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Analysis of kinetic parameters of 3 MW TRIGA Mark-II research reactor using the SRAC2006 code system

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ABSTRACT

The aim of this study is to evaluate the kinetic parameters of 3 MW TRIGA Mark-II research reactor at AERE, Savar, Dhaka, Bangladesh from the viewpoint of reactor safety. The most important kinetic parameters of nuclear reactors are the effective delayed neutron fraction (β_{eff}), the effective decay constant for *i*th family of delayed neutron precursor ($\lambda_{eff,i}$), the prompt neutron lifetime (l_p) and the mean neutron generation time (Λ). These parameters are calculated using the 3-D diffusion code SRAC-CITATION of the comprehensive neutronics calculation code system SRAC2006 based on the evaluated nuclear data libraries JENDL-3.3 and ENDF/B-VII.0 in both cases. The calculated results of reactor kinetic parameters are compared to the available safety analysis report (SAR) values of 3 MW TRIGA Mark-II reactor by General Atomic as well as the MCNP5 values (numerically benchmark) based on the evaluated nuclear data library ENDL/B-VII.0. It was found that in most cases, the calculated results of kinetic parameters demonstrate a good agreement between the JENDL-3.3 and the ENDF/B-VII libraries as well as the SAR and the MCNP5 values respectively. Therefore, this study will be essential to improve the basic nuclear data of reactor kinetic parameters for safe operation of 3 MW TRIGA Mark-II research reactor.

1. Introduction

Generally, neutrons are emitted through fissions. Over 99% of neutrons are emitted promptly at fissions and the others are emitted by decay of precursors in a few tens of seconds after the fissions. The former are called "prompt neutrons" and the latter are "delayed neutrons". Reactivity of the delayed neutron is called "effective delayed neutron fraction", β_{eff}. Average time from birth of a neutron until it induces fission is called "neutron generation time, Λ. Both β_{eff} and Λ are called kinetic parameters and they can determine time dependent behavior of reactor power after reactivity insertion.

In reactor kinetics, the most important kinetics parameters of nuclear reactors are the effective delayed neutron fraction (β_{eff}), the effective decay constant for *i*th family of delayed neutron precursor ($\lambda_{eff,i}$), the prompt neutron lifetime (l_p) and the mean neutron generation time (Λ), because they are used to assess transient/accident behavior of a nuclear reactor. In research reactors (like TRIGA) these parameters are usually provided by the manufacturer in the design phase and are not calculated for various core conditions. For example, the recommended value of β_{eff} for the 3 MW TRIGA Mark-II research reactor by General Atomic (Safety Analysis Report, 1981) using 20 wt.% low enriched uranium (LEU) fuel and featuring graphite reflector is 0.007 and the recommended the prompt neutron lifetime (l_p) is 30 μs. The reactor kinetic parameters strongly depend on the fuel type (enrichment of ^{235}U) and core configuration (size) and these parameters are very important in nuclear power plant operation due to determination of different kinds of safety measures.

It is very difficult to measure β_{eff} and Λ separately (only their ratio, β_{eff}/Λ, can easily be measured), hence these parameters are usually determined only by calculation. In 1965 Keepin provided the theoretical foundations for such calculations (Keepin, 1965), however, related to deterministic transport equation. His approach requires calculation of flux and its adjoint function as weighting function and this method is very difficult to apply as it requires detailed multi-group transport calculations. In this study, the most important kinetic parameters of 3 MW TRIGA Mark-II research reactor are calculated based on the evaluated nuclear data libraries JENDL-3.3 (Shibata et al., 2002) and ENDF/B-VII (Chadwick et al., 2006) using the deterministic 3-D diffusion code SRAC-CITATION (Fowler et al., 1969). All of the fuel rods (each containing 20 wt.% of uranium) are 19.7% enriched of ^{235}U and fresh (zero burn-up). In addition, the differences among the calculated values of β_{eff}, $\lambda_{eff,i}$, l_p and Λ are investigated only for different libraries JENDL-3.3 and ENDF/B-VII.0.

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2. Calculation techniques/tools

2.1. Reactor simulation codes

Two reactor engineering codes were used to perform this analysis and these are (i) the collision probability method lattice transport code SRAC-PIJ (Beardwood, 1966) was used for the generation of group constants or cross-sections data sets for various core regions of TRIGA reactor and (ii) the SRAC-CITATION code (Fowler et al., 1969) was used to perform global core calculations of 3 MW TRIGA Mark-II reactor. The calculation scheme of the SRAC2006 code system is shown in Fig. 1.

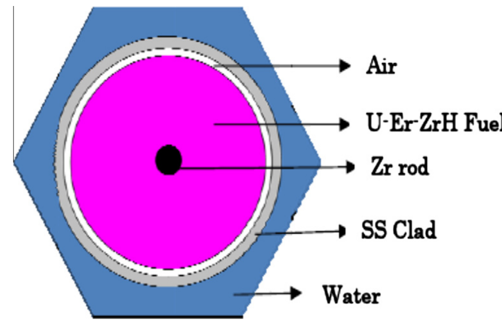

Fig. 2. Hexagonal cell model of TRIGA fuel.

