Research on Modeling and Control Method of Nuclear Power Plant Based on Gas Turbine

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Abstract: Many designers have proposed advanced nuclear power plant solutions with many different designs. However, modeling and control strategies for handling load following operations are still lacking. This study explores possible modeling methods and load tracking control strategies for gas turbine nuclear power plants. With a view to providing an evaluation method for conceptual design, and studying the frequency control strategy and average temperature control mechanism of nuclear power plant tracking load, the suitability of the control strategy and conceptual design was evaluated by a linear stability analysis method. An advanced molten salt reactor, the concept is taken as an example, and the numerical results of the nuclear power plant system are given to verify the correctness and effectiveness of the proposed modeling and load tracking control strategy.

1. Introduction

Currently, several advanced reactor concepts on the market are being developed to enhance the role of nuclear energy supply, including manageable nuclear waste, efficient fuel utilization, increased environmental benefits, competitive economy, recognized safety performance, and assurance of nuclear energy systems and the safety of nuclear materials [1]. GIF selected the six most promising systems (fourth-generation reactors) were for further research to address these challenges and provide new products such as hydrogen, compared to renewable energy sustainability, reduction of processed nuclear waste and improvement of anti-proliferation capability [2]. These six systems are: gas-cooled fast reactor (GFR), ultra-high temperature reactor (VHTR), supercritical water-cooled reactor (SCWR), sodium-cooled fast reactor (SFR), Lead-cooled fast reactors (LFR) and molten salt reactors (MSR) [3]. In addition, small modular reactors (SMR) designed based on current advanced reactor and fourth-generation reactor concepts can economically meet the energy needs of many countries in the world, applicable in remote/isolated areas, suitable for specific applications (such as seawater desalination or heating) [4]. The practical and effective use of the above nuclear reactor concept depends on sharing nuclear power with other power sources on the same power grid, which requires nuclear power plants to have load tracking capabilities or automatic plant-wide control power generation capacity [5].

The goals of an effective plant-wide control system include automated control, safe and smooth process operation, and high-quality control in the face of disturbances [6]. To achieve these goals, nuclear power plants can have basic loads or open loops (no cycles) and load following or closed loops (Loop) configuration [7]. Although these two structures have advantages and disadvantages, modern design methods are based on closed-loop structures. Open-loop structures mean that the units are arranged in series without cycles, so that the control problem of the whole plant can be effectively decomposed into control of the operation of each unit. Because there is no feedback effect, the operation of a single unit controls the dynamic behavior of the object, and the only path of interference propagation is a linear path along the process [8]. On the other hand, in a closed-loop structure, due to the existence of feedback significantly changes the dynamic and steady-state behavior of the controlled object, making the control problem of the entire object much more complicated. The main feedback effect of the controlled object is described by the overall time constant change and snowball effect. Feedback, the overall time constant may be different from the sum of the time constants of the individual units [9]. The snowball effect means that small changes

in plant inputs Plant-induced yields vary greatly due to propagation around the feedback loop. These characteristics limit the control configuration of integrated power plants. Therefore, a plant-oriented approach based on heuristic rules based on plant understanding and experience is used in nuclear power plants effective plant-wide control is necessary.

In power plants, power supply and demand must be balanced by power generation or load, because the energy storage provided by the transmission system can be ignored [10]. The method of balancing power supply and demand is to use load frequency control to provide inherent load tracking capabilities. Currently the nuclear reactor technology has sufficient regulation margin and responsiveness of load tracking operation. The problem here is that due to the slow response of the unit, it is often difficult for the thermal power generation system to track the load. Therefore, controlling the dominant object variable through the local unit controller can provide a universal solution. To handle the overall feedback effect, complex objects require simple control strategies.

The main purpose of this paper is to provide a global modeling and control method for advanced gas turbine nuclear power plants. Most of the nuclear power plant modeling methods in the literature are based on mass-energy-momentum balance (mixed model or two-fluid model), so they are quite complicated. Many assumptions and calculations of many non-measurable parameter values are required. On the other hand, the modeling method of this study is based on a centralized parameter model with simple and convenient control, that is, the first principle and physical laws of the dominant system and the input/output model, and carry out simulation research. Based on the proposed modeling method, considering reactor temperature control and turbine generator system load frequency control, a control strategy for load tracking of nuclear power plants is proposed. A simple and effective control strategy is designed for the next generation nuclear power plant. And analyzed the overall stability of different plant-wide control strategies.

