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Multiple condensation induced water hammer events, experiments and theoretical investigations

We investigate steam condensation induced water hammer (CIWH) phenomena and present experimental and theoretical results. Some of the experiments were performed in the PMK-2 facility, which is a full-pressure thermalhydraulic model of the nuclear power plant of VVER-440/312 type and located in the Atomic Energy Research Institute Budapest, Hungary. Other experiments were done in the ROSA facility in Japan. On the theoretical side CIWH is studied and analyzed with the WAHA3 model based on two-phase flow six first-order partial differential equations that present one dimensional, surface averaged mass, momentum and energy balances. A second order accurate high-resolution shock-capturing numerical scheme was applied with different kind of limiters in the numerical calculations. The applied two-fluid model shows some similarities to RELAP5 which is widely used in the nuclear industry to simulate nuclear power plant accidents. New features are the existence of multiple, independent CIWH pressure peaks both in experiments and in simulations. Experimentally measured and theoretically calculated CIWH pressure peaks are in qualitative agreement. However, the computational results are very sensitive against flow velocity.

Experimentelle und theoretische Untersuchungen zu Kondensationsschlägen. *Die spontane Kondensation von Dampfblasen führt in technischen Systemen zu starken Kondensationsschlägen, die unerwünscht sind und deren Auftreten und Vermeidung untersucht wird. Experimente zu Kondensationsschlägen werden u. a. in der vom Atomic Energy Research Institute Budapest in Ungarn betriebenen Versuchsanlage PMK-2 und in der Versuchsanlage ROSA in Japan durchgeführt. Für die theoretischen Untersuchungen von Kondensationsschlägen wurde das Programm WAHA3 entwickelt. Dieses basiert auf der Beschreibung der Zweiphasenströmung mit Hilfe von sechs Differentialgleichungen 1. Ordnung, die in eindimensionale über den Querschnitt gemittelte Massen-, Impuls- und Energiebilanzen überführt werden. Das zur Lösung der Differentialgleichungen eingesetzte numerische Verfahren ist zweiter Ordnung und so hochauflösend, dass auftretende Kondensationsschläge erfasst werden. Das Hauptaugenmerk dieses Beitrags liegt in der Messung und Berechnung von verschiedenartigen unabhängigen Kondensationsschlagereignissen. Dabei zeigte sich eine gute qualitative Übereinstimmung bei den gemessenen und berechneten Druckpeaks, allerdings reagieren die berechneten Daten sehr sensitiv auf den Parameter Strömungsgeschwindigkeit.*

1 Introduction

Safety of nuclear reactors is a fundamental issue. Nuclear and thermo-hydraulic processes in the active zone of modern reactors are well known and well-controlled, any kind of nuclear-type explosion is out of question. However, violent unwanted thermo-hydraulic transients in the primer loop may cause serious derangement or pipe breakage. Such an unplanned transient is the steam condensation induced water hammer. In thermal loops of atomic reactors or in other pipelines where water steam and cold water can mix, quick and dangerous transients can occur causing some mechanical damages in the applied pipe system.

In the following we will introduce the ROSA [1] and the PMK-2 facility [2] which are integral experimental devices and capable to produce CIWH effects.

On the other side we present the WAHA3 [3] model we use, which is a complex physical model suitable to simulate various quick transients in single and two-phase flows.

In the last two decades the nuclear industry developed a few complex two-phase flow-codes like RELAP5 [4], Trac [5] or Cathare [6] which are feasible to solve safety analysis of nuclear reactors and model complicated two-phase flow transients.

The model, WAHA3 [7] is very similar to RELAP5. This means that the conservation equations and all the applied correlations are essentially the same. The main difference between the above mentioned models and our WAHA3 code is basically the applied numerical scheme; other commercial codes have a ratio of spatial and time resolution $\Delta x/\Delta t$ which describes usual flow velocities. This code, however is capable of capturing shock waves and describes supersonic flows. To our knowledge there is no special model and computer code for water hammer simulation in the field of nuclear thermal-hydraulics.

