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Development of a sub-channel code for liquid metal cooled fuel assembly

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ABSTRACT

Extensive study has been carried out for liquid metal cooled reactors (LMR). One of the challenges for the design of LMR is to keep the cladding temperature below the design limit. Thus, accurate prediction of coolant and fuel cladding temperature is highly required. In this study, a sub-channel analysis code COBRA-LM is developed for thermal-hydraulic analysis of liquid metal reactor. Based on COBRA-IV code, the development work of COBRA-LM can be divided into two steps. Firstly, Sodium and Lead–Bismuth properties calculation is introduced; secondly, pressure drop models, turbulent mixing models and heat transfer correlations are investigated and implemented into the code. Furthermore, verification and validation study of the new developed COBRA-LM code is performed. The ORNL-19 pin tests are chosen to assess the code's capability. Comparisons are performed to demonstrate the accuracy of the code by the results of CFX, MATRA-LMR and ORNL-19 test data. According to the results, it can be concluded that a reliable tool for sub-channel analysis of LMR is developed, model recommendation is also given for future study. Finally, an analysis for PHENIX reactor is simulated by COBRA-LM code.

1. Introduction

In the framework of Generation IV International Forum (GIF), technology goals were set for the middle-long term of nuclear reactor projects. In order to achieve diverse objectives among which the waste minimization, three types of fast reactors (Gas-cooled Fast Reactor, Lead-cooled Fast Reactor, Sodium-cooled Fast Reactor) were considered. Liquid metals are foreseen to be used as coolants for GEN IV fast reactors, as well as for Accelerator Driven Systems. Thanks to the operating experience already gained in the past years, one possible mid-term available fast reactor seems to be Sodium-cooled Fast Reactor.

Due to the higher operating temperatures of fast reactors (outlet temperature over 500 °C) with respect to LWRs, many constraints about corrosion phenomena by liquid metals are posed. In order to meet economical and safety considerations, temperature distributions are key-point in the design of liquid metal reactor cores. Several design limits are concerning fuel, cladding and coolant temperatures, thus an accurate prediction of the thermal-hydraulic behavior of the core assemblies is an essential prerequisite to reactor design. Therefore, considerable experimental and theoretical studies are necessary to acquire a detailed knowledge of LMR assemblies and fuel pins conditions (Gen IV Roadmap, 2002). In the LMR design, wire wrap spacer is adopted to enhance the coolant mixing flow between sub-channels and to maintain the cooling geometry by prevention the fuel rod contacting adjacent rods.

Nowadays, the sub-channel analysis approach is usually adopted to evaluate the local thermal-hydraulic behavior of the fuel assemblies. In a sub-channel analysis code, mass, momentum and energy conservation equations are modeled and solved together with initial and boundary conditions (Steward et al., 1977). In the past, great effort has been devoted to develop reliable sub-channel analysis tools for thermal-hydraulic analysis providing coolant temperature fields in the bundle, e.g., MATRA-LMR, SLTHEN, SABRE4 and COBRA-WC.

MATRA-LMR (Kim et al., 2002) was developed at Korea Atomic Energy Research Institute (KAERI) by adaptation of MATRA to sodium coolant reactor. SABRE4 (Dobson and O'Neill, 1992) is widely used in the UK; it is a 3-D sub-channel code designed to predict the thermal-hydraulics of a sodium fast reactor fuel assembly. SLTHEN is a modified version of ENERGY, specifically developed for LMR (e.g., sodium fast reactor), providing improved computational efficiency and simplified energy equation mixing model (Yang, 1997). COBRA-WC is derived from COBRA-IV to analyze liquid metal fast breeder reactor assembly transients. It was

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Nomenclature

C_P	specific heat/(J kg^{-1} K^{-1})
D	diameter/m
D_e	equivalent diameter/m
F	correction factor/–
f	friction factor/–
G	mass flux/(kg m^{-2} s^{-1})
H	wire wrap pitch/m
LEB	lead-bismuth eutectic
LMR	liquid metal cooled reactors
LWR	light water reactor
Nu	Nusselt-number/–
P	pitch of the rod/m
p	pressure/MPa
Pe	Peclet number/–
Pr	Prandtl-number/–
q''	heat flux/(W m^{-2})
Re	Reynolds number
S	gap between two adjacent sub-channels/m
SSG	Speziale–Sarkar–Gatski Reynolds stress model
T	temperature/°C
ω'	turbulent mixing flow rate/(kg/m-s)
V	velocity/(m/s)
β	turbulent mixing coefficient/–
σ	surface tension/(N m^{-1})
λ	thermal conductivity/(W m^{-1} K^{-1})
μ	dynamic viscosity/(Pa s)
ρ	density/(kg m^{-3})

