Progress in Nuclear Energy 103 (2018) 151–164

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Review

Review on heat transfer and flow characteristics of liquid sodium (2): Two-phase

Yingwei Wu*, Simin Luo, Liu Wang, Yandong Hou, G.H. Su, Wenxi Tian, Suizheng Qiu

Shaanxi Key Laboratory of Advanced Nuclear Energy and Technology, Shaanxi Engineering Research Center of Advanced Nuclear Energy, Xi'an Jiaotong University, Xi'an City 710049, China

ARTICLE INFO

Keywords:
Liquid sodium
Two-phase
Heat transfer
Flow
Friction factor

ABSTRACT

With the development of sodium-cooled reactor (SFR), the research on the flow and heat transfer characteristics of liquid sodium becomes increasingly important and necessary. Through decades of experimental and theoretical research under different conditions, certain achievements have been obtained. To further carry out relevant studies, a review work on this topic is of necessity. As pointed out by scholars, sodium boiling might occur during hypothetical accidents, which might cause great damage to the SFRs. For safety analysis, the boiling of liquid sodium has already become a focus in this research area. Based on an extensive review on the published literature, this paper concerns two major topics, i.e. the boiling heat transfer and two-phase flow characteristics of liquid sodium. The former topic focuses on the characteristics of sodium boiling in pools, round tubes and annuli, including the correlations for the boiling heat transfer coefficient, the incipient boiling wall superheat and critical heat flux of dryout type. Also, the correlations for the single-phase flow in bundle channels are presented here due to the fact that relevant research on the sodium boiling in rod bundles is relatively insufficient and the relevant correlations are rarely found. For the part of two-phase flow characteristics, the correlations for the two-phase pressure drop multiplier are presented and analyzed. Finally, conclusions and prospects in this research field are summarized and proposed.

1. Introduction

The sodium-cooled fast reactor (SFR) is one of the Generation IV reactors, which is considered by several countries as the prime candidate for the large-scale implementation of breeder reactor technology in the near future. In SFR, the liquid sodium is used as coolant. But the heat transfer to liquid metals significant from the heat transfer to water as liquid metals have a very low Prandtl number (*Pr*). It has been pointed out by Farmer (2012), Waltar and Padilla (1977), Waltar and Reynolds (1981), Hennies et al. (1990) and Maschek and Struwe (2000), that during some hypothetical accidents, such as the unprotected loss of flow, loss of piping integrity, loss of heat sink, anticipated transient without scram and subassembly blockage etc., the boiling of sodium in the reactor may appear, leading to dryout and even the melting of material. Thus, in the safety studies of SFR, it's of great importance to develop the accurate model of sodium two-phase flow.

Compared with water boiling, the boiling of liquid metal has following characteristics (Sorokin et al., 1999): the complex interaction of the internal factors in the system makes it difficult to accurately determine the incipient boiling superheat of liquid sodium under actual conditions; in liquid sodium, large vapor bubbles are formed at several nucleation sites and the formation time of most vapor bubbles is within waiting period; the growth of liquid sodium vapor bubbles can be explosive, at a rate of about 10 m/s; the major two-phase flow patterns of liquid sodium are as the same as that of conventional fluids and dispersed annular flow pattern dominates around barometric pressure; the phase change of dispersed annular flow of liquid sodium in pipe is realized through the evaporation of liquid film rather than the formation of vapor bubbles (formed by boiling) on the wall surface, and the corresponding heat transfer coefficient can reach to a magnitude of 100,000 W/(m^2·K).

When the boiling of liquid sodium happens, the two-phase flow of sodium will appear. So research on the two-phase flow pressure drop characteristics of liquid sodium boiling is very important for the fast reactor accident analysis involved as the two-phase flow pressure drop is directly related to the determination of the cooling flow rate of liquid sodium in these accidents. Due to the significant difference of the physical properties between liquid sodium and conventional fluid, the existing correlations for two-phase flow pressure drop of water cannot be used for liquid sodium directly (Kottowski et al., 1985). Specifically,

* Corresponding author.
E-mail address: wyw810@mail.xjtu.edu.cn (Y. Wu).

https://doi.org/10.1016/j.pnucene.2017.11.016
Received 26 June 2017; Received in revised form 30 September 2017; Accepted 27 November 2017

