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Analysis of control rod drop accident in PWRs with multipoint kinetics method

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ABSTRACT

A major group of accidents in PWRs is related to improper operation of control rods. Malfunction in control and protection system (CPS) of control rods or operator's error can be a cause of such events. This paper presents a study of the drop of one control rod accident with multipoint kinetic approach. The main aim of this work is the introduction of multipoint kinetics method as an accurate model for accident analysis. In this way, a coupled dynamic model based on point and multipoint kinetics formulation is developed for transient analysis. Also for reactivity feedback calculation, a simple thermal hydraulic simulation of the core is employed. To illustrate the capability of the proposed scheme, Bushehr Nuclear Power Plant, BNPP, which is a WWER-1000 reactor, is chosen and multiples of the 1/6 symmetry slices of the core is considered as nodes. Thus, the reactor core is divided into two, three, and six nodes in each case study, and the drop of one control rod accident, CRDA, is simulated. These results demonstrate that the multi point kinetics model can reflect not only the average information but also the distribution information of the core such as the spatial power, fuel temperature and coolant temperature distribution in the core. In addition, the accuracy, simplicity, and fast computation speed of the multipoint kinetics model make it a reliable method for simulation and design of the control system for WWER-1000 reactors.

1. Introduction

Accident analysis is an important tool for confirming the adequacy and efficiency for the safety of nuclear power plants (NPPs). The most essential safety analyze is to consider anticipated operational transients and postulated accidents in the design of NPPs. Reactivity-initiated accidents (RIAs) consist of postulated accidents which involve a sudden and rapid insertion of positive and negative reactivity. These accident scenarios include a control rod ejection (CRE) and a control rod drop accident (CRDA) for PWRs. Both CRE and CRDA relate to improper operation of control rod and these are in postulated accidents. Malfunction in CPS of control rods or operator's error can be a cause of such events. This type of accidents has been extensively studied and analyzed until now. For instance, Simulation of rod ejection accident in a WWER-1000 Nuclear Reactor by using PARCS code (Noori-Kalkhoran et al., 2014); simulation of a control rod ejection accident in a VVER-1000/V446 using RELAP5/Mod3.2 (Tabdar et al., 2012); transient analysis of tehran research reactor (Aghaie et al., 2012a); coupled neutronic thermal–hydraulic transient analysis of accidents in PWRs (Aghaie et al., 2012b); REA analysis of the Iranian VVER-1000 core by modification of PRORIA code for annular pellets (Altaha and Pazirandeh, 2011) were published in past years.

Reliable transient analysis of nuclear reactors is important to understand the effects of events on nuclear reactor safety (Norouzi et al., 2013). In addition, complete understanding of reactor behavior is necessary to control of the nuclear reactor during the accidents and normal operation (Nazari et al., 2013). Thus estimating a reliable analysis method, which can predict safety margins and their associated uncertainties, is essential. Such a reliable analysis method should be capable to reflect the details of all interacting physical phenomenon like core neutron kinetic and thermal hydraulic phenomenon.

In this study, the CRDA has been simulated in the WWER-1000 (Bushehr Nuclear Power Plant—BNPP) reactor. The uncontrolled movement of single control rod into core is a CRDA that it results in a prompt negative reactivity insertion. It is an operational occurrence accident with postulated failure of reactor control and protection system which it can decrease local and global core power. So the development of coupled thermal–hydraulic and neutron kinetics code is an important step to do best estimate calculations for plant transients. In the work, regarding the spatial effects of this accident, the multipoint kinetics method has been implemented and the accuracy, simplicity, and fast computation speed of this

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Nomenclature

A	the interface area (m^2)
C_{cp}	specific heat of coolant
C_f	specific heat of fuel
C_k	k'th group of delayed neutron precursor
D	diffusion coefficient
E_{eff}	energy released per fission
N	the number of regions
P	power (MW)
R	thermal resistance between fuel and coolant
T_c	coolant inlet temperature (°C)
T_f	fuel temperature (°C)
V	region volume
W	coolant mass flow rate (kg/s)
d	the distance between regions (m)
l	prompt neutron life time (s)
m_f	mass of fuel (kg)
m_c	mass of coolant (kg)
m_d	number of delayed neutron precursor groups

Greek characters

α	coupling factor between regions
β	delayed neutron fractional yield
δ	differential singe
λ	decay constant
ϕ	neutron flux
Σ_a	macroscopic absorption cross section
Σ_f	macroscopic fission cross section

Subscripts and superscripts

c	coolant
f	fuel
eq	equilibrium
i	region number
in	input
o	output
pu	perturbation

method is illustrated. Also, this is shown that the multipoint kinetics could determine the spatial power, fuel and coolant distribution in the reactor core under steady state and transient operating condition. Finally, the sensitivity and accuracy of method against number of nodes is evaluated and results compared with Final Safety Analysis Report (FSAR).

