International Journal of Heat and Mass Transfer 87 (2015) 201–211

Contents lists available at ScienceDirect

International Journal of Heat and Mass Transfer

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

Critical heat flux for downward-facing saturated pool boiling on pin fin surfaces

Dawen Zhong, Ji'an Meng *, Zhixin Li, Zengyuan Guo

Key Laboratory of Thermal Science and Power Engineering of Ministry of Education, Department of Engineering Mechanics, Tsinghua University, Beijing 100084, PR China

ARTICLE INFO

Article history:
Received 20 January 2015
Received in revised form 24 March 2015
Accepted 2 April 2015
Available online 17 April 2015

Keywords:
Downward facing boiling
Critical heat flux
Pin fin surface
Enhancement

ABSTRACT

In-vessel retention is a key severe accident management strategy now being adopted by some nuclear power plants which has been proposed for some advanced light water reactors. Saturated pool boiling heat transfer coefficients and the critical heat flux (CHF) were measured from three downward facing pin fin surfaces in deionized water with the fins added to enhance the CHF. The inclination angles were 5°, 30°, 45°, 60° and 90° (vertical). The results show that the nucleate boiling heat transfer coefficients and the local CHF for the pin fin surfaces were consistently higher than those for a plain surface. The enhancements of the local CHF and the nucleate boiling heat transfer coefficients were mainly due to the pin fins that effectively provide sufficient liquid to the vaporization sites on the heated surfaces, reduce the wall temperature and delay the boiling crisis. The CHF of the pin fin surface with a fin length of 1 mm, a fin width of 1 mm, a fin height of 2 mm, and a fin spacing of 2 mm, were the best with more than 200% CHF enhancements at all inclination angles. The CHF of the pin fin surface increases as the number of fins per unit area increases and even more as the fin height increases.

1. Introduction

Since the Three Mile Island accident, researchers have found that a pressure vessel immersed in coolant can provide sufficient heat transfer to achieve in-vessel retention of a core melt. In-vessel retention is a key severe accident management strategy which has been used in some new nuclear power plants and proposed for some advanced light water reactors. One method to achieve in-vessel retention is external reactor vessel cooling which involves flooding the reactor cavity to submerge the reactor vessel to cool the core debris which has relocated in the vessel lower head during a severe accident [1–3]. To remove the heat load generated by the nuclear fission by the pool boiling, the wall heat flux from the core melt should not exceed the CHF for boiling on the vessel outer surface. The vessel outer surface temperature could be kept near the water saturation temperature to maintain the vessel structured integrity. Therefore, the variations of the CHF with the inclination angle must be known, meanwhile, the boiling heat transfer and the CHF on the outer surface of the pressure vessel must be enhanced.

There have been many experimental and theoretical studies to investigate the external reactor vessel cooling including the effect of inclination angle on the CHF in recent years. Many means have been developed to enhance the external reactor vessel cooling performance for in-vessel retention with the focus mainly on surface structures and thermal insulation.

Henry et al. [4] investigated the heat removal capabilities for external cooling of the reactor pressure vessel lower head to prevent lower head failure with insulation in terms of (a) the water inflow through the insulation, (b) the two-phase heat removal in the gap between the insulation and the vessel, and (c) the steam flow through the insulation. Transient quenching boiling on the outer surface was tested using molten thermite as the transient high heat flux heat source. The results indicated that the external cooling method can prevent the core debris from failing the nuclear reactor vessel, which would ensure the reactor vessel remains intact.

Chu et al. [5–7] performed laboratory and reactor-scale external vessel downward facing boiling experiments in the CYBL facility at Sandia National Laboratories. A total of ten steady-state boiling experiments were performed with uniform as well as edge-peaked heat flux distributions. The bottom center temperature was the highest even when the edge heat flux was higher than that at the bottom center. The boiling flow pattern outside the lower head showed an inner cyclic/pulsating region and an outer steady two phase boundary layer region. The scale and geometry effects resulted in substantially different heat fluxes in the near-horizontal,

* Corresponding author. Tel./fax: +86 10 62773776.
E-mail address: mja@tsinghua.edu.cn (J. Meng).

http://dx.doi.org/10.1016/j.ijheatmasstransfer.2015.04.001
0017-9310/

Nomenclature

A	surface area (m^2)
g	gravitational acceleration (m/s^2)
H	height of pin fin (m)
I	electric current (A)
L	width of the hot surface (m)
N	number of pin fins
q	heat flux (W/m^2)
Q	heat transfer rate (W)
S	spacing of pin fin (m)
W	width of pin fin (m)
T	temperature (°C)
V	voltage (V)
Greek letters	
θ	inclination angle (°) (90°: vertical, 0°: downward-facing)
Subscripts	
w	wall
sat	saturation

downward facing configurations. However, the CHF phenomenon was not observed in their experiments.

