Proceedings of the Korea Nuclear Society Autumn Meeting
Taegu, Korea, October 1997

A Simple Dynamic Model and Transient Simulation of the Nuclear Power Reactor on Microcomputers

Gee Yang Han and Cheol Park

Korea Atomic Energy Research Institute
150 Dukjin-dong, Yusong-gu, Taejon 305-353, Korea

Abstract

A simple dynamic model is developed for the transient simulation of the nuclear power reactor. The dynamic model includes the normalized neutron kinetics model with reactivity feedback effects and the core thermal-hydraulics model. The main objective of this paper demonstrates the capability of the developed dynamic model to simulate various important variables of interest for a nuclear power reactor transient. Some representative results of transient simulations show the expected trends in all cases, even though no available data for comparison. In this work transient simulations are performed on a microcomputer using the DESIRE/N96T continuous system simulation language which is applicable to nuclear power reactor transient analysis.

1. Introduction

Until recently, nuclear power plant transient analyses had been performed exclusively on large mainframe computers. Because of the low cost and fast progress of the microcomputer technology, the use of the microcomputer such as a personal computer (PC) is receiving much attention and increasing rapidly in the areas related to the nuclear power plant system simulation. In view of the computational capabilities of the currently available microcomputer, this work is initiated to develop a simple dynamic model to conduct quick and inexpensive transient analyses of a nuclear power reactor on microcomputers.

The dynamic model of the nuclear power reactor requires the reactor kinetics and the core thermal-hydraulics models which are generally applicable for simulating transients. The normalized neutron kinetics model, based on the point kinetics equations, with six groups of delayed neutrons and reactivity feedback effects is used to provide the reactor power behavior in the core. The core thermal-hydraulics model is developed by a set of simultaneous differential equations based on a lumped parameter approach for simulating the time rate of fuel and coolant temperature changes at points of interest in the core during operational transients. Reactivity feedback effects in this work consider only due to fuel and coolant temperature changes which are handled using reactivity feedback coefficients. Figure 1 shows a simplified the primary coolant loop of a nuclear power reactor.

Transient events in the nuclear power reactor involve very complicated interactions between the reactor core and the primary coolant loop. Especially, interactive physical processes between the reactor power and temperature effects on reactivity that determine the dynamic behavior of a reactor is central to the understanding of the transient responses. The main objectives of this paper are to present the first phase of a research to develop a dynamic model for the transient analysis of a nuclear power reactor on a microcomputer and to provide some representative results obtained from various transient simulations of interest using the DESIRE/N96T simulation language [1].

2. Mathematical Modeling

2.1 Reactor Kinetics Model

1. Normalized Neutron Kinetics Model

It is often convenient to use normalized variables. The point kinetics equations [2]

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n + \sum_i \lambda_i c_i \tag{1}$$

$$\frac{dc_i}{dt} = \frac{\beta_i}{\ell} n - \lambda_i c_i \tag{2}$$

have equilibrium at $n = n_0$ and $c_i = c_{i0} = \beta_i n_0 / \lambda_i \ell$.

Let $N = n/n_0$ and $D_i = c_i / c_{i0} = \lambda_i \ell \, c_i / \beta_i n_0$. The precursor equations become

$$\frac{dD_i}{dt} = \lambda_i (N - D_i) \tag{3}$$

Similarly, the normalized neutron equation becomes

$$\frac{dN}{dt} = \frac{\beta}{\ell} [(R - 1)N + \sum_i F_i D_i] \tag{4}$$

where $R = \rho/\beta$ in dollars and $F_i = \beta_i/\beta$.

2. Reactivity Feedbacks

As a linear approximation, the Doppler feedback reactivity may be written

$$\rho_f = -\alpha_f (T_f - T_{f0}) \tag{5}$$

The coolant feedback reactivity may be given

$$\rho_c = -\alpha_c (T_c - T_{c0}) \tag{6}$$

where α_f, α_c are the Doppler and coolant reactivity coefficients, T_f, T_{f0} are the average and steady-state fuel temperatures, T_c, T_{c0} are the average and steady-state coolant temperatures, respectively. Coupling to the normalized neutron kinetics model is accomplished by

$$R = R_0 - \alpha_f (T_f - T_{f0}) - \alpha_c (T_c - T_{c0}) \tag{7}$$

where R_0 is a constant, or a function representing control rods or boron concentration.

The feedback loop must be closed by equations that predict T_f and T_c in terms of the neutron density or the reactor power.