Nomenclature

A	flow area, m^2
c_p	specific heat at constant pressure, J/kg
D, d	diameter, m
D_h	hydraulic diameter, m
f	friction factor
Gr	Grashof number
g	acceleration due to gravity, m/s^2
G	mass flow rate, kg/s
H	heat transfer coefficient, $W/(m^2 \cdot K)$
I	latent heat, J/kg
L	length of flow channel or perimeter, m
M	molecular weight
N	number of rods
Nu	Nusselt number
p	pressure, Pa
P	pitch, m
PD	pitch-to-diameter ratio P/D
Pe	Peclet number
Pr	Prandtl number
q	heat flux, W/m^2
Q	flow rate, m^3/s
r	radius, m
Re	Reynolds number
S	slip ratio
T	temperature, K
T_w	wall temperature, K
T_s	saturated temperature, K
x	mass quality

Greek symbols

α	void fraction
λ	heat conductivity, W/(m·K)
μ	dynamic viscosity, Pa·s
ρ	density, kg/m^3
θ	angle of flow direction with the vertical (rad)
σ	surface tension, N/m
ϕ	two-phase friction multiplier
Φ	two-phase local pressure drop multiplier
1ϕ	single-phase
2ϕ	two-phase

Subscripts

C	critical
CHF	critical heat flux
d	dryout
g	gas
in	inlet
l	liquid
lg	liquid-gas
ONB	onset of nucleate boiling
out	outlet
s	saturated
sub	sub-cooled
w	wall

such boiling two-phase flow pressure drop can be determined by following two modes: first, the correlations under different working conditions can be given; second, the kinematic viscosities of liquid sodium and water are at the same order of magnitudes, so the existing correlation for the single-phase flow pressure drop of water can be adopted for liquid sodium and thereby the two-phase pressure drop calculations can be converted into the calculations of two-phase friction multiplication factor. The latter method is widely used by scholars for the calculation of two-phase pressure drop.

Due to the difficulties brought by the special characteristics of sodium to relevant research, scholars have established different models for sodium boiling mechanism and meanwhile provided the corresponding solutions. However, there are lots of discrepancies and even conflictions between the proposed models and such problems have increasingly aroused people's concern. Thus, a more general model for accurate calculation and analysis is of necessity to be obtained through future research.

This study mainly discusses the characteristics of liquid sodium boiling heat transfer and the calculation approaches of two-phase flow pressure drop in wire-wrapped bundles, the general geometry of flow channel in SFR. The part of liquid sodium heat transfer focuses on the characteristics of sodium boiling in pools, round tubes and annuli,

Table 1
Research of sodium boiling heat transfer characteristics.

Reference	Major target of research	Geometry of research object	Research method
Mostinskii (1963)	heat transfer coefficient	pool	theoretical analysis
Holtz and Singer (1967)	heat transfer coefficient	pool	theoretical analysis
AladevI et al. (1968)	heat transfer coefficient	round tube	experiment
Subbotin et al. (1970)	heat transfer coefficient	pool	experiment
Schleisiek (1970)	heat transfer coefficient and incipient boiling superheat	round tube	experiment
Dhir and Lienhard (1971)	heat transfer coefficient	spherical surface	theoretical analysis
Kikuchi et al. (1974)	incipient boiling superheat	–	semi-empirical correlation
Ferguson et al. (1976)	various boiling parameters under accidents of SFR	geometry of core model	calculation by programming
Grolmes and Fauske (1981)	incipient boiling superheat and the critical heat flux of dryout type	round tube	theoretical analysis
Galati (1981)	various boiling parameters	annular channel	modeling calculation
Ishii and Fauske (1983)	critical heat flux	rod bundle	experiment
Carbajo (1983)	incipient boiling superheat and the critical heat flux of dryout type	fuel assembly	modeling calculation
Carbajo (1985)	incipient boiling and critical boiling	round tube	calculation by programming
Weber and Briggs (1986)	various boiling parameters under accidents of SFR	geometry of fuel element	numerical simulation
Bottoni et al. (1990)	various boiling parameters	rod bundle	modeling calculation
Shah (1992)	heat transfer coefficient	pool	experiment
Qiu et al. (1993)	heat transfer coefficient, incipient boiling superheat, critical heat flux	annular tube	experiment, theoretical analysis
Sorokin et al. (1999)	critical heat flux	round tube and rod bundle	experiment
Martsiniouk and Sorokin (2000)	correlations for heat transfer	bundle channels	modeling calculation
Zeigarnik (2001)	critical boiling	round tube	experiment
Xiao et al. (2006)	incipient boiling superheat	annular tube	experiment, theoretical analysis