2. Dynamic modeling of nuclear power plants

The most advanced dynamic modeling of a nuclear power plant requires knowledge, approximation, and realistic interpretation of the dynamic behavior of its subsystems. The modeling method for this study is based on a lumped parameter model that includes several assumptions, such as critical reactivity, transients around incompressible flow, constant mass flow rate, constant heat capacity, and heat transfer coefficient. Dynamic models will be developed based on the reference nuclear power plant shown in Figure1, but these models will be applicable to different nuclear reactor designs, which may be included and may not include secondary and tertiary circuits. The nuclear power plant consists of nuclear reactors, heat exchangers, and power conversion devices that use the closed-loop Brayton cycle recommended by most high-temperature reactors. Used to control and simulate the nuclear power plant under consideration the dynamic model is given in the following sections.
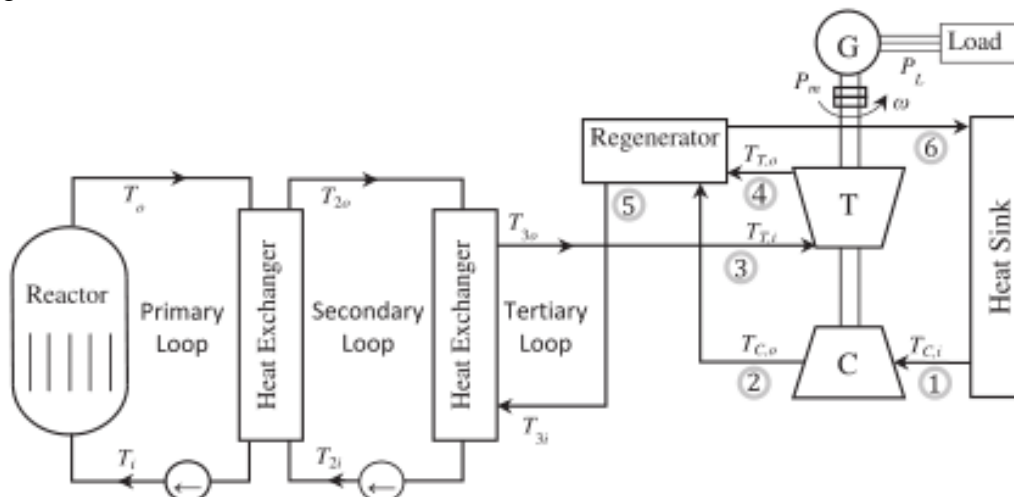

Figure. 1 A simplified scheme of an advanced nuclear power plant concept

2.1 Mathematical Modeling of a Nuclear Reactor

Nuclear reactors are modeled by coupling neutronics and thermodynamics to capture the physics complexity of nuclear reactors. Figure2 shows the reference scheme for nuclear reactor modeling. Nuclear reactors are modeled by neutron and thermals, temperature controllers, and control rods. The drive mechanism (governor) and hot/cold leg (for considering the dynamic characteristics of the pipeline) model.

The neutron flux or reactor power is calculated based on the point reactor kinetic equation (PRK), which assumes that the reactor transients are analyzed under slightly subcritical or supercritical reactor conditions:

$$\frac{dP(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda}P(t) + \sum_{i=1}^{6}\lambda_i C_i(t)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda}P(t) - \lambda_i C_i(t)$$

For a fuel cycle (such as MSR) reactor, the residence time of the fuel outside the core should be included in the neutron model. Based on this, a modified PRK equation for the fuel cycle reactor (MSR) is given:

$$\frac{dP(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda}P(t) + \sum_{i=1}^{6}\lambda_i C_i(t)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda}P(t) - \lambda_i C_i(t) - \frac{1}{\tau_c}C_i(t) + \frac{1}{\tau_c}e^{-\lambda_i \tau_L}C_i(t - \tau_L)$$
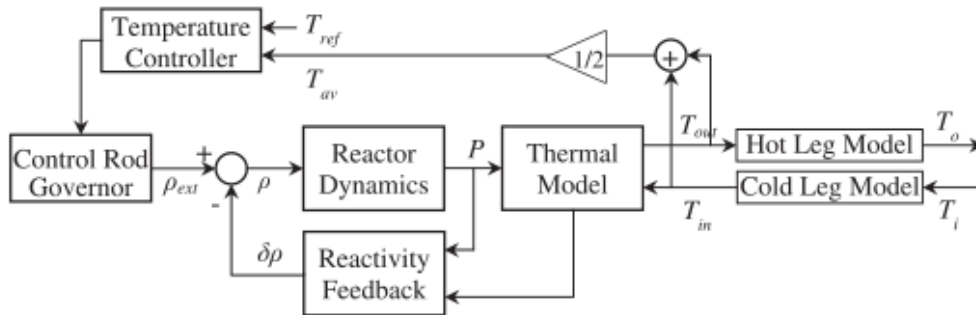