2.2. Core Thermal-Hydraulics Model

1. Fuel

The energy balance equation for the fuel elements [3] may be written as

$$\frac{dT_f}{dt} = \frac{P}{m_f c_f} - \frac{P_0}{m_f c_f}\left(\frac{T_f - T_c}{T_{f0} - T_{c0}}\right) \tag{8}$$

where m_f is total fuel mass, c_f is the specific heat of fuel, and P, P_0 are the thermal and steady-state power, respectively. The equation for the time rate of change of the fuel temperature is given

$$\frac{dT_f}{dt} = K \cdot N - \frac{(T_f - T_c)}{\tau_1} \tag{9}$$

where K is the reciprocal of the reactor heat capacity and τ_1 is the core time constant for heat loss.

2. Coolant

The energy balance equation for the coolant within the core may be written as

$$\frac{dT_c}{dt} = \frac{P_0}{m_c c_c}\left(\frac{T_f - T_c}{T_{f0} - T_{c0}}\right) - \frac{2W}{m_c}(T_c - T_p) \tag{10}$$

where m_c is the mass of coolant in the core, c_c is the coolant specific heat, W is the mass flow rate, and T_c, T_p, T_{ex} are the average, core inlet, and core outlet coolant temperatures, respectively. The final energy balance equation for the coolant yields

$$\frac{dT_c}{dt} = \frac{(T_f - T_c)}{\tau_2} - W_{rel}\frac{(T_c - T_p)}{\tau_3} \tag{11}$$

where τ_2 is the time constant for heat gain by coolant and τ_3 is the time constant for heat removal by convection.

For predicting a flow transient through the primary coolant loop, the basic balance equation for the mass flow rate is presented as

$$\frac{dW_{rel}}{dt} = \frac{(W_{sp} - W_{rel})}{\tau_4} \tag{12}$$

where W_{sp} is the set point of mass flow rate (normal $W_{sp} = 1.0$) and τ_4 is the time constant which is experimentally determined based on the loop circulation time.

3. Results and Discussion

Transient simulations are best performed to the simplified model of a nuclear power reactor for the main variables of interest which can be reactor power, reactivity, fuel and coolant temperatures. The principal transient simulation selected for the presentation of those variables in this paper are the reactivity insertion, primary inlet temperature drop, and reduced primary inlet flow rate. Most of the variables have a characteristics response time of the order of 60 sec. Input data are used of typical characteristics of the pressurized water reactor for this work.

We first consider the perturbation where there is a step change in reactivity applied to the dynamic model that is initially just critical. This sudden reactivity insertion causes to raise the reactor power and fuel and coolant temperature in the first 60 sec of the transient. When a step change in reactivity of 40, 60, and 80 cents are introduced to the reactor, reactor power shoots up rapidly, until inherent feedback characteristics take over without any control action, and bring reactor power back to a new steady state level. If it has no feedback effect, the response goes to infinity. The peak power occurs at the time when reactivity is close to prompt critical, then accompanies by the delayed neutron tail. The fuel temperature increased very fast and then began to level off, whereas the coolant temperature began to increase very slowly. The temperature coefficients for reactivity feedback effects have two parts of effects which are a prompt component as well as a delayed component. The prompt component is a effect that depends on the instantaneous state of the fuel such as Doppler effect. The delayed component depends primarily on the coolant. Hence their different roles as the prompt and delayed component of the temperature coefficient are noted. The feedback effects of most interested in nuclear reactor systems are based on the changes in power level. Because as core temperature increases the water expands and becomes less dense and thus not as good a moderator of neutrons, producing fewer thermal neutrons to cause fissions in the fuel. Thus, as reactor core temperature increases it tends to insert negative reactivity temperature coefficient of reactivity in thermal reactors. The responses of simulated transients induced by reactivity insertions are shown in Figures 2 through 5. All plotted values are indicated transient responses from the initial steady-state.

Cases with a step changes in the primary inlet temperatures and flow rates being changed at constant power are simulated to understand the effects of the temperature coefficients. When the primary inlet temperature is decreased in this model, it simulates a case where some cold water has been added. The coolant temperature drops in this case and negatively brings up the reactivity which leads to an increase in reactor power. The responses of the nuclear power reactor to a step change in primary inlet temperature are shown in Figures 6 and 7. When the transient with a decrease in primary inlet flow rate increases the core void fraction, and hence, reduces the reactor power and reactivity which are expected trends. The responses to a step change in primary inlet flow rate are shown in Figures 8 and 9.

4. Conclusions

The nuclear power reactor simulation in DESIRE/N96T simulation language provides a great potential to conduct scoping and parametric studies for transient analyses. The intent of presenting the results is to assure the proper trend in the transient simulation. Transient responses are predicted as the expected trends. But for the cases in which there are no available data for comparison. The future work for the dynamic model is planned to be further enhanced to include improved steam generator model and other main component models continuously.

