CONTROL OF LARGE NUCLEAR REACTORS BY STATE AND OUTPUT FEEDBACK TECHNIQUES

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Contents

Summary

This chapter presents the modeling and control of a large nuclear reactor by state and output feedback control techniques. Different types of nuclear reactors are now well established for production of energy which is economically competitive with energy from other conventional and nonconventional sources. The major challenge in production of nuclear energy is the design of the reactor and its control system so that fission reactions are controlled as desired and any possibility of occurrence of unsafe situations should be minimized. Hence, the control of nuclear reactors represents an important aspect of nuclear reactor design. The chapter begins with a review of important physics aspects of fission reactions taking place in a reactor. The well known point kinetics model is introduced and a space–time kinetics model is derived for large thermal reactors. The space–time kinetics model has been represented in standard form for accomplishing the task of controller design based on state and output feedback. It has been shown that the application of two time–scale methods simplifies the design problems considerably. The design techniques have been applied to a Pressurized Heavy Water Reactor.

1. Introduction

In 1938, Otto Hahn and Fritz Strassmann discovered the process of nuclear fission. They showed that bombarding the nucleus of the uranium atom with neutrons changes some of the uranium into barium. In 1939, Austrian physicists Lise Meitner and Otto Frisch recognized Hahn's experiment as the splitting of the uranium atom, and named the process nuclear fission. Later in 1942, a group of noted physicists, headed by Enrico Fermi, directed the construction of the world's first successful nuclear reactor, built at the University of Chicago, Illinois. It produced the first artificial fission chain reaction. This experiment opened up the prospect of an entirely new source of power, utilizing the internal energy of the atomic nucleus. In 1951, the Experimental Breeder Reactor I in Idaho became the first reactor to produce electrical power. Today, nuclear fission has been established as a practical means for production of energy which is economically competitive with energy from fossil fuel.

2. On Certain Preliminaries on Nuclear Reactor

2.1. Components of a Nuclear Reactor

A nuclear reactor is a device designed to convert the energy released by a controlled nuclear–fission chain reaction into heat energy which can be used to generate electricity. Nuclear fission takes place in a heavy tanklike structure called the reactor vessel, which is designed to withstand a high pressure, chosen according to thermohydraulic aspects of energy transfer taking into account the neutronic fluence and the embrittlement of the vessel material during different startup and shutdown conditions of the reactor. The several different reactors operational today, have different design types, sizes and objectives but all share in common a reactor vessel with a central part called the core, which contains the fuel, the moderator, the reflector and the coolant.

The fuel most commonly used in a reactor is uranium, natural having the $^{235}\mathrm{U}$ concentration of about $0.7\,\%$, or enriched, having $2-4\,\%$ $^{235}\mathrm{U}$. In some reactors artificially produced $^{233}\mathrm{U}$ and $^{239}\mathrm{Pu}$, in mixed oxide fuel type assemblies, have also been utilized.

Each fission reaction is accompanied with generation of two or three neutrons, which have very high energy. If it is desired, as is often the case, that most of the fissions result from the absorption of slow neutrons, *i.e.*, neutrons having low energy, there must also be present a moderator. The function of the moderator is to slow down the high energy neutrons liberated in fission reactions, mainly as a result of elastic scattering reactions. Slow moving, low energy neutrons have a greater probability of causing fission than do fast moving, high energy neutrons. Thus a moderator is used to yield higher neutron economy. The best moderators are materials consisting of elements of low mass number with relatively little tendency to capture neutrons. Most nuclear reactors use ordinary, or light, water as moderator. Heavy water, a form of water that contains the hydrogen isotope deuterium, graphite, beryllium, and beryllium oxide are also used as moderators. The relative amounts and nature of the fuel and moderator determine the energies of most of the neutrons causing fission.

The core is surrounded by a neutron reflector of a material determined largely by the energy distribution of the neutrons in the reactor. The purpose of the reflector is to decrease the loss of neutrons from the core by scattering back many of those which would have escaped.

