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Kinetics solution with iteration scheme of history-based neutron source in accelerator driven system

Song Hyun Kim *, Masao Yamanaka, Cheol Ho Pyeon

Nuclear Engineering Science Division, Research Reactor Institute, Kyoto University, Asashiro-nishi, Kumatori-cho, Sennan-gun, Osaka 590-0494, Japan

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ABSTRACT

In the time-dependent neutronic analyses of the accelerator driven system (ADS), one of the technical issues is the investigation method of reactor transients for validating why detector responses or fission rates are varied in functions of space and time. In previous studies, numerical analyses for ADS have been conducted by using forward transport methods, focusing on reconstruction of the time behavior. In this study, a source history-based kinetics equation, which is superior to cause analysis of time transients comparing with general forward transport methods, is derived by using the time contributions from each source into responses. For the validation of the applicability, through the re-evaluation of ADS experiment at the Kyoto University Critical Assembly, the time transients of fission rates and detector responses are quantitatively analyzed with the time contributions directly obtained by the proposed method. The expanded analysis ability provided by the proposed method will contribute to investigate the reactor transient of ADS including verification of reactor experiments, analysis of space-time detector responses and establishment of estimation strategy for obtaining the kinetics parameters.

1. Introduction

The accelerator-driven system (ADS) is a nuclear reactor system that uses an accelerator as external neutron source. ADS maintains a subcritical state with by the external source, thus strengthening safety against reactivity accidents. It is also used for the transmutation of radioactive waste, minor actinides and long-lived fission products. For analyzing safety of ADS, as well as establishing a reactor operation procedure, reactor transient, which is a time behavior of the neutron population in a reactor core and an analysis field to safely control the reactor power, needs to be analyzed by numerical simulation and experiments.

For the experimental analyses of reactor time transients, the kinetics parameters have been obtained from detector signals obtained in correspondence of particular positions in a reactor core. To conduct pulsed neutron source (PNS) experiments, pulsed neutrons are injected into a reactor, and detector signals are obtained in function of time. PNS methods (Persson et al., 2008; Pyeon et al., 2008, 2009; Yamanaka et al., 2016) play an important role to obtain the kinetics parameters in ADS. The area ratio method (Gozani, 1962), the Feynman-alpha method (or variance-to-mean) (Feynman et al., 1956) and the Rossi-alpha method (Kitamura et al., 2005; Taninaka et al., 2011) have been also used for obtaining the statics or kinetics parameters from PNS experiments. An important issue of PNS experimental analyses is whether or not the signals obtained from detectors located in some core particular positions can represent or less the time transient behavior of the whole core.

For space-time transient analyses in PNS experiments, the Monte Carlo simulations using the MCNP™ code were conducted in a previous study (Pyeon et al., 2008). The MCNP™ results showed that the delayed neutron precursors were not properly accumulated, and some biases of the detector responses in the MCNP™ simulations were found in comparison with the experimental results. A methodology to take into account the accumulation of delayed neutrons for the MCNP™ simulation was proposed and verified by Talamo et al. (2009). On the basis of this methodology, Bell and Glasstone spatial correction factors (Bell and Glasstone, 1970), which is a method to correct measured reactivity obtained from detector responses depending on their positions, were evaluated by the space-time transport analyses using the MCNP™ code (Talamo et al., 2012a). Also, the deterministic code PARTISN was adopted to obtain the correction factors by space-time reactor analyses (Talamo et al., 2012b). To investigate the position dependent transients analysis, kinetics methods and their applications were reported in previous studies (Gabrielli et al., 2016; Prince et al., 2016; Ganapol et al., 2016; Ott and Meneley, 1969;

* Corresponding author.
E-mail address: kim@rri.kyoto-u.ac.jp (S.H. Kim).

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Devooght and Mund, 1980; Dahmani et al., 2001; Kobayashi, 1992; Ravetto et al., 2004; Ban et al., 2012), and reconstructions of space-time transients have been performed by the kinetics methods. The kinetics simulations in previous studies are conventionally based on the forward transport simulations. Such approaches have some limitations for thorough analyses of the ADS kinetics because they do not directly provide any information about the causes of the time dependent detector responses.

This study aims at developing a reactor kinetics method allowing to provide in-depth analyses of reactor transients. First, neutron sources are classified into each birth history (prompt, delayed, and external sources) having an equivalent neutron energy distribution. A source iteration scheme is proposed by using time-dependent transfer functions for connecting the neutron sources and responses. After the re-evaluation of the experimental benchmark at the Kyoto University Critical Assembly (KUCA), detailed analyses of the time transients for fission sources and detector response are performed to validate the applicability for the analysis of reactor transients.

