Progress in Nuclear Energy 106 (2018) 20–26

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Coupling of neutronics and thermal-hydraulics codes for the simulation of reactivity insertion accident for LFR☆

Dongmei Yang, Xiaojing Liu*, Jinbiao Xiong, Xiang Chai, Xu Cheng

Shanghai Jiao Tong University, 800 Dong Chuan Road, Shanghai, China

A R T I C L E I N F O

Keywords:
Lead-bismuth eutectic cooled fast reactor
Thermal-hydraulics
Neutronics
Coupled simulation

A B S T R A C T

As one of the advanced reactors proposed by the Generation IV International Forum (GIF), Lead-bismuth eutectic (LBE) coolant cooled fast reactor has shown great advantages over the other reactors. To analysis this reactor, several steps are carried out covering the code development and the coupled analysis. Firstly, the physical property of lead-bismuth, heat transfer correlation, turbulent mixing and pressure drop models are implemented into the thermal-hydraulics code COBRA-YT, where the inlet flow rate boundary condition is calculated by Computational Fluid Dynamics (CFD). The neutronics model has seven neutron energy groups and six groups of delayed neutron precursors. The neutronics parameters such as cross-section and feedback coefficient are calculated by the DRAGON code with endfb7 library. Secondly, based on sub-channel code COBRA-YT and neutron code SKETCH-N, the coupled of neutronics and thermal-hydraulics code has been developed via Parallel Virtual Machine (PVM) software platform. COBRA-YT code performs the thermal-hydraulics calculation and transfers its results such as coolant density and fuel temperature to the neutronics code SKETCH-N to update the cross-section; then SKETCH-N carries out the neutron-physical simulation of the reactor and provides the power density to the thermal-hydraulics code COBRA-YT as boundary conditions. Finally, this coupled code platform is exploited to the lead-bismuth fast reactor design to simulate some transient and control rod withdrawal accidents. The results achieved so far indicates that an increase of the reactor power and cladding temperature occurs after the control rod withdrawal, but the thermal-hydraulics results exceed their design limits. Time step size effect and withdrawal velocity analysis are also discussed.

1. Introduction

This research is based on a pool type modular fast reactor SVBR-75/100, which is cooled by lead-bismuth eutectic (LBE) and used as propulsion system of Russia submarine (Ingersoll, 2015). This kind of reactor has an integrated design and all its equipment places in a monoblock which enhances passive heat removal during Loss-of-Coolant Accident (LOCA) type accidents. Besides, the lead-bismuth eutectic coolant has following features to secure inherent safety (Zrodnikov et al., 2008). Compared to sodium-based liquid metal coolants with low boiling temperature, the boiling temperature of LBE is 1943K at atmospheric pressure and the value can reach up to 2300 K at the high reactor core operation pressure. The high melting point makes the possibility of boiling and explosion owing to heat transfer crisis negligible. Meanwhile, the coolant possesses a more negative void reactivity effect than sodium. Besides, when coolant leakage or steam generation (SG) leak occurs, the chemical inertness of LBE avoiding reactivity with water and air is the most important advantage over sodium. Among the above advantages, LBE cooled fast reactor appears to be one of the most promising candidates for Generation IV reactors.

In the LBE cooled fast reactor, control rods (CRs) play more important roles in power adjustment and reactor control owing to no boron dissolved in coolant. So if control rods withdrawal accident occurs, a large positive reactivity will be inserted into the reactor active core and then significant reactor power burst occurrence has an obvious influence on the thermal-hydraulics parameters, such as the cladding temperature and coolant density etc.. In return, the variable parameters have an effect on reactivity due to the negative coolant density and fuel temperature feedback coefficient (Doppler effect). In nuclear power plant safety analysis, the interaction between reactor reactivity and thermal-hydraulics characteristic of the LBE cooled fast reactors can be effectively predicted by coupled neutronics/thermal-hydraulics codes, which contributes to getting less conservative safety limits and more realistic prediction of the physical phenomenon (Titouche Kouidri et al., 2015).

Due to the lack of experimental data, more accurate computational

☆ The 18th International Conference on Emerging Nuclear Energy Systems (ICENES 2017), Hefei, Anhui, China, 24–27 April 2017.

* Corresponding author.
E-mail address: xiaojingliu@sjtu.edu.cn (X. Liu).

https://doi.org/10.1016/j.pnucene.2018.02.023
Received 14 August 2017; Received in revised form 25 December 2017; Accepted 24 February 2018
0149-1970/

tools should be developed to update the reactor design and to make analysis work more efficient. The coupling work implemented by universities and research centers are as follows: the two-dimensional code extended SIMMER-IIIM (Kondo, 1992; Yamano et al., 2007) has been developed and verified on LBE-cooled Accelerator Driven Sub-critical System (ADS). Vazquez (Vazquez et al., 2012) has coupled the Monte Carlo code MCNP with the subchannel code COBRA-IV in order to analyze steady state and transient on an experimental LBE cooled fast reactor MYRRHA in engineering design stage. FDS team (Wang et al., 2015; Wu, 2016) has developed the Neutronics and Thermal-hydraulics Coupled simulation program NTC regarding to 80 MWth LBE-cooled XADS. In this paper, the coupling of neutronics and thermal-hydraulics codes has been performed via Parallel Virtual Machine (PVM) software platform (Geist et al., 1994). The platform, consisting of sub-channel code COBRA-YT and neutron code SKETCH-N, is exploited to the LBE cooled fast reactor to simulate some transient and control rods withdrawal accidents.

