Annals of Nuclear Energy 87 (2016) 157–161

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Technical note

Pressure wave propagation phenomena in the pipe system of a nuclear power plant

Sung-Jae Yi [a], Moon-Sun Chung [b,*], Ivan Toth [c]

[a] Thermal-Hydraulics Research Team, Korea Atomic Energy Research Institute, Republic of Korea
[b] HERC, Korea Institute of Energy Research, Republic of Korea
[c] KFKI, Hungary

ARTICLE INFO

Article history:
Received 5 February 2015
Received in revised form 25 August 2015
Accepted 26 August 2015

Keywords:
WAHA3 code
PMK-2 test facility
Two-phase flow
Pressure wave propagation
Pipe system
Nuclear power plant

ABSTRACT

The aim of this research is to compare the experimental data obtained by using PMK-2 test facility which was equipped as an experimental apparatus for water hammering of the pipe system in the KFKI research center in Hungary with some results of numerical calculations (V&V test) based on WAHA3 code on the pressure wave propagation phenomena in a pipe system of a nuclear power plant. In order to simulate pressure wave propagation in the two-phase flow by using WAHA3 code, we used two node systems for numerical calculations: For the first preliminary calculation, we used a simplified node system by assuming all the same area in the entire pipe system. But for the second calculations, we differently used a fully described node system for a real PMK-2 test facility. And then, we finally compare the calculated results using WAHA3 code with the experimental data produced by PMK-2 test facility.

1. Introduction

Most of the nuclear power plants in the world have complex pipeline systems and they are operated under high temperature and high pressure conditions. Therefore, the plant is at an accident condition or under the normal operation, the possibility of pipe break still exists. Some of them may be caused by the frequent exposure to pressure wave oscillations like water hammering phenomena.

Especially, the "steam bubble collapse" mechanism can be identified which is responsible for steam condensation induced water hammer in horizontal pipes. Steam bubble collapse induced water hammer events happen if the following six conditions meet (Griffith, 1996) (1) the pipe must be almost horizontal (max. pipe inclination must be less than 5°), (2) the subcooling must be greater than 20 °C, (3) the L/D (length-to-diameter ratio of the tube) must be greater than 24, (4) the velocity must be low enough so that the pipe does not run full, i.e. the Froude number must be less than one, (5) there should be a void nearby and (6) the pressure must be height enough so that significant damage occurs, that is the pressure should be above 10 atmospheres. For these reasons, many researchers have investigated condensation induced water hammer in the pipe system of the nuclear power plants.

In the previous study, Barna et al. (2010) investigate the steam condensation induced water hammer phenomena. The experiments were performed in the PMK-2 test facility and compared with the WAHA3 code analysis. As a result, they show that the narrow pressure peaks existing in the experiments are in good agreement with that of numerical calculation with WAHA3 code.

Beuthe (2009) introduced the water hammer simulation capabilities of the CATHENA one-dimensional two-fluid thermal–hydraulics code by showing the single-phase and one- and two-component two-phase water hammering phenomena. He also indicated that the results of water hammer phenomenon in the presence of noncondensable gas and condensation induced water hammer. Though the calculated results showed some discrepancies in magnitude of pressure peaks and periods, but they also showed that the CATHENA code has a capability for simulating many kinds of water hammer phenomena.

To investigate the hydraulic effects on the pipe system, the analytical research has to be carried out by using a system analysis code. WAHA3 code has been modeled to simulate one-dimensional two-phase flow water hammering phenomena with a second-order accurate high-resolution, shock-capturing numerical scheme (Imre et al., 2010). For this reason, first, we have done for the verification and validation tests of WAHA3 code with a simplified nodalization and the calculation results should be compared

* Corresponding author at: HERC, Korea Institute of Energy Research, 71-2, Jang-dong, Daejeon 305-343, Republic of Korea.
E-mail address: moonschung@kier.re.kr (M.-S. Chung).

http://dx.doi.org/10.1016/j.anucene.2015.08.024
0306-4549/© 2015 Published by Elsevier Ltd.

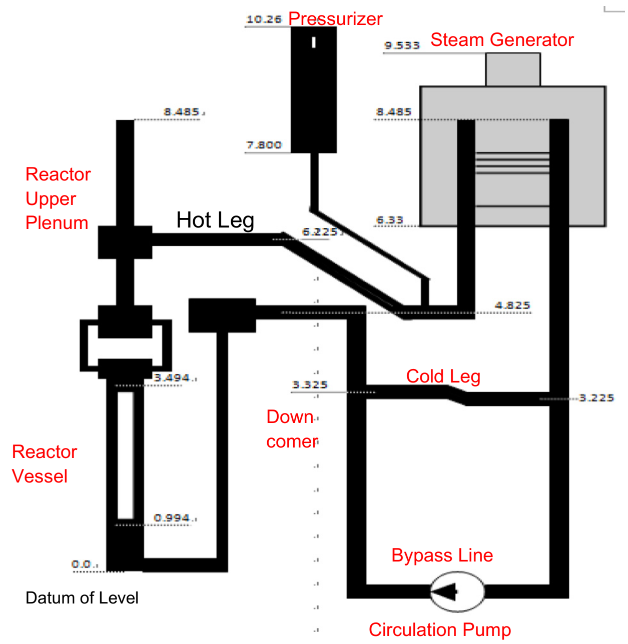

Fig. 1. Schematic diagram of PMK-2 test facility.

with the experimental data produced by operating PMK-2 test facility on pressure wave propagation in the downcomer (Szabados et al., 2007). PMK-2 test facility was equipped as an experimental apparatus for water hammering phenomena on the pipe system in the KFKI research center of Hungary.

