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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

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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 $W/(m^2 \cdot K)$.

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).

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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

including the boiling heat transfer correlations, the incipient boiling wall superheat and critical heat flux of dryout type. As the relevant research on the boiling of sodium in rod bundles is relatively insufficient in published literature, the correlations for boiling heat transfer of liquid sodium in bundle channels are rarely found. So, the correlations presented here for the flow in bundle channels focus on the single-phase flow of liquid sodium. For the part of two-phase flow pressure drop, the correlations for the two-phase pressure drop multiplier are presented and analyzed.

2. Research on two-phase heat transfer characteristics of liquid sodium

Table 1 shows the research status of sodium boiling heat transfer characteristics. Incipient boiling is defined relatively to the stable boiling. For any liquid to be boiled at heating surface, the surface temperature T_w is certainly higher than the saturated temperature T_s, and the difference value T_w-T_s is called as wall superheat. Relevant research shows that for liquid sodium, the wall superheat of incipient boiling is higher than that of stable boiling by six times and also about five times higher than that of water's incipient boiling due to following reasons: 1) the saturated vapor curve pressure gradient dp/dT of liquid sodium is relatively small within the universal pressure and temperature ranges; 2) the liquid sodium with active chemical property has strong "self-cleaning" effect on the heating surface, and furthermore the high invasion property of liquid sodium makes it difficult for the activation of surface nucleation; 3) inert gases usually retain in wall-surface holes, thus favorable for vapor bubble nucleation, while high-temperature sodium tends to eliminate gas retention. There are many factors influencing the incipient boiling wall superheat, e.g. heat flux density, pressure, flow rate, undercooling degree, retaining gas concentration, impurity content in sodium, heating surface roughness, physical and chemical properties and invasion property of fluid, etc.

Critical heat flux refers to the heat flux density when the boiling heat transfer mechanism is changed to reduce the heat transfer coefficient suddenly. In forced convection boiling, it may be the heat flux density of DNB (Departure from Nucleate Boiling) type or that of dryout type. All types of heat flux density are accompanied with the extreme deterioration of heat transfer and the rapid increase of wall-surface temperature. Fast reactor has large specific power, and its fuel element needs to bear a specific power which is higher than that of light water reactor by 4 times and a greater temperature gradient. If the reactor suffers from flow loss, hot trap loss, transient overpower, transient under-heating, local blockage or other accidents, once the coolant is boiled, the positive reaction effect of sodium bubbles may make the fuel element melt or burn out. Specifically, the melting and burning damage of fuel element is closely related to the critical heat flux of liquid sodium (Subbotin et al., 1970). Therefore, the research on the critical heat flux of liquid sodium boiling has drawn great attention.

When liquid sodium is taken as a working medium, due to its high heat conductivity coefficient and large vapor-liquid specific volume difference (relative to common fluids such as water), the in-tube boiling process mainly includes annular flow and high vapor content, and critical boiling usually appears in the form of dryout. Therefore, it is necessary to analyze dryout during the sodium boiling research.

Dryout, also called as slow evaporation, is one form of critical boiling. Under high vapor content, when the coolant flows in annular pattern, if excessively strong evaporation happens due to boiling, the liquid layer will be damaged, thereby cause critical boiling, and such kind of critical boiling is sometimes called as "burnt-out" (or dryout). When critical boiling happens, the temperature of heating surface is increased, but due to the rapidly-flowing nucleation site in annular flow, the heating surface has large heat transfer coefficient and thus the increase rate of wall-surface temperature of such critical boiling is lower than that of critical boiling under small vapor content, and the metal materials are not burnt out immediately under such conditions. However, the element cladding may be damaged if wetting and drying alternates on its surface.

This part aims to introduce heat transfer correlations, incipient wall superheat, and critical heat flux etc., during sodium boiling process under different working conditions (pool type, round tube, annular tube, rod bundle, etc.).

2.1. Pool type boiling heat transfer characteristic

At present, the theoretical research of the pool type liquid sodium nucleate boiling mainly aims to establish the corresponding boiling heat transfer models on the basis of general boiling mechanism and to find relevant coefficients experimentally in order to obtain the correlation with certain application scope.

(1) Subbotin et al. (1970) have given the following relation according to the data of the Na, K, Cs boiling experiments done on horizontal stainless steel plates.

$$\frac{h}{q^{\frac{2}{3}}} = B\left[\frac{k_s i_{lg} \rho_s}{\sigma T_s^2}\right](p/p_C)^m \tag{1}$$

When the ratio of system pressure to the critical pressure p/p_C is less than 0.001, B and m are 8 and 0.45 respectively; when the ratio is larger than 0.001, B and m are 1 and 0.15 respectively; where p is system pressure in MPa, T_s the saturated temperature in °C, p_C is the critical pressure. In the experiments, p/p_C set between 0.003 and 0.012.

(2) Ishii and Fauske (1983) have proposed a calculation relation for the critical heat flux of pool boiling based on experimental research:

$$q_{CHF} = 0.14\left[1 + 0.94\left(\frac{p}{p_C}\right)^{-0.4}\right]\rho_g i_{lg}\left[\frac{\sigma_g(\rho_l - \rho_g)}{\rho_g^2}\right]^{1/4}$$

(3) Mostinskii (1963) has obtained the following simplified correlation which has wider application scope and is applicable to many common fluids:

$$h = 0.1 p_C^{0.69} q^{0.7}(1.8\mathrm{Pr}^{0.17} + 4\mathrm{Pr}^{1.2} + 10\mathrm{Pr}^{10}) \tag{2}$$

where p_C is in bar.

Collier (1981) and Palen et al. (1972) found that the special characteristics of liquid metals make the above correlations inapplicable to the heat transfer of liquid metals. Mostinskii (1963) and Cooper (1984) believed that correlations excluding the flow characteristics may be found applicable.

(4) Shah (1992) has obtained the following relation on the basis of extensive experimental data:

$$h = Cq^{0.7}\mathrm{Pr}^m \tag{3}$$

When Prandtl number is less than 0.001, C and m are 13.7 and 0.22 respectively; while Prandtl number is larger than 0.001, C and m are 6.9 and 0.12 respectively.

The above correlation can be verified by the data of the experiments for Na, K, Cs, Li and Hg on a flat plate when Prandtl number is in the range of 4.2×10^{-6} to 1.5×10^{-2}.

Moreover, Shah also proposed a standby correlation:

$$h = Bq^{0.7}\left[\frac{k_s i_{lg} \rho_s}{\sigma T_s^2}\right]^{1/3}\mathrm{Pr}^m \tag{4}$$

where B and m are 2.1 and 0.31 respectively when Prandtl number is less than 0.001; B and m are 0.67, 0.15 respectively when Prandtl number is larger than 0.001. The application scope is as the same as that of Eq. (3). Compared with Eq. (3), this correlation has a deviation of 25% for lithium and has the deviation for other working medium within 10%.

(5) In the opinion of Holtz and Singer (1967), the growth of bubbles in early period is due to the heat transfer from the thin liquid layer near the wall-surface to the vapor, and the bubbles do not directly contact the wall-surface; after the bubbles detach from the wall-surface, cool liquid extrudes into the holes left by the bubbles and accept the heat from the wall-surface. According to Holtz' theory, the following correlation is established:

$$h = ap_s^{0.27}q^{0.68} \tag{5}$$

The above correlation can be converted into:

$$q = bp^{0.844}(T_w - T_s)^{3.125} \tag{6}$$

(6) Dhir and Lienhard (1971) proposed the following correlations for the heat transfer to liquid film on spherical surfaces in a pure fluid system under quasi-stable state:

$$Nu = \frac{hd}{2\lambda_l} = 0.785\left[\frac{g\rho_l(\rho_l - \rho_g)i_{lg}^* d^3}{\mu_l\lambda_l(T_l - T_w)}\right]^{1/4} \tag{7}$$

where i_{lg}^* is the latent heat of vaporization when vapor is condensed into liquid film:

$$i_{lg}^* = i_{lg} + 0.68c_{p,l}(T_l - T_w) \tag{8}$$

The physical properties in Eq. (7) are the values at the temperature of $T_w+0.3(T_l-T_w)$.

