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## 1.1 Importance of the present work

The idea of the green and peaceful use of nuclear energy is in the heart of the nuclear reactor technology. The nuclear reactions are the base of the energy production in the nuclear reactors. Most of the reactor uses fission of U, Th and Pu isotopes mixture as fuel. On the other hand, fusion reactor uses the hydrogen isotopes deuterium and tritium as fuel, and fusion of these two nuclei at high temperature produces neutrons. In fission reactors, neutrons have energy from thermal to few MeV, and in a fusion reactor, the neutrons have energy 14.6 MeV, which is scattered from the different reactor materials and get a spectrum of energy from thermal to 14.6 MeV [1-7]. These neutrons induce different nuclear reaction channels, such as (n, γ), (n, p), (n, n'), (n, d), (n, t), (n, α), (n, 2n) etc. These reaction channels transmute the base reactor material, which may have different mechanical, thermal and chemical properties. The mechanical properties are the strength of the material, Young's modulus, limit of elasticity etc. Thermal properties are directly related to the physical state of the material at a different temperature, which is also necessary for the mechanical strength. A reactor contains fuel, controlling, structural, shielding, and many other types of materials with specific purposes. In the case of a fusion reactor, the materials are selected as per different components, such as divertor, limiter, blanket, superconductor magnet, structural etc. The selection of these materials should be done very carefully taking into account the transmutation criteria [8]. The different materials such as Na, Ti, V, Cr, Co, Fe, Cu, Zn, Mn, Ni, Zr, W, Pb, Bi etc., are used in the reactor at several places, mostly as structural materials [9, 10]. These materials should have experimentally measured nuclear data at different energies of incident neutrons. The neutron and photon data are more important as they are the major products of the reactor fission/fusion reactions. Most of the reactor materials have measured reaction data at thermal neutron energy, 2.5 MeV and 14 MeV [11]. Using several natural as well as accelerator based neutron sources, several authors have reported nuclear data up to 20 MeV, which can be found in the experimental data library EXFOR [11]. These data are not only important for the nuclear reactor development but they are the key tools to validate the nuclear models [12-14]. The data required for the reactor development has been discussed by R. Forrest [15, 16].

In the present thesis, the nuclear data obtained from the measurements on the various structural materials are described below.

Tungsten is selected as a diverter material in the International Thermonuclear Experimental Reactor (ITER) as well as it is also being used in Accelerator Driven Subcritical System (ADS) [16]. In ITER, the neutrons are ranging from thermal energy to 14.6 MeV, and in ADS they can have higher energies. Tungsten will be irradiated with neutrons produced by the DT fusion in ITER [11, 17]. Thus the isotopes of tungsten $^{180,182\text{-}184,186}$W will be transmuted into different isotopes though different reaction channels. Hence it is necessary to know all the reaction data for all the possible reactions of tungsten isotopes. Most of the data for tungsten isotopes are available at thermal neutron energy, 2.5 MeV (DD neutrons) and 14 MeV (DT neutrons) in EXFOR data library [11]. The energy region, 1 MeV to 13 MeV and 14 to 20 MeV contains very few experimental data points. Therefore, it is important to have more data in this energy range to complete the data set.

In fission and fusion reactors, there are several transition elements used as structural materials such as V, Cu, Zn, Mn, Ni, and irradiated during the operation condition of the reactors and transmute into other materials. The measured data for the nuclear reactions for these elements are scarce and have large errors [11]. The rareearth materials, which are always present along with these structural materials (As, Gd) interact with neutrons. This leads to transmutation that can change their electronic, mechanical and thermal properties. The nuclear data for the capture reaction on the isotopes of these elements are available in a good amount below 1 MeV, but few data points are available beyond this range of neutron energy [11]. At neutron energies higher than 1 MeV, other reaction channels become important, but the data for all isotopes are not available. There are several evaluated nuclear data libraries available which provide the data for nuclear reactions. Some of these libraries are ENDF-B/VII.1 [18], JENDL-4 [19], FENDL [20], ROSFOND [21], CENDL-3.1 [22], JEEF-3.2 [23] etc. There are several nuclear models that are used for the prediction of these evaluated nuclear data libraries. The measured data can help to verify the nuclear reaction models [12, 13]. In this thesis, measurement of the neutron induced nuclear reaction cross sections in MeV energy range has been presented to enhance the nuclear data libraries.