2.2. Simulation methodology

The simulation methodology was developed in two steps (i) the SRAC-PIJ model and (ii) the SRAC-CITATION model. In case of SRAC-PIJ model, cell calculation of the LEU (Low Enriched Uranium) fuel rod was performed by the lattice physics transport code SRAC-PIJ with 1-D hexagonal model (Fig. 2) using geometry type IGT = 6 (hexagonal cell). For simulating actual spectrum, fuel cell was divided into several annular rings. Same model was also used for fuel part of the control rod. The cross section obtained from this model was used in SRAC-CITATION to model the 3-D TRIGA fresh core. Graphite dummy element, control rod element, pneumatic transfer tube, transient rod with boron carbide part and the transient rod with air follower part were modeled with the geometry type IGT = 12 (hexagonal assembly with asymmetric pin rods) as shown in Fig. 3. In this model the rod of interest is in the center surrounded by six fuel rods. This model was employed to simulate actual situation in the TRIGA LEU core so that reliable cross section data could be generated.

Central thimble (CT) cell was modeled using geometry option 12 where the central rod is CT surrounded by six graphite dummy elements and twelve fuel rods in the first and second layer of the model respectively to simulate the surrounding of the CT in the TRIGA core (Fig. 4). The effective macroscopic cross section for the B-ring used in the global calculation was obtained from the CT cell model. PEACO routine (Ishiguro and Takano, 1971) was selected for the calculation of effective resonance cross section in the resonance energy range. PEACO solves a multi-region cell problem by the collision probability method using an almost continuous (hyper-fine) energy group structure for the resonance energy range. The interaction of resonances can be accurately treated by the PEACO routine.

For the material outside the active core region such as top and bottom fitting, grid plate, top and bottom reflector, lazy Susan, graphite blanket and lead shielding, asymptotically collapsed cross sections were used in the CITATION.

The SRAC-PIJ code uses the 107 energy group (73 fast and 34 thermal) for both nuclear data files JENDL-3.3 and ENDF/B-VII.0 respectively. The fast (73) group is divided into 4 energy groups and the thermal (34) group is also divided into 3 energy groups. The total (107) energy group is condensed into 7 energy groups. All calculations were performed in seven energy groups shown in Table 1. Special consideration was made in development of energy group structure. First group was based on above threshold fission of ^{238}U and no delayed production. Second group was based on the average energy of delayed neutron group produced. Group 1 and 2 were considered as fast energy flux. Group 3 and 4 are considered as epithermal group. Groups 5, 6 and 7 were considered as

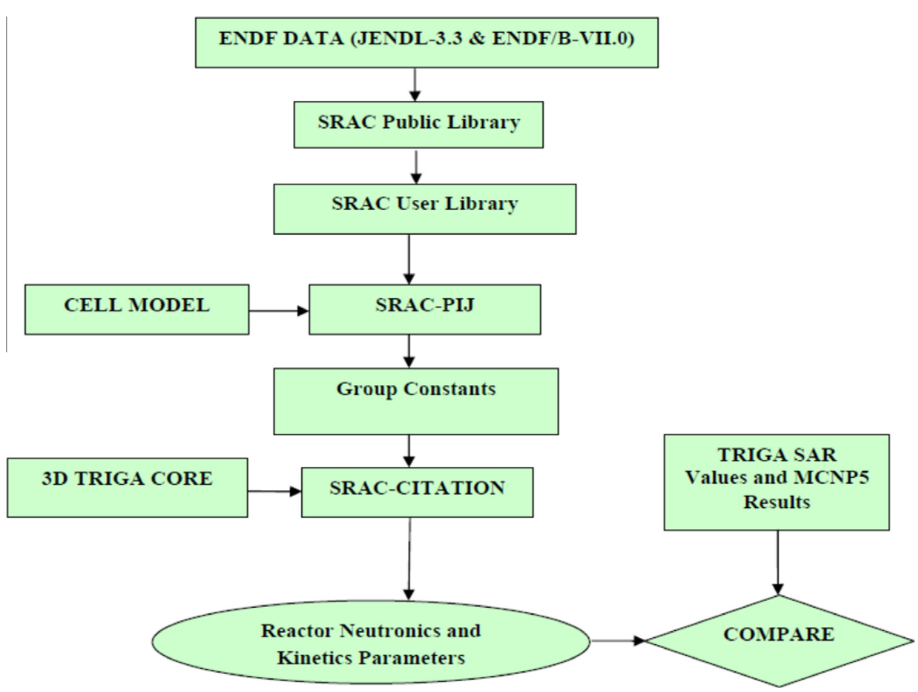

Fig. 1. Calculation flow chart of kinetics parameters of TRIGA Mark-II research reactor.