Figure. 2 Typical representative of nuclear reactor system and its average temperature control mechanism

As we all know, the PRK equation can be simplified by a group delayed neutron method into two first order differential equations. This can be an option in numerical simulation because reactive feedback makes PRK non-linear. Temperature and density's effect of change is the main consideration for reactive feedback. Due to the relationship between density and temperature in a fluid, changes in density can usually be expressed as a function of temperature, so total temperature reactive feedback can be considered.

In order to deal with the dynamic characteristics of a piping system where fluid loses energy due to friction, the temperature model of the hot and cold arms can be included in the system modeling. The dynamics of the piping together with the control valve can play a role of variable resistance and simple capacity leading to time delays in energy transfer systems. The hot-arm and cold-arm models can simply be represented by the next-order differential equation:

$$\tau_{hl}\frac{dT_e}{dt} = -T_0 + T_{out} \tau_d \frac{dT_i}{dt} = -T_i + T_{in}$$

The control rod governor (control rod drive mechanism) adjusts the position of the control rod group in the reactor core to obtain small incremental reactivity changes at each step. Although the reactivity control mechanism can be composed of unconventional methods such as carbon reflector

or gasket gas, but in this study, all types of reactive control systems are considered as control rod regulators. Control rod governors can be simply modeled with first-order differential equations as:

$$\tau_{hl} \frac{dT_e}{dt} = -T_0 + T_{out} \tau_d \frac{dT_i}{dt} = -T_i + T_{in}$$

2.2 Modeling of Flow Control System

The coolant flow at rated power is controlled by the reactor coolant pump (or flow control valve). More specifically, the total mass flow is calculated and compared with a reference flow, and a flow controller is designed to operate the flow control system. When the coolant is operating at rated power, the total mass flow through the reactor is given by:

$$\dot{m} = \frac{P_{th}}{C_p \Delta T}$$

Nuclear power plants usually use motor-driven centrifugal pumps to adjust the speed of mass flow. As shown in Figure3, the pump curve provided by the manufacturer shows the performance of the pump, namely head (H), flow (_M), and power (P). In pumping applications, variable frequency (or speed) drive is an effective flow control method used with throttling and bypass methods. The torque-power relationship is given by P=2pNM/60, where P is the power (in watts) the power characteristics of the pump are determined by the pump curve and the affinity law. The affinity law states that for a given pump, the flow is directly proportional to the speed, the head is directly proportional to the square of the speed, and the required electrical power is proportional to the square of the speed Proportional.
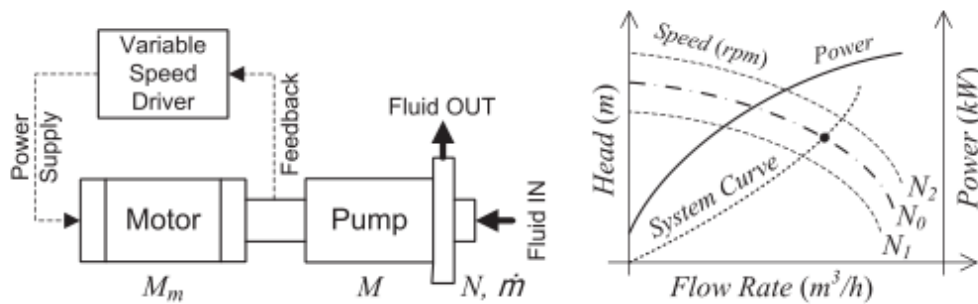