The fission reactions are also accompanied with release of tremendous amount of heat, which is removed by circulation of a suitable coolant. In addition to cooling fuel assemblies to prevent their overheating, the coolant also carries heat out of the reactor, to a site where it can be used to produce power. Although carbon dioxide gas and liquid metals, such as sodium, are sometimes used as coolant, most reactors use water, which is abundant and cheap. If the energy released in the reactor is to be converted into electrical power, the heat must generally be transferred from the coolant to a working fluid to produce steam or hot gas, which can then be used in a conventional turbine–generator system. In some reactors, water is boiled within the reactor core, so that fission heat is utilized directly to produce steam. In other reactors, the heat carried by the primary coolant is transferred to secondary coolant.

2.2. Reactor Types

All neutrons emerging from fission have potential to continue the chain reaction in ^{235}U, ^{233}U, and ^{239}Pu nuclei. However, the probability of fast neutron induced fission is less than that of thermal neutron induced fission. Therefore, majority of current commercial reactors rely on thermal neutrons, and consequently they are called thermal reactors. Some reactors, called fast reactors, rely on fast neutrons. Fast reactors have been designed and built, but these are mostly prototypes and are not operating in large numbers.

Among the different types of thermal reactors that have been designed and installed, Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR), Gas Cooled Reactor (GCR), and Pressurized Heavy Water Reactor (PHWR) technologies are most commonly employed. Most of the western countries, except Canada, have adopted the PWR and BWR designs. In Canada and a few other countries such as India, Pakistan, Argentina, and Korea, the PHWR design has been adopted. The various different technologies available have their relative merits and demerits and choice of a particular technology is governed by several factors.

The PWR and BWR principles consist of a vertical compact core using enriched uranium with cooling and moderation using the same light water fluid. In a PHWR, natural uranium is used as fuel while heavy water is used both as moderator and coolant. Besides, the PHWR employs the pressure tube concept, in which moderator is maintained in a physically separate circuit at a low temperature and pressure and the pressurized coolant flows in pressure tubes. The gas cooled reactor has a fixed moderator, usually graphite, and a gas coolant. The fuel is made of slightly enriched uranium oxide.

2.3. Neutron Balance

The major challenge in the production of nuclear energy is to sustain and control the nuclear fission. The chain reaction is sustained by balancing the production and loss of neutrons. The chain reaction has a multiplicative nature as each fission event induced by a single neutron can produce two or three neutrons. Therefore, the neutron population has the potential to grow exponentially in an uncontrolled chain reaction. When the balance between production and loss is reached in a nuclear reactor, the number of neutrons produced is equal to the number of neutrons lost for a given time interval and the system is said to be critical. The overall neutron population within the reactor is then maintained, and both the neutron flux and the resulting neutronic power are held constant. The control of the neutron population in a reactor is achieved by varying what is called the “reactivity” of the reactor core. Adding or subtracting reactivity contributions by the action of specially designed “reactivity devices” enables to increase or decrease the neutron population.

2.4. Reactivity Control

When a neutron is born in fission, it can, firstly simply escape from the reactor, *i.e.*, leak out from the system. If it does not escape, it starts to diffuse through the reactor core

and can, first, be absorbed as a fast neutron in various core materials. In most such absorptions the neutron will be lost. However, in a small number of fast neutron absorption in $^{238}\mathrm{U}$, fission occurs, making a positive contribution to the chain reaction and to the neutron economy.

If the neutron does not escape and is not absorbed as fast neutron, it will be slowed down in collisions with the moderator nuclei. As the slowing down process takes place, the neutron can, however, still be lost through leakage or absorption. If the neutron survives slowing down through the resonance region without being captured, and becomes a thermal neutron, it can leak out from the system as a thermal neutron, or it can be absorbed in a non–fuel material. Thus it is lost. If it is absorbed in fuel, thermal neutron will, in a certain fraction of such absorptions, induce fission and thus produce a new generation of neutrons to keep the chain reaction going.

In the nuclear reactor, at any instant of time, there are, of course, a large number of neutrons undergoing the processes just described. Whether the chain reaction is self sustaining, and to what degree, depends on the relative number of neutrons from one generation which succeed in giving birth to a new generation. Considering the number of neutrons in two successive fission neutron generations, the ratio of these numbers can be defined. This ratio is called the effective multiplication factor, k, and characterizes the chain reaction.

