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The application of transient analysis code to estimation of total released energy in a two-fuel-solution-tank system

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ABSTRACT

A calculation code was developed for transient analysis of the total released energy in a criticality accident with two fuel-solution tanks. The calculation method was confirmed to be effective for transient analysis in a fuel-solution-tank system. The verification of the code was performed by comparing the calculation results with those of TRACY experiments. In the code, neutronic coupling between the tanks is treated. Using this code, transient analyses were performed for a two-tank system using different feedback models. The analyses confirmed that the total energy released in criticality accident with two fuel-solution tanks can be estimated using the knowledge of the total energy released in a single-fuel-solution-tank system.

1. Introduction

In nuclear fuel fabrication facilities and reprocessing facilities, various types of nuclear fuels are processed, including powders, solids and solutions. Historically, most of the criticality accidents that have occurred have involved fuel solutions (McLaughlin et al., 2000). Criticality accidents in the case of fuel solutions have unique behaviors in terms of the feedback mechanism (Yamane, 2012). It is important to analyze the kinetic behavior of fuel solutions in such circumstances to evaluate radiation doses and establish safety measures for criticality accidents.

Experiments and analysis on criticality accidents were performed using the Transient Experiment Critical Facility (TRACY) and Static Experiment Critical Facility (STACY) of the Japan Atomic Energy Agency (JAEA). To obtain data for fuel-solution criticality accidents, factors such as transient changes of power, production of radiolytic dissociation gas voids, reactivity feedback effects and other reactions were investigated at TRACY. The code for analyzing the transient behavior including total released energy for a single-tank was previously developed by JAEA (Nakajima et al., 2002). The criticality volume of the fuel-solution, prompt neutron lifetime and data about criticality safety in reprocessing facilities have also been obtained using STACY; specifically, criticality experiments for coupled systems in static condition have been performed. The reactivity effect of the distance between the tanks and neutron absorbers placed between tanks was estimated by these previous experiments. However, those experiments and analyses were performed for static conditions in a coupled system and also for transient conditions in a single-fuel-solution-tank system; the total released energy in transient condition with several-fuel-solution-tank has not yet been analyzed.

In general, kinetic behavior and total released energy depend on the geometry and distance between the tanks even if the total amount of the fissile material is the same. So the conventional point kinetic method cannot be applied to analyze such systems.

During criticality accidents of fuel solutions, energy is released during the first pulse of power, oscillation of power after the pulse and stable conditions with constant power which depends on the solution temperature. When we think about the dose from an accident, the dose from the first pulse is important because it is expected that the workers are evacuated immediately after the accident. So in this study, the total released energy by the first pulse is going to analyzed.

In a criticality accident in a fuel-solution tank with a other fuel-solution tanks, the total released energy may be higher than that in an isolated single-tank system because of the neutronic coupling between the fuel-solution tanks (Nuclear Criticality Safety Handbook of Japan, 1995). This means that the total released energy may be underestimated if the neutronic coupling between the tanks is ignored. It was shown that an integral kinetic model

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based on space-dependent kinetic analysis was useful for the analysis of a weakly coupled system (Takezawa and Obara, 2010, 2012).

In the previous work, a kinetic analysis method for a reactor pumped laser experiment was developed (Takezawa and Obara, 2010, 2012). In that work, the calculation results by a newly developed code were verified and validated for solid-fuel cores. It is expected that this method can be applied to the analysis of the kinetic behavior of a two-fuel-solution-tank system by introducing proper feedback models for the solutions.

In criticality accidents of fuel solutions, the first pulse is found in the super prompt critical condition. So the kinetic analyses for prompt neutrons are important to analyze the first pulse. Also to predict the energy during the first pulse has important meaning, because the energy released by the first pulse is dominant in total released energy in step-wise reactivity insertion (Nomura et al., 2004).

The purpose of this study was to apply an integral kinetic model to kinetic analysis of the supercritical condition in a system with two-fuel-solution-tank and to develop a method to estimate the total released energy based on previous estimate of the energy released in the case of a single-fuel-solution-tank system. In this study, the analysis code for transient analysis of a weakly coupled system in a criticality accident is modified and the results are verified. Analyses are performed for the first pulse by prompt neutrons. The aim to estimate the released energy is to estimate the dose by the first pulse. Hence, the accuracy of the estimation shall be within one or two digits.

2. Calculation method

2.1. Integral kinetic model for two solution tanks

In this analysis, a kinetic analysis method based on an integral neutron transport equation developed for a weakly coupled system was used (Takezawa and Obara, 2010, 2012). The method is effective for time-dependent kinetic analysis of weakly coupled systems because of its applicability to the flexible system geometry. In this study, criticality accidents with two fuel-solution tanks were considered.