Figure. 3 Reactor coolant-pump system and pump curve

3. Control system design

A typical gas turbine model consists of three control loops: load frequency control, temperature control, and accelerator control. The main control loop under normal operating conditions is load frequency control. Other control loops, temperature and accelerator control, are active under abnormal conditions. Acceleration control was used during the start-up process, which was ignored in this study. The reactive input is limited by its predefined maximum and minimum limits. The control mechanism of the gas turbine nuclear reactor is shown in Figure4. The "low value selection" module is used. In order to select different reactor power control signals, such as power control and average temperature control. Load frequency control is effective under normal operating conditions and uses speed deviation as a control input. On the other hand, when the temperature of the reactor exceeds the specified limit, the temperature control will act.
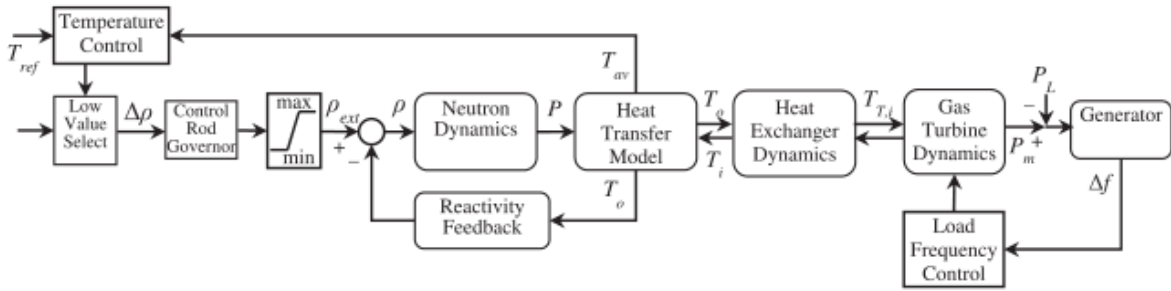
Figure. 4 Simplified diagram of nuclear power plant and load following control mechanism

3.1 Load Frequency Control

When the power balance is completed under steady-state conditions, the frequency specifications are met. Any differences in power balance are immediately responded by changes in frequency (speed). Load frequency control is the main control mechanism under normal operating conditions. Measurement speed and reference speed deviation between speeds is the input of load frequency control. Many complex load frequency control methods have been studied in the literature. Figure5 shows a typical load frequency control structure of a power plant. The controller consists of a primary (proportion) and secondary ((Integral) control loop.
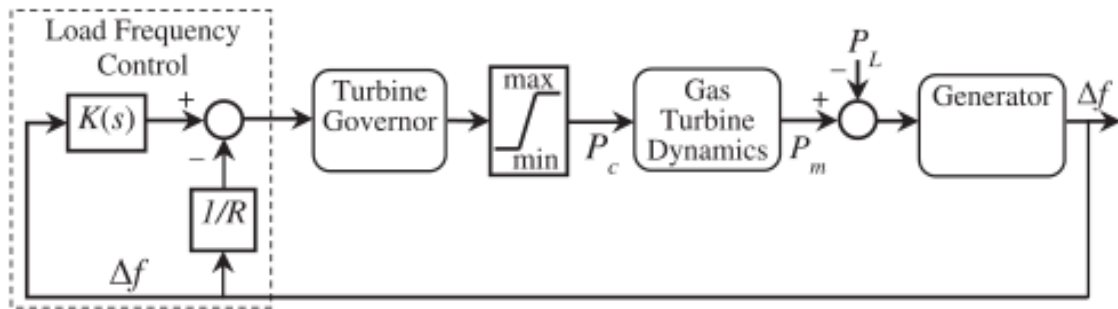

Figure. 5 Load frequency control mechanism

The main control loop adopts the speed reduction R (unit: Hz/MW), which is proportional to the generator output power. The speed reduction is a governor function, which decreases the governor reference speed as the load increases. Permitted the speed range is 95-107% of the rated speed, which means that in nuclear and thermal power plants, the typical value of R is about 4% of the rated generator frequency. The secondary control loop for load frequency control can designed as integral control to handle steady-state frequency tracking errors. By comprehensively considering these controllers, it is clear that the load frequency controller is a proportional-plus-integral (PI) controller, as shown below.

$$\Delta P_c(t) = -\frac{1}{R}\Delta f(t) - K_i \int_0^t \Delta f(t)dt$$

3.2 Temperature Control

Temperature control is used in the gas turbine control mechanism to limit the gas turbine output temperature to the highest predetermined temperature. In nuclear power plant control, when the reactor temperature exceeds a constant maximum, temperature control is used to control the reactor output temperature. When the load demand increases, due to with load frequency control, the output power will increase, which will increase the reactor temperature. If the temperature is higher than the maximum rated output temperature of the reactor, the temperature control output will be lower than the load frequency control output. Therefore, the temperature control action will take over control action.