Further, during the plasma shot in a fusion reactor, the runaway electrons can be generated and they can reach the energy from few keV to several MeV [24]. The

higher energy electrons can form a beam. Such runaway electron beam can strike on the wall of the vacuum vessel. It can either damage the vessel or partially damage it [25]. During this interaction, these electrons can produce high energy photons of several MeV energy. At lower photon energies, the photoelectric, Compton's scattering and pair production reactions dominate. The photons can reach beyond 10 MeV energy and they can open photonuclear reaction channels with the target nuclei. The lowest threshold among all photonuclear reactions is (γ, n) reaction. This interaction is energy dependent. The interactions are (1) Giant Dipole Resonance (6 – 30 MeV) – GDR, (2) Quasi – deuteron (30 – 150 MeV) and (3) Intra-nuclear Cascade (above 150 MeV) [26]. In a fusion reactor, it is important to study the GDR mechanism, as the most of the photons lie in this energy range. It can produce a considerable amount of neutrons, which can affect the total neutron yield, ultimately to the fusion power. Further, the reaction itself is a neutron source, and in designing a neutron source based on this reaction, target design and photon energy optimization are the main objectives. There are several experimental and evaluated data available in the EXFOR and evaluated data libraries respectively [11]. Due to the unavailability of several targets and other complications in experiments, it is not possible to have complete experimental data set. Alternatively, with the help of nuclear modeling, it is possible to get the evaluated data, which can complete the data set. Further, the development of suitable empirical formula based on primary nuclei properties can explain the photonuclear reactions. There are number of empirical formula at specific neutron energies that can be found in the literature [27-34]. The work done in this field of photon induced reactions can be found elsewhere [35,36]. In the present thesis, a new empirical formula has been developed which can explain the (γ, n) reaction cross section near to GDR peak energies for the isotopes with $Z \geq 60$. Systematic development of the formula using the different terms, with their physical interpretation has been discussed. Further, the applications of this formula in the development of a nuclear database for future reactor technology have been discussed.

## 1.2 Nuclear reactions

### 1.2.1 Overview

When a particle (with sufficient energy) incidents on a target nucleus, its interactions with the nucleus is called nuclear reaction. This interaction may be either elastic or inelastic. If the kinetic energy before and after interaction remains same, it is called elastic, and if not, then it is inelastic. Both interactions give information about the nuclear structure. The different incident particles are used worldwide to learn different nuclear reactions. When a neutron is an incident on a target atom, it directly reaches to the nucleus, as it doesn't possess any charge and hence no Coulomb barrier. Once it enters within the nuclear radius, it is under the strong force field of the target nucleus. For a moment, a neutron inside the nucleus loses its initial information and behaves as a nucleon of the target nucleus. At this moment a momentarily stable state called the compound nucleus of the target and the entered neutron is produced. This compound nucleus doesn't contain any information of its formation, and ultimately the transient state of compound nucleus decays into different channels as per its excitation energy. If the ejected particle is a neutron and has the same energy as incident one, it is called an elastic scattering and if the ejected neutron has different kinetic energy then the reaction is called inelastic scattering denoted as (n, n'). If in place of neutron a gamma photon is ejected, and the incident neutron is absorbed, then the reaction is called neutron capture. The other neutron induced nuclear reactions are (n, p), (n, d), (n, t), (n, $\alpha$), (n, 2n) etc. If neutron energies are below 2-3 MeV the compound nuclear reaction mechanism is important. Above this neutron energy, another mechanism called pre-equilibrium mechanism gradually becomes important. In this mechanism, the nucleus does not go completely into the compound nucleus stage, but before it, it ejects some particles. And above 20 MeV, the mechanism is called direct reaction dominates. In this mechanism, neutron interacts to a nucleon in place of the whole nucleus. The details of all these three-reaction mechanisms are discussed in Chapter – 2.