The paper is organized as follows. Section 2 gives the reactor core main design parameters. After that, the models in two codes and coupling process will be introduced in section 3. Section 4 shows the analysis results using the coupling simulation tool. Finally, a summary and conclusions are presented.

2. Core main parameters and layout

The modified SVBR-75/100 with square lattice main core parameters are shown in Table 1 (Zrodnikov et al., 2008). The reactor core consists of 57 homogenized fuel assemblies with a width of 20.834 cm. The thickness of top and bottom axial reflectors is 35 cm. The core is surrounded by a row of 20.834 cm homogenized reflectors made of stainless and LBE. Two fuel assembly designs depending on whether control rods are contained or not have been specified. To avoid a non-uniform power distribution due to the same fuel enrichment in the whole core, control rod assemblies are mainly placed in the central region. Each fuel assembly is composed of fuel rods arranged with a pitch of 13.596 mm square lattice. Fig. 1 shows the core radial and axial layout of one assembly for this calculation. The white zones refer to assemblies without control rods, the blue zones standing for the fuel assemblies with control rods and the grey ones are radial and axial reflectors.

Table 1
Main parameters of LBE-cooled fast reactor.

Designation	Value	Units
Thermal power	280	MW
Coolant	LBE (44.5%Pb, 55.5%Bi)	
Active core dimensions height/diameter	0.9/1.645	m
Number of fuel pin per assembly	212	–
pressure	0.1	MPa
Primary coolant		
-flow rate	11 760	kg/s
-Core inlet temperature	320	°C
-Core outlet temperature	482	°C
Fuel:		
-Type	UO_2	–
-Average U-235 enrichment	16.1%	wt.%
Fuel pin:		
-OD	12	mm
-Length	1638	mm
-Lattice type	Square	–
-Pitch-to-diameter ratio	1.133	–
Cladding:		
-Material	EP-823 (12% Cr)	–
-Thickness	0.4	mm
Control rod		
-type	B_4C	–
-B10 enrichment	90	wt.%

3. Models and coupling method

3.1. SKETCH-N model

The SKETCH-N (Zimin et al., 1999; Asaka et al., 2001; Zimin et al., 2001) code can solve neutron diffusion equations in Cartesian geometry for steady and kinetics problems with arbitrary number of neutron energy groups and delayed neutron precursors. This code can be a standalone code or an external neutronics module coupling with other codes. The geometry model utilized in neutronics calculation is the same as thermal-hydraulics one shown in Fig. 2. The active core is divided into 18 axial layers and each layer is with the height of 5 cm from bottom to top. In SKETCH-N, 2 different material compositions are specified with corresponding cross-sections, namely fuel assemblies with the same enrichment of U-235 and reflectors. The arrangement of control rods in control assembly is shown in the right of Fig. 2. Along the axial direction, the length of control rod is the same as the core active height.

In SKETCH-N, macroscopic cross-sections are generated by a lattice cell code DRAGON with endfb7 library (Marleau et al., 2008). This code simulate the neutron behavior by an octant of a fuel assembly shown in Fig. 3. Reflective, diagonal and symmetry boundary conditions are imposed on the assembly surface. DRAGON code gives transport, absorption, scattering, fission and production cross-section of two material compositions by solving the neutron transport equation. These values are functionalized on the reference macroscopic cross-section, coolant temperature and square root of Doppler fuel temperature as:

$$\Sigma = \Sigma_0 + \left(\frac{\partial \Sigma}{\partial \rho}\right)_0 (\rho - \rho_0) + \left(\frac{\partial \Sigma}{\partial T_D}\right)_0 \sqrt{T_D} - \sqrt{T_{D0}} \tag{1}$$

where index 0 stands for the reference value; ρ: the density of LBE, g/cm^3; T_D: Doppler fuel temperature, K. The reference value and derivative cross-sections of the thermal-hydraulics feedbacks are illustrated in Table 2.

The Doppler temperature is computed by interpolating the fuel rod centerline temperature $T_{f,c}$ and surface temperature $T_{f,s}$ as equation (2).

$$T_D = 0.3T_{f,c} + 0.7T_{f,s} \tag{2}$$

Except for the material compositions cross-sections on reference value, the cross-sections with and without control rods should be prepared. When the control rod is completely inserted, the lower edge of the control rod is coincident with the upper side of the lower reflector. The location of the lower edge is 90 cm higher than the lower reflector for a complete withdrawal. The macroscopic cross-section of the node with a control rod are expressed as:

$$\Sigma^{with\ \ CR} = \Sigma^{without\ \ CR} + \Delta\Sigma^{CR} \tag{3}$$

where $\Delta\Sigma^{CR}$ is the differential cross-section independent on feedbacks. The differential macroscopic cross-sections involved in control rods are only determined by the height of nodes occupied by control rods, not the feedbacks. Therefore, macro cross sections are dependent on the control rod position and values of thermal-hydraulics feedbacks.

This simulation uses seven prompt neutron groups and six delayed neutron groups. Table 3 shows the 7 groups prompt neutron spectrum and time constants and fractions of 6 groups delayed neutrons. Vacuum boundary conditions are imposed on the outer fuel assemblies.

3.2. COBRA-YT model

The thermal-hydraulics calculation has been carried out by a sub-channel code COBRA-YT that can predict the flow and temperature distributions in fuel rod bundles and channels for steady state and transient conditions in LBE cooled fast reactors. COBRA-YT has been modified by implementing the physical property of lead-bismuth, heat transfer correlation, turbulent mixing and pressure drop models into the