If the number of neutrons, leaking out from the system, is completely ignored, as would be the case in a reactor core of infinite size, the ratio of number of neutrons resulting from fission in each generation to those absorbed in the preceding generation is called as the infinite multiplication factor, k_∞, and the ratio, $P_{NL} \equiv \frac{k}{k_\infty}$, is called the nonleakage probability of the system.

A quantity closely related to the effective multiplication factor is the reactivity, denoted by the symbol, ρ and defined as

$$\rho = \frac{k-1}{k}. \qquad (1)$$

It is obvious that the reactivity is a measure of the deviation of the core multiplication factor from its critical value $k = 1$, which signifies that the neutron population in the system is constant, generation after generation. If $k > 1$, or alternatively if $\rho > 0$, a greater number of neutrons are born in every successive generation, *i.e.*, the chain reaction is more than self–sustaining. The neutron population would increase with time. The reactor is said to be supercritical. On the contrary if $k < 1$, or equivalently if $\rho < 0$, the chain reaction is not self–sustained. The reactor is said to be subcritical. In this case, the neutron population decreases with time.

Most of the time it is desired to maintain the reactor at steady–state. Then it must be kept critical. When it is desired to increase or decrease the neutron population, the reactor must be made supercritical or subcritical for some length of time until the desired neutron population is reached, after which criticality must be regained. This

refers to control of criticality.

Another condition of importance in reactor operation and control is the shut down in which the neutron population is very low and is not increasing with time. To shut the reactor down, subcriticality must be achieved and maintained. What must be avoided are the situations where the reactor is uncontrollably supercritical for too long a time period since a divergent power "excursion" may lead to overheating of the fuel and other core components.

Thus, it is important for effective reactor control that there be means of changing and controlling the reactivity of the system, which may be achieved by changing production, absorption and/or leakage. Most power reactors rely on one or more of the neutron absorption procedures, *e.g.*, by using solid, movable absorbers called control rods, by dissolving absorber material (soluble poison) in moderator or coolant, or by employing solid, fixed absorbers (burnable poison), which "burn out" gradually with neutron reactions over the life time of the reactor fuel. In a PWR, burnable poisons are used to obtain an optimum power distribution at the beginning of the fuel cycle and negative coolant reactivity coefficient.

From the relation $k = k_{\infty} P_{NL}$, it is obvious that the condition for criticality falls into two parts. The first is the evaluation of infinite multiplication factor which is a function of the materials, *e.g.*, fuel, moderator, coolant, structure, *etc.* of the reactor, and the physical arrangement of the materials. The second involves the determination of the nonleakage probability, which is largely dependent on the geometry, *i.e.* shape of the system. The nonleakage probability increases with size, for a reactor of given composition. Among different shapes of the same volume, the surface area of the sphere is the least. Hence, the nonleakage probability of a spherical reactor will be the highest but incorporating engineering features for control, safety, and fuel handling, *etc.* in spherical system would be difficult. Hence, a cylindrical geometry, which is the next best from the point of view of achieving high nonleakage probability, and which at the same time is convenient for incorporating the various engineering features, is adopted in almost all nuclear reactors.

2.5. Cross section for Neutron Reactions

The description of the interactions of neutrons with atomic nuclei can be made quantitative by means of concept of cross sections. If a given material is exposed to the action of neutrons, the rate at which any particular nuclear reaction occurs, depends upon the number of neutrons, their velocity, and the number and nature of nuclei in the specified material, *i.e.*,

$$\text{Rate of interaction} = \sigma N V n v \qquad \text{s}^{-1} \tag{2}$$

where n denotes the neutron density or the number of neutrons per unit volume, v is the velocity of neutrons, N is the atom density or number of nuclei per unit volume of specified material, V denotes the volume of the material and the symbol, σ, denotes the microscopic cross section of the material. The product $N\sigma$ is equivalent to the total

cross section of the nuclei per unit volume of the material, and is called the macroscopic cross section, often denoted by Σ. The product nv, denoted by ϕ, is called the neutron flux.