Through the analysis of a single sphere, Bird et al. (1960) obtained the following correlations:

Under natural convection working condition:

$$Nu_g = 2 + 0.60Gr_g^{1/4}\mathrm{Pr}_g^{1/3} \tag{9}$$

$$Sh_g = 2 + 0.60Gr_g^{1/4}Sc_g^{1/3} \tag{10}$$

Under forced convection working condition:

$$Nu_g = 2 + 0.60\,\mathrm{Re}_g^{1/2}\mathrm{Pr}_g^{1/3} \tag{11}$$

$$Sh_g = 2 + 0.60\,\mathrm{Re}_g^{1/2}Sc_g^{1/3} \tag{12}$$

From Eq. (9) to Eq. (12), the values of physical properties are taken at the qualitative temperature of 0.5 ($T_g + T_l$) and the length L is set as sphere diameter. Under natural convection condition, the Grashof number of gas Gr_g depends on two-phase temperature and components and is determined by the total density of the liquid film:

$$Gr_g = \frac{g\rho_{g,\infty}(\rho_{g,\infty} - \rho_{g,i})L^3}{\mu_g^3} \tag{13}$$

This chapter mainly talks about the pool type boiling heat transfer characteristic and some correlations for heat transfer coefficient or heat flux are given. As can be seen from above, in pool type boiling, the correlations for heat transfer calculation seem to be more expressed in a way that correlates heat transfer coefficient to heat flux directly. Nd additionally, the critical heat flux is a major concern in the research of pool type boiling heat transfer.

2.2. Heat transfer characteristics in round tube channels

As for sodium boiling heat transfer, many scholars have carried out experimental research and theoretical analyses and calculations under the round tube working condition, thus accumulating many valuable empirical correlations and calculation models. Meanwhile, the calculation of the sodium boiling heat transfer in round tube is also the basis for complex working conditions.

(1) Schleisiek (1970) performed a series of sodium boiling experiments under forced convection working condition in the sodium boiling loop NSK. This test section was a nickel tube heated through high-frequency induction, wherein the inner diameter of the tube was 9 mm, the heating length 168 mm and the experiment working conditions as follows: heat flux density of 170–600 W/cm^2; pressure of 0.3–1atm; mass flow rate of 0.06–0.12 kg/s; inlet sub-cooling of 85–170 °C;

A boiling heat transfer correlation was obtained under conditions of the heat flux density less than 720 W/cm^2 and the correlation takes its form as:

$$Nu = 5.3 + 0.018Pe^{0.85} \tag{14}$$

After the experiments, the authors drew the following conclusions: the wall superheat of incipient boiling is between 40 °C and 125 °C, and it increases with the increase of heat flux density and the reduction of sodium flow rate, accompanied with strong fluctuations of flow rate and wall-surface temperature in the whole parameter range; the appearance of boiling is mainly determined by the pressure drop and the heat flux density of the liquid phase before the test section; under small pressure drops, dryout occurs when the heat flux density is about 250 W/cm^2, while under large pressure drops, dryout occurs when the heat flux density is increased to 400 W/cm^2.

(2) AladevI et al. (1968) analyzed the experimental data and gave the following correlation for the heat transfer of liquid sodium boiling in round tubes:

$$h = 0.57q^{0.7}p^{0.15} \tag{15}$$

(3) Zeigarnik (2001) reported that the liquid metal boiling in a round tube under low pressure forms annular flow (and then forms reverse annular flow) at extremely low vapor quality (several percentages). At this moment, the liquid film is distributed on the heating surface and the vapor containing liquid drops is located at the flow center. In most cases, due to the high thermal conductivity of liquid sodium and the low thickness of the liquid film, the wall superheat is insufficient to nucleate the vapor, so the heat generated from the heating wall-surface is conducted out through the liquid film on the wall-surface and then the surface of the liquid film is evaporated. Therein, the total temperature difference ΔT_α is the sum of the temperature difference ΔT_f in the liquid film and the temperature difference ΔT_{ph} determining the phase change:

$$\Delta T_\alpha = \Delta T_f + \Delta T_{ph} \tag{16}$$

$$\Delta T_f \cong \frac{qD(1-\varphi)}{4\lambda} \tag{17}$$

$$\Delta T_{ph} = \frac{qT_s v''(2\pi\widetilde{\mathrm{R}}T_s/M)^{0.5}}{i^2\beta} \tag{18}$$

where q is the wall-surface heat flux density, D is the channel diameter, φ is the real void fraction, λ is the liquid heat conductivity, T_s is the saturated temperature, v'' is the specific volume of vapor, i is the latent heat of vaporization, $\widetilde{R}$ is the universal gas constant, M is the molecular weight, and β is the evaporation coefficient.

(4) Sorokin et al. (1999) proposed the following correlation to predict the critical heat flux:

$$q_c = 0.312(G^{0.95}/(L/D_r))r \tag{19}$$

The application ranges of the above equation are: G = 1.69–402kg/(m^2·s); p = 0.001–0.003 MPa, D_r = 4–9 mm; length of heating section 200–1000 mm; q_c = 32.3–7370 kW/m^2.

(5) Carbajo (1983) used LOOP-1 code for the one-dimensional calculation of boiling and the comprehensive analysis of the transient power and time of dryout in the closed loop of liquid sodium,

wherein the flow loss and the heat dissipation failure were respectively considered in the analysis. Specifically, the flow loss analysis result shows: under natural and forced convection conditions, through the increase of power and entrance temperature or the decrease of pressure in the test section, the incipient boiling time can be slightly advanced while the advancement of dryout time is much more obvious. Under forced convection, the same stable and transient working conditions can be obtained through the correction of the thermal power of the pump, no matter whether to improve the valve value or reduce the bypass value, thus realizing a long boiling process before dryout; under natural convection, the above operations can accelerate incipient boiling and dryout. The larger the flow rate before transient flow loss is, the later the incipient boiling and the dryout occur. Under heat dissipation failure, the higher the entrance temperature is, the smaller the power for boiling and dryout is needed. The power difference between boiling and dryout is increased with the entrance temperature. These conclusions have been verified by relevant experiments.

2.3. Heat transfer characteristics in annular channel

By virtue of sodium loop experiment platforms, extensive experimental research has been performed for the thermo-hydraulic characteristics of liquid sodium in annular channels, and comprehensive theoretical analysis has also been carried out according to the experimental data and some achievements were obtained.

The boiling mechanism and characteristics of liquid sodium during its vertical flow in annular channel under negative pressure have been obtained through experimental research and theoretical analysis and massive experimental data and curves under various working conditions have also been obtained.

2.3.1. Experimental research

Qiu et al. (1993) performed a serious of relevant experiments and the experiment parameters were as follows: heat flux density of 12.77 W/cm^2-78.73 W/cm^2; liquid sodium flow rate of 0.047m/s-0.674 m/s; saturated pressure of 850Pa-9870Pa; inlet sub-cooling of 63.12°C–270.97 °C. The following conclusions were obtained from the experiments:

(1) For this experiment system, once there is any small vapor bubble on the heating surface, it will have influence on the friction pressure drop, which indicates that the vapor bubble of liquid sodium is larger than that of water.
(2) The sub-cooled boiling process of liquid sodium is shortened along with the increase of the heat flux density and the decrease of inlet sub-cooling and flow rate.
(3) When the flow rate is reduced at the stable nucleation point on the wall surface and the stable onset of sub-cooled boiling, the wall-surface temperature is suddenly increased.
(4) The wall-surface temperature of the electrical heating element is suddenly increased and rapidly dropped at the onset of stable sub-cooled boiling, namely: the wall-surface temperature at the onset of stable saturated boiling is lower than the maximum of the wall-surface temperature increased suddenly at the onset of stable sub-cooled boiling. At this moment, since large latent heat of vaporization of the heating wall-surface is needed for the continuous evaporation and separation of many vapor bubbles on the heating wall-surface after the occurrence of saturated boiling, thus the heat transfer capability between the wall-surface and the liquid is significantly strengthened and the wall-surface temperature is reduced.
(5) In the experiment, the temperature of the liquid mainstream tends to drop at the onset of stable sub-cooled boiling and then increases to the saturated temperature for stable saturated boiling.
(6) After the onset of saturated boiling, the wall temperature and the liquid temperature both slightly fluctuate around the stable value and the flow is reduced during significant fluctuation.
(7) After the occurrence of stable sub-cooled boiling, there was noise and vibration in the experiment channel. At the saturated boiling, obvious cracking sound appeared.

Based on further experiments, Qiu et al. (1993) further proposed the following relation:

$$h = 0.832q^{0.768}p^{0.253} \tag{20}$$

where h is the heat transfer coefficient (W/(m^2·°C)); q is the heat flux density (W/m^2); p is the saturated pressure of the system (Pa).

The experiment parameter range of the relation is as follows: heat flux density of 1.577 × 10^5 W/m^2-4.45 × 10^6 W/m^2; pressure of 850Pa-50000Pa; mass flow rate before boiling of 40kg/(m^2·s)-320kg/(m^2·s); inlet sub-cooling before boiling of 63.12°C–214.5 °C; The average discrete degree between the calculated values by Eq. (20) and the experimental data is 14.7%.

