The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14
Toronto, Ontario, Canada, September 25-30, 2011

NURETH14-xxx

Steam Condensation Induced Water Hammer Simulations for Different Pipelines

I.F. Barna , Gy. Ézsöl
KFKI Atomic Energy Research Institute (AEKI) of the Hungarian
Academy of Sciences, Thermo-hydraulic Department,
Address P.O. Box 49, H-1525 Budapest, Hungary,EU
Tel : +36-1-392-2222/1472 , Fax:+36-1-395-9293,
Email : barnai@aeki.kfki.hu

Abstract

We investigate steam condensation induced water hammer (CIWH) phenomena and present theoretical results for different kind of pipelines. We analyze the process 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. At first, we present calculations for various pipelines in the VVER-440-312 type nuclear reactor. Our recent calculation clearly shows that the six conditions of Griffith are only necessary conditions for CIWH but not sufficient. As second results we performed calculations for various geometries and compare with the theory of Chun.

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, explosions are out of question. However, violent unwanted thermo-hydraulic transients in the primer loop may cause serious deformation or pipe breakage. Such an unplanned transient is the CIWH. In thermal loops of atomic reactors or in other pipelines where water steam and cold water can mix, quick and dangerous transients can happen causing pressure surges, which mean high financial expenses or even cost human lives.

We simulate CIWH with the WAHA3[1] model we use, which is a complex physical model suitable to simulate various quick transients in single and two-phase flows, such as ideal gas Riemann problem, critical flow of ideal gas in convergent-divergent nozzle, rapid depressurization of hot liquid from horizontal pipes and column separation water hammer or even CIWH.

In the last two decades the nuclear industry developed a few complex two-phase flow-codes like Relap5[2] , Trac[3] or Cathare[4] which are feasible to solve safety analysis of nuclear reactors and model complicated two-phase flow transients.

The model, WAHA3 has some similarities with Relap5. This means that the conservation equations are the same but the applied correlations are partially different [1]. 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 which describes usual flow velocities. WAHA3, however is capable of capturing shock waves and describe pressure waves which may propagate quicker than the local speed of sound. As a second point WAHA3 has a quick condensation model which is not available for Relap5 and Cathare.

To our knowledge WAHA3 is the only model which is capable to simulate CIWH phenomena. There is only one theoretical study available from Chun and Yo [5] which gives analytical formulas for the lower and the upper critical feed water flow rate for an effective pipe length to produce CIWH.

We tried to simulate CIWH with Relap5 and Cathare, unfortunately in vain. These two large system codes have different numerical procedure which is unable to reproduce large and narrow pressure pikes in problems where long pipe length is combined with sonic velocity.

According to our knowledge, which is based on the study of the Relap5 and Cathare Manuals there are no systematic study about CIWH phenomena with these codes.

In our following study we are going to present calculations concerning the amplitude and duration of the pressure peak generated by CIWH. The calculation of other quantities like void fraction and local temperatures are neglected, because these quantities cannot be measured properly and their calculated values are strongly model-dependent, while the pressure peak can be measured with high accuracy.

1. Theory

There are large number of different two-phase flow models with different levels of complexity which are all based on gas dynamics and shock-wave theory[6,7]. 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:

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

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

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

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

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

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

Index l 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 the 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. The uncertainties of steady-state correlations in fast transients are very high.

A detailed analysis of the source terms can be found in Tiselj et al.[8]

Two additional equation of states(eos) are needed to close the system of Eqs. (1-6.) Here the subscript k can have two values 'l' for liquid phase, and 'g' for gas phase

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

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[9].

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

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

where Ψ represents a vector of the non-conservative variables $\Psi(p,\alpha,v_f,v_g,u_f,u_g)$ and **A, B** are 6-times-6 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($\underline{\underline{A}}-\lambda\,\underline{\underline{B}}$). 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 the work of LeVeque[10].

3 Results and Discussion

3.1 Simulations for the VVER 440 Nuclear Reactor

In the following we will present calculations for 5 different pipelines which are in the national nuclear power plant of VVER-440/312 type and located in Paks, Hungary. We will see that there are 5 problematic pieces exist which might cause CIWH effects.

According to safety reasons we must not mention the alphanumerical code or the positions of these pipes in the original national nuclear power plant. All the following five pipes are horizontal. We calculate pressure-time functions 40 cm appart from the cold water inlet.

