Section Data σ (mb) | S of $\sigma_{(x)}$ with Ref. | References |
|---|---|---|---|---|
| $^{64}Zn(n,g)^{65}Zn$ | Thermal | 1.980E+03 | - | Calculated σ(n,) for this work |
| | | 0.719E+03 | 891.6617 | Nakamura et al., 2024 |
| | | 0.717E+03 | 893.0759 | Farina et al., 2013 |
| | | 1.840E+03 | 98.9950 | Jinxiang et al., 1995 |
| | | 0.726E+03 | 886.7119 | De Corte,1988 |
| | | 1.150E+03 | 586.8986 | Koester et al., 1985 |
| | | 1.720E+03 | 183.8478 | Garg et al., 1981 |

*S = Standard Deviation for Sample $\sigma_{(x)}$= Cross Section Determined in this Work*

The available data in Table 1 are required for Equation (1). In Table 2 the data obtained in the spectrum measurement with HpGe include the value of Full Energy Peak Efficiency at the H1 = 2cm geometry reported elsewhere is given as ε = 1.235 x $10^{-3}$ and compares suitably with Sadiq and Jonah (2007).

The comparison of the results of the cross-section data obtained in this work with literature data is given in Table 3. For the reaction investigated, the data obtained was found to be in varying degree of acceptable limit with the measurements of Jinxiang et al., (1995) and Garg et al. (1981) as shown with the uncertainty in measurement provided by the determined Standard Deviation (S). With other measurement, more significant disagreement is noticed up to uncertainty of about 891.6617 Standard Deviation. Overall, the no significant discrepancy beyond 1000 standard can be reported.

It is noticed that the cross-section data obtained using NIRR-1 is consistently lower than all the other measurement. This suggests the likelihood of reduced neutron flux in the inner irradiation channel B2 compared to other facility used for the other reports. The extensive spread of existing experimental data also shows need for further interrogation targeted at harmonizing outcome of different methodology. This observation however has not diminished the accomplishment of the principal goal of this study which is related to the establishment of credible investigation framework, standard procedure and methodology for the measurement

of the production of cross section of medical radioisotopes in NIRR-1 that can be extended to other Miniature Neutron Source Reactors (MNSRs).

Comparing our determined cross section, 1.980E+03 for $^{64}Zn(n,\gamma)^{65}Zn$ reaction with the retrieved EXFOR data from nds-iaea website, our result agrees most with Jinxiang et al. (1995) with value at is 1.840E+03 mb.

**4.0 Conclusion**

The data from the study shows that NIRR-1, as well as other MNSR facilities, are suitable for the determination of the production cross section of medical radioisotopes in the thermal neutron energy range. The cross-section for $^{64}Zn$ (n, γ) $^{65}Zn$ fall within same order of magnitude and same range with all other retrieve measurements. The noticeable large uncertainty margin indicates the likelihood of diminished neutron flux in the B2 bare inner irradiation channel. The data obtained is however sufficiently in good agreement to explore further improvement in thigh quality cross section determination for the production cross section of Zn-65 and other important radioisotope for medical diagnostic and therapeutic and theragnostic applications.

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