2.6. Fission Rate and Reactor Power

The concept of cross section discussed in the preceding section can now be applied to the fission process taking place in the nuclear reactor. Thus, the fission rate can be expressed as

$$\text{Rate of fission} = \Sigma_f V\phi \qquad \text{s}^{-1}\ , \tag{3}$$

where, $\Sigma_f = N_f \sigma_f$, is the macroscopic fission cross section and σ_f is the microscopic fission cross section and N_f the number of fissile nuclei per unit volume. The average energy released in thermal fission of ^{235}U is approximately 3.2×10^{-11} J. Hence, the power, P, of a nuclear reactor using ^{235}U as fuel is equal to $3.2\times10^{-11}\Sigma_f V\phi$ W. Generally speaking, the fission power in a nuclear reactor is expressed as

$$P = E_{eff}\Sigma_f V\phi \qquad \text{W}\ , \tag{4}$$

where E_{eff} is the average energy released in thermal fission of the fuel material in the reactor core. Thus, for a reactor of given active volume, the power is proportional to the product of the macroscopic fission cross section and the neutron flux. As the reactor operates, Σ_f, which is equal to $N_f\sigma_f$, must decrease somewhat, since there is a gradual decrease in N_f. For practical purposes, however, N_f and hence Σ_f, may be regarded as constant. Consequently, the power level of a reactor is proportional to the neutron flux. The power output of the reactor is thus altered by changing the neutron flux or neutron density.

2.7. Prompt and Delayed Neutrons

The neutrons released in fission, can be divided into two categories, namely prompt neutrons, which constitute more than 99% of the total fission neutrons and delayed neutrons. Prompt neutrons are released within 10^{-14} s (or less) of the instant of fission. The emission of the prompt neutrons ceases immediately after fission has occurred, but the delayed neutrons continue to be expelled from the fission fragments over a period of few hours, their intensity falling off rapidly with time. Experimental studies of the rate of emission of the delayed neutrons have shown that these neutrons fall into six groups, each characterized by a definite exponential decay rate. It is thus possible to associate a specific half life with each group.

2.8. Neutron Life time

The average time elapsing between the release of a neutron in a fission reaction and its

loss from the system by absorption or escape, is called the neutron lifetime. It consists of two parts, namely, the slowing down time , *i.e.*, the mean time required for the fission neutrons to slow down to thermal energies, and the diffusion time, also called the thermal neutron lifetime, *i.e.*, the average time the thermal neutrons diffuse before being lost in some way.

In an infinite medium, thermal neutrons are lost by absorption only and the thermal neutron life time can be expressed mathematically as

$$\ell_\infty = \frac{1}{v\Sigma_a} \quad \text{s} , \tag{5}$$

where Σ_a is the total macroscopic absorption cross section for thermal neutrons.

It is found that the average slowing down time in commonly used moderators is usually much less than the thermal neutron diffusion time. For this reason, it is a common practice to neglect the slowing down time and to refer to ℓ_∞ as the neutron lifetime in an infinite medium. In order to distinguish this from the effective lifetime, which takes into account the delayed fission neutrons, it is also called the prompt neutron lifetime.

In a system of finite size, the average lifetime is somewhat less that ℓ_∞ because some neutrons are also lost by leakage. In such a system, the average prompt neutron lifetime is expressed as

$$\ell = \ell_\infty P_{NL} \quad \text{s.} \tag{6}$$

2.9. Neutron Flux Measurement

Neutron flux in a reactor spans several decades. Flux in a fresh core prior to first criticality is approximately 10^{14} times lower than the flux in an equilibrium core at full power. Because of this large range, different types of instruments are required depending on the power level. These measurements are required both for control of reactor power and for reactor protection.

At very low power levels (below 10^{-6} of full power), power is measured by boron trifluoride (BF_3) counters located temporarily in-core or out-of -core, depending on the source range neutron flux and detector sensitivity. In some reactors a neutron source may deliberately be loaded in the core to bring the neutron flux in sensitive range of the detector. They are needed only for the initial reactor start-up. These highly sensitive detectors, called start-up counters, may be needed for restarts following extended reactor shutdowns as well as for the initial start-up. At power levels of about 10^{-7} times full power, the permanently installed out-of-core ion chambers start coming on scale. At 10^{-6} times full power, the ion chambers are well on scale and the start-up counters are reaching the end of their range. The ion chambers span a range of operation from 10^{-7} to about 150 % full power.