Table I contains the tested pipe geometries with thermohydraulic parameters.

The first pipe is L= 5289 mm long with a d = 100 mm diameter. The steam pressure is p = 58 bar. The temperature of the saturated steam is 546 K and the temperature of the cooling water is T = 294 K. The mass flow of the cooling water is 5kg/s which is equivalent with v = 0.636 m/s. The approximate Froude Number Fr = 0.64.

Figure 1 presents the pressure history of the first pipeline. We can clearly see that the maximal pressure peak is about 530 bar which is huge. Our former experience clearly tells us that, the absolute magnitude of the pressure can vary with about 50 percent[11,12], but the reliability of the model is satisfactory. Which means, that such kind of large pressure peak will happen in such geometrical and flow conditions.

Our institute is planning to build the former pipeline in the PMK-2 facility which is located at the KFKI Atomic Energy Research Institute (AEKI) Budapest, Hungary . It is a full-pressure scaled down thermohydraulic model of the primary and partly the secondary circuit of the nuclear power plant of VVER-440/213 type (VVER is a Hungarian 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 are 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 thermohydraulical 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 which gives a power ratio of
1:2070 (39.312:19 ~ 2070) and, therefore, the overall volume scaling ratio is also 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 VVER's in the world like the American LOFT, the ROSA-IV in Japan, the PACTEL facility in Finland or the Hungarian PMK-2.

Now back, to our simulations.

The second pipe is L= 3643 mm long with a d = 50 mm diameter. The steam pressure is p = 58 – 64 bar. The temperature of the saturated steam is 546 K and the temperature of the cooling water is T = 294 K. The mass flow of the cooling water is 5kg/s which is equivalent with v = 2.55 m/s. The approximate Froude Number Fr = 0.64. The pressure history has a very complex shape, but without a well-defined CIWH pressure peak. Hence we do not include any figure. We must emphasize that parallel to the pressure history we also consider the time propagation of the steam void fraction, if we cannot see any “bubble-capture” mechanism than no CIWH happens. For a better description see[11,12].

The third pipeline has the following geometrical and flow properties: L = 8753 mm, d = 50 mm, Pressure p = 58- 64 bar. The temperature of the saturated steam is 546 K and the temperature of the cooling water is T = 294 K. The mass flow of the cooling water is 5kg/s which is equivalent with v = 2.55 m/s. The approximate Froude Number Fr = 0.64. The pressure history is very similar to the former case. Wild and quick pressure oscillations can be seen which comes from the numeric, and no overpressure peaks are present. Hence we do not include any figure as well.

The fourth pipeline has the following features L = 8753 mm, d = 50 mm, Pressure p = 110 bar. The temperature of the saturated steam is 591 K and the temperature of the cooling water is T = 294 K. The mass flow of the cooling water is kg/s which is equivalent with v = 2.55 m/s. The approximate Froude Number Fr = 0.64. At t = 0.1 sec there is a large pressure peak which might be a CIWH. We checked the dynamics of the steam void fraction and it is clear that for this flow for such a short time (0.1 s) no steam bubble can be formed and captured. The reason of the pressure peak is the following, at such a large steam pressure in the interface relatively large mass of hot steam is immediately condensed, causing a very local large pressure. At the same time the vicinity of the cold water inlet is called down causing no further large pressure variation.

In the fifth pipeline the conditions are more different, the steam pressure is small and the flooding velocity is much larger. L = 6950 mm, d = 233 mm, Pressure p = 7 bar. The temperature of the saturated steam is 438 K and the temperature of the cooling water is T = 294 K. The mass flow of the cooling water is 40-50 kg/s which is equivalent with v = 0.8 m/s. The approximate Froude Number Fr = 0.6-0.7. No spurious pressure peaks, or other kind of pattern can be seen and no CIWH happens in this case.

