MODELING AND SIMULATION OF ADVANCED NUCLEAR REACTORS

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ABSTRACT

Advanced nuclear power plants are currently being proposed with a number of various designs. However, there is a lack of modeling and control strategies to deal with load following operations. This research investigates a possible modeling approach for advanced nuclear power plants in order to provide an assessment way to the concept designs. The modeling approach covers linear and nonlinear reactor modeling and linear modeling of heat exchanger-turbine-generator systems. Numerical results are presented on an example molten-salt type nuclear power plant system to demonstrate the validity and effectiveness of the modeling approach.

Index Terms— Nuclear reactor, modeling, kinetics, dynamics, dynamic simulation.

1. INTRODUCTION

Several advanced reactor concepts are being proposed to enhance the future role of the nuclear energy supply with manageable nuclear waste, effective fuel utilization, increased environmental benefits, competitive economics, recognized safety performance, and secure nuclear energy systems and nuclear materials [1]. To deal with these challenges and provide new products such as hydrogen for transportation applications, reduced nuclear wastes for disposal, and increased proliferation resistance, six most promising systems (Gen-4 reactors) are selected by the GIF for further study: gas-cooled fast reactor (GFR), very-high-temperature reactor (VHTR), super-critical-water-cooled reactor (SCWR), sodium-cooled fast reactor (SFR), lead-cooled fast reactor (LFR), and molten salt reactor (MSR). Furthermore, small modular reactor (SMR) designs, which are based on current advanced reactors and the Gen-4 reactor concepts, can be an economically viable option for energy needs of many countries in the world, specifically for remote/isolated areas and for specific applications (e.g. water desalination or heat production). The practicality and effective utilization of the above nuclear reactor concepts are depend on sharing nuclear power with other power sources on the same electricity grid, which necessitates load following capabilities or plant-wide automation control generations in nuclear power plants.

The goals for an effective plant-wide control system include automated control, safe and smooth process operation, and high quality control in the face of disturbances. To achieve these goals, the nuclear power plants can have base-load or open-loop (no recycle) and load-following or closed-loop (recycle) configurations. In the open-loop configuration there is no feedback effects, the individual unit operations governs the dynamic behavior of plant and the only path for disturbance propagation is linear along the process [2]. On the other hand, in the closed-loop configurations, the plant-wide control problem becomes much more complex because the presence of feedback significantly changes the dynamic and steady-state behaviors of plant. The main feedback effects in a plant are described with overall time constant change and snowball effect [2], [3]. The overall time constant can be different from the sum of the time constants of the individual units due to possible positive feedbacks. The snowball effect means that a small change in the plant input can produce a large change in plant output due to propagation around the feedback loop. These properties limit the control configurations in an integrated power plant. Therefore, a plant oriented approach which uses heuristic rules based on plant understanding and experience is necessary for an effective plant-wide control in nuclear power plants.

In power plants, the power supply and demand must be balanced by either generation or load since transmission systems provide negligible energy storage [4]. The approaches for balancing power supply and demand utilize the load frequency control to provide inherent capability of load following. The problem in here is that the thermal generation systems often have difficulty in following the load due to the slow response of the units [5]. Hence, controlling dominant plant variables through local unit controllers can provide a general solution. It is also pointed out that simple control strategies are desirable for complex plants in order to deal with overall feedback effects.

The load following control studies for nuclear power plants have mostly been carried out in steam turbines due to dominancy of pressurized and boiling water reactors [6], [7], [8], [9], [10], [11], [12], [13]. Recently, some studies have also been demonstrated for dynamic modeling, control and transient analysis in gas turbine nuclear plants [14], [15], [16], [17]. Plant dynamics and control for super light-water and super-fast reactors have been studied in [18], and safety simulations in nuclear power systems based on computer codes have been collected in [19]. More detailed and complete models and control configurations have been provided for space nuclear reactors in [20], [21]. On the other hand, the modeling and control studies of the gas turbine power systems have been given for thermal power plants other than nuclear power plants, such as fossil fuel power plants. Many different types of gas turbines are currently in use in power systems worldwide. References [22], [23], [24], [25] have been overviewed the gas turbine systems. In general, a typical model of gas turbine consists of load-frequency control, temperature control and acceleration control loops [22]. The applicability of these control approaches to nuclear power plants requires a detailed study. Since there are no detailed nuclear plant models and control studies by considering gas turbine nuclear power plants, it is significantly necessary to develop system models for evaluating stability and feasibility of possible control approaches and for assessment of plant safety limits.

