Synopsis on

Study of Neutron Induced Reaction Cross Sections Upto 18 MeV for Advanced Reactor Design

By

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INTRODUCTION:

Several countries all over the world have different programs for utilizing the nuclear energy and some of them are developing the advanced technology known as ADSs (Accelerator Driven Sub-critical System), in order to utilize the potential of fertile ^{238}U and ^{232}Th [1-3]. In December 1999, in India a co-ordinator committee was formed for the research and development and construction of ADSs [2]. Primarily, the advancement of the technology which is mandatory for building ADSs was discussed in details and construction of linear proton accelerator (up to 20 MeV proton energy and 30 mA current) and study of the necessary maps and drafts of subcritical reactor as well as understanding of codes and development of befitting nuclear model codes, all these aspects are taken care of [1-9]. Actual design and construction of ADSs was looked after in the later phase. APSARA is the first swimming pool running reactor with highly enriched ^{235}U made in India at BARC, which has been decommissioned. Recently, a new swimming pool type of reactor with low enriched ^{235}U has been made at BARC. So investigations regarding coupling such a pool type reactor with a linear accelerator are going on [1-4].

Some of the striking features of ADSs are superior safety characteristics, its potential to incinerate long lived actinides, transmutes long lived fission products and of course nuclear energy production (electricity generation plus research and development). Rich reservoirs of thorium in India, add an additional dimension to this advanced technology. Also, as compared to uranium, thorium produces much less amount of the long lived radio actinides. However, thorium is a fertile material and not the fissile one. This fertile thorium can be transmuted into fissile isotope ^{233}U by neutron irradiation. This ^{233}U can be further used as a fuel and a part of the energy generated from it can be used to govern the accelerator connected with the sub-critical reactor and this loop can go on. The construction of ADSs was divided in several stages. These stages are discussed below in details:

1. Reactor physics experiments using AHWR reactors and 14 MeV neutron generators- building well designed Cyclotron (up to 350 MeV) and Linac (up to 100 MeV) were taken care in first stage.
2. Second stage if decided n few other sub-stages in order to test the modules of ADSs, spallation target and fission power.
3. Final stage will have the appropriate designing of a proto-type ADSs with proton beam up to 1 GeV. Proper testing and further research and development will be done in this stage.

Neutron cross-section data is required for the design of different components of advanced reactor, i.e., shielding design, waste estimation, estimation of radiation damage, nuclear heating, ttransmutation effects, radiation dose. Reaction cross-section data such as (n,γ), (n,2n), (n,p) and (n,α) of the spallation target (^{nat}Pb and/or ^{209}Bi) and structural materials (e.g., Zr, Nb, Fe, Ni, Cr, Mn, Y and Ag) induced by fast neutron are needed to reduce the risk related to radiation leakage. Nuclear data up to high energies for structural materials (e.g., Zr, Nb, Fe, Ni, Cr, Y, Mn and Ag) are required as hydrogen and helium production occurs during the irradiation, which can lead to swelling of the fuel as well as structural material and can cause the failure of the mechanical assembly. Besides this, the neutron induced reaction products are important for the use of medical diagnosis such as positron emission tomography (PET). However, its cross-sections database is scarce. In view of this, it is important to study the fast neutron induced reaction/fission cross-sections of stable fertile actinides (e.g., ^{232}Th and ^{238}U), structural and cladding materials (e.g., Zr, Nb, Fe, Ni, Cr, Mn, Y and Ag) [1-4].

Literature Survey:

Literature survey shows that lot of experimental work has been performed for the neutron induced reactions in the lower energy region of structural as well as fuel materials. There are no data available for these materials at some neutron energy regions. Also there is discrepancy among the available measured data of numerous authors in the measured as well as computed data [4-6].

On the basis of rigorous literature survey, the author has found that for ^{197}Au(n,γ) there are measured reaction cross section data available in between the neutron energy range of 0.025 eV to 3 MeV and at 14.7 MeV, compiled in the EXFOR data base [5]. The ^{197}Au(n,γ) reaction cross section data are available for 0.024 to 3 MeV and 14.7 MeV neutron energies. Except for these data, there is a zero availability of any other experimental data in 3 to 14.7 MeV neutron energy range [10-22].

For ^{55}Mn(n,γ)^{56}Mn reaction, measured cross sections are made available within 4 MeV neutron energy range and around 13.4 to 15 MeV. From 0.97 to 19.4 MeV energy range, only one data set is reported by Menlove et al [34]. Within 4 MeV neutron energy, there is discrepancy in the available data of various authors [23-33].

