Nuclear Engineering and Design 316 (2017) 75–88

Contents lists available at ScienceDirect

Nuclear Engineering and Design

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

Experimental study on two-phase flow natural circulation in a core catcher cooling channel for EU-APR1400 using air-water system

Ki Won Song [a,b], Thanh Hung Nguyen [c], Kwang Soon Ha [b], Hwan Yeol Kim [b], Jinho Song [b], Hyun Sun Park [a], Shripad T. Revankar [a,c,*,1], Moo Hwan Kim [a,d]

[a] *Division of Advanced Nuclear Engineering, POSTECH, Pohang 790-784, Republic of Korea*
[b] *Korea Atomic Energy Research Institute, Daejeon 34057, Republic of Korea*
[c] *School of Nuclear Engineering, Purdue University, West Lafayette, IN 47906, USA*
[d] *Korea Institute of Nuclear Safety, Daejeon 305-338, Republic of Korea*

HIGHLIGHTS

- Two-phase flow regimes and transition behavior were observed in the coolant channel.
- Test were conducted for natural circulation with air-water.
- Data were obtained on flow regime, void fraction, flow rates and re-wetting time.
- The data were related to a cooling capability of core catcher system.

ARTICLE INFO

Article history:
Received 1 September 2016
Received in revised form 18 January 2017
Accepted 4 March 2017

Keywords:
Core catcher
Two-phase natural circulation
Flow regimes in large channels
Local void fraction
Flow transition

ABSTRACT

Ex-vessel core catcher cooling system driven by natural circulation is designed using a full scaled air-water system. A transparent half symmetric section of a core catcher coolant channel of a pressurized water reactor was designed with instrumentations for local void fraction measurement and flow visualization. Two designs of air-water top separator water tanks are studied including one with modified 'super-step' design which prevents gas entrainment into down-comer. In the experiment air flow rates are set corresponding to steam generation rate for given corium decay power. Measurements of natural circulation flow rate, spatial local void fraction distribution and re-wetting time near the top wall are carried out for various air flow rates which simulate boiling-induced vapor generation. Since heat transfer and critical heat flux are strongly dependent on the water mass flow rate and development of two-phase flow on the heated wall, knowledge of two-phase flow characteristics in the coolant channel is essential. Results on flow visualization showing two phase flow structure specifically near the high void accumulation regions, local void profiles, rewetting time, and natural circulation flow rate are presented for various air flow rates that simulate corium power levels. The data are useful in assessing the cooling capability of and safety of the core catcher system.

1. Introduction

External cooling by natural circulation has been considered especially in advanced reactor core cooling systems because of its high reliability and safety during severe accident scenarios. The European versions of pressurized water reactor EU-APR1400 has adopted a passively actuating core catcher system which arrests and retains the corium discharged from the reactor vessel as a countermeasure for the severe accident (Song et al., 2011). The schematic of the coolant loop in the core catcher system is shown in Fig. 1. Molten core is cooled and the decay heat is extracted by direct contact with water at the upside of the core catcher steel plate and at the downside indirectly through the plate to the coolant channel water supplied from the IRWST (In-containment Recirculation Water Storage Tank). The water in the coolant channel of the core catcher heats up and boils. The flow in the coolant channel is driven by a thermally induced density difference of working fluids to generate natural circulation flow

* Corresponding author at: Division of Advanced Nuclear Engineering, POSTECH, Pohang 790-784, Republic of Korea.

E-mail addresses: shripad@postech.ac.kr, shripad@purdue.edu (S.T. Revankar).

[1] Permanent address: 400 Central Drive, School of Nuclear Engineering, Purdue University, West Lafayette, IN 47906, USA.

http://dx.doi.org/10.1016/j.nucengdes.2017.03.009
0029-5493/

Nomenclature

T	total measuring time [sec]
N	total number of bubbles
Q	volumetric flow rate [m^3/sec]
q''	heat flux [W/m^2]
A	area [m^2]
j	superficial velocity [m/s]
DNB	departure from nucleate boiling
CHF	critical heat flux

Greek symbols

ρ	density
$\tilde{\alpha}$	time averaged void fraction

Subscript

g	gas
l	liquid
fg	difference between gas and fluid
j	jth bubble
v	vapor

without external power and removes decay heat of the discharged molten core. An ultimate goal of the core catcher system is to cool and stabilize molten core without molten core concrete interaction (MCCI) and/or over-pressurization of the containment that may threaten the containment integrity of the reactor.