Subscripts

ave	average
B	boiling point
in	inlet
M	melting point
out	outlet
W	wall

specifically developed to analyze a core flow coastdown to natural circulation cooling. The detailed information about COBRA-WC code and the difference between COBRA-IV and COBRA-WC can be found in the report from Pacific Northwest Laboratory (George et al., 1980). However, as pointed out in a recent paper (Wu et al., 2013), there are still some limitations for the current sub-channel codes, e.g., limitation for the calculation speed and number of the fuel rod and sub-channels. Moreover, it should be noted that an intensive investigate of various thermal hydraulic models in sub-channel code, e.g., heat transfer and mixing correlations are required to recommend proper models for LMR simulation. Therefore, a model sensitivity analysis for sub-channel code is also needed.

In the frame of this study, the development of a sub-channel analysis code COBRA-LM for liquid metal reactors is presented. The sub-channel code COBRA-IV was adapted to deal with sodium and lead-bismuth eutectic cooled bundles, and confirmed by the benchmark analysis. The results achieved in this paper provide a reliable tool for the sub-channel analysis of the LMR, some model recommendations are also given for the future study.

2. Development of COBRA-LM

The development of the sub-channel code consists in the adaptation of a reference code toward the analysis of liquid metals application in nuclear reactors. As mentioned in the introduction, COBRA-IV-I (Steward et al., 1977) is the reference code for this purpose. While COBRA-IV-I deals with water in PWRs, COBRA-LM aims to describe the thermal-hydraulic behavior of liquid metal flows in reactor bundles. Therefore, the properties of the coolant and the models related to flowing and heat transfer of LMR should be adapted. Thus, this study entailed the modification of following issues:

- Properties of the liquid metal coolant
- Pressure drop models for liquid metal
- Turbulent mixing models for liquid metal
- Heat transfer correlations for liquid metal

In order to develop a both reliable and flexible tool open to improvements, a wide range of models are analyzed in this paper. The following subsections describe in detail the modifications to the reference code.

2.1. Liquid metal properties

Intensive studies have been performed in different countries aiming at a better understanding of liquid metals properties needed for design and safety analysis of the nuclear installations. The thermo-physical properties of these coolants were measured in many laboratories, but mainly at atmospheric pressure and at relatively low temperatures (except for sodium). In general, the reliability of data is satisfactory, especially for sodium.

For the analysis of liquid metal cooled fuel assemblies, several modifications to COBRA-IV-I were necessary as the original code does not allow direct calculation of the thermo-physical properties of liquid metals. In fact, COBRA-IV-I is provided with water properties in tabular form. A review and compilation of data available in open literature for the main thermo-physical properties of liquid metal (sodium and lead–bismuth eutectic: LBE) is provided by Sobolev (2010). Table 1 summarizes the main property of these two liquid metal. Eventually, a full set of both sodium and LBE property correlations was implemented in COBRA-LM.

From the correlation of the specific heat, it is possible to find an equation for the enthalpy increment, taking the melting point as reference value. Thus, the specific enthalpy per unit mass as a function of temperature at the given pressure is obtained by integration of the isobaric specific heat capacity over temperature, as illustrated in Eq. (1):

$$h(T,p) = h(T_M,p) + \int_{T_M}^{T} C_p(T,p)dT \tag{1}$$

2.2. Pressure drop models

Liquid metal reactors fuel assembly presents tight arrangement of thin fuel pins together with hexagonal geometry. This design feature can provide a high heat transfer coefficient for liquid metals. However, proper spacing between the fuel pins is necessary; therefore, helical wire-spacers are frequently used. This implies that a suitable pressure drop correlations for wire-wrapped rod bundles should be implemented into the sub-channel code.