The WAHA3 model can successfully reproduce the experimental data of different one- or two-phase flow problems such as ideal gas Riemann problem, critical flow of ideal gas in convergent-divergent nozzle, column separation or cavitation induced water hammer or even rapid depressurization of hot liquid from horizontal pipes [3].

2 Experimental setups and theory

In the following section we give a brief overview of the Hungarian PMK-2 and the Japanese ROSA where the water hammer experiments were performed. After that our theoretical model WAHA3 will be briefly introduced.

2.1 PMK-2 test facility

The PMK-2 facility is located at the KFKI Atomic Energy Research Institute (AEKI) Budapest, Hungary [2]. It is a full-pressure scaled down thermalhydraulic model of the primary and partly the secondary circuit of the nuclear power plant of VVER-440/213 type (VVER is the Russian abbreviation of the water-water energetic reactor). It was primarily designed for the investigation of off-normal transient processes of small-break loss of coolant accidents.

Between 1985 and 2007 there were 55 different experiments performed on the apparatus. The group of transients is as follows 7.4 % cold leg breaks (15 tests), cold leg breaks of different sizes (10 tests), hot leg breaks and primary to secondary leaks (10 tests); tests for natural circulation characteristics and disturbances (10 tests); plant transients and accidents (10 tests). Results of experiments were used to validate thermalhydraulic system codes as ATHLET, CATHARE and REALP5 for VVER applications.

Considering the scaling ratio interval and the financial possibilities of the country, a 19 rod core model with 2.5 m heated length was selected and gives a power and volume scaling ratio to 1:2070. The operating pressure of the PMK-2 is 12.3 MPa and the core thermal power is 664 kW. The heat loss for the PMK-2 facility is about 3.6 percent of the nominal heat power. Due to the importance of gravitational forces in both single- and two-phase flow the elevation ratio is 1:1. Other important similarity properties like the Richardson, Stanton, Froude and the Nusselt numbers are 1:1 as well. There are 10 integral type facilities for PWR's (Pressurized Water Reactors) and VVERs in the world like the American LOFT, the ROSA-IV in Japan, the PACTEL facility in Finland or the Hungarian PMK-2. VVERs are slightly different from PWRs of the usual design and have a number of special features like: 6-loop primary circuit, horizontal steam generators, and loop seal in hot and cold legs, safety injection tank set-point pressure higher than secondary pressure. Figure 1 presents the PMK-2 integral facility from a bird's eye view.

The steam pressure on the steam generator side is 4.6 MPa. The CIWH experimental setup is connected into the steam line of PMK-2 and located on the top of the integral facility. The experimental setup is basically a horizontal pipe section of 5 m length and 193 mm inner pipe diameter initially filled with vapor that is supplied from the dome of the steam generator of the PMK-2. The other side of the test device is connected to the condenser unit of PMK-2 which substitutes turbine of the real power plant. Both ends of the CIWH tube are further equipped with inertia blocks of 200 kg each serving a 90 deg bend in the same time. The test section can be isolated by two valves; one is located in the connection with the head of the steam generator, and the other in the connecting line towards the condenser. For the flooding, a cold water tank with a volume of 75 l is installed and pressurized with air. Figure 2 shows the recent water hammer experimental device.