The main purpose in this paper is to provide an overall plant-wide modeling approach for advanced nuclear plant plants. The modeling approach is based on lumped parameter models, i.e. first principles and physical laws of dominant systems as well as input/output models. In the modeling, temperature control of the reactor is considered.

The paper first gives dynamic models for the nuclear plant systems. Next, some numerical simulation results are provided. Finally, conclusions are given.

2. DYNAMIC MODELING

Dynamic modeling and using a system dynamics model require art as well as knowledge since the approximations and realistic interpretations are indispensable parts of a model. The modeling approach in this study is based on the lumped parameter models that include several assumptions, e.g. transients during close neighborhood of critical reactivity, incompressible flow, constant mass flow rates, constant heat capacity and heat transfer coefficients. The reference nuclear power plant consists of two loops as shown in Fig. 1. The nuclear power plant is composed of a nuclear reactor, heat exchangers and power conversion unit that can utilize closed-loop Brayton or Rankine cycles. The dynamic models are used to simulate the nuclear power plant under consideration are given below. The reference for modeling a nuclear reactor is given in Fig. 1.

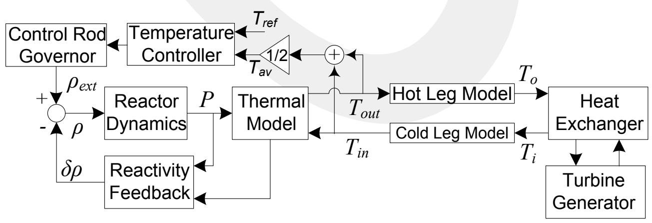

Fig. 1 A representation of nuclear power plants with reactor control system.

2.1. Dynamic Modeling of the Nuclear Reactors

The nuclear reactor is modeled based on coupling neutronics with thermodynamics to capture the complexity of the physics of the nuclear reactors. Considering the reference system illustrated in Fig.1, the nuclear reactor consists of neutronics, thermodynamics, temperature controller, control rod drive mechanism (governor), and hot and cold legs dynamics to account for dynamic characteristics of pipeline.

Neutron flux or reactor power is calculated from point reactor kinetics equations (PRK) that assumes that the reactor transients are analyzed during slightly subcrital or supercrital reactor conditions.

$$\begin{aligned} \frac{dP(t)}{dt} &= \frac{\rho(t)-\beta}{\Lambda}P(t)+\sum_{i=1}^{6}\lambda_i C_i(t) \\ \frac{dC_i(t)}{dt} &= \frac{\beta_i}{\Lambda}P(t)-\lambda_i C_i(t) \end{aligned} \tag{1}$$

where P (in watts) is the reactor power, C_i (in watts) is the ith precursor or latent power, ρ (in Δk/k) is the reactivity, β_i is the delayed neutron faction of the ith group, β is total precursor group fraction $\beta = \sum_{i=1}^{6}\beta_i$, Λ (in s) is mean neutron generation time and λ_i (in s^{-1}) is the decay constant for the ith precursor.

The effects of temperature and density changes are primarily concerns in reactivity feedback [26]. Due to relationship between density and temperature in fluids, the density variations can usually be expressed as a function of temperature, and thus a total temperature reactivity feedback can be considered. Then, the time dependency of reactivity $\rho(t)$ can be written as a function of external reactivity input, $\rho_{ext}(t)$, and the total temperature feedback is [26], [27]

$$\rho(t) = \rho_{ext}(t) - \alpha\left(T_{out} - T_r\right) \tag{2}$$

where T_{out} (in °C) is the reactor outlet temperature, T_r (in °C) is an equilibrium reference temperature and α (in °C^{-1}) is the total temperature feedback coefficient.