The temperature control can also be designed as a pi controller, used to control the deviation between the reference temperature and the measured temperature. The pi controller for temperature control (represented in the Laplacian domain in Figure6) is given by:

$$\Delta\rho(t) = K_p e(t) + K_i \int_0^t e(t)dt$$
$$e(t) = T_{ref} - T_{av}$$

4. Analysis of simulation results

In this study, the concept of MSR was used to control and simulate advanced nuclear power plants. Since the experiments at the Oak Ridge National Laboratory Molten Salt Breeder Reactor (MSBR) and preliminary calculations of the kinetics and dynamic characteristics of the new MSR concept. It has been shown that since MSR has high controllability and security, MSR is one of the most promising concepts. Several MSR systems are being studied, including deep-burning MSR, thorium MSR, small MSR, and advanced high-temperature MSR. In MSR, the fuel is dissolved in fluoride or chlorine salt coolants, which are circulated in a primary circuit. Fuel temperature, fuel salt density, fission product poisons and delayed neutrons are the main factors affecting reactor dynamics during MSR operation. The MSR concept has several attractive features: strong negative Doppler feedback, negative expansion reactivity feedback for molten fuel fluoride/chloride fuel, and negligible xenon concentration levels due to the use of the injection system. These attractive features can develop an efficient automatic load tracking control mechanism.

Molten salt acridine element recovery and deformer (Mosart) system was selected as an example of an advanced nuclear power plant to work with. Figure1 shows one possible arrangement of a simplified Mosart plant. The reactor core is designed to provide 2400 trillion thermal energy from the combustion of uranium in watts. The fuel salt is a mixture of molten 15LiF-27BeF2-58NaF (mol%) and indium trifluoride. It is used to transfer heat from the primary circuit to the secondary circuit and limit the secondary release of plutonium salt is NaF-NaBF4. So far, the power conversion unit of the Mosart system has not been designed, but a closed-loop Brayton cycle power conversion system is a natural solution. For numerical simulation based on MATLAB/SIMULINK, the main parameters of the Mosart concept as Table1 shows. Table2 shows the delayed neutron precursor parameters of the Mosart concept calculated and verified by the MCNP5/MCNPX program through the END /BV and ENDF/B-VI nuclear databases.

Table. 1 Basic parameters of Mosart system

Core diameter/height (m)	3.4/3.6
Thermal/electrical power (MW)	2400/1200
Core fuel salt mass flow rate, $\dot{m}$ (kg/s)	10,000
Core fuel salt mass, m (kg)	69,946
Fuel salt density, d (kg/m^3)	2.140
Heat capacity of fuel salt, c_p (J/kg/°C)	2.087
Core transit time, τ_c (s)	7
Loop transit time, τ_l (s)	3.94
Total temperature coefficient, α (pcm/°C)	−3.86

Table. 2 Delayed neutron data

Group	Decay constant, λ_i (1/s)	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, $\lambda\Lambda$ (s):			8.3e−6