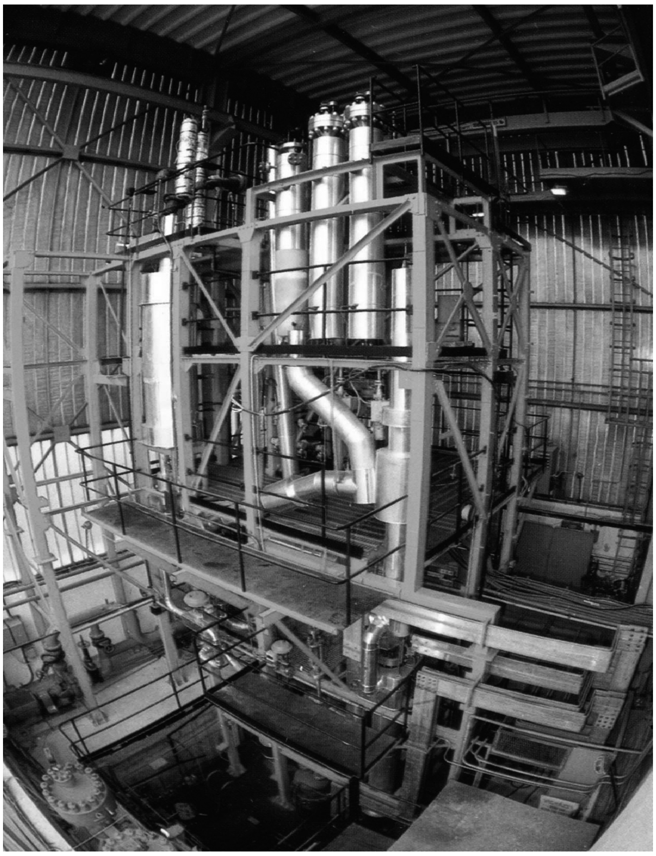

Fig. 1. PMK-2 experimental facility

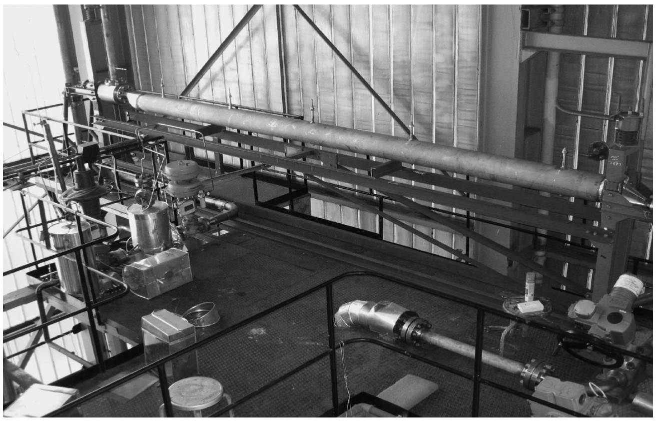

Fig. 2. Water hammer experimental device

2.2 ROSA test facility

The second experimental facility which will be introduced is the Japanese Rosa. OECD/NEA ROSA Project Test 2 (ST-WH-05, 06, 07, 08, 09, 10 and 11, conducted by JAEA) was performed on April 11 and 12, August 28 and 29, and September 5 and 6, 2007 by using the Large Scale Test Facility (LSTF) [1] in the Japan Atomic Energy Agency (JAEA). The objective of this test is to obtain detailed thermal-hydraulic transient data concerning condensation-induced water hammer (CIWH) in a horizontal branch pipe connected to the LSTF vessel downcomer. The schematic view of a CIWH in a horizontal pipe is shown in Fig. 3. The data is in particular used to study the effect of the system pressure on the CIWH characteristics such as the intensity of the CIWH pressure pulse. It is important for the nuclear safety, since room-temperature water is injected by ECCS (Emergency Core Cooling System) including passive safety system even at high pressure condition.

This study covers the CIWH induced at two-phase counter current flow in a horizontal pipe. The liquid phase flow simulates the room-temperature water flow injected by ECCS. The vapor phase flow simulates the saturated steam flow driven by the condensation on the room-temperature water. Such two-phase condition may be appeared at the ECCS injection line, when the water injection rate is decreased. A horizontal pipe is employed as the test section for CIWH tests. The dimensions of the test section, made of stainless steel, are 2050 mm in

