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Core Power Control of the fast nuclear reactors with estimation of the delayed neutron precursor density using Sliding Mode method

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HIGHLIGHTS

- We present a S.M.C. system based on the S.M.O for control of a fast reactor power.
- A S.M.O has been developed to estimate the density of delayed neutron precursor.
- The stability analysis has been given by means Lyapunov approach.
- The control system is guaranteed to be stable within a large range.
- The comparison between S.M.C. and the conventional PID controller has been done.

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M. Instrumentation and control

ABSTRACT

In this paper, a nonlinear controller using sliding mode method which is a robust nonlinear controller is designed to control a fast nuclear reactor. The reactor core is simulated based on the point kinetics equations and one delayed neutron group. Considering the limitations of the delayed neutron precursor density measurement, a sliding mode observer is designed to estimate it and finally a sliding mode control based on the sliding mode observer is presented. The stability analysis is given by means Lyapunov approach, thus the control system is guaranteed to be stable within a large range. Sliding Mode Control (SMC) is one of the robust and nonlinear methods which have several advantages such as robustness against matched external disturbances and parameter uncertainties. The employed method is easy to implement in practical applications and moreover, the sliding mode control exhibits the desired dynamic properties during the entire output-tracking process independent of perturbations. Simulation results are presented to demonstrate the effectiveness of the proposed controller in terms of performance, robustness and stability.

1. Introduction

Considering the reducing fossil energy resources, pollution, the greenhouse gas and the economic efficiency, nuclear power is taken into consideration. Since thermal reactors are able to use such a small percentage of uranium fuel, the design, construction and operation of fast reactors is underway (Driscoll et al., 1979). By greatly simplifying the nuclear fuel cycle, fast reactors such as travelling wave reactor could improve the cost, safety, social acceptability, and long term sustainability of nuclear energy as a source of emissions-free electricity (Weaver et al., 2009). On the other hand, to use a nuclear power, reactor energy must be controlled. According to the reactor period dependency to the neutron lifetime and the crucial importance of this quantity in reactor power, it is necessary to have control on the reactor power (Hetrick, 1965). Due to the very short life time of prompt neutrons in the fast nuclear reactor, it is difficult to control its power. Therefore, the delayed neutrons which are produced indirectly by precursors must be considered to increase the reactor period. Besides, it seems that a simple and high performance control system is needed. A successful strategy to control uncertain nonlinear systems is Sliding Mode Control (SMC). The sliding mode controller is an attractive robust control algorithm because of its inherent insensitivity and robustness to plant uncertainties and external disturbances (Choi, 1999; Furat and Eker, 2012). In this paper, a SMC system is designed to control a fast nuclear reactor based on the point kinetics equations with one group of the delayed neutrons.

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Since the measurement of the delayed neutron precursor density is practically difficult and it should be measured to design the controller, a sliding mode observer which has the robust characteristics facing the external disturbances and parameters uncertainties is proposed based on the reactor power measurement to estimate the densities of delayed neutron precursors.

Finally, a sliding mode control system based on the sliding mode observer is presented for controlling the fast nuclear reactor core power. Simulation results are provided to show the effectiveness of the proposed control system.

2. The reactor core model

2.1. The point kinetics equations

Reactor kinetics equations for both fast and thermal reactors are identical. Point kinetics approximations can be used more effectively for fast reactors than for thermal reactors because fast reactors are more tightly coupled neutronically. Tighter coupling implies that the neutron flux is more nearly separable in space and time, which is a necessary condition for point kinetics approximations to be valid (Walter et al., 2012). To simulate the nuclear reactor core, point kinetics equations with one group of the delayed neutrons is used (Hetrick, 1965). The model assumes feedback from lumped fuel and coolant temperatures. The normalized model, with respect to an equilibrium condition, based on point kinetics equations with one delayed neutron group is as follows:

$$\frac{\mathrm{d}n_r}{\mathrm{d}t} = \frac{\rho-\beta}{l}n_r + \frac{\beta}{l}C_r \quad (1)$$

$$\frac{\mathrm{d}c_r}{\mathrm{d}t} = \lambda n_r - \lambda C_r \quad (2)$$

where n_r, ρ, c_r, λ, l and β refer to relative (normalized) power, reactivity, relative precursor density, decay constant of precursors, prompt neutron lifetime and effective delayed neutron fraction, respectively. Reactor temperatures vary as a function of power generated in fuel and heat transfer from (or to) the system. The reactor power can be represented as

$$P(t) = n_r P_0 \quad (3)$$

where $P(t)$ is reactor power at time t (MW), and P_0 is the nominal power (MW).