When a neutron is able to break the target nucleus into small fragments, the reaction is called fission reaction. Heavy nucleus once captures the incident neutron, they get such excitation energy, so that they break into small parts (as separate nuclei), which is known as fission fragments. These reactions can produce 2-3 neutrons, along with these fission fragments. These neutrons can further incident on new targets and can

continue the reaction in chain form called a chain reaction. During this process, energy is produced by conversion of some mass into energy according to Einstein's formula. The controlled chain reaction is the principal of the nuclear reactor. Along with the neutron, the photonuclear reactions are the part of this thesis work. The details of the photonuclear reactions have been discussed in Chapter – 3.

### 1.2.2 The reaction kinematics

The energy released or consumed in a nuclear reaction is called Q value of the reaction. Suppose a nuclear reaction X(x, y)Y, in which the x is an incident particle (projectile), X is a target, y is ejected particle (ejectile), and Y is a residue nucleus. The masses of the each particles are $m_x$, $m_X$, $m_y$ and $m_Y$ respectively. $E_x$, $E_X$, $E_y$ and $E_Y$ are the kinetic energies of the each respectively If we think about the masses, then there is a difference in the mass on the either sides of the reaction, i.e. the total mass of the projectile and target before the reaction may not equal to the total mass of the ejectile and residue nucleus. This indicated some mass to energy conversion or vice versa. This energy is called Q value. Its mathematical expression can be written for the above reaction as follows,

$$Q = \{(m_X + m_x) - (m_Y + m_y)\} \cdot c^2 \quad 1.1$$

$$Q = (E_Y + E_y) - (E_X + E_x) \quad 1.2$$

The Q value energy either positive or negative. The positive value indicates exothermic reaction, whereas the negative value represents endothermic reaction.

In case of reaction kinematics, two frames of references are taken in general: the laboratory system (LS) and the center of mass system (CM). In laboratory system, the target is considered to be at rest, and incident particle is moving with certain velocity towards the target, and after interaction the ejectile deflects with certain velocity at certain angle. In contrast to this, in the CM system, the center of the mass of the target and projectile is to be considered at rest. Hence before the reaction both projectile and the target are moving towards each other, and after collision, both are moving at different angle with different speeds. Suppose in LS that the velocity of projectile and ejectile are $v_x$ and $v_y$, the velocity of residue is $v_Y$. The velocities projectile, ejectile, target and residue in CM system are $v_x$ - Vcm, $v_y$', -Vcm and $V_Y$' respectively.

The visualization of kinematics for both systems is given in Fig 1(a-b).