Table 1. A table of the tested pipe geometries with thermohydraulic parameters

Pipe Section	**Steam Pressure (bar)**	**Steam Temperature (K)**	**Pipe Length (m)**	**Pipe Diameter (m)**	**Flooding Velocity (m/s)**	**CIWH Event Yes/No**
1	58	546	5.3	0.10	0.63	Yes
2	58	546	3.6	0.05	2.55	No
3	58	546	8.7	0.05	2.55	No
4	110	591	8.7	0.05	2.55	No
5	7	438	6.9	0.23	0.80	No

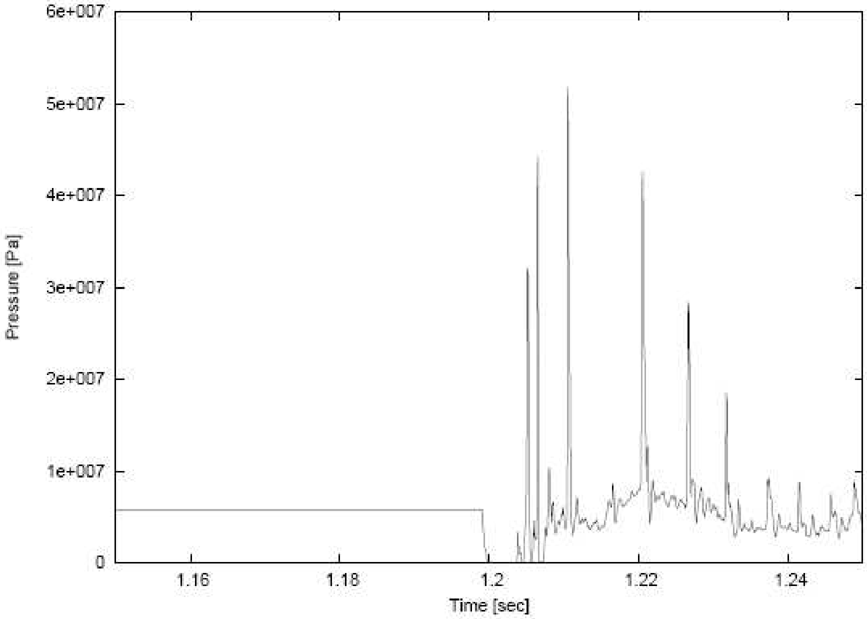

Figure 1. The pressure history for the first pipeline.

3.2 Simulations to test the theory of Chun

In the second part of our study we present some calculations where we compare our WAHA3 calculations with the results of Chun and Ho [5]. In their work - they cite 17 different experimental results - they state that if in a horizontal pipe a slug is formed then the CIWH event will happen. In their study they give formulas for the upper and lower bound, however the derivation is not clear for us. Unfortunately, they do not present any numerical value for the maximal pressure peaks during CIWH events. Our experience shows that slug formation is not a sufficient condition for CIWH. They present large number of critical feedwater flow rate versus L/D ratio results for the upper and lower bound for CIWH for various pressure.
The L/D ratio runs from 24 up to 300 and the pressure runs from 0.1 MPa to 16 MPa.
In the following we concentrate on thick pipes where the L/D ratios are 30 or 50 .
We also examined low initial steam pressure 0.1 and 4 MPa There are large number of calculations needed to get the exact upper and lower feedwater flow rate for CIWH.
For 0.1 MPa our calculations are still inconsistent, in some cases where an exact slug is formed no CIWH happens, but in some other cases a 1.8 MPa peak occurs.
Therefore we present our results only for the 4MPa case. Table 2 shows our upper and lower feedwater flow rate data together with the data of Chun for L/D = 30 and for L/D = 50.
Note, that our results define a much broader parameter range for CIWH then the results from Chun.
In both cases (L/D = 50 and L/D = 70) our WAHA3 pressure peaks are about of 30 MPa which are very large. Further work is in progress to build up a solid database for upper and lower limits of CIWH.

Table 2. Comparison of WAHA3 results with the results of Chun

L/D ratio	**lower flow rate from Chun (kg/s)**	**upper flow rate from Chun (kg/s)**	**our lower flow rate (kg/s)**	**our upper flow rate (kg/s)**
30	0.15	0.155	0.24	3.24
50	0.12	0.27	0.26	7.3

1. Conclusion

We presented five simulations for five different pipeline elements which are part of the national Nuclear Power Plant equipped with the VVER-440/312 type.

The numerical analysis was done with the help of a one dimensional two-phase flow model WAHA3. 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 CIWH in horizontal pipes.

Our theoretical investigation confirm the following six conditions of Griffith[13] which have to be fulfilled to produce CIWH events:

1) the pipe must be almost horizontal (max. pipe inclination must be less than 5 degree)
2) the sub-cooling must be greater than 20 C^o
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
6) the pressure must be height enough so that significant damage occurs, that is the pressure should be above 10 atmospheres.