   The numerical simulation results are shown in Figure6 and Figure7. As shown in Figure6, the average temperature control system makes the average reactor temperature reach its reference temperature value of 665C in a short time through the control rod driving mechanism. The total reactivity margin of the MSR is equivalent to adding a reactivity of about 1000 PCM (or 0.01 Dk/k) over a period of several minutes. As shown in Figure6, the reactivity input is about 0.01 Dk/k margin. Because the mass flow is constant, so reactivity, power and average temperature remain constant after a short transient. In Figure7, the response of the load frequency control system is shown. When the load power changes, the load frequency control is achieved by the steam turbine regulator. During the load demand change, the generator frequency remains at its reference value of about 60 Hz. As the load on the slope changes, the generator frequency has a constant but small steady-state error. It can be observed that the change in load demand has no significant effect on the average temperature control system. This can be explained by the fast response of the local unit controller and the slow response of the thermal system (low-pass filter characteristics of the thermal system).
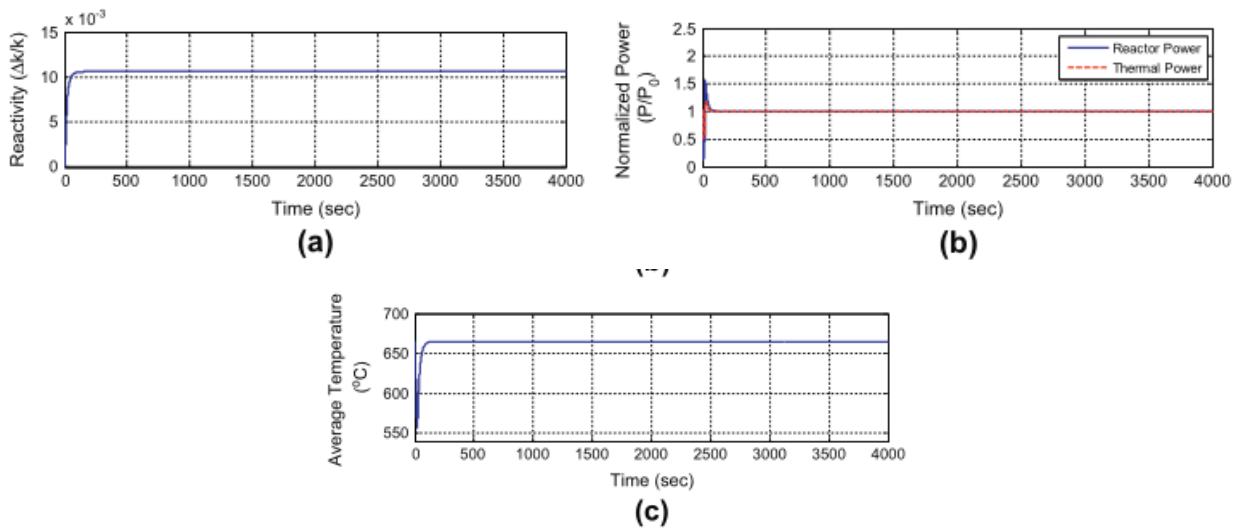

Figure. 6 Time response of the average temperature control system, (a) reactivity provided by the control rod system, (b) normalized reactor power and extracted thermal power, and (c) time response of the average temperature
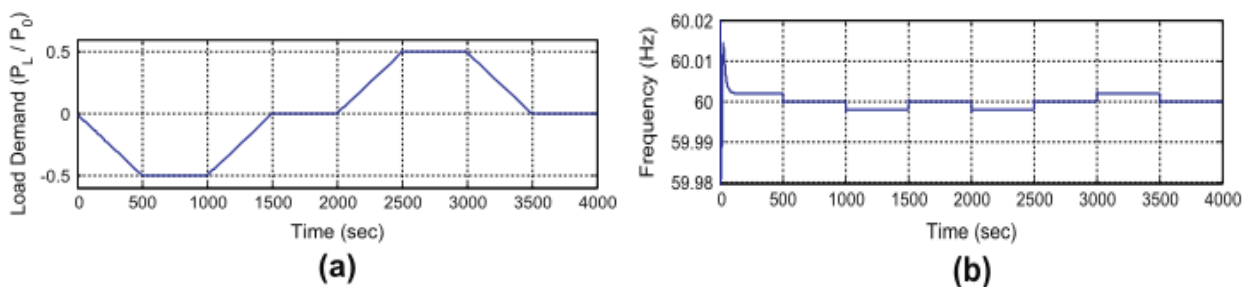

Figure. 7 Time response of load frequency control system, (a) load demand, (b) frequency change.

5. Conclusion

This paper studies the dynamic modeling and control of gas turbine nuclear power plants. The purpose of this research is to determine the impact of load tracking control strategies on the stability of advanced nuclear power plants. Through simple dynamic models and mature controller design, the frequency of nuclear power plants following loads is given control and temperature control strategies. The complexity of nuclear power plant design (i.e. the number of coolant circuits) and plant-wide control strategies (i.e., centralized reactor control or local unit controllers) have a significant impact on the stability and response time of nuclear power plants. Studies have shown that the effect of load following in the control of the pants width system is twofold:

(1) Load-tracking control of multiple coolant circuits in a power plant directly centered on a nuclear reactor results in low stability margins and long total time constants.

(2) Local unit controller improves stability and overall time constant

In addition, the proposed modeling method eliminates the limitations of complex hybrid or two-fluid models, allowing us to design effective control systems and provide appropriate stability analysis. Since many components of the power plant are similar, the proposed modeling and control methods can be applied to a variety of nuclear power plants, research reactors, and space nuclear reactors with a few modifications. This research can be extended to benchmark verification and model validation, and interconnected power systems.

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