Development of Versatile Methods to Estimate Kinetic Parameters for Various Kinds of Reactor

Background

In nuclear reactors, neutrons are emitted via fissions. Over 99% of neutrons are radiated 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 since they determine time dependent behavior of reactor power after reactivity insertion. In commercial light water reactors (LWRs) where UOX fuels are loaded, estimated values of β_{eff} and Λ are used to measure control rod worth.

β_{eff} and Λ of a core are conventionally calculated employing adjoint flux as weight function for yield of delayed neutrons and for lifetime of all neutrons. Accordingly, we need to solve neutron transport and adjoint equations to obtain β_{eff} and Λ. For this purpose, specific multi group cross section libraries must be made based on point-wised cross section data set and some kinds of approximation are introduced to solve the equations in many cases. For a commercial reactor, applicability of the libraries made for assemblies of ^{235}U fuels are verified in conjunction with approximations utilized to solve the equations. However, they should be verified again for cores where MOX fuel assemblies are partially loaded because a specific multi group cross section library must be made for assemblies of each isotopic composition and rigorous treatment of neutron transport is required to predict fission density distribution that varies steeply around interface between the MOX and UOX assemblies. For such cores in which new kinds of fuel are loaded, verification of specific multi group cross section libraries and of approximations will be a cumbersome task.

Objectives

To develop estimation method of the kinetic parameters for various kinds of core without making specific multi group cross section libraries and any approximation.

Principal Results

1. Proposal of Estimation of Kinetic Parameters with Number of Fission Neutrons in Next Generation

Here we define M as expected number of fission neutrons in next generation produced by a neutron in current generation. The number of fission neutrons in next generation produced by delayed ones, N_d, is evaluated by integration of energy - angle differential delayed neutron emission density multiplied by the M. We can also evaluate total number of fission neutrons in next generation, S_{next}. We propose to estimate β_{eff} by ratio N_d/S_{next} as shown in Fig.1. M is used as the weight function to the neutron yield density instead of adjoint flux used in conventional methods. In the same manner, we propose to estimate Λ by using M as the weight function. In the proposed methods, the specific cross sections are not necessary since we do not solve the adjoint equations.

2. Implementation of Proposed Method in Continuous Energy Monte Carlo Transport Calculation Code

Continuous energy Monte Carlo method is known to solve neutron transport equation precisely for any kind of core with point-wised cross section data set. We implement functions to calculate the kinetic parameters based on our proposed methods into MCNP-4C code, which is one of the most popular Monte Carlo calculation codes for neutron transport. By the implementation, the parameters can be estimated directly without any approximation.

3. Verification of Proposed Method

With the enhanced MCNP-4C code, we estimated kinetic parameters of critical cores of light water moderated UOX and MOX fuel lattices, fast assemblies and U nitrate solution. In Fig.2 - 3, the estimated β_{eff} and Λ are compared with those measured in room temperature. In the figures, the estimated kinetic parameters based on the proposed methods are also compared to those calculated by conventional methods with a specific multi group cross section library elaborated for each core through experimenters. The comparison indicates that kinetic parameters are predicted by the proposed methods in comparable accuracy with that done by the conventional methods.

Future Developments

Point-wised cross section data set will be prepared for operation temperatures of LWRs. With the data set, kinetic parameters will be estimated for those cores for the control rod worth measurements, etc.

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Reference

Y. Nauchi and T Kameyama, “Proposal of Direct Calculation of Kinetic Parameters β_{eff} and Λ based on Continuous Energy Monte Carlo Method”, Journal of Nuclear Science and Engineering, 42(6), 503-p514, 2005.