Our former experimentally measured and theoretically calculated pressure peaks are in good agreement[11,12].

As outlook we mention that construction of a new CIWH experimental facility is in progress in the Hungarian PMK-2 integral experimental device right now. The geometry is basically the same as mentioned for the first pipeline.

As second results we presented some calculations where we compared our WAHA3 results with the theory of Chun. Further work is in progress to build up a solid database from a series of WAHA3 simulations for a large range of flow parameters and geometries.

5. References

[1] I. Tiselj, A. Horvath, G. Cerner, J. Gale, I. Parser, B. Marko, M. Gigot, J.M. Sinhalese, B. Kucienska and H. Lemonnier, *WAHA3 code manual*, Deliverable D10 of the WAHALoads project, March 2004

[2] K.E. Carlson, R.A. Riemke, S.Z. Rouhani, R. W. Shumway and W. L. Weaver RELAP5/MOD3.3Beta Code Manual, Vol 1-7, NUREG-CR/5535, EG\&G Idaho, Idaho Falls 2003

[3] TRAC-PF1/MOD1: *An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Thermal-Hydraulic Analysis*, NUREG/CR-3858, L.A-10157-MS, 1986

[4] D. Bestion and G. Geffraye, "*The CATHARE code*" CEA Grenoble Report, DTP/SMTH/LMDS/EM/22001-63, April 2002

[5] M-H. Chun and S-O. Hu, "A parametric study and a guide to avoid condensation-induced water hammer in a horizontal pipe " *Nucl. Eng. and Des*. Vol. 201, 2000, pp 239

[6] H.B. Stewart and B. Wendroff, "Two-Phase flow: Models and Methods" *J. Comp. Phys*. Vol. 56**,** 1984, pp. 363

[7] R. Menikoff. and B. Plohr, "The Riemann Problem fluid flow of real materials" *Rev. Mod. Phys*. Vol 61, 1989, pp 75-130

[8] I. Tiselj and S. Petelin, "Modeling of Two-Phase Flow with Second-Order Accurate Scheme" *Journal of Comput. Phys*. Vol. 136, 1997 pp 503- 521

[9] J.M. Seynhaeve, *Water properties package*, Catholic University of Louvain (1992) Project Built with IAPS from Lester, Gallaher and Kell, McGraw-Hill 1984

[10] R. J. LeVeque *Numerical Methods for Conservation Laws,* Lecture in Mathematics, ETH, Zurich, (1992)

[11] I. F. Barna, G. Baranyai and Gy Ezsol " 8095 Theoretical and Experimental Study of Steam Condensation Induced Water Hammer Phenomena" Proceedings of the 2008 International Congress on Advances in Nuclear Power Plants, Anaheim CA USA, June

[12] I. F. Barna, G. Baranyai and Gy Ezsol " 9003 Experimental and Theoretical Study of Steam Condensation Induced Water Hammer Phenomena " Proceedings of the 2009 International Congress on Advances in Nuclear Power Plants, Tokyo, May 10-14 2009

[13] P. Griffith *Screening Reactor System/Water Piping Systems for Water Hammer* Repaired for Division of Systems Technology Office of Nuclear Regulatory Commissions Washington DC, 20555-0001 NRC Job Code J6008 NUREG/CR-6519

6. APPENDIX

7. NOMENCLATURE

A pipe cross section (m^2)

C_i internal friction coefficient (kg/m^4)

CVM virtual mass term (N/m^3)

e_i specific total energy [e = u + v^2/2] (J/kg)

$F_{g,wall}$ wall friction per unit volume (N/m^3)

g gravitational acceleration (m/s^2)

h_i specific enthalpy [h = u + p/ρ] (J/kg)

p pressure (Pa)

p_i interfacial pressure p_i = pα(1-α) (Pa)

Q_{ij} interf.-liq./gas heat transf. per vol. rate (W/m^3)

t time (s)

u_i specific internal energy (J/kg)

v_i velocity (m/s)

v_r relative velocity (v_r = v_g -v_f) (m/s)

w pipe velocity in flow direction (m/s)

x spatial coordinate (m)

Γ_g vapor generation rate (kg/m^3)

α vapor void fraction

ρ_i density (kg/m^3)

ϑ pipe inclination