2.2. Fast reactor reactivity equation

One of the most important parameters of the dynamics behaviour of the reactor is reactivity (Hetrick, 1965). Due to the fast neutron spectra in fast reactors, fast reactors reactivity equation is very different from thermal reactors, and appears as follows (Tentner et al., 2010):

$$\rho = [P_r(t)-1]A + \left[\frac{P_r(t)}{F_r(t)} - 1\right]B + \delta T_{in}C + \rho_r \quad (4)$$

$$\frac{d_{\rho r}}{\mathrm{d}t} = G_r z_r \quad (5)$$

$$A = \alpha_D \Delta T_F(0)$$

$$B = \left[\alpha_D + \alpha_{Na} + \alpha_{Axi} + 2\left(\frac{\mathrm{XMC}}{\mathrm{XAC}}\right)\alpha_{\mathrm{Radi}} + 2\alpha_{\mathrm{Crod}}(aL_{\mathrm{CRD}} - bL_{\mathrm{VHP}})\right]\frac{\Delta T_c(0)}{2}$$

$$C = \alpha_D + \alpha_{Na} + \alpha_{Axi} + \alpha_{\mathrm{Radi}} + \alpha_{\mathrm{Crod}}\cdot(aL_{\mathrm{CRD}} - bL_{\mathrm{VHP}} - bL_{\mathrm{VCP}})$$

Table 1
Values of constants used for control analysis and simulations (Walter et al., 2012; Kim and Taiwo, 2012).

Parameter	Values	Parameter	Values
β	0.00334	$\lambda(1/s)$	0.0337
$M(\mathrm{MW}/^\circ C)$	0.085	$l(s)$	0.00000038
$P_0(\mathrm{MW})$	1000	G_r	0.0405
f_f	0.92	A (¢)	−73
T_{in} (°C)	355	B (¢)	−41
C (¢/°C)	−0. 4	μ_f (MWs/°C)	0. 1802
μ_c (MWs/°C)	0.06	Ω (MW/°C)	0.447

where $P_r(t)$ is the normalized reactor power, $F_r(t)$ is the normalized core coolant flow, A, B, C is the reactivity feedback parameters, α_D is the doppler coefficient (pcm/°C), α_{Na} is the sodium density coefficient (pcm/°C), α_{Axi} is the fuel axial exp ansion coefficient (pcm/°C), α_{Radi} is the core radial expansion coefficient (pcm/°C), α_{Crod} is the control rod driveline expansion coefficient (pcm/m), XMC is the grid plate to core midplane distance (m), XAC is the grid plate to above – core load plane distance (m), a is the thermal expansion coefficient of the control rod driveline (1/°C), b is the thermal expansion coefficient of the vessel wall (1/°C), L_{CRD} is the length of control rod driveline in contact with the hot pool (m), L_{VHP} is the length of vessel wall in contact with the hot pool (m), L_{VCP} is the length of vessel wall in contact with the cold pool (m), ρ is the reactor reactivity change, ρ_r is the control rods reactivity (pcm), G_r is the reactivity worth of rod per unit length (pcm/cm), z_r is the control input, control rod speed (cm/s), $\Delta T_F(0)$ is the steady state Temperature difference, fuel to coolant, $\Delta T_c(0)$ is the steady state Coolant temperature rise, inlet to outlet.

2.3. Thermal-hydraulics model of the reactor core

The thermal-hydraulics model of the reactor core can be represented with the following equations

$$\frac{\mathrm{d}T_f}{\mathrm{d}t} = \frac{f_f P_0}{\mu_f}n_r - \frac{\Omega}{\mu_f}T_f + \frac{\Omega}{2\mu_f}T_{\mathrm{out}} + \frac{\Omega}{2\mu_f}T_{\mathrm{in}} \quad (6)$$

$$\frac{\mathrm{d}T_{\mathrm{out}}}{\mathrm{d}t} = \frac{(1-f_f)P_0}{\mu_c}n_r + \frac{\Omega}{\mu_c}T_f - \frac{(2M+\Omega)}{2\mu_c}T_{\mathrm{out}} + \frac{(2M-\Omega)}{2\mu_c}T_{\mathrm{in}} \quad (7)$$

where f_f is the fraction of reactor power deposited in the fuel, Ω is the heat transfer between fuel and coolant (MW/°C), M is the mass flow rate times heat capacity of Na (MW/°C), μ_f is the fuel mass times specific heat (MWs/°C), μ_c is the coolant mass times specific heat (MWs/°C), T_f is the fuel temperature (°C), T_{out} is the Temperature of coolant leaving the core (°C), T_{in} is the Temperature of coolant entering the core (°C).

The parameter values of the reactor model are given in Table 1 as follows:

In Table 1, kinetics parameters for the fast spectrum neutrons such as prompt neutron lifetime and effective delayed neutron fraction were taken from Walter et al. (2012) which are consistent for all the fast nuclear reactors and the integral reactivity parameters A, B, C and detailed reactivity coefficients of fast reactors depend to the nuclear reactor type and fuel type which in this paper they were taken from Kim and Taiwo (2012) for ABR-1000 with oxide fuel which is a fast nuclear reactor. Therefore, kinetics parameters and reactivity parameters are consistent to each other.