Second, a realistic numerical calculation is carried out by using WAHA3 code with a more detailed nodalization than that using in the preliminary calculation. And then we compare the calculated results with the experimental data produced by KHJRL (Korea-Hungary Joint Research Laboratory) program.

The calculation results of pressure in the downcomer have been compared with the experimental data from PMK-2 test facility when loss of coolant accident (LOCA) was initiated to break a rupture disk. The PMK-2 facility is composed of a high pressure and high temperature test loop as shown in Fig. 1.

2. PMK-2 test facility

Because we just focus on the pressure wave propagation phenomena in the downcomer region, the external downcomer and reactor vessel parts of PMK-2 test facility have been considered separately, and the other regions of reactor coolant system are considered as boundary conditions in the numerical calculation. The entire nodalization configuration for calculation of WAHA3 code is shown in Fig. 2.

For simplicity, we consider the complex three dimensional geometry of the downcomer and reactor vessel system as a constant area piping system with the same dimensions and we do not consider an elastic interaction between water and structure. For simplifying boundary conditions, the pressure condition for the outside of a break nozzle (Tank 010) is assumed to be under the atmospheric pressure.

The connections to a hot leg and a cold leg are modeled as Tank 011 and Tank 012. Total 71 nodes used in the simulation based on the WAHA3 code. For the initial conditions, the pressure and the temperature are 12.4 MPa and 541 °C respectively, which are the same conditions of pre-experimental test. The break time of rupture disk is assumed to be almost zero second at the beginning of numerical calculation.

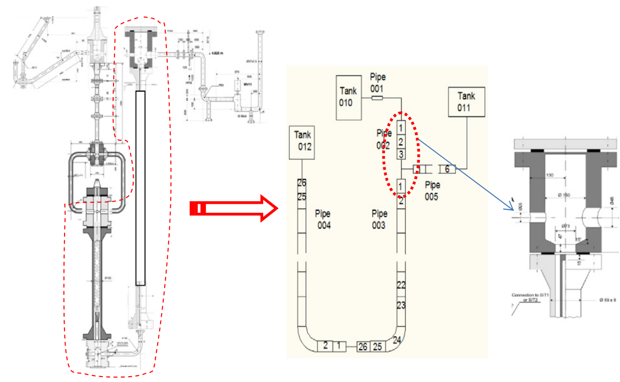

Fig. 2. A simplified nodalization on the downcomer region of PMK-2 test facility.

3. Preliminary calculation for a pressure wave propagation

After numerical calculation using WAHA3 code, we can compare the calculated results with the former experimental data. Fig. 3 shows a different pressure behavior between upper downcomer and reactor lower plenum by comparing a LOCA test result using PMK-2 test facility with a numerical calculation using WAHA3 code. At the beginning of disk rupture, from 0 to around 0.075 s, results of calculation and experiments show some discrepancies with different behaviors. During the short period of transient time, downcomer pressure wave produced by critical flow at the nozzle is propagated as shown in Fig. 4.

In case of calculated results using WAHA3 code as shown in Fig. 5, all thermo-hydraulic values reached to steady-state condition up to 0.150 s after the initiation of flow transient. In Fig. 6, the water and the vapor velocity reach critical flow velocity at the end of the nozzle. The velocities of water and vapor increase with void fraction reduction as time goes by. However, the pressure undershoot at the nozzle exists due to the overestimated critical velocities of each phase in the early stage right after flow initiation.

That is to say, the downstream of break nozzle, the fluid status in this period of downcomer and reactor vessel are still single-phase water as shown in Fig. 5. The void fraction at the nozzle exit is about 0.075 after 0.12 s from the time of flow initiation. In that case, the critical velocity at the nozzle exit should be about 70–80 m/s for bubbly flow regime as Henry et al. pointed out in

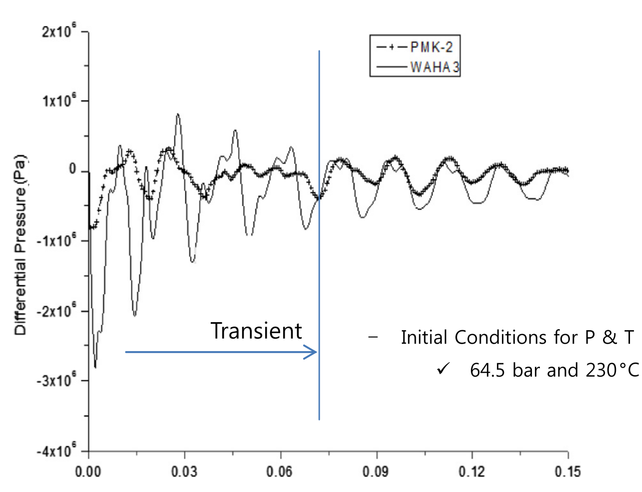

Fig. 3. Comparison result of differential pressure between test and WAHA3.