The detailed literature survey on ^{232}Th(n,γ) and ^{238}U(n,γ) reaction cross-section data indicates that there are many experimentally measured data are available for the ^{232}Th(n,γ) and ^{238}U(n,γ) reactions over wide range of neutron energies from thermal to 3 MeV based on physical measurements and activation technique. Beyond 3 MeV, only few measured data of the ^{232}Th(n,γ) reaction at the neutron energies of 3.7, 5.9, 8.04, 9.85, 11.9, 13.5, 14.55, 14.8, 15.5 and 17.28 MeV are available [34-45] and in the case of ^{238}U(n,γ) reaction, measured data are available at 3.033, 3.5, 3.7, 4, 5, 5.9, 6, 7, 7.2, 7.6, 8.04, 8.2, 9.2, 9.85, 10.2, 11.2, 11.9, 12.2, 13.2, 14, 14.2, 14.5, 13.5, 17.28, 14.8, 15.5, 17, 18, 19, 20 MeV [46-60].

MOTIVATION:

Most of the nuclear data in the neutron-induced reactions from the compilation [10-60] are based on average neutron spectrum of reactor. Experimental nuclear data of neutron induced fission/reactions of actinides up to high energy, which are quite important for the design of ADSs are rare and very much limited. Therefore, there is strong need to accurately measure fission as well as reaction cross sections of Th and U along with the structural material in the medium energy region (1-20 MeV) with mono-energetic neutrons apart from reactor neutrons. Measurements of the different types of reaction cross sections in the above mentioned neutron energy region will help us to study and analyze the excitation functions in detail. This will provide a proper database which will lead to utter understanding of mechanisms happening in the nuclear reactions [4].

OBJECTIVE:

Keeping above aspects in mind, we have measured the reaction cross sections in fast neutron induced reaction of ^{232}Th and ^{238}U isotopes related to ADSs. We have also measured the reaction cross section of neutron flux monitor (Au) and structural material (Mn) induced by fast neutrons.

The experimental work consists of the irradiation of the actinides (^{232}Th and ^{238}U), structural materials (Mn) and flux monitor (Au) followed by their gamma ray spectrometric analysis. The off-line gamma ray counting of the activated samples was done using HPGe detector connected to 4096 channel analyzer.

Theoretical data was computed using TALYS and EMPIRE nuclear model codes. Analyzed nuclear data was compared to the theoretically computed data with necessary graphs drawn in ORIGIN. These data were published in renowned journals as well as conferences.

Content of the present thesis:

The contents of the present work are divided in the five chapters as follows:

Chapter -1

This chapter starts with the introduction of the present work. Detailed literature survey is done regarding the present work and is explained in detail. Some basic information about ADSs, neutron induced reactions, fission reaction, cross section, flux monitor, neutron activation technique and neutron sources are also described in the present chapter. The motivation and objective of the present work are also explained in detail in this chapter. Content of the present thesis are explained in brief at the end of the chapter 1 [1-4].

Chapter -2

Experimental details like experimental set up, target preparation, generation of mono energetic and quasi energetic neutrons based on different reaction, irradiation details, cooling and gamma-ray counting of activated samples are covered in this chapter. Experimental facilities like BARC-TIFR Pelletron Facility at TIFR, Colaba, Mumbai and FOTIA facility at Van-De-Graff, BARC, Anushaktinagar, Mumbai are explained in details in this chapter. These are the facilities, used to carry out present experimental work. These facilities are explained in detail followed by target preparation method [40-43, 45, 57, 58, 60, 61].

Chapter -3

Theoretical aspects regarding present work are covered in this chapter. TALYS [7-8], EMPIRE [8-9], EXFOR [5] and ENDF [6] are explained in elaboration with the different nuclear models in this chapter. This chapter begins with the introduction to codes of nuclear models and proceeds with its applications, functioning, computations, etc. When there is a scarcity of experimental data in the literature, theoretical data can provide the useful insight of reactions of interest. Also if the experimental data are available one can compare it with simulated data and can view the physics part more clearly. Hence, presently, there is a need of theoretical as well as experimental data. Theoretical calculations based on nuclear models are explicit for modern nuclear data evaluation methodology. In the present chapter, author has discussed two nuclear model codes, TALYS and EMPIRE [7, 9]. Both TALYS and EMPIRE are Linux based programmable simulation codes, which are quite helpful in predicting the nuclear reaction cross section data theoretically. Both the codes use RIPL data library in order to get different parameters like level density, reaction models, etc [8]. Also these model codes have different samples where parameters are set as default and hence they are user friendly. For someone who wants to explore their expertise there is a facility to set the input files as well as parameters manually. EXFOR and ENDF are international data libraries, which provide experimental and theoretical nuclear reaction cross section data, respectively [5-6].