Since the ion chambers are measuring the thermal neutron flux in their vicinity, the power indication cannot therefore represent very accurately the total power generated in the reactor. Also, large fluctuations may occur in the ion chamber signals in local refueling operations or reactivity devices movement. Shutoff rod movement will also affect ion chamber signals. Moderator poison also depresses the ion chamber signals. Boron poison, dissolved in the moderator to compensate for fresh fuel, will decrease the ion chamber current for a given power level. Hence, it is necessary to calibrate the ion-chamber signal so that it corresponds with the reactor thermal power.

In large reactors, neutron flux is also sensed by several in-core detectors. In-core flux detectors, typically self–powered neutron detectors, are used during normal operation at high power levels. This type of detector is essentially a coaxial cable consisting of an inner emitter electrode, and an outer collector electrode separated from each other by an annular insulator. It is self-powered because it does not require an applied bias voltage to separate and collect ionization charge to derive a signal.

In-core flux detectors consist of vanadium, platinum, cobalt, or rhodium emitters. Vanadium and rhodium detectors are almost totally neutron sensitive but have a slow dynamic response, while platinum and cobalt detectors have a response which is mainly prompt. Platinum or cobalt detectors are therefore employed for the reactor control and protection systems, while the vanadium or rhodium detectors are used for neutron flux mapping.

3. Modeling of Nuclear Reactors

In a practical reactor system, the production, diffusion and absorption of neutrons take place at different energies, and for accurate description of the system, the various different processes should be modeled as continuous function of neutron energy but the mathematical treatment will be quite complicated. However, essential identical results to those obtained by detailed modeling, may be obtained by a one group model, which assumes that the production, diffusion, absorption, and leakage take place at a single energy.

3.1. Neutron Diffusion Equation

In the absence of a primary source, a noncritical system is not in a steady state and the neutron density, n, is changing with time, as given by

$$\frac{dn}{dt} = D\nabla^2\phi - \Sigma_a\phi + S, \qquad (7)$$

where D is called the diffusion coefficient and S, called the source term, is the rate of production of neutrons per unit volume. The above equation, generally referred to as diffusion equation, is basic to thermal reactor theory. In it, the first term on the right hand side expresses the balance of number of neutrons per unit volume per second due to leakage out of the volume and leakage into the volume and the second term gives the number absorbed per unit volume per second. S consists of two terms. The first, called the prompt neutron production rate, is equal to $\nu\Sigma_f\phi(1-\beta)$, where ν denotes the total

number of neutrons produced, on average, in the fission reaction, β denotes the fraction of the delayed neutrons; consequently $(1-\beta)$ denotes the fraction of prompt neutrons. The second term contributing to production, called the delayed neutron production term, is dependent on the rate of radioactive decay of various different delayed neutron precursors. Let $\bar{C}_i$ be the concentration of the ith kind of delayed neutron precursor, characterized by the decay constant, λ_i, then the total rate of formation of delayed neutrons is equal to $\sum_{i=1}^{m_d} \lambda_i \bar{C}_i$, where m_d is the total number of delayed neutron precursor groups. Hence, in Eq. (7), we have $S = \nu\Sigma_f \phi(1-\beta) + \sum_{i=1}^{m_d} \lambda_i \bar{C}_i$. Further, from the definition of k_∞ given in Section 2.4, it follows that $k_\infty = \frac{\nu\Sigma_f}{\Sigma_a}$ and therefore $\nu\Sigma_f$ may be substituted by $k_\infty \Sigma_a$ to obtain

$$\frac{dn}{dt} = D\nabla^2\phi - \Sigma_a\phi + k_\infty\Sigma_a\phi(1-\beta) + \sum_{i=1}^{m_d} \lambda_i \bar{C}_i. \qquad (8)$$

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Biographical Sketches

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