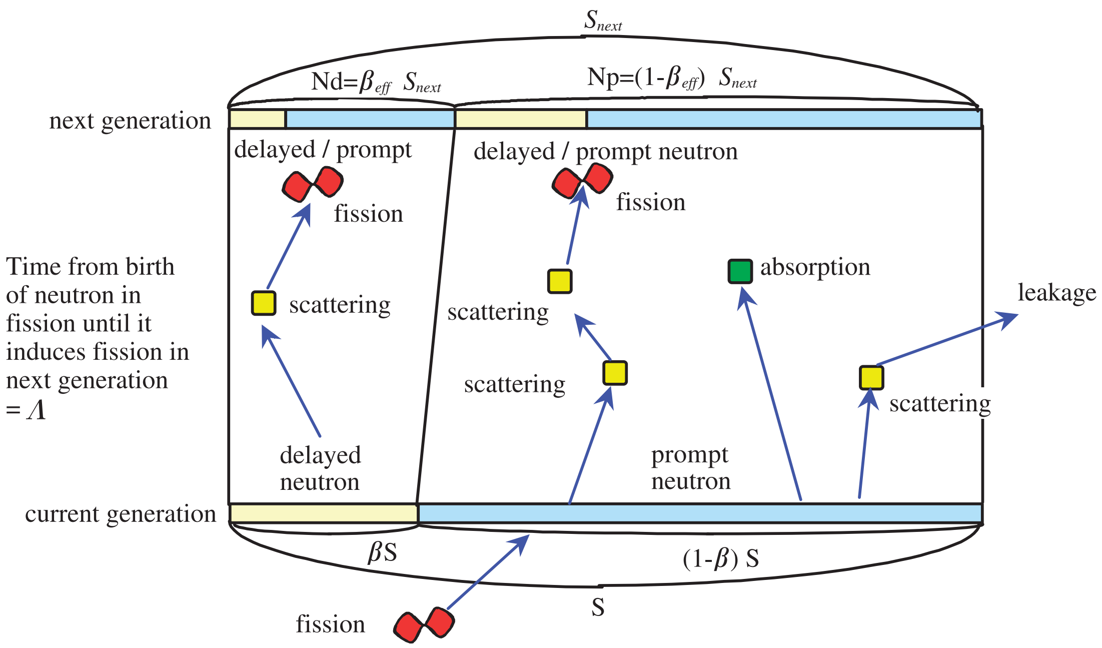

Fig.1 Schematic view of proposed estimation methods of kinetic parameters

In eigenvalue calculations with MCNP code, effective multiplication factor k_{eff} is estimated by the ratio of neutron yield of next generation, S_{next}, to that of current, S: $k_{eff}=S_{next}/S$. In the proposed method, β_{eff} is estimated as N_d/S_{next}, where N_d is the number of fission neutrons in next generation produced by delayed neutrons. Whereas, Λ is estimated by averaging lifetime of neutrons in current generation multiplied by the number of fission neutrons in next generation produced by the neutrons in current generation.

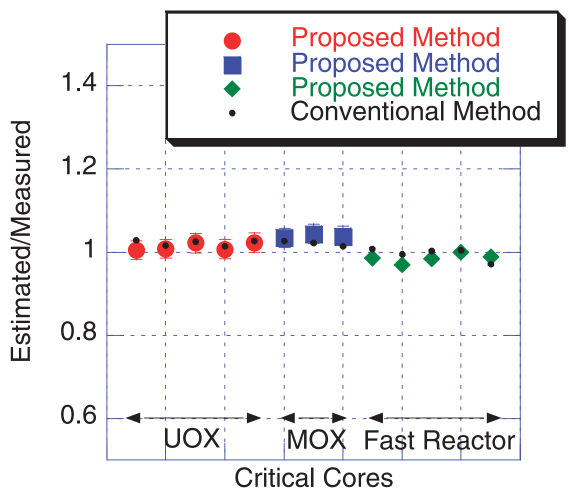

Fig.2 Comparison of prediction accuracy of β_{eff} by proposed method to that by conventional method for light water moderated UOX and MOX fuel lattices and fast reactor cores

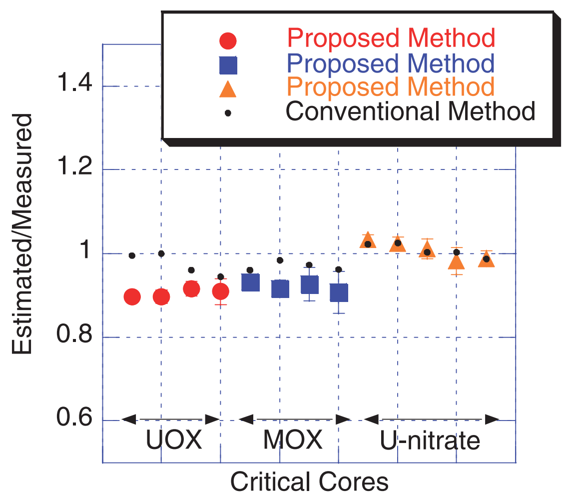

Fig.3 Comparison of prediction accuracy of β_{eff}/Λ by proposed method and to that by conventional method for light water moderated UOX and MOX fuel lattices and U-nitrate cores