The experiment result shows that the heat transfer coefficient of liquid sodium boiling has strong exponential relations with heat flux density q and weak exponential relations with system pressure, which is consistent with the qualitative conclusion of Kottowski and Savatteri (1984) and Kovalev et al. (1973). The saturated boiling heat transfer of liquid sodium is completely substituted by its evaporation process, and the heat transfer coefficient is irrelevant to the inlet sub-cooling and mass flow rate but is strongly related to the heat flux density of the heating element and the system pressure.

2.3.2. Heat transfer correlation

Qiu et al. (2013) performed experimental research on the incipient boiling wall superheat of liquid sodium. Through the comparison of the experiment data and relevant literature, the following qualitative conclusions are obtained: (1) The heat flux density can increase the incipient boiling wall superheat of liquid sodium. Compared with literature (Edwards, 1965; Deane and Rohsenow, 1969; Logan et al., 1969), the tendency conformance is good; (2) The flow rate (Reynolds number) can reduce the incipient boiling wall superheat and the tendency is well compared with literature (Logan et al., 1969; Pezzilli et al., 1970; Kikuchi et al., 1974); (3) The inlet sub-cooling can increase the initial boiling wall superheat degree; (4) The system pressure can reduce the initial boiling wall superheat degree. In comparison with the results of literature (Deane and Rohsenow, 1969; Edwards, 1965; Chen, 1968), the tendency conformance is good.

Based on the data points obtained experimentally and multi-linear regression method, the authors established the following correlation for the incipient boiling wall superheat of sodium:

$$(T_w - T_s)_{ONB} = 3.2765 \times 10^{-5} q^{1.188} p^{-0.098} \Delta T_{sub}^{0.296}\, \mathrm{Re}^{-0.257} \tag{21}$$

The above correlation is applicable for the range of heat flux from 128 to 846 kW/m^2, inlet sub-cooling from 63.1 to 287.8 °C, Reynolds number less than 13,000 and system pressure from 0.85 to 28.79 kPa. The deviation between the measured incipient boiling wall superheat of sodium and the calculated values is within ± 20%.

The authors (Qiu et al., 2015) performed further experiments to investigate the heat transfer to liquid sodium in annulus and they proposed a new correlation on the basis of the obtained data

$$h = 5q^{0.7}p^{0.15} \tag{22}$$

The deviation between the calculated values by Eq. (22) and the experimental data is within ± 25%.

2.3.3. Incipient boiling superheat

Xiao et al. (2006) analyzed the incipient boiling superheat of liquid sodium through the combination of experiment research and theoretical derivation. In the experiment, thirty groups of sodium boiling heat transfer data were obtained, and the experiment parameters were as follows: heat flux density of 8–13 W/cm^2; system pressure of 109–126 kPa; incipient boiling flow rate of 0.02–0.14 m/s; wall superheat of 8–35 °C; oxide impurities content in sodium of 20 × 10^{-6}-50 × 10^{-6}. From the experiment and theoretical analysis, the authors proposed a correlation for incipient boiling superheat:

$$T_w - T_s = 0.921[q - 8348.1 - 31(\psi Pe)^{0.779}]^{0.32}p^{-0.27}Pe^{-0.544} \tag{23}$$

The discrete degree of the experiment data is within ± 19%.

After post-process of the experimental data, the authors further reported that three factors, the heat flux q, the system pressure P and the flow rate, have significant influence on wall superheat $T_w - T_s$. Their respective influences are summarized as follows:

(1) Usually, the larger q is, the higher T_w will be. In such case, not only the large cavities can be easily nucleated, but also the small ones can be activated into bubbles. Under such conditions, large energy input is needed, and the wall superheat is increased with the heat flux q.
(2) The increasing system pressure P raises the saturated temperature T_s on one hand, and reduces surface tension on the other hand. While according to bubble dynamics, the wall superheat declines with pressure reduction, and is equal to zero under critical pressure.
(3) Increase of the flow rate can strengthen the heat transfer effect of incipient boiling, so sufficient energy can be transferred for the vapor bubble nucleation under relatively low wall temperature. Meanwhile, the increase of the flow rate can increase the probability for transferring massive inert gases from the expansion box to the wall-surface. All above factors can cause the reduction tendency of the wall superheat along with the increase of flow rate.

2.3.4. Critical heat flux

Qiu et al. (1993) carried out relevant experiment research for the critical heat flux of the liquid sodium boiling in a sodium loop, and the thermal parameters are as follows: pressure 850-5 × 10^{4}Pa; mass flow rate 40–570kg/(m^2·s); heat flux density 0.128–1.07 MW/m^2; inlet sub-cooling 43.2–307.5 °C. When heating wall-surface temperature reaches up to 1400 °C, the heat transfer is deteriorated, leading to critical heat flux and the burnout of heating element.

The qualitative relationship between CHF and several major factors concerned can be obtained by analyzing the experimental data: CHF increases with inlet sub-cooling, mass flow rate and system pressure, and the flow instability causes an earlier occurrence of CHF.

Through experimental research and theoretical study, Qiu et al. (1993) reported that the occurrence of the critical heat flux of liquid sodium is related to mass flow rate, inlet sub-cooling and system pressure, etc., and that the system pressure can influence the physical property of liquid sodium and its vapor. Accordingly, the correlation for the critical heat flux of liquid sodium was proposed as follows:

$$q_{CHF} = 0.969G^{0.728}\Delta h_{in}^{0.72} + 0.035\rho_g i_{lg}\left[\frac{\sigma_g(\rho_l - \rho_g)}{\rho_g^2}\right]^{1/4} \tag{24}$$

The average discrete degree of the experiment data and the calculation value of Eq. (24) is 19.7%. The comparison of the experiment data and the calculation value is as shown in Fig. 1. The critical heat flux is also related to geometrical condition, experiment condition and experiment method. Among the published experiment data for the critical heat flux of liquid sodium, the system pressure of the experiment is relatively high and the data were mainly obtained under conditions of turbulence in rod bundle channels, so the experimental data obtained thereby are significantly higher than the value calculated by Eq. (24).

2.4. Heat transfer characteristics in rod bundle channel

As the rod bundle channel is most approximate to the actual condition in the reactor and the transverse flow and the heat transfer among channels cannot be considered in the single-rod research, it is necessary to research the flow and heat transfer in rod bundle channels in order to more truly simulate the heat transfer process in the reactor. Along with the in-depth research of single rod, the improvement of experiment condition, the gradual perfection of sub-channel model and the rapid improvement of computer calculation capability, the experimental and theoretical analysis of rod-cluster becomes possible.

(1) Sorokin et al. (1999) adopting 22%Na-%78 K as working medium, carried out the experiments in two rod bundle channels in which 7 vertical rods with the length of 3 m are arranged in triangle respectively, and observed three boiling flow patterns (as shown in Fig. 2):
  a) Nucleate boiling in the initial stage: In this pattern, the coolant temperature, element pressure drop and the coolant flow are stable. When heating are increased, nucleate boiling can be converted into jetting boiling;
  b) Jetting boiling (q = 125–170 kW/mm): the fluctuation and the generation & ascending of large bubbles cause the intensive reduction of inlet flow rate and the significant fluctuation of measurement constant. The surface temperature of fuel element is not higher than the saturated temperature, thus indicating the liquid film on the surface; the power increase causes the increase of the jetting rate and the decrease of the temperature fluctuation amplitude. For q = 210–230 kW/mm, the transition from the jetting flow to the annular flow is observed;
  c) Annular flow: the measurement constant is stable, and the liquid evaporation and the movement thereof from the surface of the heating element can cause surface dryout. Therefore, the annular flow can be regarded as the limiting boiling flow pattern, which can provide sufficient cooling condition.