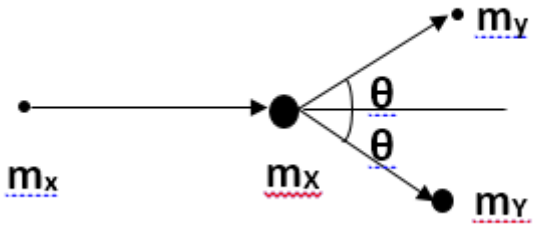


**FIG 1.1** Laboratory System

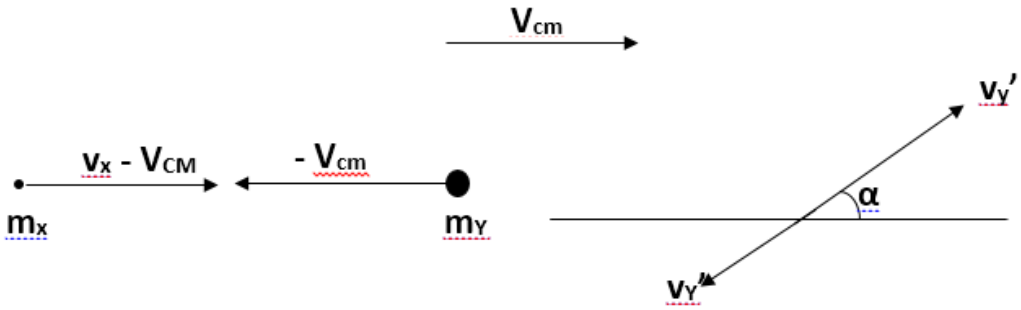


**FIG 1.2** Center of Mass System

By applying the collision theory and energy conservation the Q value of the reaction can be given as,

$$Q = E_y\left(1+\frac{m_y}{m_Y}\right) - E_x\left(1-\frac{m_x}{m_b}\right) - \frac{2\sqrt{m_x m_y E_x E_y}}{m_Y}\cos\theta \qquad 1.3$$

From the present equation one can calculate the energy of the ejected particle as well as the energy distribution as per the angle.

There is one more term called threshold energy has been defined by following equation,

$$E_T = -Q\frac{m_x+m_X}{m_X} \qquad 1.4$$

This is the minimum energy required to initiate a nuclear reaction. For a charge particle reaction in addition with this, the Coulomb's barrier also need to be considered.

### 1.2.3 Neutron activation analysis

#### 1.2.3.1 Introduction

Georg Hevesy and Hilde Levi were the first who reported a new method of activation analysis [37]. It was their observation that when materials are irradiated with neutrons they become radioactive. With the radioactive emission, one can identify the product isotopes. This method, which is based on neutron irradiation, activation and then analysis, is called the Neutron Activation Analysis (NAA). At that time, because of unavailability of high neutron flux sources, this method of NAA was not much explored. Later on with the development of the high neutron flux sources such as research reactors, this method becomes an important tool for research purpose. The most significant advantages are, the process is non-destructive and used on any material with any physical state [38]. NAA can simultaneously analyze multiple elements with high accuracy and sensitivity, while also only requiring a minimal amount of material for a sample, makes the NAA process very attractive for research purposes [39]. Its primary use is to determine the elemental concentrations within a sample of interest. Additionally, isotopes contained within a sample can be detected and separated from the base elements allowing the researcher to determine what percent of the sample is comprised of certain isotopes [40]. In this method after the irradiation, the parent isotope produces radioactive daughter isotopes, which emits the alpha, beta or gamma rays. After the development of the High Purity Germanium detector (HPGe), which has very high resolution, the method becomes very effective. The gamma photon emitted from the daughter can be measured using an HPGe detector, and the peak area of the interested peak energy contains the information of the reaction parameter – cross section. This method is very important for the cross section measurements and has been widely used by the researchers since its discovery.

In this method, a material is irradiated with the neutrons. The neutrons produce activation in the sample. The transmuted isotope must be radioactive and should have not too small and not too large half-life. Further, it must follow the radiation decay, which can be measured with the help of detector spectrometry. With these data, one can estimate the unknown parameters of the reaction. The mathematical formulism for this method is discussed in the next section.

### 1.2.3.2 Derivation of reaction rate

If we are dealing with a neutron induced reaction, which is a capture reaction, then it requires no threshold energy. But for the other reaction, the threshold energy is non-zero and this is in the MeV range as the binding energy of a nucleus is in MeV range. The derivation of the reaction rate can be found in the literature [41].