In the method, the time- and space-dependent fission rate function in each region is obtained in advance by integration of fissions in the region caused by the fissions in all regions from the past to the present. The general theory is described elsewhere (Takezawa and Obara, 2010, 2012). For the analysis of the first pulse, the fission rates with no contribution from delayed neutrons in a two fuel-solution-tank system are shown below:

$$N_1(t) \approx \sum_{j=1}^{2} \int_{t-t_c}^{t} \alpha_{1j}(t-t')N_j(t')dt' \tag{1}$$

$$N_2(t) \approx \sum_{j=1}^{2} \int_{t-t_c}^{t} \alpha_{2j}(t-t')N_j(t')dt' \tag{2}$$

where, for example, $N_1(t)$ is the fission rate at time t in fuel-solution tank 1, and the expression $\alpha_{12}(\tau)$ represents the secondary prompt-fission probability rate function in fuel-solution tank 1 provided by the first fission in fuel-solution tank 2 with time difference τ. t_c is the time when fissions have no effect on the fissions at time t from the viewpoint of prompt neutron lifetime, which is 2×10^{-2} [s] throughout the analysis. $\alpha_{ij}(\tau)$ can be calculated based on the fission source distribution, which can be obtained using the Monte Carlo neutron transport method (Kukharchuk and Gulevich, 2000).

The concept of function $\alpha_{ij}(\tau)$ is shown in Fig. 1. For example, $\alpha_{12}(\tau)$ indicates the neutronic coupling from tank 2 to tank 1 as shown in the figure. The four functions $\alpha_{ij}(\tau)$ were used in the analysis of a two-tank system.

The fission rate of tank i induced by fissions in tank 1 and 2 after the reactivity insertion $N_i^a(t)$ is written by the following equation:

$$N_i^a(k\Delta t) = \sum_{j=1}^{2} \int_{0}^{k\Delta t} \alpha_{ij}(k\Delta t - t')N_j(t')dt', \tag{3}$$

where k is the number of time steps, and Δt is the width of the time steps, which is 1×10^{-6} [s] throughout the analysis. It was confirmed that this width is small enough by comparison with the results of the analysis using smaller time-step width. If we assume that the system is critical before reactivity insertion, the fission rate of tank i $N_i^0(t)$, induced by fissions in tanks 1 and 2 before reactivity insertion, is written by the following equation:

$$N_i^0(k\Delta t) = \sum_{j=1}^{2} \int_{(k-k_c)\Delta t}^{0} \tilde{\alpha}_{ij}(k\Delta t - t')N_{j0}dt', \tag{4}$$

where $\tilde{\alpha}_{ij}(\tau)$ means $\alpha_{ij}(\tau)$ at critical condition, N_{j0} is the steady-state fission rate of fuel-solution in tank j at critical condition, and k_c represents the time steps that satisfy $k_c\Delta t > t_c$.

The fission rate of tank i can be obtained by the summation of $N_i^a(t)$ and $N_i^0(t)$.

$$\begin{aligned} N_i(k\Delta t) &= N_i^0(t) + N_i^a(t) \\ &= \sum_{j=1}^{2} \left\{ \int_{(k-k_c)\Delta t}^{0} \tilde{\alpha}_{ij}(k\Delta t - t')N_{j0}dt' + \int_{0}^{k\Delta t} \alpha_{ij}(k\Delta t - t')N_j(t')dt' \right\} \end{aligned} \tag{5}$$

In the practical calculation, function $C_{ij}(\tau)$, which is the ratio between the number of source fissions in fuel-solution tank j and the number of secondary fissions in fuel-solution tank i, is used for kinetic analysis instead of function $\alpha_{ij}(\tau)$. Eq. (5) can be modified as shown below:

$$N_i(k\Delta t) = \sum_{j=1}^{2} \left\{ N_{j0}[\tilde{c}_{ij}(\tau')]_{k\Delta t}^{k_c\Delta t} + \sum_{k'=0}^{k-1} N_j(k'\Delta t)\left[c_{ij}(\tau')\right]_{(k-k'-1)\Delta t}^{(k-k')\Delta t} \right\}, \tag{6}$$

where τ' is the time delay between the fission reactions in tank j and tank i, which is equal to $k\Delta t - t'$ in the analysis. The first term of the equation gives the rate of fissions caused by the fissions before reactivity insertion and the second term gives the rate of fissions caused by fissions after reactivity insertion. $C_{ij}(\tau)$ is defined as follows:

$$C_{ij}(\tau) \equiv \int_{0}^{\tau} \alpha_{ij}(\tau')d\tau'. \tag{7}$$

The function $C_{ij}(\tau)$ is calculated using the modified Monte Carlo code MVP2.0 with nuclear data library JENDL-4.0 (Takezawa and Obara, 2010, 2012).

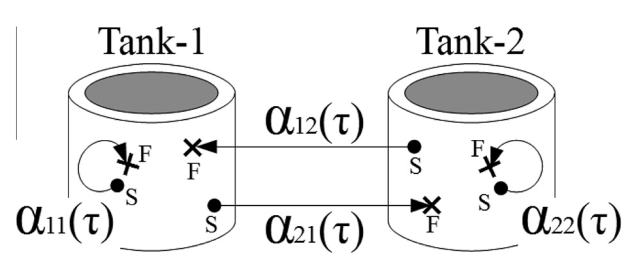

Fig. 1. Concept of α_{ij} functions (S, point of source neutron; F, point of fission reaction; τ, time from the creation of source neutron to fission reaction).