The flow rate increase factor of the surface coolant in the experiment is about 3 during the conversion from nucleation to annular flow. It can be explained as follows: when mass quality is low, the two-phase friction pressure drop is relatively small and can be ignored when compared with vapor-phase dynamic effect. As for q > 250 kW/mm, the coolant flow rate is reduced and dispersed annular flow transforms into dispersed flow (trans-critical heat transfer) because of the slightly increase of real vacuole volume fraction and the increased friction

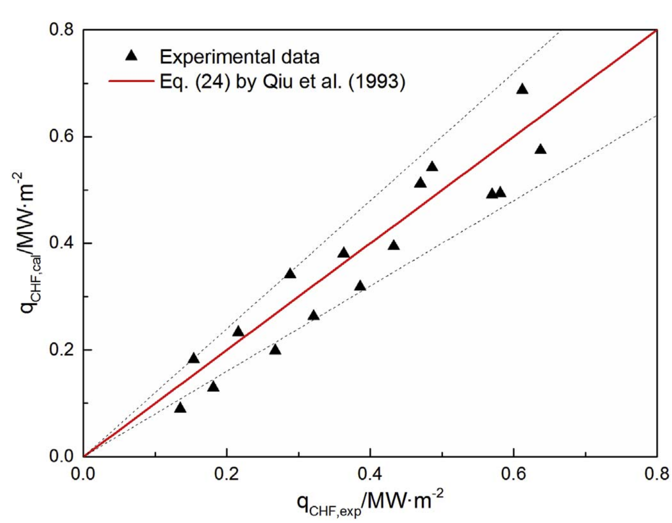

Fig. 1. Comparison of calculated values and experimental data for the critical heat flux of sodium boiling (Qiu et al., 1993).

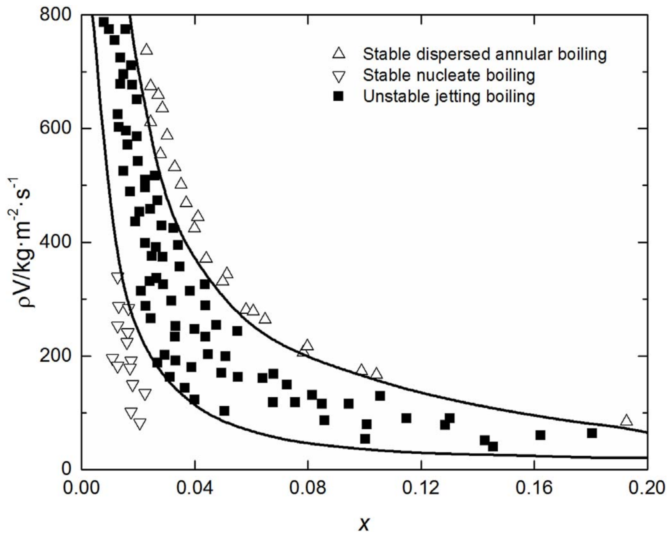

Fig. 2. Schematic diagram of boiling process (Sorokin et al., 1999).

pressure drop of two-phase flow.

(2) For the calculation of the critical heat flux of liquid sodium boiling under forced convection in rod bundle channel, Ishii and Fauske (1983) proposed the following correlations on the basis of experiment research:

$$q_{CHF} = G\Delta h_{in} + \left(\frac{1}{C_0} - 0.11\right) i_{lg}[g\rho_g(\rho_l - \rho_g)D]^{1/2} \tag{25}$$

$$C_0 = 1.2 - 0.2\left(\frac{\rho_g}{\rho_l}\right)^{1/2} \tag{26}$$

where q_{CHF} is the critical heat flux; ρ_g is the sodium vapor density; ρ_l is the liquid sodium density; G is the mass flow rate; Δh_{in} is the inlet subcooling enthalpy; D is the inner diameter of the round tube; C_0 is the density correction factor.

(3) By the review of the relevant experiment data, Sorokin and Efanov (Kirillov et al., 1984) indicated that the heat transfer coefficient can be expressed by the following correlation for the pool boiling and the in-tube boiling of potassium and the boiling of Na-K alloy in rod bundles (Kirillov et al., 1984):

$$h = Aq^m p^n \tag{27}$$

where $m = 0.7$, $n = 0.1$–0.15 and $A = 4$–7.

Based on the experimental data, the liquid metal boiling heat transfer coefficient can be well expressed as the following equation:

$$Nu = 8.7 \times 10^4 Pe^{0.7} K_P^{0.7} \tag{28}$$

where:

$$Nu = \frac{\alpha}{\lambda_l}\sqrt{\frac{\sigma}{\rho_l - \rho_g}} \tag{29}$$

$$Pe = \frac{qc\rho_l}{r\rho_g\lambda_l}\sqrt{\frac{\sigma}{\rho_l - \rho_g}} \tag{30}$$

$$Kp = \frac{p}{\sqrt{\sigma g(\rho_l - \rho_g)}} \tag{31}$$

(4) By taking the fuel element of the reactor core as the calculation object, Martsiniouk and Sorokin (2000) found that the heat transfer from the wall-surface to the two-phase mixture can be calculated by the following correlation:

$$Nu = 0.017\,\mathrm{Re}_{cm}^{0.8}\mathrm{Pr}_w^{0.8}Y \tag{32}$$

where:

$$\mathrm{Re}_{cm} = \left(\frac{\rho wd}{\mu_g}\right)\left[1 + x\left(\frac{\rho_l}{\rho_g} - 1\right)\right]$$

$$\begin{cases} Y = 1 + 0.5\left(\frac{\rho_l}{\rho_g} - 1\right)^{0.8}(1 - x) & \frac{\rho_l}{\rho_g} < 450 \\ Y = 1 + 70(1 - x) & \frac{\rho_l}{\rho_g} > 450 \end{cases}$$

They also obtained the correlation for the heat transfer from the wall-surface to the vapor as follows:

$$Nu = 0.023\,\mathrm{Re}_g^{0.8}\mathrm{Pr}_g^{0.4} \tag{33}$$

(5) Carbajo and Rose (1984) believed that it is very important to predict the time from the incipient boiling to the dryout, and they have also obtained a correlation for dryout time according to the experiment data of 19-pin bundle (Wantland et al., 1979) and 61-pin bundle (Rose et al., 1980). In the experiment, the power was 4.1–15.3 kW per rod, the inlet velocity 0.22–1.35 m/s, the inlet flow rate 4.6×10^{-6}-26.3×10^{-6} m^3/s/pin, the inlet temperature of the test section 386–450 °C, the boiling temperature 913–982 °C, the cladding perimeter of bundle 0.11 and 0.2 m, and the flow area 3.7×10^{-4} and 12.9×10^{-4} m^2.

For appearance of boiling, the outlet temperature was set as 700 °C, and the flow rate was gradually reduced.

Denote Q as the corresponding flow rate of incipient boiling and two factors are defined firstly in order to obtain the correlation for dryout time.

$$K_1 = \frac{P}{Q\rho_{in}\Delta h_{sub}} \tag{34}$$

$$K_2 = 1000\frac{L}{Nv} = \frac{1000LA}{NQ} \tag{35}$$

where P is the total heating power (kW), Q is the flow rate of incipient boiling (m^3/s), v is the inlet flow rate (m/s), ρ_{in} is the inlet sodium density (kg/m^3), L is the outer cover perimeter (m) of the rod bundle, A is the flow area (m^2) and N the number of rods.

K_1 obtained thereby is dimensionless and K_2 has the dimension of time(s), wherein K_1 is the specific value of the total input power and the power needed for saturating all inlet sodium at the end of the heating section.

Through test and error analysis, relation factor $I_d = K_2^{0.5}/K_1$ can be obtained. The relationships between I_d and the time of dryout from incipient boiling are as follows:

For forced convection:

$$t_d = 10^{(0.76I_d - 0.32)},\ 1.6 \le I_d \le 2.5,\ 8s \le t_d \le 39s \tag{36}$$

For natural convection:

$$t_d = 10^{0.32I_d^3 - 0.98I_d^2 + 27},\ 2.5 \le I_d \le 3.15,\ 39s \le t_d \le 1000s \tag{37}$$

In order to apply the above correlation, it is necessary to determine K_1 first: if K_1 is less than 1, liquid sodium will be not boiled (without considering the sub-cooled boiling); if K_1 is more than 1, K_2 and I_d should be calculated. If I_d is less than 1.6 and t_d is less than 8, the dryout phenomenon rapidly appears; if I_d is larger than 1.6 but less than or equal to 2.5, the correlation for forced convection should be adopted; if I_d is larger than but less than or equal to 3.15, the correlation for natural convection should be adopted; if I_d is larger than 3.15 and t_d is larger than 1000, the boiling will last for a long time and it can be considered that dryout does not occur.

Besides the above analysis, many scholars have also obtained many correlations for the heat transfer characteristics of liquid sodium in bundle channels, specifically as follows:

(1) Correlations by Dwyer and Tu (1960), and Friedland and Bonilla (1961).

