STUDY OF NEUTRON INDUCED REACTION CROSS SECTION FOR REACTOR BUILDING MATERIALS

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Neutron induced reaction cross section is a measure of the interaction probability between the neutron and the atomic nucleus. This probability is a quantitative value and the interaction can be of different types depending on the energy of the incident neutron. The quantitative value of the neutron interaction probability (or cross section) is one of the main ingredients of any simulation code used in the nuclear applications. The methodologies used to determine the interaction probability or cross sections can differ depending on the type of incident neutrons, such as slow or fast neutrons. The reason for this shift in strategy is that the process of producing neutrons has changed. In nuclear reactors, the fusion-fission both processes occurred. It produces both slow and fast neutrons, which interact with the reactor's surrounding materials, reactor walls and creates various reactions depending on their energy, causing the activation of different structural materials, the effect of decay heat, gas productions and radiation damage. To meet the massive energy demand worldwide, current fission reactors and future fusion reactors must be more reliable and efficient. Scientists and engineers are working hard in developing new ideas and concepts to improve dependability and reduce the waste output. An accurate nuclear reactions database, especially of neutron interactions with complete experimental information and detailed uncertainty quantification, is needed by the evaluators. This information is an essential piece, which is used to build up the evaluated nuclear data files that the user directly uses in their simulation code of any nuclear physics application and to understand the different types of neutron interactions related to the reactor operation and the nuclear fuel cycle. This neutron interaction data is one of the primary requirement for these new reactor systems. In the existing literature, so many neutron cross sections measurements had been carried out; however, the specific cross sections with their complete experimental information and detailed uncertainty quantification of the cross sections data, including the covariance (or correlation) matrix, were missing and in some cases, for a particular neutron induced reaction the cross section data is missing in the fast neutron energy region and there are differences between the different measurements for the specific activation cross section values in the existing database.

This doctoral thesis work has been constructed by following the above information and requirements in mind, and the thesis comprises the quantitative result of nuclear physics investigations that includes the fast neutron induced reactions (n,γ), (n,p), (n,α), and $(n,2n)$ cross section for the different elements (Sodium, Potassium, Chromium, Iron, Copper, Zinc, Molybdenum, Iodine, and Lead) having different isotopes used in reactor building materials. These quantitative results are useful for practical applications such as accelerator-driven sub-critical systems (ADSs), reactors, astrophysics, radiation therapy, nuclear medicine, radiation safety, waste management and environmental research, materials analysis, and so on. This nuclear data is also required to validate the various statistical model codes and sensitivity of result on the different sets of the phenomenological parameters used for predicting the cross section.

The charged-particle accelerator based neutron sources were used in the present exper-

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imental thesis work. The available fast neutron production facilities in India have been used to perform the experiments. For the low-energy neutrons (i.e., 0.6 to 3.2 MeV range), FOTIA facility at BARC in Mumbai was used, where the proton beam was accelerated and bombarded on a natural metallic ^{7}Li$-$target, creating the neutrons through the reaction ^{7}Li(p,n)^{7}Be and the D$-$T fusion reaction based neutron generator of the PURNIMA facility at BARC, Mumbai was employed to produce 14.92 ± 0.02 MeV energy neutrons.

The cross section measurement has been carried using the neutron activation technique followed by the offline γ-ray spectroscopy. For the gamma-ray spectroscopy, we have used the latest nuclear decay data properties of the residue product retrieved from the ENSDF library and the γ-ray activity produced from the residue product were detected with the precalibrated lead-shielded HPGe detector. The cross sections were measured relative to ^{115}In($n,n'\gamma$)^{115m}In and ^{27}Al(n,α)^{24}Na monitor reactions cross section. IRDFF-1.05 standard monitor cross section library was used to retrieve the cross sections for these monitor reactions. For the low energy neutron range, ^{115}In($n,n'\gamma$)^{115m}In reaction was used as standard monitor and at 14.92 ± 0.02 MeV neutron energy, the ^{27}Al(n,α)^{24}Na reaction was used for monitor.

All the necessary corrections related to the geometric correction factor, coincidence summing effect, gamma-ray self attenuation, and low-energy background neutrons have been taken care of in the measurements to improve the precision and accuracy of the cross sections result. Also, the detailed uncertainty propagation of the measured cross sections was carried out for the first time in all the reactions studied in the thesis work. Besides these, detail covariance analysis were also performed for the first time in all these reactions.

Also, the cross section for all the nuclear reactions had been reproduced theoretically using the nuclear reaction model codes EMPIRE and TALYS. Besides calculating the cross sections, we have also studied the impacts of different input parameters such as nuclear level density parameters, optical model potentials, and gamma strength functions, etc. on the cross sections results. Moreover, we have also studied the contribution from the different reaction mechanism (compound nucleus, direct reaction, and pre-equilibrium emission) in the different reaction production cross sections which varies with the incident neutron energy [1, 2]. The measured cross sections have been compared with the literature data reported in the IAEA-EXFOR database, theoretically predicted result from EMPIRE and TALYS codes, and the different evaluated nuclear data (ENDF, TENDL, JENDL, JEFF, IRDFF) library. The results of this thesis work generate a new and extended nuclear database (8 reactions for the 4 elements have been studied and 13 data points, in total generated experimentally in this thesis work), and all the experimental data were also compiled in the IAEA-EXFOR database (EXFOR Id: 33150 & 33154 & 33162 & 33163) [3–6].

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