Chapter 5

Summary & Conclusions

This chapter provides the summary and conclusions of the investigations discussed in chapters 3 & 4. The chapter is subdivided into two parts, which the first contains the conclusions drawn from the neutron-induced reaction cross-section measurements, second gives the conclusions drawn from the analysis of improvement in reactor shielding materials. In summary, the data reported in the present thesis are vital for the advancement of the current reactor/accelerator technology, for dose estimation of nuclear structural materials, nuclear medicine, in the refinement of control rods, etc.

The measurement of neutron-induced reactions for reactor structural materials is reported in the present thesis. The experimental work has been performed at BARC-TIFR Pelletron and BARC-Purnima accelerator facilities. The neutrons of required energies were generated by using $^{nat}Li(p,n)$ and $^{3}H(^{2}H,n)$ reactions at BARC-TIFR Pelletron and BARC-Purnima accelerator, respectively. The generated neutrons from these methods have been found very much suitable for this type of experimental measurement. Further, the off-line γ-ray spectroscopy is adopted for the measurement of the γ-activity of irradiated samples. The irradiated samples were counted after the irradiation, using a p-type single crystal HPGe detector.

Here, the γ-detection and detector efficiency of the samples were measured from a common detector setup, hence, the covariance analysis was adopted to measure the correlations and uncertainties in the neutron reaction cross-sections. The overall uncertainties in neutron-induced reaction data were found to be below 19%. The reported data were found to be consistent with the literature data for the entire range of consideration, however, at few energies, negligible variation can be noticed. The outcomes of this work present a better data set for the measured reactions, especially where the literature data contained discrepancies. Moreover, the results were also found to be in agreement with the evaluated and theoretical data from the nuclear code. Another main objective of this thesis was to improve reactor shielding materials by adding low-Z and high-Z additives in concrete. The samples were prepared according to the local manufacturing standards. Total, twenty-one concrete samples were prepared to investigate the shielding parameters of γ-rays and neutron. The experimental work has been performed using ^{60}Co and ^{Cf}Cf sources at Defence Laboratory Jodhpur. The γ-rays and neutron attenuation parameter were measured using NaI(TI) scintillation and BF_3 detectors. This experimental measurement compared with various theoretical prediction codes, like MCNP, XCOM, Auto-Z_{eff}, and NXcom. The parameters μ_m, Z_{eff}, N_{eff}, HVL, TVL, MFP, and Σ_R were calculated to investigate the present work. The outcomes of the present work show that the modified samples have more advantages as compared to the pristine concrete. These also revealed that shielding parameters strongly rely on the atomic composition and density of the prepared samples. Moreover, the results of the parameters were also found to be in good agreement with the theoretical codes. The specific details for both the study are provided in the following sections, respectively.

5.1 Neutron Induced Reactions

The excitation function for the $^{159}Tb(n,\gamma)^{160}Tb$, $^{113}In(n,n)^{113m}In$, $^{115}In(n,2n)^{114m}In$ and $^{115}In(n,n')^{115m}In$, and $^{181}Ta(n,2n)^{180}Ta$ reactions were measured for the incident quasi-monoenergetic neutron energies in the span of 5 to 20 MeV varying differently for each case. Basically, the present work pointing that the uncertainties in the monitor reaction data are crucial in the uncertainty calculations. Moreover, The conclusions drawn from each reaction measurement are as follows:

- The $^{159}Tb(n,\gamma)^{160}Tb$ reaction cross-sections were measured relative to the $^{232}Th(n,f)^{97}Zr$ monitor reaction at the neutron energies of 5.08 ± 0.165, 12.47 ± 0.825, and 16.63 ± 0.95 MeV. The measured cross-sections match well with the literature as well as theoretical data. The uncertainties in the data were obtained with the help of covariance analysis was found to be within the span of 11-14%. Among the total, the major contribution was coming from the counting statistics of the monitor reaction. The results were compared and found to be in agreement with the evaluated data from

ENDF/B-VII.1 and JENDL-4.0 libraries. The theoretical prediction from TALYS-1.9 using different level density model, reproduced the trend of the data very well, except for 5.08 ± 0.165 MeV, which was found lower compared to the theoretical predictions.