Dwyer and Tu (1960), Friedland and Bonilla (1961) derived two equations for heat transfer to liquid metal flowing in triangle bundle of circular rods as follows:

$$Nu = 0.93 + 10.81PD - 2.01PD^2 + 0.0252PD^{0.273}(\psi Pe)^{0.8} \tag{38}$$

$$Nu = 7.0 + 3.8PD^{1.52} + 0.027PD^{0.27}(\psi Pe)^{0.8} \tag{39}$$

where ψ is the ratio of the eddy diffusivity of heat to the eddy diffusivity of momentum; in the later part of discussion, ψ is assumed to be one.

The main difference between the above two equations lies in the different assumptions about the velocity profile. For Peclet numbers between 70 and 10^4 and pitch-to-diameter ratios P/D of 1.375 up to 2.2, Eq. (38) is recommended, while the range of the applicability of Eq. (39) is specified to be Pe $= 0 \sim 10^5$ and $PD = 1.3$–10.

(2) Correlation by Mareska and Dwyer (1964).

Mareska and Dwyer carried out experiments for 13 rods of 13 mm outer diameter, which were arranged in an equilateral triangular lattice with the pitch-to-diameter ratio of 1.75. The working medium was liquid mercury (Pr ~ 0.02). To describe the test data, they proposed the following semi-empirical correlation:

$$Nu = 6.66 + 3.126PD + 1.184PD^2 + 0.0155(\psi Pe)^{0.86} \tag{40}$$

Eq. (40) is recommended for triangular bundles in the range of Peclet numbers between 70 and 10^4 and pitch-to-diameter ratios P/D of 1.3 up to 3, and ψ is again the ratio of the eddy diffusivity of heat to the eddy diffusivity of momentum and is assumed to be one for later discussion.

(3) Correlation by Subbotin et al. (1965).

For the flow and heat transfer of liquid metal in a triangle lattice of rods with the pitch-to-diameter ratio P/D of 1.1 up to 1.5 and Peclet numbers of 80-4,000, Subbotin et al. (1965) recommended the following correlation:

$$Nu = 0.58\left(\frac{d_h}{d}\right)^{0.55} Pe^{0.45} \tag{41}$$

where d_h and d are the hydraulic diameter and the rod diameter respectively. For the triangle lattice, the above equation becomes:

$$Nu = 0.58\left(\frac{2\sqrt{3}}{\pi}PD^2 - 1\right)^{0.55} Pe^{0.45} \tag{42}$$

Although Eq. (41) was derived for the triangle lattice, it is also applicable to the square rod-cluster and becomes the following form:

$$Nu = 0.58\left(\frac{4}{\pi}PD^2 - 1\right)^{0.55} Pe^{0.45} \tag{43}$$

(4) Correlation by Borishanskii et al. (1969)

Borishanskii et al. (1969) carried out the experiment with working sections consisting of seven tubes of 22 mm outer diameter arranged in equilateral triangular bundles with different pitch-to-diameter ratios (1.1, 1.3, 1.4 and 1.5). The heated length of the test section was 800 mm, and the temperatures varied from 206 °C to 236 °C. The results of three groups of the experiments are presented in the following paragraph. In their paper, the working fluids are not explicitly specified though, the Prandtl numbers for the three groups of the presented results are given: 0.007, 0.03 and 0.024. On the basis of the experiment data, Borishanskii et al. (1969) obtained the following correlation:

$$Nu = 24.15\log(-8.12 + 12.76PD - 3.65PD^2) + 0.0174(1 - e^{-6(PD-1)})B \tag{44}$$

where $B = \begin{cases} 0, & Pe < 200 \\ (Pe - 200)^{0.9}, & Pe \geq 200 \end{cases}$.

Eq. (44) is applicable to triangular bundles in the range of Peclet numbers of 60-2200 and pitch-to-diameter ratios P/D of 1.1–1.5.

(5) Correlation by Gräber and Rieger (1972)

Gräber and Rieger (1972) measured three groups of experimental data, wherein the test section consisted of 31 tubes of 12 mm outer diameter arranged in equilateral triangular bundles with pitch-to-diameter ratios P/D of 1.25, 1.6 and 1.95. The working medium was 44% Na-56%K at temperatures of 100–425 °C, and the variation of Prandtl number with temperatures was 0.011–0.024. Gräber and Rieger totally gave 246 data pairs of Nu against Pe and they fitted the data into following correlation.

$$Nu = 0.25 + 6.2PD + (0.032PD - 0.007)Pe^{0.8-0.024PD} \tag{45}$$

(6) Correlation by Ushakov et al. (1977).

For the flow of liquid metal in a triangle lattice of rods with the pitch-to-diameter ratio P/D of 1.3–2.0 and Peclet numbers up to 4,000, Ushakov recommend the following correlation:

$$Nu = 7.55PD - \frac{20}{PD^{13}} + \frac{0.041}{PD^2}\left[1 - \frac{1}{PD^{30} - 1/6 + \sqrt{1.15 + 1.24\varepsilon_6}}\right] \times Pe^{0.56+0.19x-0.1/PD^{80}} \tag{46}$$

where ε_6 is the "approximate criterion of thermal similarity of the fuel rods in the triangular assembly", d_h and d are the hydraulic diameter and the rod diameter respectively. For simplicity, Eq. (46) can be reduced as follows:

$$Nu = 7.55PD - \frac{20}{PD^{13}} + \frac{0.041}{PD^2}Pe^{0.56+0.19PD} \tag{47}$$

(7) Correlation by Zhukov et al. (2002).

In the framework of BREST lead-cooled reactor project, Zhukov et al. (2002) conducted experiments extensively to study the heat transfer characteristics of liquid metal in tubes arranged in square bundles. The test section consisted of 25 tubes with the inner diameter of 12 mm, and four groups of experimental data were measured for pitch-to-diameter ratios P/D of 1.25, 1.28, 1.34 and 1.46. The working medium was 22%Na-78%K at a temperature of about 50 °C, and the heated length of the assembly was 980 mm. Totally, Zhukov gave 36 data pairs of Nu vs. Pe and obtained the following relation according to the experimental data:

$$Nu = 7.55PD - 14PD^{-5} + 0.007Pe^{0.64+0.246PD} \tag{48}$$

(8) Correlation by Mikityuk (2009).

Through an extensive review of data of heat transfer to liquid metal for rod bundles, Mikityuk found that the correlation proposed by Gräber, i.e. Eq. (45), as well as the correlation given by Ushakov, i.e. Eq. (46), has the highest quality among the correlations considered in predicting the experimental results, and he further derived a new correlation as a best fit for the data analyzed (with mean absolute error of −0.1 and root-mean-square error of 1.9):

$$Nu = 0.047(1 - e^{-3.8(PD-1)})(Pe^{0.77} + 250) \tag{49}$$

The correlation is recommended for Peclet numbers of 30–5000 and pitch-to-diameter ratios of 1.1–1.95.

(9) Correlation by Ma et al. (2012)

Ma et al. (2012) proposed a method theoretically for prediction of liquid metal heat transfer for rod bundles based on annuli. The authors suggested that an annulus can be seen as a bundle with a single rod without complicated interaction among rods so the bundles arranged in an equilateral triangular pattern or square pattern may be treated as a special equivalent annulus which will consider those interaction effects in the bundles. Based on these assumptions, four corrections, i.e. equivalent hydraulic diameter correction, temperature difference correlation, interaction correction and shape correction, are carried out to make the correlation for annuli applicable to the bundle cases. For different arrangements of rod bundles, the authors have deduced corresponding correlations and the predicted values are in good agreement with relevant experimental data.

The correlations have a unified form, as follows:

$$Nu = \left(4.82 + 0.697y + 0.0222(\overline{\psi}Pe)^{0.758y^{0.053}}\right)C_h C_t C_i C_s \tag{50}$$

where $Nu = (4.82 + 0.697y + 0.0222(\overline{\psi}Pe)^{0.758y^{0.053}})$ is the correlation given by Dwyer and Tu (1965) for annulus; Factors, C_h, C_t, C_i and C_s, are the equivalent hydraulic diameter correction, temperature difference correlation, interaction correction and shape correction, respectively.