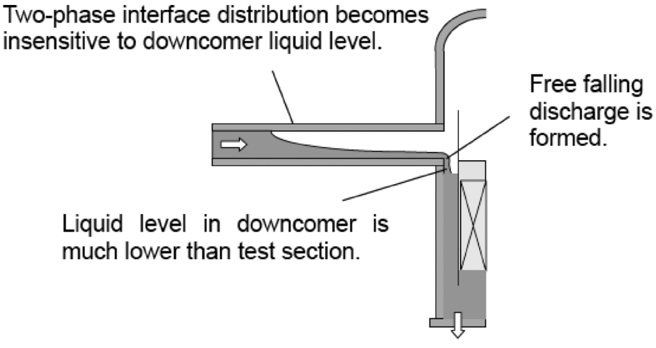

Fig. 3. A schematic sketch of the ROSA CIWH experimental tube

length, 66.9 mm in inner diameter and 11 mm in pipe wall thickness. One end of the test section is horizontally connected to the LSTF horizontal nozzle named N-18c. The nozzle length is 290 mm from the LSTF downcomer inner surface, and the inner diameter is the same as the test section inner diameter. Accordingly, they form an about 2.3 m long horizontal pipe. The other end of the test section is closed using a sealing plate. The room-temperature water stored in the LSTF RWST tank is injected to the bottom of the test section near the closed end using the LSTF high pressure injection system (HPI). The water is discharged to the LSTF downcomer through the test section. When the downcomer liquid level is much lower than the bottom elevation of the test section, the water falls freely into the downcomer would affect the water fall under high water supply conditions.

The CIWH tests were performed at the system pressure of 0.35, 1.0, 2.8, 4.4, 5.5 and 7.0 MPa. The maximum system pressure of 7.0 MPa is determined in consideration for the result of the past LSTF test. The minimum system pressure of 0.35 MPa is the lowest system pressure that is controllable. Therefore, since the heat removal of the LSTF primary loop is performed by the steam generators, the primary pressure has to be comparatively larger than the secondary pressure which is set to atmospheric pressure. The flow rates about 0.1, 0.3 and 0.9 kg/s at the room-temperature were employed as test conditions. According to the LSTF tests experience, the CIWH may be induced near about 0.3 kg/s. The flow rates about 0.1 and 0.9 kg/s are lower and upper bound at supply flow rate; respectively, in order to keep the balance between the supply flow rate and the discharge flow rate. The flow rate is controlled by the pump operation of the high pressure injection system (HPI). The water temperature of the former and the latter is about 290 K and about 305 K, respectively. Unfortunately, additional technical details of the experiments cannot be found in the report, even private communications could not helped us to lighten all details.

2.3 Theory

There exist a large number of different two-phase flow models with different levels of complexity ([8], [9]) which are all based on gas dynamics and shock-wave theory. In the following we present the one dimensional six-equation equal-pressure two-fluid model.

The density, momentum and energy balance equations for both phases are the following:

$$\frac{\partial A(1-\alpha)\,\rho_l}{\partial t}+\frac{\partial A(1-\alpha)\,\rho_l(v_l-w)}{\partial x}=-A\Gamma_g \tag{1}$$

$$\frac{\partial A\alpha\rho_g}{\partial t}+\frac{\partial A\alpha\rho_g(v_g-w)}{\partial x}=A\Gamma_g \tag{2}$$

$$\begin{aligned}&\frac{\partial A(1-\alpha)\,\rho_l v_l}{\partial t}+\frac{\partial A(1-\alpha)\,\rho_l v_l(v_l-w)}{\partial x}\\&\quad+A(1-\alpha)\frac{\partial p}{\partial x}-A\cdot CVM-Ap_i\frac{\partial\alpha}{\partial x}=AC_i|v_r|v_r-A\Gamma_g v_l\\&\quad+A(1-\alpha)\rho_l g\cos\vartheta-AF_{l,wall}\end{aligned} \tag{3}$$

$$\begin{aligned}&\frac{\partial A\alpha\rho_g v_g}{\partial t}+\frac{\partial A\alpha\rho_g v_g(v_g-w)}{\partial x}\\&\quad+A\alpha\frac{\partial p}{\partial x}+A\cdot CVM+Ap_i\frac{\partial\alpha}{\partial x}=-AC_i|v_r|v_r+A\Gamma_g v_g\\&\quad+A\alpha\rho_g g\cos\vartheta-AF_{g,wall}\end{aligned} \tag{4}$$

$$\begin{aligned}&\frac{\partial A(1-\alpha)\,\rho_l e_l}{\partial t}+\frac{\partial A(1-\alpha)\,\rho_l e_l(v_l-w)}{\partial x}+p\frac{\partial A(1-\alpha)}{\partial t}\\&\quad+\frac{\partial A(1-\alpha)\,p(v_l-w)}{\partial x}=AQ_{il}-A\Gamma_g(h_l+v_l^2/2)\\&\quad+A(1-\alpha)\,\rho_l v_l g\cos\vartheta\end{aligned} \tag{5}$$

$$\begin{aligned}&\frac{\partial A\alpha\rho_g e_g}{\partial t}+\frac{\partial A\alpha\rho_g e_g(v_g-w)}{\partial x}+p\frac{\partial A\alpha}{\partial t}\\&\quad+\frac{\partial A\alpha p(v_g-w)}{\partial x}=AQ_{ig}+A\Gamma_g(h_g+v_g^2/2)+A\alpha\rho_g v_g g\cos\vartheta\end{aligned} \tag{6}$$

