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Kinetic behavior in criticality accident of two fuel solution tanks

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ABSTRACT

This work aimed at criticality safety in fuel-fabrication or reprocessing facilities. To estimate the expected released energy in criticality accidents of several fuel-solution tanks based on the energy released in a single-fuel-solution-tank accident, the total energy released in a two-tank system was evaluated with different compositions of fuel solutions. The code used for the analysis took into account the neutronic coupling between the tanks. The results made clear that as the fuel enrichment increases, neutronic coupling becomes stronger. The calculated increase of the total energy released in a criticality accident of a system with two fuel-solution tanks compared to that of a system with just one was roughly the same regardless of different feedback model used for the analysis. The results show that the difference in energy release between multiple-tank and single-tank systems of similar composition may be evaluated using the developed transient analysis code.

1. Introduction

In nuclear fuel fabrication facilities and reprocessing facilities, various types of nuclear fuels are processed, including powders, solids and solutions. However, most actual criticality accidents have occurred with the use of fuel solutions (Yamane, 2012). Criticality accidents in 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). Fundamental data about transient changes of power, production of radiolytic dissociation gas voids, reactivity feedback effects and other reactions have previously been collected at TRACY, for the purpose of investigating fuel-solution criticality accidents (Nakajima et al., 2002a). Fundamental data about criticality safety in reprocessing facilities have also been obtained using STACY; specifically, criticality experiments for coupled systems have been performed (Tonoike et al., 2003). The reactivity effects of the distance between the tanks and neutron absorbers placed between the two tanks were estimated by these previous experiments. The AGNES code for analyzing the transient behavior including total released energy for a single-tank system has been previously developed by JAEA based on TRACY experimental results (Nakajima et al., 2002b). However, these experiments and analyses were performed for static conditions in systems of several fuel-solution tanks; thus, the total energy released in the transient condition has not yet been analyzed.

In a criticality accident in a fuel-solution tank surrounded by several other fuel-solution tanks, the total released energy becomes higher than that in an isolated single-tank system because of the neutronic coupling between the fuel solution tanks. This means that the total released energy is underestimated if the neutronic coupling between the tanks is ignored. The usefulness of the transient analysis code in weakly coupled systems has been confirmed in previous studies, and its methodology has been verified (Takezawa and Obara, 2010, 2012). The calculation results confirmed the increase of total energy released in 10% enriched fuel-solution tank system. However, the effect of changing the enrichment or concentration of the fuel solution has not yet been clarified.

The purpose of this study was to perform analyses of the supercritical condition using the developed code for a several-tank

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system for different compositions of fuel solution and to estimate the increase of total energy released in a two-tank system compared with that of a single-tank system.

2. Analysis method

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 any system geometry.

In the method, time- and region-dependent fission density functions are obtained in advance by integration of all fission contributions from the past to the present (Takezawa and Obara, 2010, 2012). The fission density $N_i(t)$ is shown below:

$$N_i(t) = \sum_{j=1}^{n}\left(\int_{-\infty}^{t}\alpha_{ij}^{p}(t-t')N_j(t')\mathrm{d}t' + \int_{-\infty}^{t}\alpha_{ij}^{d}(t-t')N_j(t')\mathrm{d}t'\right) \tag{1}$$

where $N_i(t)$ is the total fission density at time t in fuel-solution in tank i. $\alpha_{ij}^{p}(\tau)$ and $\alpha_{ij}^{d}(\tau)$ are secondary prompt-fission and delayed-fission probability density functions in fuel-solution tank i provided by the first fission of fuel-solution in tank j with time difference τ. We are considering the super-prompt critical accident, so the delayed neutrons which the little effect in the first pulse in the accidents are ignored. So the Eq. (1) can be reduced to Eq. (2)

$$N_i(t) \approx \sum_{j=1}^{n}\int_{t-t_c^p}^{t}\alpha_{ij}^{p}(t-t')N_j(t')\mathrm{d}t' \tag{2}$$

where t_c^p is the time that satisfies $\alpha_{ij}^{p}(\tau > t_c^p) \approx 0$. $\alpha_{ij}^{p}(\tau)$ in Eq. (2) can be calculated based on a fission source distribution, which is reasonable for problems solved using the Monte Carlo neutron transport method (Kukharchuk, 2000).