Chapter -4

The $^{197}Au(n,\gamma)^{198g}Au$ reaction cross-section data at different neutron energies with high accuracy is important because of its application as a neutron flux monitor during the determination of other reaction cross-section measurements. In IAEA, the excitation function of (reaction cross section vs. energy graph) $^{197}Au(n,\gamma)^{198g}Au$ is standard reaction monitor among all the nine standard reaction monitors. Besides this, the experimentally measured $^{197}Au(n,\gamma)^{198g}Au$ Au reaction cross-section data is useful in testing the theoretical TALYS model [7, 8].

In the present work we have successfully determined the $^{197}Au(n,\gamma)^{198g}Au$ reaction cross-section at 1.12, 2.12, 3.12 and 4.12 MeV neutron energies with the use of neutron activation technique and the off-line γ-ray spectrometric technique. The values for the neutron energies of 3.12 and 4.12 MeV are reported for the first time. Our data at the neutron energies of 1.12 and 2.12 MeV matches well with the available literature data [10-22]. The $^{197}Au(n,\gamma)^{198g}Au$ reaction cross-section were also computed theoretically using the computation code TALYS 1.6. All the present measurements as well as computed data were plotted along with the available experimental data in the literature.

Chapter -5

Because of its properties like hardenability, deoxidizer and sulphide former, manganese is used in making steel and also used in making structural and shielding materials in a reactor. In a reactor, neutrons interact with manganese present in the structural and shielding materials. When fast neutrons come in the contact with the ^{55}Mn, it forms unstable isotope ^{56}Mn. The radioactive ^{56}Mn has a half life of 2.5789 hours, which decays to a stable product with the emission of 846.8 keV γ-rays. In any conventional reactors, the neutron spectrum varies from thermal to 15~20 MeV. Thus it is quite important to know the cross sections of $^{55}Mn(n,\gamma)^{56}Mn$ reaction for various neutron energies. This will enable us to learn reaction mechanism more clearly.

At the four different neutron energies of 1.12, 2.12, 3.12 and 4.12 MeV, the $^{55}Mn(n,\gamma)^{56}Mn$ reaction cross section are determined by using neutron activation technique and off-line gamma ray spectroscopic technique. These data are in compliance with one set of experimentally measured literature data but not with the other two sets of data available in literature [23-33]. The $^{55}Mn(n,\gamma)^{56}Mn$ reaction cross sections were computed using the two nuclear model codes, TALLYS and EMPIRE [7-9]. Both theoretical and measured data fall inside the spectrum of available literature data.

Chapter -6

This chapter holds the information of neutron induced cross sections for $^{232}Th(n,\gamma)$ and $^{238}U(n,\gamma)$ reactions. Further, this chapter gives the detailed descriptions of the experiments done at both the institutes, BARC and TIFR including the description regarding the detailed analyses of the results. So many data of various authors are there in the lower neutron energy region. There is some data available in the literature below 3 MeV neutron energy and a few data within 13-15 MeV. Nuclear fission and reaction data from medium to higher energies are of extreme importance in ADSs In view of this, $^{232}Th(n,\gamma)$ and $^{238}U(n,\gamma)$ reaction cross-sections are determined in the present work for 5.08±0.165, 8.96±0.77, 12.47±0.825 and 16.63±0.95 MeV neutron energies. Also to check the reliability of the measured data, reaction cross-sections in between the neutron energies of 1-20 MeV was computed using the nuclear code TALYS 1.6 [14-15]. The $^{232}Th(n,\gamma)$ and $^{238}U(n,\gamma)$ reaction cross-sections obtained from the present work at the neutron energies of 5.08±0.165, 8.96±0.77, 12.47±0.825 and 16.63±0.95 MeV are measured for the first time by using the neutron activation technique and it falls inside both upper and lower limits of already available literature and theoretical data [34-60].

Chapter -7

This chapter contains the results from the present experimental work, theoretical calculations and their discussion part. The summary of the complete research work done by the author is covered in this chapter with the future outlooks of the present work [1-4].

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