Index f refers to the liquid phase and index g to the gas phase. Nomenclature and variables are described at the end of the manuscript. Left hand side of the equations contains the terms with temporal and spatial derivatives.

Hyperbolicity of the equation system is ensured with the virtual mass term CVM and with he interfacial term (terms with p_i). Terms on the right hand side are terms describing the inter-phase heat, mass (terms with Γ_g vapor generation rate) volumetric heat fluxes Q_{ij}, momentum transfer (terms with C_i), wall friction $F_{g,wall}$, and gravity terms. Modeling of the inter-phase heat, mass and momentum exchange in two-phase models relies on correlations which are usually flow-regime dependent.

The system code RELAP5 has a very sophisticated flow regime map with a high level of complexity. WAHA3 however has the most simple flow map with dispersed and horizontally stratified regimes only, because the uncertainty of steady-state correlations in fast transients are very high.

A detailed analysis of the source terms can be found in Tiselj et al. ([3], [7]).

Two additional equation of states (eos) are needed to close the system of equations

$$\rho_k=\left(\frac{\partial\rho_k}{\partial p}\right)_k dp+\left(\frac{\partial\rho_k}{\partial u_k}\right)_p du_k. \tag{7}$$

Partial derivatives in Eq. (7) are expressed using pressure and specific internal energy as an input. The table of water and steam properties was calculated with a software from UCL [10].

The system of Eqs. (1–6) represents the conservation laws and can be formulated in the following vectorial form

$$\underline{\underline{A}}\frac{\partial\overline{\Psi}}{\partial t}+\underline{\underline{B}}\frac{\partial\overline{\Psi}}{\partial x}=\overline{S} \tag{8}$$

where Ψ represents the non-conservative variables $\Psi(p,\alpha,v_f,v_g,u_f,u_g)$, **A, B** are matrices and **S** is the source vector of non-differential terms. These three terms can be obtained from Eq. (1–6), with some algebraic manipulation.

In this case the system eigenvalues which represent wave propagation velocities are given by the determinant det(**B**-

λ**A**). An improved characteristic upwind discretization method is used to solve the hyperbolic equation system (Eq. 8). The problem is solved with the combination of the first-and second-order accurate discretization scheme by the so-called flux limiters to avoid numerical dissipation and unwanted oscillations which appear in the vicinity of the non-smooth solutions. Exhaustive details about the numerical scheme can be found in *LeVeque* [11].

2.4 Results

Fig. 4 presents one of the experimentally measured CIWH pressure peaks in the OECD/NEA ROSA Project. There are various measurements done, we analyse only one of them in the following.

The stem pressure was 2.8 MPa with 503 K temperature. The cold water temperature was 305 K with a flooding velocity of 0.088 m/s.

The measurement took 1320 s. In contrast to other CIWH experimental setups(like in the PMK2), here the horizontal tube is opened at both ends, hence it is possible to have flow conditions when the tube cannot be filled up with water and there is a relatively large interface surface at the top of the horizontally stratified flow regime in the tube for a long time. Our detailed analysis showed that there is a continuous "steam bubble capture" mechanism on the surface of the horizontally stratified flow – as time goes on – which is responsible for the large number of WAHA pressure peaks. Fig. 5 shows our results where numerous WAHA peaks can be detected for the above mentioned flow system. Our experience shows that the results are extremely sensitive to the flow velocity. We could not perform calculations up to 1320 s only up to 120 due to the extreme number of data. We tried to simulate 5.5 MPa ROSA experimental data as well. This system was even more sensitive for the flooding velocity therefore we cannot carry out simulations more than 30–40 s, the experimental tube was always filled up.