For triangular bundles:

$$\overline{\psi} = 1 - \frac{1.82}{\Pr(\varepsilon_M/v)_{\max}^{1.4}}, \quad \log(\varepsilon_M/v)_{\max} = 0.887\log(\mathrm{Re}) - 2$$

$$y = 2.373(P/D) - 1.311, \quad C_h = \left(\frac{2\sqrt{3}}{\pi}(P/D)^2 - 1\right)/(y-1), \quad C_t = 0.088\ln y + 1.359$$

$$C_i = 2(P/D-1)/(y-1), \quad C_s = \begin{cases} 4\sqrt{3}(P/D)^2/\pi - 1 & P/D < 1.176 \\ 1.05 & P/D \geq 1.176 \end{cases}$$

For square bundles:

$$\overline{\psi} = 1 - \frac{1.82}{\Pr(\varepsilon_M/v)_{\max}^{1.4}}, \quad \log(\varepsilon_M/v)_{\max} = 0.887\log(\mathrm{Re}) - 2, \quad y = 2.593P/D - 1.374.$$

$$C_h = \left(\frac{4}{\pi}(P/D)^2 - 1\right)/(y-1), \quad C_t = 0.088\ln y + 1.359,$$

$$C_i = 2(P/D-1)/(y-1), \quad C_s = \begin{cases} 4(P/D)^2/\pi - 1 & P/D < 1.293 \\ 1.128 & P/D \geq 1.293 \end{cases}$$

To evaluate the heat transfer models, three sets of experimental data are used here for discussion, one measured by Gräber and Rieger (1972), the other two given by Zhukov et al. (2002). Compared with Peclet number, the variation range of pitch-to-diameter ratios is relatively narrow and doesn't exert as much impact as Peclet does on the Nusselt number. Two typical pitch-to-diameter ratios, 1.25 and 1.3, are chosen here for the comparison of the models presented above. As there are two more dimensionless numbers, Reynolds number and Prandtl number in the model proposed by Ma et al. (2012), for sake of simple discussion, a constant value of 0.007 for Prandtl number of liquid sodium is assumed. Also, since the triangular lattice is widely used in SFRs, the correlations established for square lattice, i.e. Eq. (43) and the square bundle model by Ma et al. (2012), are not discussed here. The comparison results are shown in the Fig. 3 and Fig. 4. From Figs. 3 and 4, we can see that there is a similar trend for all models presented above that given a constant pitch-to diameter ratio, the calculated Nusselt number all increases with Peclet number and it grows faster when Peclet number is relatively small. Despite the similar trend, the calculated results by different correlations differ much from each other. The correlations by Dwyer and Tu (1960) and Mareska and Dwyer (1964) generally give a larger Nusselt number than the other correlations. By comparison of the correlations and experimental data from Borishanskii et al. (1969), Gräber and Rieger (1972) and Zhukov et al. (2002), it can be seen from the figures that the correlations by Gräber and Rieger (1972), Ushakov et al. (1977), Mikityuk (2009), Ma et al. (2012) give a better prediction than the other correlations when Peclet number is within the discussed range as shown in the figures. Of course, further evaluations of the correlations are still in need in the future to determine their applicability.

2.5. A brief summary

This section mainly discusses the characteristics of sodium boiling in pools, round tubes and annuli, including the boiling heat transfer correlations, the incipient boiling wall superheat and critical heat flux of dryout type. In the part of pool type boiling, we can see that the heat transfer coefficient is often directly correlated to heat flux instead of being expressed in dimensionless number, i.e. Nusselt number. Additionally, the critical heat flux is a major concern in the research of pool type boiling heat transfer. As for the heat transfer in round tube bundles, there is relatively more research, in which the heat transfer coefficient, the incipient boiling superheat and critical heat flux are the focus. Currently, the relevant research on the boiling of sodium in rod bundles is still relatively insufficient in published literature. This paper summarizes some known research done by Sorokin et al. (1999), Ishii and Fauske (1983), Martsiniouk and Sorokin (2000), Carbajo and Rose (1984). Their work made some achievements, like obtaining the boiling flow patterns in rod bundle channels, a correlation for critical heat flux and an equation for boiling heat transfer coefficient etc. As a supplement, some typical models for heat transfer to liquid metals are also presented and discussed here. And it is found that, within the discussed range of Peclet number, correlations by Gräber and Rieger (1972), Ushakov et al. (1977), Mikityuk (2009), Ma et al. (2012) give a better prediction than the other correlations. But a further evaluation is necessary in the future.

3. Research on two-phase flow characteristics of liquid sodium

The relevant research of sodium two-phase flow is summarized in Table 2. From Table 2, we can see that for the study of system flow

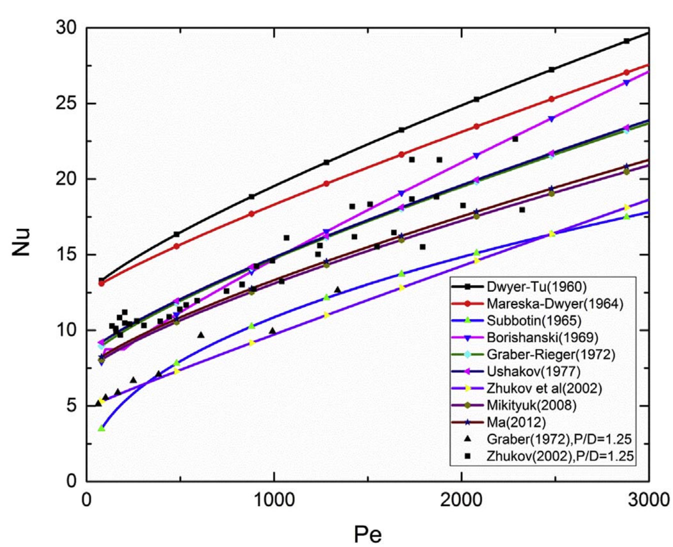

Fig. 3. Comparison of heat transfer models (P/D = 1.25).

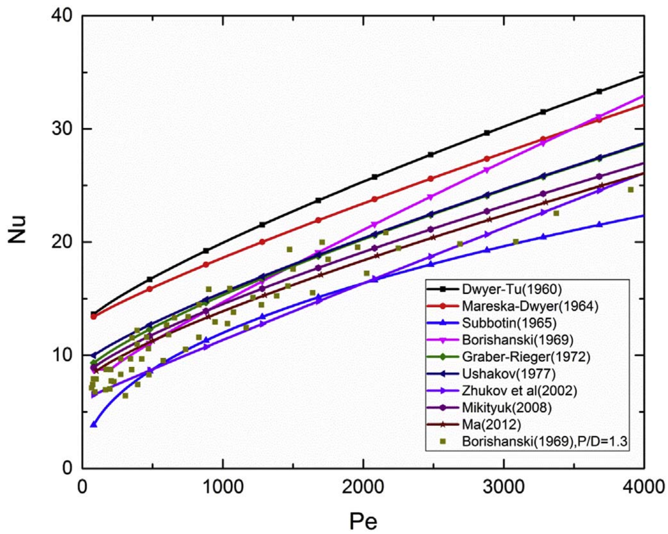

Fig. 4. Comparison of heat transfer models (P/D = 1.3).

characteristics, besides the two-phase friction multiplier, the flow instability research is also one of the important parts. In a heating flow system, if the fluid has phase change, namely: the two-phase flow appears, the significant fluid volume change in a non-uniform form may cause flow instability. The flow instability is harmful to the operation of heating flow system in the following aspects: the mechanical force caused by flow and pressure fluctuation can lead to the harmful mechanical motion, and the continuous mechanical vibration can cause the fatigue damage of the system components; the flow fluctuation can disturb the control system and cause the periodic change of the local thermal stress of the system components, thus leading to the fatigue damage of them; the flow instability can damage the heat transfer performance in the system and significantly reduce the heat transmission capability of the system and critical heat flux, thus accelerating the appearance of critical boiling. Generally the flow instability refers to the flow excursion generated by the slightly-disturbed fluid or the flow fluctuation with constant or variable amplitude at a certain frequency in a two-phase system in which mass flow rate, pressure drop and void fraction are coupled with each other. As pressure drop plays an important part in a two-phase flow system, this chapter presented here is aimed at introducing the investigation results of the experimental and theoretical research on the two-phase flow pressure drop in rod bundle channel, the typical flow channel in SFR.

3.1. Computational models for pressure drop

Generally, the pressure drop across the inlet and outlet of a flow channel can be expressed in the following form:

$$p_i - p_o = \Delta p_{Acc} + \Delta p_{Grav} + \Delta p_{Fric} + \Delta p_{Form} \tag{51}$$

where Δp_{Acc}, denotes the acceleration pressure drop due to a change in velocity, void or density, Δp_{Grav} the pressure drop for different elevation, Δp_{Fric} the frictional pressure loss caused by shear stress at the wall, and Δp_{Form} the pressure drop owing to an sudden change in flow direction or geometry.