Under constant thermal properties and mass flow rate $\dot{m}$, $T_{out}(t)$ can be found from

$$\begin{aligned} \frac{dT_{out}}{dt} &= k_f P(t) - \gamma_f\left(T_{out} - T_{in}\right) \\ P_{th} &= \dot{m}c_p\left(T_{out} - T_{in}\right) \end{aligned} \tag{3}$$

where k_f is the reciprocal of the reactor heat capacity $k_f = 1/m_c c_p$, $1/\gamma_f$ is the mean time for heat transfer $1/\gamma_f = m_c/\dot{m}$, c_p (in J/kg·°C) is the heat capacity coefficient, P_{th} (in watts) is the thermal power generated by reactor and T_{in} (in °C) is the reactor inlet temperature. The thermal power generated by the reactor given in (3) equals to the reactor power given in (1) under steady state conditions.

The dynamics of the pipeline together with control valves can act as a variable resistance and a simple capacity, which results in time delay in energy transfer systems [28]. The hot leg and cold leg models can simply be expressed with the following first order differential equations

$$\tau_{hl}\frac{dT_o}{dt} = -T_o + T_{out}\ ,\ \ \tau_{cl}\frac{dT_i}{dt} = -T_i + T_{in} \tag{4}$$

where τ_{hl} and τ_{cl} are the time constants (in seconds) for hot leg and cold leg temperature models, respectively.

Control rod governors adjust the position of the control rod banks in the reactor core to obtain smaller incremental reactivity changes per step. The control rod governor can simply be modeled by

$$\tau_g \frac{d\theta}{dt} = -\theta + \Delta V \tag{5}$$

where τ_g is the time constant of the rod drive mechanism (about several milliseconds), θ is the governor output position and ΔV is the governor input voltage. The output of control rod governor is then converted to reactivity input ρ_{ext} by the integral rod worth function [29]. The average rod worth is used for control calculations in power reactors due to critical reactor conditions.

2.1.1. Model Linearization

The system models of nuclear power plant can be represented with simple input/output models for stability analysis and dynamic simulations. For models described in Section 2.1, the simplified linear models are as follows. The rod control governor (5) has the following transfer function

$$G_G(s) = \frac{1}{1+\tau_g s} \tag{6}$$

Nuclear reactor model for one group critical reactor without reactivity feedback is given by

$$G_R(s) = \frac{s+\lambda}{\Lambda s(s+\beta/\Lambda)} \tag{7}$$

This linear equation is an excellent approximation to reactor modeling, but do not include reactivity feedback. However, the nuclear reactors are inherently safe (stable) with reactivity feedback because they must be designed to have negative reactivity feedback. The hot/cold legs models are

$$G_L(s) = \frac{1}{1+\tau_l s} \tag{8}$$

where τ_l is the average time constant for legs. In (8), it is assumed that both hot and cold legs have the same time constants due to their similar structures.

2.2. Modeling Heat Exchanger and Turbine-Generator Systems

A good transfer function representation of heat exchangers (HEXs) is given by the first order plus dead time (FOPDT) model which is commonly used to design its temperature controller as follows,

$$G_H(s) = \frac{1}{1+\tau_h s} e^{-\theta s} \tag{9}$$

where τ_h is the time constant for HEX (around 5 to 120 seconds) and θ is the dead time (around 1 to 30 seconds) [30]. The approximate values of parameters τ_h and θ can be determined through process reaction curve approach. The time delay term can be approximated to $e^{-\theta s} \approx 1/(1+\theta s)$ using Taylor series expansion. It should be noted that the HEX dynamics may be ignored when it has the bypass control mechanism [2]. The turbine system can be expressed with

$$G_T(s) = \frac{1}{1+\tau_t s} \tag{10}$$

where τ_t (in seconds) is the time constant of turbine system. The generator-load model has the following transfer function

$$G_{GL}(s) = \frac{1}{D+2Hs} \tag{11}$$

where D is the load damping factor (in megawatts per Hz) and H represents inertia constant (about 2 to 9 seconds).