Theofanous et al. [8–11] conducted a full-scale experimental study of the boiling crisis phenomenon on the outer surface of a hemispherical reactor vessel using a two-dimensional copper slice with independently heated zones in the ULPU facility at UCSB. They tested five configurations with configurations I, II and III fabricated to investigate downward facing boiling and provide support for the assessment of in-vessel retention for AP600, meanwhile, configurations IV and V fabricated to enhance heat removal capability for higher power reactors (AP1000). Configuration I simulated downward facing boiling on the lowermost spherical segment ($-30° < \theta < 30°$) where the orientation θ was measured from the bottom center of the hemispherical lower head. Configuration II simulated a full side of a reactor lower head from the bottom center to the equator ($0° < \theta < 90°$). The CHF increased from 500 to 1500 kW/m^2 as the orientation changed from the bottom center to the equator in configuration II. On the basis of the above two configurations, the effect of a channel-like geometry created by the reactor vessel thermal insulation was studied in configuration III [8,9]. The configuration III results indicated that the thermal insulation did not significantly affect the CHF. In order to improve the in-vessel retention for AP1000, Theofanous et al. [10] then developed configuration IV (ULPU-2000) by incorporating a baffle as the thermal insulation, which streamlined the flow path around the lower head. The results showed that the flow path streamlining improved the in-vessel retention for higher power reactors. The experimental data showed that the highest CHF was 1880 kW/m^2 on the upper edge of the vessel, while the CHF on the bottom center was 940 kW/m^2. However, the flow channel geometry in ULPU-2000 was not specifically designed to match the AP1000 geometry and flow conditions. The effect of the AP1000 geometry on the CHF was then explored in thirty-six burnout experiments conducted on configuration V (ULPU-2400). This configuration had four major modifications compared to ULPU-2000 with an inlet baffle and a baffle entry, an adjustable baffle around the vessel lower head, smooth transition from the baffle to the riser, and a nozzle that simulates the transition from the riser to the inclined duct. The CHF at the upper region ($66° < \theta < 90°$) of the lower head was 1800–2000 kW/m^2. However, the copper slices with independently heated zones in the ULPU facility was very large with large heat loss, which may cause the CHF measurement in this experiment to be significantly larger than in other facilities.

Rouge et al. [12,13] conducted a series of experiments to study the effects of pressure, inlet temperature, mass flow velocity, flux, gap size and inclination on the CHF and to measure the main characteristics of two dimensional, two phase flow in the SULTAN facility. A rectangular flat stainless steel plate $4 \times 0.15 \times 0.0015$ m was fabricated as the heating block and was cooled on one side by a rectangular channel, the gap size of which was varied from 0.03 to 0.15 m. An important feature of the SULTAN system was that the test surface could be fixed from vertical to horizontal plane to simulate the effect of inclination angle. The results indicated favorable possibilities of the coolability of large surfaces under natural convection, the campaign showed that the heat flux larger than 1 MW/m^2 may be removed under natural water circulation conditions. An empirical CHF correlation was developed as the SULTAN CHF correlation in terms of pressure, mass velocity, local thermodynamic quality, gap size and inclination angle. The 1.5 mm thick heating plate had less thermal inertia than 4 cm thick reactor lower head wall. However, when the heat flux was more than 1 MW/m^2, the plate easily melted and the tests could not be completed.

Cheung and Haddad [14,15] and Cheung and Liu [16,17] preformed a number of experiments to study downward facing boiling and CHF on the outside surface of a 0.305 m diameter hemispherical lower head in the subscale boundary layer boiling (SBLB) facility, which provided scaled 3D simulations of the downward facing boiling with and without thermal insulation. Cheung and Haddad [14,15] conducted transient and steady state boiling experiments with both saturated and subcooled conditions to obtain a CHF database. The spatial variations of CHF in these experiments were similar to those in the ULPU experiments. The CHF was found to increase almost linearly from the bottom center toward the upper edge of the vessel. The CHF range changed from 0.4 to 0.9 MW/m^2 for $0° < \theta < 67.5°$. Cheung and Liu [16,17] also used the SBLB facility to study downward facing boiling and CHF on the external surface with thermal insulation. Two types of thermal insulation were employed for both AP600 and KNGR1300. The CHF for the case with thermal insulation were consistently higher than the case without thermal insulation. A CHF correlation was developed for KNGR 1300 experimental data to support feasibility studies of in-vessel retention for KNGR-like vessels.

In order to enhance the cooling capability of in-vessel retention for the APR1400, a three-year U.S.-Korean International Nuclear Energy Research Initiative (INERI) project was launched. Dizon et al. [18], Cheung et al. [19,20] and Yang et al. [21–24] investigated the boiling processes and CHF phenomena on the outer surface of the lower head with scaled APR1400 thermal insulation in the SBLB facility. Two different methods were employed to enhance the external reactor vessel cooling. One involved a thermal insulation while the other involved metallic micro-porous coatings. Both methods greatly enhanced the external reactor vessel cooling, the CHF for a plain surface varied from 0.453 to 0.888 MW/m^2 for $0° < \theta < 70°$; while the CHF on the micro-porous coating increased to 0.849–1.880 MW/m^2. The results showed that a local CHF enhancement of 200% to 330% could be achieved over a plain vessel using an enhanced insulation structure with micro porous coatings. However, the reliability and lifetime of the coatings for the 60 year lifetime of the generation III reactor needs to be further investigated.