Annals of Nuclear Energy 107 (2017) 82–88

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

A thermal neutronics coupling analysis method for lead based reactor core

Shuzhou Li, Jingchao Feng, Liankai Cao, Muhammad Shehzad Khan, Hongli Chen *

School of Nuclear Science and Technology, University of Science and Technology of China, Hefei, Anhui 230027, China

ARTICLE INFO

Article history:
Received 6 October 2016
Received in revised form 11 February 2017
Accepted 17 April 2017

Keywords:
Neutron kinetics
Thermal hydraulic
Coupling method
Sub-channel code

ABSTRACT

In this research paper, a sub-channel thermal hydraulic analysis code is coupled with the point reactor neutron kinetics model with six group delayed neutron. The coupling code is mainly used to perform the transient calculation of ADS/lead based alloy cooled fast reactor. The thermal hydraulic model is used for calculating temperature distribution profile and the feedback temperature information, providing input parameters for point kinetic model. This sub-channel analysis model can provide a new approach to solve the problem of one-dimension thermal hydraulic model and simulate the temperature distribution accurately. Furthermore the accuracy and reliability of calculated results are tested by another coupled code named FLUENT/PK and good agreements are achieved. To improve computational speed, one equivalent assembly is used to replace the whole core and the study shows that using of equivalent assembly which has the same average outlet temperature with the core obtained more reasonable results. The effects of fuel rods pitch diameter P/D ratio on simulation results are discussed. The code is capable to the quick calculations and safety analysis for reactivity accidents.

1. Introduction

Thermal hydraulic and neutronic simulations are of great importance in the design stage of a nuclear reactor core. The process of reactor shutdown and restart as well as the changing of core power can be achieved by introducing different reactivity, whereas inappropriate introduction may cause a reactivity insertion accident (RIA), and endanger the safety and reliability of reactor. Thus the research on dynamic response characteristics and the law of the variation of core power become an important task of reactor safety analysis (Chen et al., 2005). It is generally believed that the core power is in direct proportional to the neutron flux, thus the problem of core power calculation can be transformed into solving the neutron flux and its changing regularity with time and space dimension in the core. The fast reactor safety analysis codes typically adopts point reactor kinetics model to approximately depict the variation of core power (Hamidouche and Bousbia-Salah, 2010). The point kinetics model was derived based on the assumption that the spatial distribution of neutron flux does not change with the time. This assumption is reasonable and feasible if the effective multiplication factor is not far from critical state and disturbance is feeble. Compared with pressurized water reactors, fast reactor core configuration can be more compact, in which the time and space variations of neutron flux can be easily separated, making the point reactor model more applicable to fast reactors (Dapu and Hongqiu, 1995).

The point reactor kinetics model generally acquire a thermal hydraulic model to couple with, for the reactivity feedback is normally related to either coolant bulk temperature or the temperature of fuel rods. Many institutions have been committed to coupling method research and several coupled codes have been developed in the purpose of transient reactor safety analysis. Most of these coupled codes are applied on specific cases (Khan et al., 2013). System codes such as RELAP and ATHLET are frequently adopted in coupling technology research due to the computational stability and the modifiability of their source codes (Aghaie et al., 2012). One dimension thermal hydraulic model is generally adopted by these system codes, in which the effect of lateral mixing may not be taken into account.

In this paper, a sub-channel code named KMC-sub (Written in the C language, developed for lead base alloy cooled fast reactor) is applied as the thermal module, which is coupled with point reactor model. The approach utilized in this research to couple thermal model and kinetics model is serial integration coupling method, the source codes of KMC-sub is modified, point kinetics model is embedded as a subroutine. Another coupling approach is parallel coupling, this approach was adopted by some of the current cou-

* Corresponding author.
E-mail address: lszhrb@mail.ustc.edu.cn (H. Chen).

http://dx.doi.org/10.1016/j.anucene.2017.04.021
0306-4549/

pling investigation, for instance, the RELAP5-PARCS coupling code (Salah et al., 2006), TRACE-PARCS coupling code (Xu et al., 2009), the feedback influence from reactivity coefficients was discussed, in these cases, the thermal hydraulic code and kinetics code are interfaced but execute their own function separately, the calculation data are exchanged during simulation process. Some other similar coupling effort are TRAC-BF1/NEM/COBRA-TF (SolıS et al., 2002), COBRA-WIMS (Zare et al., 2010). This method, though popular, have a relatively low efficiency and cannot provide 3D analysis. The independently development of KMC platform make it possible to develop an efficient and accuracy coupling method. In current research, a small lead cooled reactor was considered, algorithm of transient simulation is presented and the effects of fuel rods pitch diameter ratio P/D on simulation results are discussed. The models and the calculated results are discussed both quantitatively and qualitatively in this research paper.

