Study of nuclear reactions for structural materials and investigation of reactor shielding

Bhargav K. Soni[1],* S. K. Mukherjee[1]
[1]Department of Physics, Faculty of Science,
The M. S. University of Baroda, Vadodara 390002, Gujarat, INDIA
* email: sonibhargav15@gmail.com

Introduction

Developing and emerging nations increasingly demand cleanly generated electricity. A nuclear reactor is one of the most efficient and affordable ways to generate clean and low-carbon energy. The only disadvantage is that it creates radioactive waste. The dilemma of radioactive waste can be solved by converting waste into stable nuclei through a high neutron flux accelerator. Recently, various research and development groups have been actively participating in the development of advanced reactor systems, such as ADSs and ITER, to meet the demand for clean energy [1]. Their purpose is to build more advanced fission and fusion reactors with upgraded safety features and economic enhanced resource use with a minimal amount. Apart from the upcoming upgrade of the functioning fission power plants, fusion devices, and accelerators, the concerned codes must require a wide range of nuclear data analogues to cross-section and decay properties for all the materials of interest in the device. Therefore, the section of the work targeted the data based on nuclear cross-section for a structural material for both fission and fusion devices. Due to the efficiency of radiation hardness and long durability, structural materials are an important segment of any nuclear reactor [2]. At present, the data associated with the structural materials is insufficient and has large discrepancies. Hence, an exhaustive and convenient cross-section data library is needed.

Since the beginning of the use of nuclear reactors, concrete has been widely used for the shielding of ionizing radiation. Due to its versatile properties, like attenuation (changes with chemical composition), low cost of fabrication, and easy casting with good structural and mechanical properties. Various authors have done extensive work by adding appropriate admixtures to the concrete [3]. This study is mainly associated with the γ-rays and neutron shielding parameters of ordinary cement having low-Z and high-Z additives in it. The proper knowledge must be required in the way of improving reactor shielding. The investigation was conducted by considering both experimental and simulation techniques [4]. All these efforts can only succeed if we have plentiful and explicit knowledge about nuclear reaction data for various materials applicable in reactors and their shielding materials.

Therefore, the present work is confined to nuclear reaction cross-section data for the reactor structural materials. Aside from this, the improvement in reactor shielding materials has been parallelly examined on concrete samples prepared using different amounts of supplements.

Computational Codes

The theoretical code TALYS-1.95 has been used in the investigation of neutron-induced reaction cross-section measurements with pre-defined level density parameters. For the present work, these predefined level density parameters were also adjusted to get a better theoretical prediction. Apart from this, the examination of γ-rays and neutron shielding parameters was performed with the MCNP, WinXCom, Auto-Z_{eff}, and NXcom software. Among both, the γ-ray attenuation coefficient was determined using the MCNP simulations and WinXCom software, which was further used in the calculation of other shielding quantities such as effective atomic number (Z_{eff}), electron density (N_{eff}), half-value layers (HVL), tenth value layers (TVL), and mean free path (MFP). The effective atomic number and electron density were determined using the Auto-Z_{eff} software in comparison with the data from a direct method. The software NXcom, together with the MCNP simulations, was used to study the fast neutron removal cross-section. Analysis of the present work was performed with the latest version of each software.

Reaction cross-section measurement

In the measurements of neutron induced cross-sections of ^{159}Tb(n, γ)^{160}Tb [5], ^{113}In(n, n′)^{113m}In, ^{115}In(n, 2n)^{114m}In, ^{115}In(n, n′)^{115m}In [6] reactions using quasi-mono-energetic neutrons generated by ^{nat}Li(p, n) reaction. While the ^{181}Ta(n, 2n)^{180}Ta [7] reaction cross-section measured at 14.78 MeV neutron energy generated by ^{3}H(^{2}H, n)^{4}He reaction.

The experiments were carried out at the 14UD Bhabha Atomic Research Center-Tata Institute of Fundamental Research (BARC-TIFR) Pelletron accelerator at TIFR and the Purnima Accelerator at BARC's Nuclear Physics Division in Mumbai, India. The measurements were carried out using the neutron activation technique, which was followed by off-line γ-ray spectroscopy. The reaction cross-sections have been analyzed using the neutron activation formula along with the uncertainty analysis by adopting the ratio measurement technique. The TALYS code was used with the ENDF/B-VII.1, JENDL-4.0, and JEFF-3.3 evaluated data libraries to compare the present measurements. The outcomes of the neutron-induced reaction cross-sections are significantly important for reactor structural materials and the advancement of reactor technology. The results also show that the relative measurement of reaction cross-sections up to 20 MeV incident proton energies by applying the low-energy tailing correction.

Improvement of Shielding Materials

At a working stage, a reactor generates highly penetrating γ-rays and neutrons from fission and subsequent radioactive decay. Which have been absorbed and deflected using suitable shielding materials such as concrete, lead, WC, B4C, etc. Among all the concrete mixture contains low-Z as well as high-Z constituents that can enhance shielding from both γ-rays and neutrons. With this perception, the concrete has been prepared with different amounts of Tungsten Carbide (WC) and Boron Carbide (B_4C) additives. Overall, twenty-one concrete samples were prepared in a set of 1 cm, 2 cm, and 3 cm average thickness, each set containing seven distinct WC and B_4C ratios in the concrete mixture, along with ordinary concrete (OC). These prepared samples were experimentally analyzed through γ-ray (^{60}Co) and neutron (^{252}Cf) sources at the Defence Laboratory Jodhpur, Rajasthan, India. The theoretical prediction codes XCOM, MCNP, Auto-Z_{eff}, and NXcom were used to compare the present findings [8]. In investigations shielding parameters such as; μ_m, Z_{eff}, N_{eff}, HVL, TVL, MFP, and Σ_R were calculated. The outcomes of the work show that the modified compositions are far superior as compared to the pristine concrete. The results highlight that shielding parameters strongly rely on the atomic composition and density of additives in the prepared concrete.

Summary and conclusion

The excitation functions for the ^{159}Tb, ^{113,115}In, and ^{181}Ta isotopes were measured for the incident quasi-monoenergetic neutron energies in the range of 5 to 20 MeV (varying differently for different elements). The present measurement shows that the uncertainties contributed by the monitor reaction data are important in the reaction uncertainty calculations. It suggests that for the isotopes related to structural material, more experimental data through nuclear model comparisons is needed. The uncertainties in the experimentally measured cross-sections were estimated with covariance analysis and were found to be below 19%. Whereas, the investigation of WC and B_4C additives in ordinary concrete for γ-rays and neutron shielding has been performed through parameters μ_m, Z_{eff}, N_{eff}, HVL, TVL, MFP, and Σ_R. 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.

References

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