In the space-dependent kinetic analysis method of Eq. (2), it is assumed to be applied to transient analysis of a step-wise positive reactivity insertion by an accident into a steady-state condition of very low power – for example, 1 W – with a source distribution of a fundamental mode (Takezawa and Obara, 2010, 2012).

In the practical accidents, the initial condition shall be subcritical condition. In the analysis, the initial condition is assumed to be critical condition. But its power is assumed to be very low power so it will give little effect on the results.

The integral kinetic model Eq. (2) can be solved by using forward calculation method with the time step Δt:

$$\begin{aligned} N_i(k\Delta t) &= \sum_{j=1}^{n}\left\{\int_{(k-k_c)\Delta t}^{0}\tilde{\alpha}_{ij}(k\Delta t-t',T_0)N_j\mathrm{d}t' + \int_{0}^{k\Delta t}\alpha_{ij}(k\Delta t-t',T(t'))N_j(t')\mathrm{d}t'\right\} \\ &= \sum_{j=1}^{n}\left\{N_j\left[\tilde{c}_{ij}(\tau',T_0)\right]_{k\Delta t}^{k_c\Delta t}\int_{k'=0}^{k-1}N_j(k'\Delta t)\left[c_{ij}(\tau',T_{k'})\right]_{(k-k'-1)\Delta t}^{(k-k')\Delta t}\right\} \end{aligned} \tag{3}$$

where $\tilde{\alpha}_{ij}(\tau)$ is a $\alpha_{ij}(\tau)$ at a critical steady state with $\rho = 0$, N_j is a steady state fission density in fuel-solution tank j and $T(t')$ is a temperature distribution in a system at time t'.

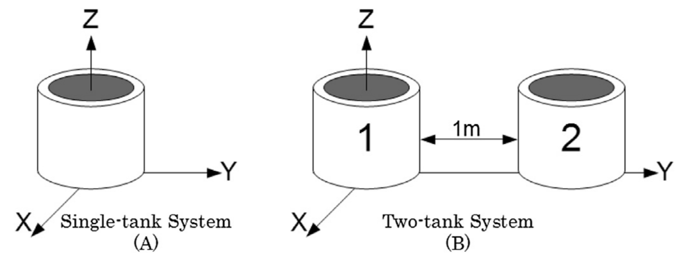

Fig. 1. Calculation geometries of single- and two-tank system.

The code using for analysis was assumed to apply the transient analysis of a step-wise positive reactivity insertion into a fuel-solution tank system.

Most of the energy during the prompt critical is released in the first pulse (Yamane, 2012, 2003). So the analysis of the first pulse by prompt neutron has important meaning.

In this study, transient analyses were performed for a situation in which positive reactivity is inserted in a critical condition with very low power (1 W); thus, it was necessary to obtain $\alpha_{ij}(\tau)$ functions that would be suitable for this purpose.

Radiolytic dissociation gas voids are created in the case of the supercritical condition in fuel solutions, when the power density exceeds a particular threshold (Ogawa and Kaminaga, 2008). Two negative reactivity-feedback models were considered. One is a model treating the change of density by the increase of the fuel-solution temperature. The other model treats the decrease of solution density and increase of volume by the creation of radiolytic dissociation gas voids when the power exceeds the threshold.

In the feedback model for radiolytic dissociation gas voids, the total volume of the gas voids created during excursion was estimated by the following equation.

$$V_G = 4.1 \times (E_P - e_c V_L) \tag{4}$$

where V_G is the volume of the radiolytic dissociation gas voids (L), E_P is the energy of the first pulse (MJ), e_c is the threshold energy for the radiolytic dissociation gas generation per unit volume (MJ/L) and V_L is the volume of the solution (L). The threshold value of radiolytic dissociation gas voids production 37 kJ/L which was given by Ogawa was used in the analysis (Ogawa and Kaminaga, 2008). In this calculation, only prompt neutrons were taken into account. It was assumed that the radiolytic dissociation gas voids were created homogeneously in the solution. The changes in density and volume of the fuel solution by the radiolytic dissociation gas voids were taken into account in the reactivity feedback.

In a previous study, it was confirmed that the results calculated by the code were suitable for a weakly coupled system (Takezawa and Obara, 2010, 2012). Using this code, it's possible to consider