2. Models and coupling method

2.1. Thermal hydraulics model

Heat transfer in reactor core is quite complex. It might occur through several different ways (e.g. heat conduction, convection, radiation). The energy from fuel rod is transferred as heat flux throughout the solid materials and coolant. Detailed analysis models combining the local bulk temperatures with coolant flow are required for accurate determination of associated system properties. Most of the major aspects of thermal hydraulic simulation are well-defined such as turbulent cross flow between fuel rods and channels (Rosenkra, 2012). The thermal hydraulic model adopted in this work is based on the KMC-sub code models. The rightness and feasibility of these models have been validated by experimental data (Pacio et al., 2016; Fontana et al., 1974; Lyu et al., 2016). The main difference between this code and traditional single channel analysis method is the existence of lateral momentum conservative model and turbulent mixing model. The details of these models are presented as follows.

2.1.1. Lead physical properties

In the present study, the thermal physical properties of liquid lead are all the functions of temperature, the main properties included in the coupled code are density, normal melting point, heat capacity, thermal conductivity, viscosity. Empirical correlations suggested in OECD/NEA handbook were adopted, formulas of these properties are presented in Table 1.

2.1.2. Friction resistance model

The Novendstern resistance model, which has been developed for hexagonal wire wrapped fuel rod assembly, was adopted for friction resistance calculation. This model was derived from bare rod friction correlations, the effect of wire and bundle were taken into account by using a modifying factor. Basic form of Darcy formula is:

$$\Delta P = fL\rho v^2/2D_H \tag{1}$$

If Reynolds number is smaller than laminar critical number:

$$f_L = 64/Re \tag{2}$$

If Reynolds number is larger than turbulent critical number:

$$f_T = 0.3164/Re^{0.25} \tag{3}$$

If the flow is in transition region:

$$f_{tr} = f_L(1-A)^{0.5} + f_T A^{0.5} \tag{4}$$

where $A = (Re - 400)/4600$

And if the resistance from wrap wire have been taken into account, f becomes $f_T * f_w$, where

$$f_w = (1.034/Re^{0.124} + 29.6(P/D)^{6.94}Re^{0.086}/(H_{wire}/(D_{rod}+D_{wire}))^{2.239})^{0.885} \tag{5}$$

where H_{wire} is wire lead length, D_{wire} is wire diameter.

2.1.3. Heat transfer model

The main research contents in this paper include the effect of fuel rods pitch diameter ratio on simulation results, for the heat transfer mechanism of liquid metal is quite different from that of water and other fluid (Wang et al., 2013), and the reactivity feedback for neutronic module originated from temperature profile. A lot of heat transfer correlations have been presented (Handbook of single-phase convective heat transfer, 1987). In most of the cases, Calamai correlation calculate the smallest heat transfer coefficient and the results are the most conservative (Mikityuk, 2009; Cheng and Todreas, 1986). All results presented in this paper were calculated by this correlation.

H.Calamai correlation:

$$\mathrm{Nu} = 4.0 + 0.16(P/D)^5 + 0.33(P/D)^{3.8}(\mathrm{Pe}/100)^{0.86}, \quad 10 \leqslant Pe \leqslant 5000, \quad 1.1 \leqslant P/D \leqslant 1.4 \tag{6}$$

where Nu is the Nusselt number, P is the pitch of fuel rods, D is the diameter of fuel rod, and Pe is the Peclet number.

2.1.4. Turbulent mixing model

The sub-channel equation system allows sub-channels to be connected with each other and thus the transverse momentum equation can be derived. A fully three dimensional physical situation may be represented with relative ease simply by connecting channels in a 3-D array, and turbulent mixing exist between two adjacent channels, this make the sub-channel model significantly different from one dimension single channel model and the porous zone model, the sub-channel model can give a detailed temperature distribution for both fluid and fuel rod, which make the feedback temperatures more precise and close to that of reactor core, the turbulent mixing model are embedded in energy conservation and axial momentum conservation equations, the model used in this study is:

$$w'_k = \beta s_k G_k \tag{7}$$

where w'_k is turbulent mixing mass velocity, β is turbulent mixing coefficient, s_k is contact width between adjacent sub-channels, G_k is the mean flow rate between adjacent sub-channels.

Table 1
Thermophysical properties of lead.

Thermophysical property	Correlation	Temperature range (K)
Density($kg \cdot m^{-3}$)	11441–1.2795·T	600.6–2021
Normal melting point(K)	600.6	
Heat capacity($J \cdot kg^{-1} \cdot K^{-1}$)	$175.1 - 4.961 \times 10^{-2} \cdot T + 1.985 \times 10^{-5} \cdot T^{-2} - 2.099 \times 10^{-9} \cdot T^3 - 1.524 \times 10^6 \cdot T^{-2}$	600.6–1500
Thermal conductivity ($W \cdot m^{-1} \cdot K^{-1}$)	9.2 + 0.011·T	600.6–1300
Viscosity(Pa·s)	$4.55 \times 10^{-4} \cdot \exp(1069/T)$	600.6–1200