For the pressure drop due to acceleration or a change in elevation, which is dependent on the physical property for a given geometry, the calculation methods are given as follows.

$$\Delta p_{Acc} = [\rho_l \alpha_l v_l^2 - \rho_g \alpha_g v_g^2]_{out} - [\rho_l \alpha_l v_l^2 - \rho_g \alpha_g v_g^2]_{in} \tag{52}$$

$$\Delta p_{Grav} = \int_{z_{in}}^{z_{out}} (\alpha_l \rho_l + \alpha_g \rho_g) g \cos\theta dz \tag{53}$$

where ρ, α and v represent density, void fraction and velocity respectively; z and g are the elevation and gravitational acceleration, θ the angle of flow direction with the vertical; the subscript l and g denote liquid and gas respectively.

The other two parts of pressure drop are more complicated and they are discussed more in detail in the subsequent chapters.

3.2. Two-phase friction pressure drop

It has been experimentally observed that for a given mass flow rate, the pressure drop of two-phase flow can be significantly greater than that of the corresponding single-phase flow. In order to obtain two-phase friction pressure drop, the typical method is to correlate the two-phase flow pressure drop with the single-phase flow pressure drop at the same mass flow flux by use of a multiplier ϕ_l^2 to account for the two-phase effects, i.e.

$$\Delta p_{2\phi} = \phi_l^2 \Delta p_{1\phi} = \phi_l^2 f_l \frac{L}{D_h} \frac{\rho_l v_l^2}{2} \tag{54}$$

For the calculation of single-phase pressure drop in wire-wrapped bundles, many models have been proposed by scholars, such as Novendstern (1972), Rehme (1973), Engel et al. (1979), Chiu et al. (1978), Cheng and Todreas (1986), Baxi and Donne (1981) etc. and the details of these models have been summarized by Chen et al. (2014). Thus, the focus of this section is to present a selection of correlations for computing the two-phase friction multiplier factor ϕ_l^2 as follows:

(1) Lockhart and Martinelli (1949).

Lockhart and Martinelli defined the parameter X_{LM} as follows:

$$X_{LM} = \left(\frac{1-x}{1}\right)^{0.9} \left(\frac{\rho_g}{\rho_l}\right)^{0.5} \left(\frac{\mu_l}{\mu_g}\right)^{0.1} \tag{55}$$

On the basis of measurement for air-water horizontal flow in pipes at around atmospheric pressure, the authors suggested that ϕ_1 can be correlated uniquely as a function of X_{LM}, and they proposed the following correlation for turbulent flow:

$$\phi_l^2 = 1 + \frac{20}{X_{LM}} + \frac{1}{X_{LM}^2} \tag{56}$$

(2) Lottes and Flinn (1956).

The friction multiplier factor was expressed by Lottes and Flinn as a function of vapor volume fraction, and this function is independent of

Table 2
Summary of sodium two-phase flow research.

Reference	Major target of research	Geometry of research object	Research method
Chen and Kalish (1970)	two-phase friction multiplier	round tube	experiment
Kaiser et al. (1974)	two-phase friction multiplier, flow patterns	round tube	experiment
Grolmes and Fauske (1981)	flow patterns and flow instability	round tube	theoretical analysis
Kirillov et al. (1984)	pressure drop and flow instability	rod bundle	modeling analysis and theoretical calculation
Ninokata (1985)	flow characteristics	triangular rod bundle	sub-channel model with empirical correlations
Seiler et al. (2010)	two-phase friction multiplier	rod bundle	experiment and theoretical calculation

pressure and flow rate. They proposed that the two-phase frictional effect is primarily caused by drag of the liquid phase along the wall for small vapor quality values, and:

$$\phi_l^2 = \left(\frac{1}{1-\alpha}\right)^2 \tag{57}$$

This model directly results from the triangular relationship associating the liquid film flow rate with the film thickness and wall shear stress and it has been adopted by Ninokata to predict the two-phase flow pressure drop in SABENA code (Ninokata and Deguchi, 1989).

For convenient discussion and comparison with the other models shown in Fig. 5, the above correlation is expressed as a function of the parameter X_{LM}. For the void fraction, the correlation of Nguyen (1985) is used:

$$\alpha = (1 + X_{LM}^{0.8})^{-0.378} \tag{58}$$

(3) Kottowski and Savatteri (1984).

On the basis of the round-tube quasi stable-state experiments, Kottowski adopted the least square method to fit the experimentally measured data and obtained the following correlation:

$$\log \phi_l = 0.1046(\log X_{LM})^2 - 0.5098 \log X_{LM} + 0.6252, \ \ 0.07 < X_{LM} < 30 \tag{59}$$

(4) Kaiser et al. (1989).

The Kaiser correlation was obtained from the quasi stable-state sodium boiling experiments conducted in a 7-pin test section which was electrically-heated:

$$\ln \phi_1 = 1.48 - 1.05 ln\sqrt{X_{LM}} + 0.09(\ln \sqrt{X_{LM}})^2 \tag{60}$$

(5) Qiu et al. (2015)

Qiu et al. (2015) performed boiling experiments on sodium flowing through an annulus, and the authors obtained a correlation for the two-phase friction multiplier based on the experimental data:

$$\phi_l^2 = 1 + \frac{8.57}{X_{LM}} + \frac{1}{X_{LM}^2} \tag{61}$$

(6) Other correlations

Based on the air-water horizontal flow data given by Lockhart and Martinelli, several scholars plotted ϕ_l^2 against $(1-\alpha)$ and found $\phi_l^2 = (1-\alpha)^{-n}$. Lottes and Flinn adopted $n = 2$ (see Eq. (57)), but afterwards, according to the analysis of Katsuhara (1958) and Richardson (1959), n is suggested as 1.75.

Subsequently, on the basis of the measured data of the sodium loop, in which the electromagnetic induction heating method was adopted for the round-tube test section with the inner diameter of 9 mm and the heating length of 200 mm, a correlation as follows (Kaiser et al., 1974) was derived for the two-phase friction pressure drop multiplier:

$$\phi_l = 8.2 X_{LM}^{-0.55} \tag{62}$$

Finally, the following correlation is presented here, which was proposed by Chen and Kalish (1970) on the basis of the measurements with potassium:

$$\ln\left(\frac{1}{\phi_l}\right) = -1.59 + 0.518 \ln(X_{LM}) - 0.0867(\ln X_{LM})^2 \tag{63}$$

To evaluate the different models, the calculation results of the above correlations and the experimental data are compared as follows:

As can be seen from Fig. 5, the calculation results of different models have similar trends with the existing experimental data, but there are large discrepancies among different models, which makes it hard to propose an accurate model to predict the two-phase friction pressure drop. A good explanation of such phenomena is that the different contributions to the two-phase flow pressure drop (elevation, acceleration, friction and singular) cannot be separately measured. Additionally, the data for determining the empirical models are the total pressure drops over a given length of the experimental device and most often the two-phase flow parameters such as void fraction and flow quality are required for the application of these correlations. Due to the great difficulty to measure these parameters in boiling liquid metal, an auxiliary boiling model, which provides the void fraction α and flow quality x, with associated numerical approximations, is generally used to derive the friction pressure drop correlations. Finally, it is also necessary to consider the difficulties related to experimental errors. Kottowski (1994) pointed out two main measurement errors during experiments as follows: (1) pressure measurement error related to high temperatures, temperature gradients and temperature variations; (2) the heat loss during high-temperature operations, which can strongly influence the thermal balance considered for determining the quality or void fraction.

The first group of data in Fig. 5, issued by Lurie and Noyes (1964), is based on sodium boiling experiment in an annular test section with inner diameter of 6.35 mm and outer diameter of 12.7 mm. After post-treatment of the total pressure drop measurements, the data presented in Fig. 5 correspond to the frictional pressure drop. Within the range of flow rates and pressure during the experiments (liquid Reynolds number of 6000-34,000 and pressure of 0.1–0.7 bar), when the pressure drop factor is expressed as a function of the parameter X_{LM}, the pressure drop factor will not depend on additional parameters, wherein the slight data scattering is cause by experimental variations and errors (especially in the region of high X_{LM} and low quality). As proposed by authors Lurie and Noyes (1964), most experimental data fall between the Lockhart-Martinelli correlations used for viscous and turbulent flow.