If a neutron induced reaction has threshold energy – $E_{th}$ and we are using a mono energy neutron with $E_n$ energy with $E_n > E_{th}$, then the reaction rate R can be given as

$$R = N \cdot \sigma \cdot \phi \quad 1.5$$

Where, N = number of target atoms,

$\sigma$ = cross section of the reaction

$\phi$ = neutron flux with energy $E_n$

This equation contains a parameter cross section of the reaction, which is a probability of the reaction. Its dimension is per unit area, and measured in the barn (= 1 $cm^2$). This equation is only valid if neutrons are monoenergetic. If the neutrons are not mono energy but having energy spectrum, then the following expression should be considered [42].

$$R = \int_{E_{th}}^{E_{max}} N \cdot \sigma(E_i) \cdot \phi\,(E_i) dE \quad 1.6$$

Consider an isotope A is irradiated with neutrons, which transmute it into isotope B, which is also radioactive with decay constant $\lambda_b$, decays into isotope C. Hence, the process can be written as

$$A + n \rightarrow B(\lambda_b) \rightarrow C \quad 1.7$$

The irradiation of A produces B, but during the irradiation, some of the fraction of B also converts into another isotope (other reaction channels). Further, the isotope B decays into C, hence the overall reaction rate of production of isotope B can be written as,

$$R_b = N_a(t)\sigma_a\emptyset - N_b(t)\sigma_b\emptyset - \lambda_b N_b(t) \quad 1.8$$

$$\frac{dN_b}{dt} = N_a(t)\sigma_a\emptyset - N_b(t)\sigma_b\emptyset - \lambda_b N_b(t) \quad 1.9$$

Where, $R_b$ is the reaction rate, $N_a(t)$ is the number of targets of isotope A at time t, $N_b$ is the number of atoms of nuclei B at time t,

The solution of this reaction can be given as,

$$A_b = N_b(t_i)\lambda_b = \frac{N_a(0)\sigma_a\emptyset}{1+(\sigma_a-\sigma_b)\emptyset/\lambda_b}[e^{-\sigma_a\emptyset t_i} - e^{-(\lambda_b+\sigma_b\emptyset)t_i}] \quad 1.10$$

Practically, the target is selected such that very small fraction of destroyed in irradiation and hence it can be negligible ($\sigma_a\emptyset t_i \ll 1$). Also, it is necessary that the produced isotope must have followed the condition $\lambda_b \gg \sigma_b\emptyset$. With these conditions, the solution above can be simplified by,

$$A_b(t_i) = N_a(0)\sigma_a\emptyset[1 - e^{-\lambda_b t_i}] \quad 1.11$$

This formulism can be used to measure neutron flux or the cross section of the reaction. When a target is irradiated with the neutrons, the product isotope will be formed. The irradiation time is taken above as $t_i$. After irradiation, the activity produced in the target is measured using gamma ray spectrometer. The time duration from stopping the irradiation to the start counting is called cooling time ($t_w$). The counting time is $t_c$ and the clock time (real time) is $t_r$. The cross section of the reaction can be given by the following expression. Suppose that an irradiated sample is counted between time $t_1$ and $t_2$.

$$\int_{t_1}^{t_2} A_b(t)dt = \frac{N_a(0)\sigma_a\emptyset}{\lambda_b}[1 - e^{-\lambda_b t_i}][e^{-\lambda_b t_1} - e^{-\lambda_b t_2}] \quad 1.12$$

The left side of the equation is the measured activity by some gamma spectroscopy,

$$A_\gamma = \frac{N_a(0)\sigma_a\emptyset}{\lambda_b}\left[1 - e^{-\lambda_b t_i}\right]\left[e^{-\lambda_b t_1} - e^{-\lambda_b t_2}\right]$$

$$\sigma_a = \frac{A_\gamma\lambda_b}{N_a(0)\emptyset[1 - e^{-\lambda_b t_i}][e^{-\lambda_b t_1} - e^{-\lambda_b t_2}]}$$

Suppose we considered start time as zero then,

$$\sigma_a = \frac{A_\gamma\lambda_b}{N_a(0)\emptyset[1 - e^{-\lambda_b t_i}][1 - e^{-\lambda_b t_c}]}$$

As during cooling, the isotope B will decay, hence with the correction of cooling time,

$$\sigma_a = \frac{A_\gamma\lambda_b}{N_a(0)\emptyset[1 - e^{-\lambda_b t_i}][1 - e^{-\lambda_b t_c}]e^{-\lambda\cdot t_w}}$$