It is worth to mention that a CIWH pressure peak has a 2 ms half width which means extremely high time resolution and output. Further work is in progress to clear out all the details.

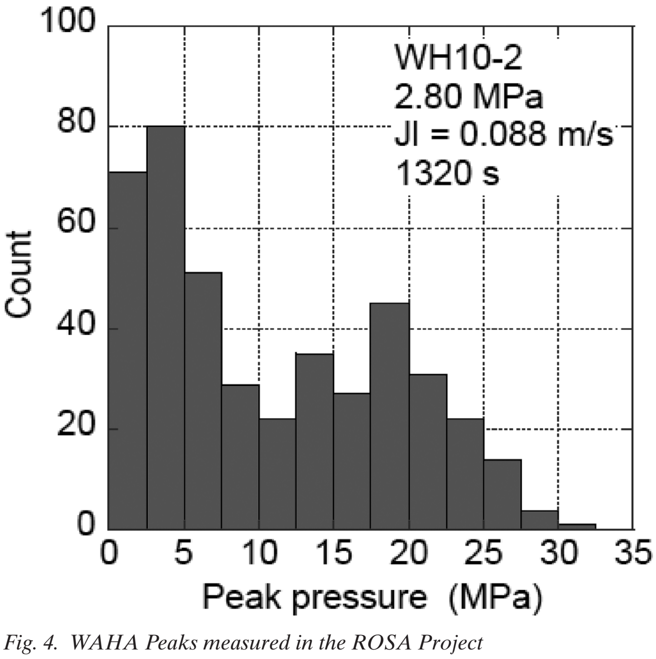

Fig. 4. WAHA Peaks measured in the ROSA Project

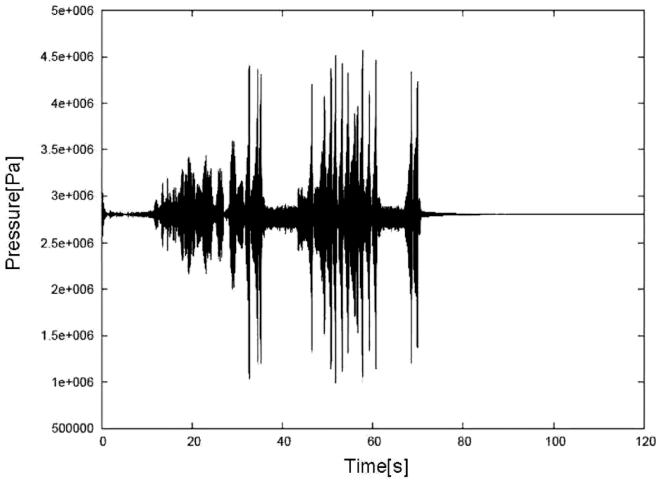

Fig. 5. Time history of the pressure peaks for the ROSA measurement

As a second system we investigated and analyzed the measurements done at PMK-2 in Budapest.

All together 9 measurements were performed at 3 different steam pressures, with 6, 10 and 15 bar. Fig. 6 shows the time history of the measured pressure peaks for 15 bar. Contrary to our former experiments [12] where a 3 m long tube was used and only a single CIWH event happened now in this 5 m long tube two (or for 6 bar even three) independent pressure peaks were measureable. The first pressure peak is 62 and the second is 28 bar. The measuring point was at 40 cm from the cold water inlet. At a different point at 80 cm the pressure peaks are much more different.

A careful investigation of the dynamics of the void fraction along the tube during the flooding clearly shows that in a longer tube there is enough room for two steam bubble formation. Our former study [12] gives a detailed analysis of the "bubble capture mechanism". Fig. 7 shows our results for the above mentioned experiment. The first pressure peak is 62 bar and the second is 72 bar. The four small wave-like structure in the pressure history before the first pressure peak indicates some quasi steam bubble formation. We tried to shift both pressure peaks to higher time points lowering the flooding velocity, unfortunately both pressure peaks disappeared. This results are extremely sensitive to the flooding velocity.