2. Methods

Generally, the time-dependent neutron transport equation is based on neutron flux depending on space, energy, angle and time as follows:

$$\frac{1}{v}\frac{\partial\varphi}{\partial t}+\hat{\Omega}\cdot\nabla\varphi+\sum_t\varphi = \int\int\sum_s(\underline{r},\hat{\Omega}',E'\rightarrow\hat{\Omega},E)\varphi(\underline{r},\hat{\Omega}',E',t)d\hat{\Omega}'dE'+S(\underline{r},\hat{\Omega},E,t) \quad (1)$$

where v is neutron velocity, φ neutron angular flux, t time, $\hat{\Omega}$ solid angle, Σ_t total macroscopic cross section, $\Sigma_s(\underline{r},\hat{\Omega}',E'\rightarrow\hat{\Omega},E)$ macroscopic scattering cross section from $(\hat{\Omega}',E')$ to $(\hat{\Omega},E)$ at location $\underline{r}$, and $S(\underline{r},\hat{\Omega},E,t)$ neutron source at $(\underline{r},\hat{\Omega},E,t)$. The Monte Carlo method uses the Eq. (1) directly, and kinetics methods are based on the forward transport equation. In this study, to analyze the time-transient behavior in detail, neutron sources are only used

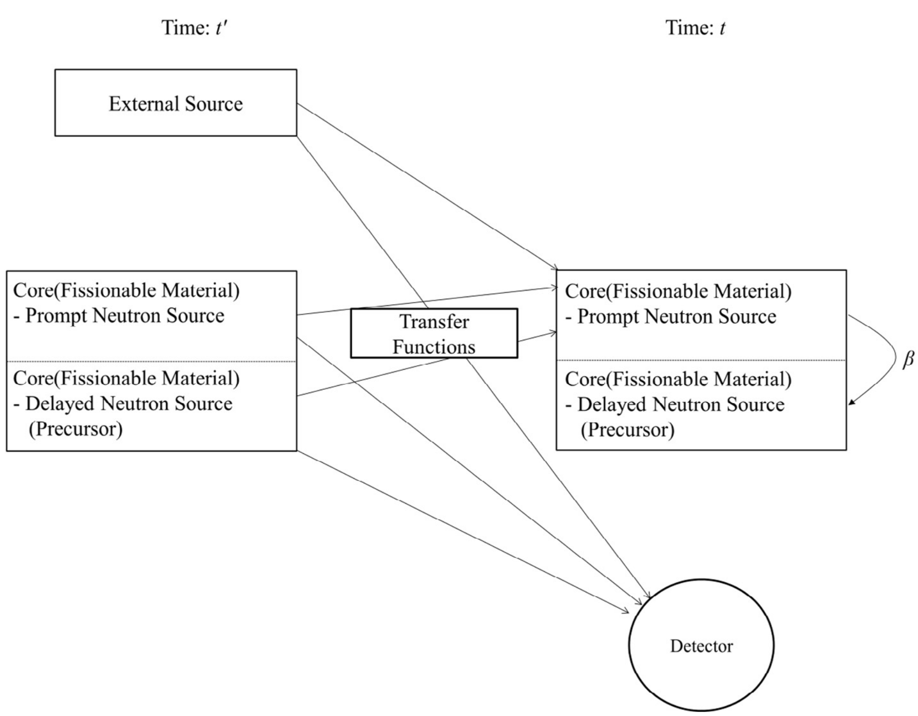

Fig. 1. Conceptual overview of the estimation scheme of neutron sources and detector response at time t with transfer function.

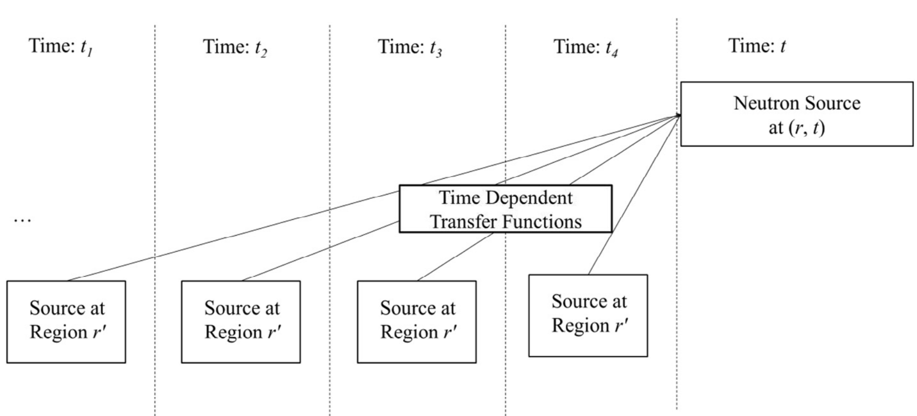

Fig. 2. Overview of time integration of neutron source with time-dependent transfer function.