Including the detector parameter gamma efficiency (ε), gamma abundance and time correction, the modified activation formula is now,

$$\sigma_a = \frac{A_\gamma\lambda_b(t_c/t_r)}{N_a(0)\emptyset I_\gamma\varepsilon[1-e^{-\lambda_b t_i}][1-e^{-\lambda_b t_c}]e^{-\lambda\cdot t_w}} \quad 1.13$$

This is the standard neutron activation equation. This equation can be used to measure the cross section of a neutron induced nuclear reaction. And for a monitor reaction, the neutron flux can also be measured. In the present thesis, the neutron induced nuclear reaction cross sections were analyzed using this above formalism.

#### 1.2.3.3 Importance of neutron activation analysis

The neutron activation analysis – NAA is a spectroscopic method, used to measure several data of nuclear reactions. The major use of this method is in measurements of the neutron flux and reaction cross section. The method is used to calibrate the yield of the neutron sources by foil activation. The different standard foils, such as In, Au, Ni, Cu, Al, Th can be irradiated with the neutron sources. The activation produced in the foil carries the information of the source from which it got activated, which can be identified (gamma spectroscopy) and analyzed with NAA method. With this one can calibrate the source. In the Joint European Torus (JET), which is the world's largest tokamak, the neutron yield and ultimately the fusion power calibration has been done using this method [43,44]. The upcoming fusion devices will also use the same method for several diagnostics point of view.

Large cross section database, which is purely experimental, has been available in experimental data libraries such as EXFOR, is because of this method. The method is more important as it is nondestructive and any physical form of the sample can be used. Also, it does require a very small amount of the material. Further, as it is based on nuclear reaction, hence it is chemically independent.

The method is useful to know the material composition, which is up to the level of isotopes constitute in the material. It allows not only identifying the unknown isotope in the material, but also the amount of it present in the material. The primary requirement of this method is a neutron source, which is discussed in the next section.

In the present thesis, the NAA method was used to analyze the neutron induced nuclear reactions. The cross sections of (n, γ), (n, p), (n, 2n) etc. were measured and calculated for different selected materials.

## 1.3 Neutron sources

Chadwick discovered neutron in 1932. The natural neutron sources are based on ($\alpha$, n) reaction. In such sources, an alpha emitter is used to incident alpha on a light element such as beryllium. These sources are potentially with large half-life and neutron yield. Some examples are polonium and beryllium, $^{241}$Am+Be, Ra+Be. The disadvantages of these sources are they have a high gamma background as well as neutrons are not mono energy. There are some potential spontaneous neutron sources, such as $^{252}$Cf. The detail of this source is given in Chapter – 5. The reactors are the good sources of the neutrons. Due to fission, they produce neutrons in each fission reactions. The neutron flux in reactors is very high.

The other way is the production of neutrons artificially. These sources are based on nuclear reactions. In these kinds of sources, a high energetic beam of the particle is made to incident on a target nucleus. The beam has sufficient high energy that the incident particle can cross the Coulomb's barrier of the target nucleus. Such energetic beam of particles can be produced by a particle accelerator, such as a cyclotron, tandem accelerator etc. The details of such accelerators can be found in any standard nuclear physics book. The major reactions used for neutron production using the accelerated beam are DD, DT, and Li(p, n) reactions.

$$^{2}_{1}\mathrm{H} + ^{2}_{1}\mathrm{H} \rightarrow ^{3}_{2}\mathrm{He} + ^{1}_{0}\mathrm{n} + 3.268\ \mathrm{MeV} \qquad 1.14$$

$$^{2}_{1}\mathrm{H} + ^{3}_{1}\mathrm{H} \rightarrow ^{4}_{2}\mathrm{He} + ^{1}_{0}\mathrm{n} + 17.589\ \mathrm{MeV} \qquad 1.15$$

$$^{1}_{1}\mathrm{H} + ^{7}_{3}\mathrm{Li} \rightarrow ^{7}_{4}\mathrm{Be} + ^{1}_{0}\mathrm{n} - 1.664\ \mathrm{MeV} \qquad 1.16$$