- The $^{113}In(n,n)^{113m}In$, $^{115}In(n,2n)^{114m}In$ and $^{115}In(n,n')^{115m}In$ reaction cross-sections were measured at neutron energies of 10.95 ± 0.67, 13.97 ± 0.97, 16.99 ± 0.88 and 20.00 ± 0.94 MeV. The uncertainties measured using covariance analysis were found to be in the span of $4-19\%$. The results were found to be in agreement with the literature, evaluated data from JENDL/AD-2017 and the theoretical prediction by using different as well as by adjusting level density models in TALYS-1.9. Default level density model (collective + effective) of TALYS were found unable to emulate the experimental as well as the literature data. It surpassed the trend of data in the threshold to 22 MeV for $^{113}In(n,n')$ and 12 to 22 MeV range for $^{115}In(n,2n)$reaction. The results were found to be satisfactory for $^{115}In(n,n')^{115m}In$ reaction.

- The production cross-section for ^{181}Ta is measured at the neutron energy of 14.78 ± 0.20. The $^{197}Au(n,2n)^{196}Au$ reaction were used as monitor for (n, 2n) reaction cross-section. The uncertainty in the present measurement calculated with the help of covariance analysis was found to be 15%. The present results were found in accordance with the both literature and the evaluated data from JEFF-3.32, ENDF/B-VII.1 libraries. Similar to the above measurement, the theoretical calculations reproduced the datum satisfactorily.

5.2 Shielding Material

The investigation of WC and B_4C additives in ordinary concrete for γ-rays and neutron shielding have been performed through parameters μ_m, Z_{eff}, N_{eff}, HVL, TVL, MFP, and Σ_R.

- The measured γ-ray attenuation parameter (μ_m) data for all the investigated samples were found to be in good agreement with the theoretical prediction XCOM and MCNP.

- The measured value of μ_m is further used to calculate Z_{eff}, N_{eff}, HVL, TVL, and MFP. Among these, the value of Z_{eff}, and N_{eff} are predicted from Auto-Z_{eff} software. The calculated and predicted (Auto-Z_{eff}) trend of N_{eff} are closely related to Z_{eff} and for all the samples their variations with respect to energy are also found similar.

- The results of HVL and TVL parameters show the prepared M1-M5 sample has required a lower volume of the material compared to M6 and M7.

Analyzing all these parameters suggested that among the prepared samples M1-M5 has a superior for γ-ray shielding.

- The experimental findings of neutron attenuation were compared with the NXcom prediction and MCNP. The comparison revel that the prepared M1-M6 samples are more effective for neutron attenuation relative to OC.
- The both γ and neutron study emphasized that the compounds of tungsten (WC) and boron (B_4C) as additive may be a good choice that can be used for the shielding.
- It is suggested that, for radiation containing γ and/or neutron, the materials M1-M5 provide an alternative source/option with ordinary concrete. Which can minimize the required space as well as the effective cost of shielding materials.

5.3 Future Plans

Due to the limitation in energy range of the accelerators, in the present experimental measurements of reaction cross-sections have a possible range of up to 20 MeV of incident neutron energies. Hence, similar work can be extended for the neutron energies above 20 MeV, to generate accurate nuclear data and also investigate the energy dependence of the reaction cross-section. The neutron spectra generated through simulation codes like MCNP, PHITS, and GEANT4 at higher proton energies above 15 MeV give more simplicity in terms of analysing the data. Aside from simulation, the n-TOF technique can be used to measure the neutron spectrum from the $7^{L}i(p,n)$ reaction experimentally. Similarly, for radiation shielding various low cost, and easy available compounds must have to tested for both the γ as well as neutron shielding. Also, similar work can be extended for the various matrix elements apart from concrete such as EPDM rubber, HDPE, etc.

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