3. NUMERICAL RESULTS

Two-loop MSR reactor configuration as shown in Fig. 1 is considered. The reactor core is designed to provide 2400 MW thermal power from transuranic burn-up. The fuel salt is 15LiF-27BeF_2-58NaF (in mole %) and trifluorides of actinides and the secondary salt is NaF-$NaBF_4$. For the MATLAB/Simulink based numerical simulations, the delayed neutron precursor parameters of the nuclear reactor which are calculated and verified with MCNP5/MCNPX codes via ENDF/B-V and ENDF/B-VI nuclear data libraries are given in Table 1.

Table 1: Delayed neutron data

Group	Decay Constant, λ_i (s^{-1})	β_i/β	Delayed Fraction, β_i
1	0.0128	0.0293	9.96e-5
2	0.0300	0.2507	8.52e-4
3	0.1103	0.1725	5.87e-4
4	0.3135	0.3832	1.30e-3
5	0.8634	0.1275	4.34e-4
6	1.3503	0.0368	1.25e-4
Total delayed neutron fraction, β			3.40e-3
Neutron generation time, Λ (s)			8.3e-6

The numerical simulation results are given in Figs. 2 and 3. Figure 2 illustrates time responses of reactivity, power and average temperature of the reactor. The total reactivity margins of the MSRs correspond to around 1000 pcm (or 0.01 $\Delta k/k$) reactivity addition in more than some minutes. By considering reactivity limitation, the average temperature control system brings the average reactor temperature to its reference temperature value 665 ^{o}C in a short time through control rod drive mechanism. In Fig. 3, the response of the load frequency control system is displayed. The load frequency control is obtained through turbine regulator during load power demand variations. The generator frequency is maintained around its reference value 60 Hz during load demand variations. During ramp load changes, the generator frequency has a constant but small steady-state error. It is also observed that the load demand variations do not have visible transient effects on the average temperature control system.

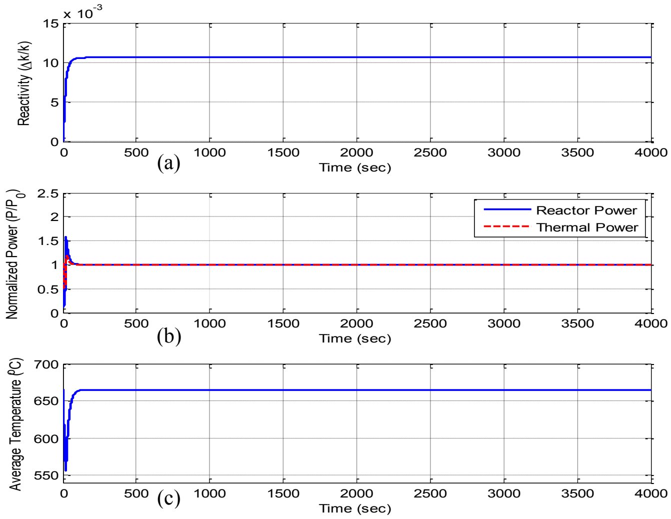

Fig. 2 Time responses for average temperature control system, (a) external reactivity response, (b) normalized power, (c) average temperature.

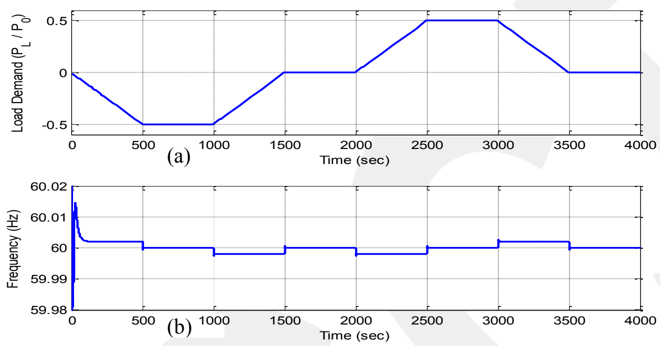

Fig. 3 Time responses for the load frequency control, (a) load demand, (b) frequency variations.

4. CONCLUSION

The dynamic modeling and simulation problems of nuclear power plants have been investigated. The study set out to determine the effect of load following control in stability of the advanced nuclear power plants. The nuclear power plant is modeled through combination of nonlinear and simple linear dynamic models for investigating stability and safety features of reactors. Future research might investigate the model verification and validation using benchmark tests and the load following control mechanisms.

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