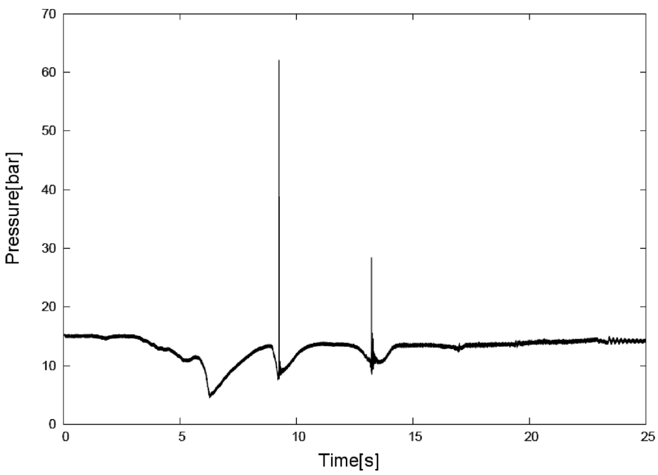

Fig. 6. Time history of the pressure measured at 40 cm from the left end of the horizontal pipe

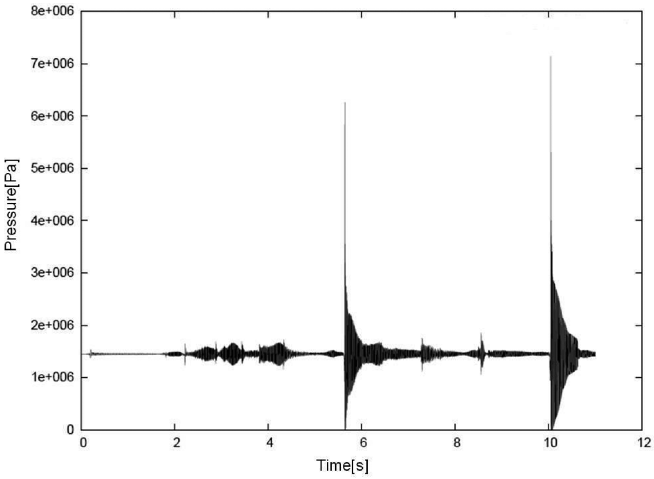

Fig. 7. Time history of the calculated pressure in the PMK-2 experiment

3 Conclusions

We presented and analyzed steam condensation induced water hammer experimental results performed at the ROSA and the Hungarian PMK-2 experimental facility. Where the later is a full pressure scaled down model of the primary and partly the secondary loop of the national Nuclear Power Plant equipped with the VVER-440/312 type.

With the help of a one dimensional two-phase flow model we investigated the steam condensation induced water hammer phenomena. With a detailed analysis of the pressure wave propagation and the dynamics of the vapor void fraction along the pipeline the "steam bubble collapse" mechanism is 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 [13]:

- the pipe must be almost horizontal (max. pipe inclination must be less than 5 degree)
- the subcooling must be greater than 20 °C
- the L/D (length-to-diameter ratio of the tube) must be greater than 24
- the velocity must be low enough so that the pipe does not run full, i. e. the Froude number must be less than one
- there should be a void nearby
- the pressure must be high enough so that significant damage occurs, that is the pressure should be above 10 atmospheres.

Contrary to past CIWH experiments, the ROSA and the new PMK-2 setups can produce more than a single CIWH event which is a new feature in this field. Unfortunately, we could not carry out a full time numerical simulation of the ROSA experiment but our investigation might give us a clearer sight into the physical phenomena which is happening behind.

As a second system we investigated the experimental data from the new CIWH experimental facility, which was built in the Hungarian PMK-2 integral experimental device right now. The geometry is basically the same as mentioned in our former study [12] but a much larger horizontal pipe was raised with 5 m lengths and 20 cm in diameters. First experiments gave water hammer events with 60–80 bar peak pressures, which are much smaller than in our previous experiments. On the other side, our simplified theoretical analysis showed that appearance of 350 bar overpressure peaks are not impossible in a 5 m long device (these results are not presented here). We explain such huge discrepancies with the fact that CIWH events are very sensitive to the initial flooding water velocity. The new experimental system has another peculiarity, two or even three independent CIWH events happen one after another separated by some seconds. A careful investigation of the dynamics of the void fraction along the tube during the flooding clearly shows that in a longer tube (now 5 meters long former was only 3) there is enough room for two steam bubble formation.