There are two experimental facilities to simulate the EU-APR1400 core catcher cooling systems. The Air-water Experiment Passive Ex-vessel corium retaining and Cooling System (AE-PECS) facility (Revankar et al., 2013) designed and installed at POSTECH is based on scaling analysis to simulate natural circulation flow in a coolant channel using an air-water system. The heated steam-water system Cooling Experiment-PECS (CE-PECS) (Rhee et al., 2012) is constructed in KAERI that actually simulates steam–water flow in the coolant channel for different decay heat of the corium. Evaluation of the core catcher facility using the CE-PECS experiment and the RELAP5/MOD3 code has been conducted especially in terms of the mass flow rate corresponding to various heating conditions and water level in the core catcher upper tank (Park et al., 2016). Both of these test facilities have provided some preliminary understanding of two-phase flow development inside a core catcher coolant channel and their impact on cooling performance of the core catcher. As prediction of two-phase flow inside of such a complex geometry with large hydraulic diameter using existing normal models has a limitation, further experimental observation and data is necessary to fully comprehend the cooling capability of the core catcher for various flow conditions.

Two-phase flow phenomena in a large diameter pipe are quite different from that in a small diameter pipe. Large diameter pipe in an air-water system under atmospheric pressure is defined as a diameter larger than about 10 cm pipe (Shen et al., 2014), while the hydraulics diameter of the core catcher channel is 20 cm. Results of small scale experiments on flow boiling and pool boiling are difficult to extrapolate to a large scale system due to scale effects on flow structure and flow parameters such as maximum bubble size (Batchelor, 1987) and vapor accumulation (Bui and Dhir, 1985). To understand two-phase flow phenomena inside a large scale channel, many experiments are carried out especially in vertical and horizontal piping with co-current flow. Prasser et al. (2007) have observed the two-phase flow development along a large vertical pipe of 19.5 cm diameter using air-water and steam-water and have collected a comprehensive database of two-phase flow parameters for various flow rate conditions. Channel inclination effects on the flow regime in various flow conditions were obtained by Oddie et al. (2003). In order to investigate flow characteristics and the coolability of the reactor system, full-scale experiments in a large channel have been conducted in many facilities such as CYBL (Chu et al., 1994), ULPU (Theofanous et al., 1994; Theofanous and Syri, 1997) and SULTAN (Rougé, 1997) facilities. The experiment by Rougé (1997) using a 15 cm × 15 cm of 10° inclined a channel with a downward facing heated wall observed a slug-type flow at low heat fluxes with periodic re-wetting of the plate. The profile of coolant temperature and void fraction is helpful to validate the two-phase flow code. In order to estimate the coolability limits of the AP600reactor vessel, Theofanous et al. (1994), Theofanous and Syri (1997) performed a full-scale simulation of the reactor vessel lower head. Critical heat flux distribution on the lower head and a correlation as a function of angular position is given in different configurations. Chu et al. (1994) reported the transition of the two-phase flow regimes and boiling patterns along the AP600 reactor vessel lower head under prototypic heat loads. These types of experiments are necessary

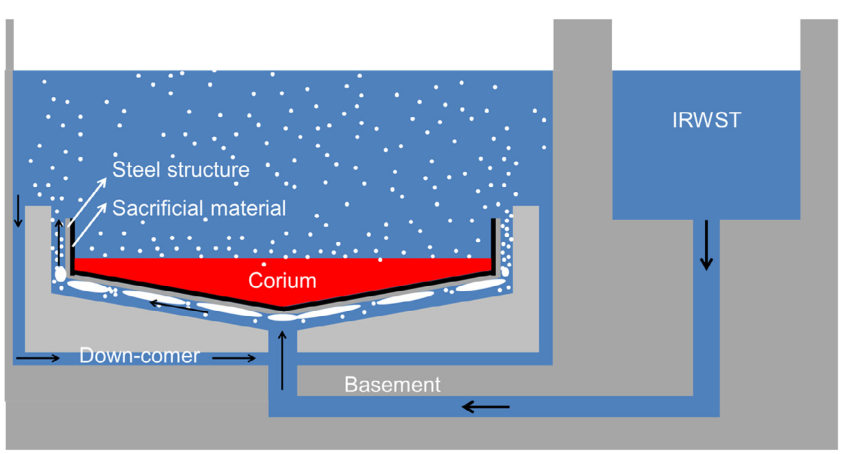

Fig. 1. Natural circulation driven two-phase coolant loop in core catcher system.