# Numerical and experimental estimation of void fraction of supersonic steam jet in sub-cooled water: a comparative study


Afrasyab Khan*, Evgeny Spiridonov[1], Alexander Podzerko[2], Khabarova Darya[3]
*Institute of Engineering and Technology*
*South Ural State University*
Chelyabinsk 454080, Russian Federation
*drafrasyabkhan7@gmail.com
*khana@susu.ru
[1]spiridonovek@susu.ru
[2]podzerkoav@susu.ru
[3]khabarovadf@susu.ru

Mohammed Zwawi[4], Mohammed Algarni[5], Ali Bahadar[6],
*[4,5]Department of Mechanical Engineering*
*[6]Department of Chemical and Materials Engineering*
*King Abdulaziz University*
Rabigh 21911, Saudi Arabia
[4]mzwawi@kau.edu.sa
[5]malgarni1@kau.edu.sa
[6]absali@kau.edu.sa

Khairuddin Sanaullah[7,8]
*[7]Informetrics Research Group*
*[8]Faculty of Applied Sciences*
*Ton Duc Thang University*
Ho Chi Minh City, Vietnam
khairuddin.sanaullah@tdtu.edu.vn

Bassem F. Felemban
*Mechanical Engineering Department,*
*Taif University*
Taif, Saudi Arabia.
b.felemban@tu.edu.sa

Atta Ullah
*Department of Chemical Engineering,*
*Pakistan Institute of Engineering & Applied Sciences (PIEAS), Nilor,*
Islamabad, Pakistan
atta@pieas.edu.pk

Andrew Ragai Henry Rigit
*Dept. of Mechanical & Manufacturing Engineering*
*University Malaysia Sarawak (UNIMAS)*
94300 Kota Samarahan, Malaysia
arigit@unimas.my



***Abstract*— Gas-liquid two-phase flows occur in a range of chemical, process, petroleum, metallurgical and power industries. Void fraction is a principal parameter. An effort has been made here to perform a comparative study in which the approximate void fraction of the supersonic steam jet into the sub-cooled water has been measured both numerically as well as experimentally. On a numerical basis, Direct Contact Condensation (DCC) model and on an experimental basis, Electrical Resistance Tomography (ERT) has been used for computing the void fraction. On an experimental basis, the over-estimation is nearly 45% when the surrounding water temperature is 30°C with a steam inlet pressure of 1.5 bars and the over-estimation goes up to 83% at 60°C and 3.0 bars. The void fraction computed by the use of the DCC model at 1.5 bars and 300C is 17.66% whereas the computed void fraction at 3.0 bars of inlet pressure and 600C of water temperature is 31.1%.**



***Keywords*— *Supersonic steam jet; Void fraction; Electric Resistance tomography; Image Processing; CFD;***


## I. INTRODUCTION

During the whole operational life of a steam-driven nuclear power reactor, it is very unlikely to ignore the vital phenomenon of direct contact condensation. Instances of such inevitable occurrences were detected in the Loss of Cooling Accident (LOCA) or heat up transients whereas in the former water comes into contact with the steam when it is injected into the steam-filled reactor and in the latter, such phenomenon occurs due to the malfunction in the main steam line isolation valve. Moreover, in the latter case, steam is injected into the pool to suppress the system pressure. However, ensuring the availability of several interaction sites for the steam to come into contact with the cold water needs valuable efforts, which are extremely useful in the handling of such situations in the case of exigencies. The issues related to the safety of the nuclear reactor rely heavily on the operational lines that contain steam and water. The intermittent flow of steam and water causes flow oscillations by the injection of sub-cooled water into the steam-filled regions due to the condensation of steam; steam discharge into a pool of stagnant water resulting into the condensation oscillations and chugging; countercurrent limitations in flooding at abrupt changes in the flow area of restrictions and water hammer caused by the sudden collapse of the steam bubble in the water which cause a sudden drift in the stagnant fluid i.e. water [1]. All the three dominant modes of mixing include injection of steam into a water pool, the interaction of steam with the water when water is sprayed into the steam-filled region and the flow of steam over the water stratified layer. Irrespective of whatever mode prevails, the affectivity of the flow controlling devices can be determined with this very fact that how fast the fresh cold water has been brought in the vicinity of the steam dominant region to establish the intermittent interface [2] and thus facilitates in the transformation of heat, mass, and energy. The safety studies related to the safe operation of the steam-driven nuclear power reactors became important since the time when the first such reactor was commercialized for the generation of electricity. The economic constraints restrict the safety-related studies of the nuclear power reactors to two facets with one related to investigations based on the mockup experiments and the other is related to the simulation studies. Based on the knowledge accumulated for the last two decades and the ever-growing processing and computational power for the theoretical studies


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