2. Multi-point kinetics method

The main problem in reactor analysis is the determination of the spatial power distribution in the reactor core under steady state and transient operating condition. The behavior of nuclear reactors is described by the transport equation (Duderstadt and Hamilton, 1983); but numerical solutions of the coupled time–space dependent transport and delayed neutron precursors' equations for reactor kinetics problems are difficult. Therefore, attempts are to use the simpler methods; since mean free paths of neutrons are fairly long and since the lifetimes of neutrons in a reactor are quiet short, the effect of local perturbation on neutron flux will quickly spread throughout a reactor. The immediate consequence of perturbing a reactor locally is thus a readjustment in the shape of flux. In many cases this readjustment is slight and is completed in a few milliseconds; after that the readjusted shape rises or falls as a whole depending on whether the initial perturbation increased or decreased K_{eff}. For reactors in which transients proceed in this manner, merely being able to predict the change in the level of the flux is sufficient to permit a very accurate prediction of consequences of perturbation. Thus, instead of having to face the very difficult problem of solving time dependent diffusion equation in full detail a simple set of equations that specify how the overall magnitude of flux changes with time is used for specified regions, multi-points kinetics equations, on using the above assumption the systems' dynamics for power is then may be represented by (Shimjith et al., 2010):

$$\frac{dP_i}{dt} = -\alpha_{ii}\frac{P_i}{\ell_i} + \sum_{j=1}^{N}\alpha_{ji}\frac{P_j}{\ell_i} + (\rho_i - \beta)\frac{P_i}{\ell_i} + \sum_{k=1}^{m_d}\lambda_k C_{ik} \quad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta}{\ell_i}P_i - \sum_{k=1}^{m_d}\lambda_k C_{ik} \quad (2)$$

$$P_i = E_{eff}\Sigma_{fi}V_i\phi_i \quad (3)$$

$$\alpha_{ij} = \frac{D_j A_{ji}\Sigma_{fj}}{d_{ij}\Sigma_{fi}V_i\Sigma_{aj}} \quad (4)$$

$$\alpha_{ii} = \frac{\nu\Sigma_{fi}P_i + \sum_{j=1}^{N}\frac{D_i A_{ij}\Sigma_{fi}}{d_{ij}\Sigma_{fj}V_j}P_j - \Sigma_{ai}P_i}{P_i\Sigma_{ai}} \quad (5)$$

where N, m_d, α_{ij} and α_{ii} are the number of regions, the number of delayed neutron precursor groups, the coupling between regions and the self coupling coefficient of the regions respectively.

Furthermore, P_i, ρ, C_{ik}, β, λ_k, Σ_a, Σ_f, ℓ_i, V, D, A, d and E_{eff} are power in region i, reactivity, k'th delayed neutron precursor group, delayed neutron fractional yield, decay constant, absorption cross section, fission cross section, prompt neutron life time, region volume, diffusion coefficient, the interface area, the distance between regions and energy released per fission respectively. In this method, a core is divided into N regions and an average quantity for each variable is defined for each region.

2.1. Thermal reactivity feedbacks

In a simple case, the lumped influences of fuel and coolant temperature are taken into account and for a system as illustrated in Fig. 1 (Anglart, 2011) the respective energy balance equations for fuel and coolant can be defined as it follows (Lewis, 1977; Tong, 1988).

$$m_{fi}C_{fi}\frac{dT_{fi}}{dt} = P_i - \frac{1}{R_i}(T_{fi} - T_{ci}) \quad (6)$$

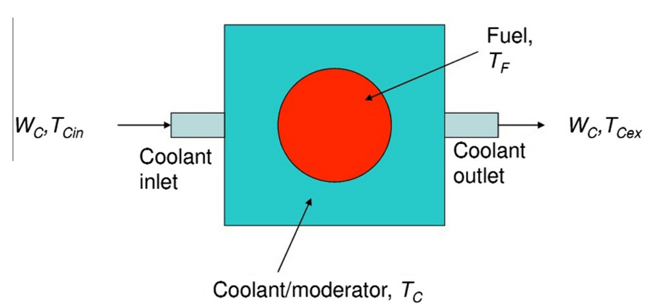

Fig. 1. Thermal hydraulic model of core.