The other groups of data in Fig. 5 were measured by Kaiser et al. (1974) and Kaiser (1979) in the sodium boiling loop NSK, Karlsruhe, wherein the loop includes different test sections: a round tube and a 7-pin bundle. More than 60 experimental points representing the two-phase pressure drop were obtained in the quasi stable-state boiling

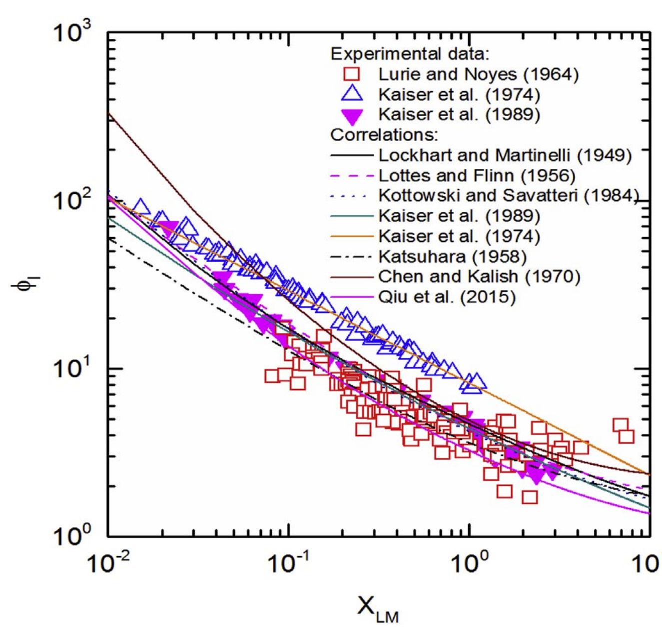

Fig. 5. Comparison of the two-phase friction multiplier for different models with experimental data.

experiments carried out in the electrically heated 7-pin bundle. From Fig. 3, we can see that the best approximation for these experimental data is Eq. (62).

From Fig. 3, it can be seen that the data measured in the round tube by Kaiser differ much from the other groups of experimental data. The measurement results of 7-pin bundle experiment carried out by Kaiser are within Lurie measurement result range. In the two groups of results, more scattering is found in low quality region.

To sum up, due to the significant differences among different models as well as experimental data, great efforts are still needed to develop a more general model for the calculation of two-phase multiplier. Before the establishment of a more accurate model, at present, the Kaiser correlation and Lockhart-Martinelli correlation, i.e. Eqs. (60) and (56) are recommended for the prediction of friction multiplier as they both give a relatively satisfactory result in different geometries except round tubes.

3.3. Two-phase local pressure drop

The two-phase pressure drop caused by local barrier such as grid spacers can be treated as the friction pressure drop. Therefore, as in Eq. (54), the local two-phase pressure drop can be obtained by the corresponding single-phase pressure drop times an appropriate two-phase multiplier:

$$\Delta P_{2\phi local} = K_{1\phi}\frac{\rho_l v_l^2}{2}\Phi \tag{64}$$

where Φ is the two-phase local pressure drop multiplier.

(1) Homogeneous model

On the basis of massive experiment data, Lahey et al. (1993) reported that the two-phase pressure drop data can be well correlated by a homogeneous phase multiplier for a wide range of grid-type spacers, and they proposed the following correlation:

$$\Phi = \left(\frac{x}{\rho_g} + \frac{1-x}{\rho_l}\right)\rho_l \tag{65}$$

(2) Slip model

Assuming a slip between the gas phase and the liquid phase, the two-phase pressure drop multiplier can be obtained by the following correlations:

$$\Phi = \frac{\rho_l}{\alpha\rho_g + (1-\alpha)\rho_l}, \quad \alpha = \frac{1}{1 + \left(\frac{1-x}{x}\right)S\frac{\rho_g}{\rho_l}} \tag{66}$$

As can be seen from Eq. (66), when the slip ratio $S = \frac{v_g}{v_l}$ is equal to 1, the slip model can be simplified into the homogeneous model.

3.4. A brief summary

In this chapter, computational models for pressure drop are discussed. And as mentioned earlier, in two-phase flow, the pressure drop calculation is one of the major concerns, in which the calculation of two-phase pressure drop multiplier is of great importance and necessity. Here, correlations for the multiplier are mainly discussed and compared. The result shows that the Kaiser correlation and Lockhart-Martinelli correlation, i.e. Eqs. (60) and (56) are suggested because they both give a better prediction of friction multiplier in different geometries except round tubes. Besides, models for calculation of two-phase local pressure drop are given.

4. Conclusions and prospects

As the sodium-cooled fast reactor is considered by countries as one of the Generation IV reactors with the most mature technology and the best application prospect in near future, it is of great significance to study the flow and heat transfer characteristics of liquid sodium. During last decades, much experimental and theoretical research has been carried out and some achievements were obtained. In the first stage of relevant research, the study mainly focused on the single-phase flow characteristics of liquid sodium and it was found that the single-phase flow characteristics of liquid metal are not quite different from that of non-metallic liquid, and can be analogized by conventional fluids, such as water. Therefore, the correlations for single-phase flow pressure drop of water are generally applicable to liquid sodium. But due to the special channels in SFR, i.e. the wire-wrapped bundle channels, additional efforts have been made to study the flow characteristics in such geometries and many models have also been proposed by scholars, and the details of these models can be referred to the paper by Chen et al. (2014) as this paper presents before. However, although there are lots of similarities between the flow characteristics of liquid metal and that of non-metallic liquid, the heat transfer characteristics between these two kinds of fluid are quite different. As discussed in the introduction part of this paper, different much from non-metallic liquid, the main feature of heat transfer to the liquid metal is the large contributions of thermal conduction. This special characteristic has been widely confirmed by extensive experiments done in various kinds of channels, such as round tubes, annular channels and rod bundles, and a number of models have been established for the heat transfer calculation of liquid metal under different conditions. However, as the relevant studies go deeper, it was pointed out by scholars (Waltar and Padilla, 1977; Waltar and Reynolds, 1981; Hennies et al., 1990; Maschek and Struwe, 2000; Farmer, 2012) that the boiling of liquid sodium might occur during some hypothetical accidents in SFR, which could do great harm to the safety of SFR. So the development of a two-phase flow and heat transfer model of liquid sodium becomes increasingly necessary and important.

Compared with the abundant research on the single-phase flow and heat transfer of liquid sodium, the studies of boiling of liquid sodium are relatively limited. The relevant experimental research having been carried out mainly focused on the boiling in round tubes and annular channels, as can be seen in Table 1, with few experiments of the sodium boiling in rod bundles due to the great difficulty to perform such kind of experiment. Thus, although some theoretical and experimental research has been carried out, a uniform conclusion in realms of boiling heat transfer is not obtained yet and even conflictions exist between the conclusions drawn by different scholars. For example, the obtained experimental data of incipient boiling wall superheat scatter overall, correspondingly leading to the great difference between correlations by different scholars. For the boiling heat transfer coefficient of liquid sodium, although there are relatively abundant experimental data and correlations by different scholars for the geometries of round tube and annular channel, certain discrepancy still exists between each other. In addition, despite many correlations for the heat transfer to single-phase liquid sodium, the correlations applicable to the boiling in rod bundles are rare, which are more representative under conditions of accidents in SFRs. Fortunately, in comparison to two-phase heat transfer characteristics, the two-phase flow characteristics of sodium in rod bundles can well evaluated by typical methods, i.e. using a pressure drop multiplier to account for the two-phase effects. Numbers of correlations have been proposed by scholars, like those given by Lockhart and Martinelli (1949), Kottowski and Savatteri (1984) and Kaiser et al. (1989) etc., and the typical ones are summarized in this paper. Although there're discrepancies between the obtained correlations, based on the comparisons of those correlations for the pressure drop multiplier, the Kaiser correlation (Kaiser et al. (1989)) and Lockhart-Martinelli correlation (Lockhart and Martinelli (1949)), i.e. Eq. (60) and Eq.

(56) presented in this paper are suggested at current stage for the prediction of friction multiplier as they both give a relatively satisfactory result in different geometries except round tubes.

To sum up, the relevant studies of sodium two phases flow and heat transfer are relatively insufficient, especially in the aspect of sodium boiling heat transfer in bundle channels. And it can be seen from Table 1 that the relevant research was mainly carried out between 1960s and 1980s, like the typical theoretical analysis done by Mostinskii (1963), experiment conducted by AladevI et al. (1968) and the experiment performed by Ishii and Fauske (1983) etc. Due to the limitations of experimental condition, as mentioned before, conclusions made by scholars are not well consistent with each other. Also, the lack of theoretical and experimental research on the sodium boiling heat transfer in bundle channels makes it hard to consider the actual boiling conditions under hypothetical accidents in SFRs. In addition, the occurrence of sodium two-phase flow might cause flow instability, which poses a threat to the safety of SFRs. Therefore, further research on the sodium boiling in rod bundles as well as the flow instability induced, is in great need for a more accurate and general model to cope with these problems. It is worth mentioning that due to the complexity in performing sodium boiling experiments, it is more necessary to take into account of the influence of the experiment conditions and any factors having not been considered yet in the previous experiments, which to some extent brought about great discrepancies between the experimental data by different scholars.

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