References

1. G. A. Korn, *DESIRE/N96T Reference Manual*, G. A. and T. M. Korn Industrial Consultants, Chelan, WA (1996)
2. D. L. Hetrick, *Dynamic of Nuclear Reactors*, University of Chicago Press, Chicago (1971)
3. E. E. Lewis, *Nuclear Power Reactor Safety*, John Wiely & Sons, New York (1977)

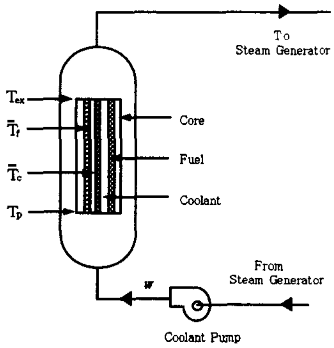

Fig. 1 A Schematic Diagram of Nuclear Power Reactor

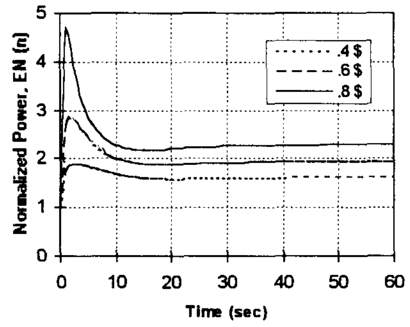

Fig. 2 Power Responses to Reactivity Insertions

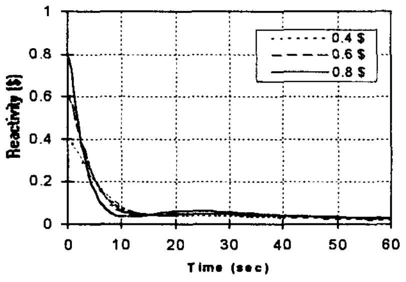

Fig. 3 Reactivity Responses to Reactivity Insertions

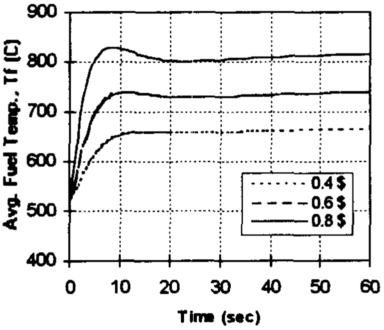

Fig. 4 Average Fuel Temperature Responses to Reactivity Insertions

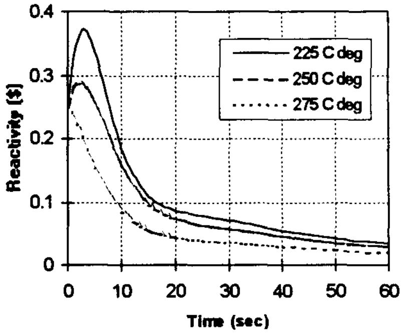

Fig. 7 Reactivity Responses to Core Inlet Temperature Changes

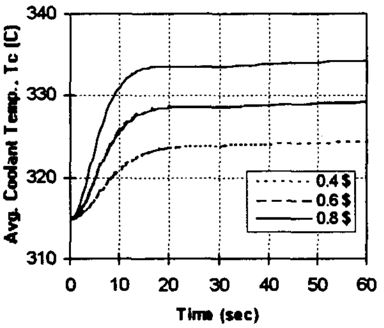

Fig 5 Average Coolant Temperature Responses to Reactivity Insertions

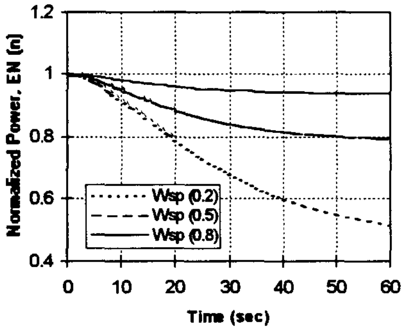

Fig. 8 Power Responses to Flow Rate Changes

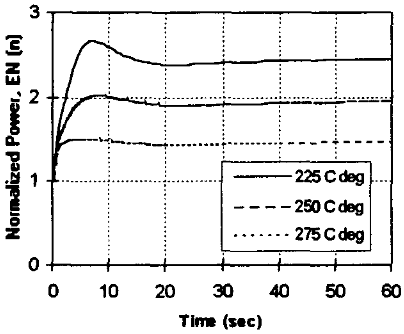

Fig. 6 Power Responses to Core Inlet Temperature Changes

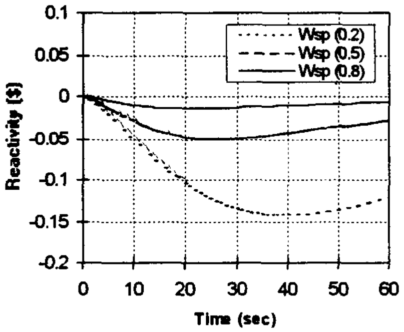

Fig. 9 Reactivity Responses to Flow Rate Changes