Further theoretical investigations are in progress to illuminate all details.

In contrast to large system codes like REALP5 or TRAC, we have the source code of WAHA3 which is transparent and flexible to apply it to other two-phase flow systems.

Recently, we modified our model and created a realistic two-phase liquid-steam table for mercury. We performed calculations to simulate pressure waves and cavitation effects in the planned European Spallation Source (ESS) [14].

As a long term interest we also plan to investigate other liquid metal (e. g. bismuth-lead eutectic) systems [15] or liquid helium which can be interesting as a cooling media for new type of nuclear reactors. Liquid metal systems can operate on low (some bar) pressure and have much larger heat conductivity than water which can radically enhance thermal efficiency.

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Nomenclature

Symbol	Unit	Denomination
A	m^2	pipe cross section
C_i	kg/m^4	internal friction coefficient
CVM	N/m^3	virtual mass term
e_i	J/kg	specific total energy ($e = u + v^2/2$)
$F_{g,wall}$	N/m^3	wall friction per unit volume
g	m/s^2	gravitational acceleration
h_i	J/kg	specific enthalpy ($h = u + p/\rho$)
p	Pa	pressure
p_i	Pa	interfacial pressure ($p_i = p\alpha(1-\alpha)$)
Q_{ij}	W/m^3	interfacial-liquid/gas heat transfer per volume rate
t	s	time
u_i	J/kg	specific internal energy
v_i	m/s	velocity
v_r	m/s	relative velocity ($v_r = v_g - v_f$)
w	m/s	pipe velocity in flow direction
x	m	spatial coordinate (m)
Γ_g	$kg/(m^3s)$	vapor generation rate
α	–	vapor void fraction
ρ_i	kg/m^3	density
θ	–	pipe inclination

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Research Reactor Allocation for Materials under High Neutron Fluence. Published by the International Atomic Energy Agency, 2011, IAEA TECDOC No. 1659, ISBN 978-92-0-116010-2, 18.00 EUR.

Research reactors (RR) have played and continue to play a key role in the development of the peaceful uses of atomic energy. Moreover these facilities are used intensively for education and training purposes of scientists and engineers. At present, a larger scale of designs is in use and they also have different operating modes, producing energy which may be steady or pulsed. Due to this, RR are very unique tools in the scientific and technological development area and they have a fairly wide spectra of applications. In principle, there is a common approach for design which is the pool type reactor in which the core is a cluster of fuel elements sitting in a large pool of water. Such adapted design allows for the reach of higher neutron fluence due to higher power density and therefore these facilities are primarily used frange of structural materials and their properties. Existing RR facilities, especially in developing countries, are often under-utilized and could be used more effectively (e. g. for material testing, radioisotope production, beam line applications, nuclear transmutation doping and analytical services), with new initiatives on a national, regional and inter-regional level. The sharing of resources can increase the utilization on one hand and pave the way for the decommissioning of under-utilized ageing reactors on the other, without depleting knowledge base and human resources.

The overall objective of this publication is to overview the activities in the area of RR applications for studies of materials under high neutron fluence. It is expected that this technical document will help to stimulate new activities by using of RRs as well as strengthening of the expertise, know-how and best practise.

The report is also focused on the specific RRs applications in irradiation of materials at high neutron flux and fluence, as well as integration issues, including:

- Available irradiation facilities and recent development of irradiation facilities,
- New material irradiation programmes and their implementation,
- Contribution to the better understanding of radiation damage at high doses and dose rates,
- Effective and optimal operation procedures for irradiation purposes,
- Fostering the advanced or innovative technologies by promotion of information exchange, collaboration and networking,
- Sharing of information and know-how.

The scope of this report is to summarise available information in the area, presented by participating IAEA Member States in order to enhance research reactors utilization for practical applications. Today's multipurpose research reactors are used for various applications with respect of individual needs of particular countries, such as irradiation services, isotope production, neutron radiography and beam research as well as material characterization and testing. This document gives brief overview of such practical applications and basic information about related infrastructure.