The first two reactions give mono-energetic neutrons. The DT reaction will be used for power generation in a fusion reactor. These reactions produce fixed energy neutrons. Compact DT neutron sources are available for the research purpose. The third reaction is used at various laboratories. This reaction is important because its threshold energy is 1.88 MeV, hence the accelerated proton beam kinetic energy can be transferred to neutrons. If $E_p$ is the proton beam energy then the ejected neutrons will have $E_p$ – 1.88 MeV neutron energy. The details of this reaction are discussed in Chapter – 4.

Apart from this, the (γ, n) neutron sources are also used. The high energy natural γ sources, as well as the bremsstrahlung γ source are made to fall high energy γ on target to do photo-fission. These sources are relatively lower yield and white spectrum.

In accelerator driven subcritical system (ADS), a very high energy proton will be targeted on heavy atomic number material, and through the spallation, reaction neutrons will be produced. Due to the impact of a high energy proton, the target nucleus breaks into fragments, this reaction is called spallation reaction. The neutrons produced in this reaction will have very high energy. The spallation reaction can be observed in earth's atmosphere by the cosmic rays.

## 1.4 Objective of the present thesis

In view of the above discussions, following are the main objectives of the present thesis:

1) The measurements of nuclear reactions cross-sections of $^{186}W(n, \gamma)^{187}W$, $^{182}W(n, p)^{182}Ta$, $^{154}Gd(n, 2n)^{153}Gd$, $^{160}Gd(n, 2n)^{159}Gd$ reactions between 5 to 17 MeV neutron energies using TIFR Pelletron facility, Mumbai, India. The neutrons were produced using $^{7}Li(p, n)$ reaction. And the analysis was done using neutron activation analysis method.
2) Development of the new empirical formula for the photoneutron production (γ, n) reaction cross-section near giant dipole resonance (GDR) for isotopes with $Z \geq 60$.
3) The measurements of nuclear reactions cross-sections of $^{183}W(n, p)^{183}Ta$ and $^{184}W(n, p)^{184}Ta$ in $^{252}Cf(sf)$ neutron field at Defense Laboratory, Jodhpur, and IPR, Gandhinagar.
4) The measurements of nuclear reactions cross-sections of $^{75}As(n, p)^{75}Ge$, $^{66}Zn(n, p)^{66}Cu$, $^{64}Zn(n, p)^{64}Cu$, $^{55}Mn(n, p)^{55}Cr$, $^{51}V\ (n, p)^{51}Ti$ and $^{58}Ni(n, p)^{58}Co$ at 14.2 ± 0.2 MeV using DT neutron source at BHU, Varanasi.
5) The calculations of reaction cross sections using various nuclear reaction codes (nuclear modular and transport) for nuclear data prediction.

## 1.5 Structure of present thesis

The present thesis contains seven chapters. Chapter – 1 introduces the topic of the present thesis. Chapters 2 – 7 contains a description of work carried out during this study. The Chapter – 2 is about the nuclear codes, which are the theoretical tools for development of nuclear data. It gives a description of different nuclear reaction models used in nuclear modular codes, which are TALYS and EMPIRE, and nuclear transport code, which is MCNP. These codes are described in this chapter.

In Chapter – 3, the development of the new empirical formula for the photonuclear reaction cross sections has been discussed. The Chapter – 4 describes the neutron induced reaction cross sections measurements performed at the Pelletron facility at Tata Institute for Fundamental Research, Mumbai, India. Chapter – 5 is about the nuclear reaction cross sections measurements using the $^{252}$Cf spontaneous neutron source at Defense Laboratory, Jodhpur, India. The Chapter – 6 describes the cross section measurement at Banaras Hindu University, Varanasi, India, using 14 MeV neutrons. Finally, Chapter – 7 describes the summary of the work done so far in this study and the outcomes.

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