

# Implementation of a graphical user interface for safety analysis of the ROSA/LSTF hot-leg break experiment

**Andrej Prošek**

*Jožef Stefan Institute, Jamova cesta 39, SI-1000 Ljubljana, Slovenia  
E-mail: Andrej.Prosek@ijs.si*

**Abstract.** Today, most software applications, also in the nuclear field, come with a graphical user interface. Recently, the Symbolic Nuclear Analysis Package (SNAP) was designed to simplify the process of performing safety analysis. The purpose of the present study was to develop an SNAP animation model for RELAP5 calculations of a nuclear installation. The facility selected was the Rig of Safety Assessment/Large Scale Test Facility (ROSA/LSTF), which is a simulator of a four loop pressurized water reactor. The source data needed for animation were obtained by the RELAP5/MOD3.3 Patch 04 computer code. A scenario of the hypothetical small-break loss-of-coolant accident in the hot-leg of the reactor coolant system was simulated. The SNAP animation of the ROSA/LSTF facility was used for presentation of the physical phenomena and processes. It can be concluded that the developed animation model significantly helps in the safety analysis and the analysis results can be presented to a wider public in an easy and understandable way.

**Keywords:** graphical user interface, animation model, SNAP, safety analysis

## 1 INTRODUCTION

Today, most software applications come with a graphical user interface (GUI). GUI provides the user with a valuable means to learn, understand, and use the application software. GUIs for visualization of the results in the safety analysis of nuclear installations were developed for computational fluid dynamic codes, system codes, severe accident codes, etc. In early 1990's, the United States Nuclear Regulatory Commission (NRC) stated that all their software products, including the RELAP5 thermal-hydraulic computer code, would henceforth have GUIs. The first GUI for the RELAP5 computer code was called the Nuclear Plant Analyzer (NPA) [1]. It focused on visualizing reactor coolant behaviour during a simulated transient. Development of the Symbolic Nuclear Analysis Package (SNAP) started in 1996 [2] and is still underway. SNAP currently supports the CONTAIN [3], COBRA, FRAPCON-3, MELCOR, PARCS, RELAP5 [4] and TRACE [5] analysis codes. In the last two decades, there have been some studies performed for the Krško nuclear power plant (NPP) using GUI for visualization. The first visualization of the Krško nuclear power plant (NPP) was done for RELAP5/MOD3.1 calculation using NPA in 1995 by the Jožef Stefan Institute (JSI) [6]. With the NPA code the general plant mask and steam generator masks were developed. The MELCOR model of the Krško NPP has been also developed at JSI together with the graphical

interface using NPA [7]. A simple hypothetical severe accident scenario in the Krško NPP has been chosen for animation with NPA. Several masks have been developed including thermal-hydraulics showing the plant with the containment, core mask showing the core materials inventory, cavity mask showing the shape (e.g. after core-concrete interaction), mask with graphs, etc. Another example of visualization of results for Krško NPP is the study [8], [9] to make an assessment of the vulnerabilities of the typical PWR cavity structures to steam explosions.

As SNAP is a Java-based computer application running on most popular computer platforms including Windows 7 and Windows XP, Linux-based systems and Mac Os X, a new SNAP animation model of the Krško NPP for transients and accidents was developed in 2010 [10]. The main motivation for developing the animation model was to prepare tools for smoother transition from RELAP5 to the TRACE computer code, better presentation of the simulated physical phenomena and processes, user friendly-tool for understanding nodalization and the detail of plant modelling, better presentation of the results, a tool to train new users of the thermal-hydraulic code, and finally, with modern tools, to attract new people to work with system codes. The new tools will so complement the traditional ones used in the safety analysis [11], [12], [13].

In the following paragraphs, the developed Rig of Safety Assessment/Large Scale Test Facility (ROSA/LSTF) animation model is presented together with RELAP5 calculations needed to verify the

animation model during transient states. The calculations were performed with the latest RELAP5/MOD3.3 Patch 04 code. Finally, examples of animations are given.

## 2 DESCRIPTION OF TOOLS

A nuclear installation has many important aspects to visualize. GUI can be used to visualize the thermal hydraulics, heat transfer, fuel rod behaviour, controls, trip logic, etc. It also aids users in preparing the input model, running the calculations and plotting the results of their calculations. There is a variety of means to post process the data. First, the visualization tool SNAP is briefly described. Then the RELAP5 code used, the ROSA/LSTF facility, the scenario and finally RELAP5 input model of ROSA/LSTF are described.

### 2.1 SNAP graphical user interface

The SNAP [14] was designed to simplify the process of performing safety analysis. It is intended for creating and editing input for safety analysis codes and it has functionality for submitting, monitoring, and interacting with the codes. SNAP currently supports the CONTAIN, COBRA, FRAPCON-3, MELCOR, PARCS, RELAP5 and TRACE analysis codes. Each code is supported by a separate plug-in. The SNAP interactive and post-processing capabilities are predominately realized within its animation displays. An animation display retrieves data from the server and represents it visually in some fashion. The data can be from an actively running calculation, a completed calculation, external data, etc.

### 2.2 RELAP5/MOD3.3 Patch 04 computer code

The basic RELAP5 thermal-hydraulic model uses six equations: two mass-conservation equations, two momentum-conservation equations and two energy-conservation equations. Closure of the field equations is provided through the use of constitutive relations and correlations. Since release of RELAP5/MOD2 in 1985, the code has been continuously improved and extended. Several new models, improvements to the existing models, and user conveniences have been added to RELAP5/MOD3.3, with the most recent RELAP5/MOD3.3 Patch 04 release in 2010 [4].

### 2.3 ROSA/LSTF test facility

The ROSA/LSTF [15] was designed to simulate thermal-hydraulic phenomena peculiar to small-break loss-of-coolant accidents (SBLOCA) and operational transients by having prototypical component elevation differences, large loop-piping diameters, prototypical

primary-pressure levels, and simulated system controls. The ROSA/LSTF has volumes scaled at 1/48 of a typical 3423 MWt 4-loop pressurized water reactor (PWR) plant. The four primary loops in the reference PWR are represented by two symmetric loops in the ROSA/LSTF, each one including an active steam generator and an active reactor coolant pump. The component elevations are preserved full scale to simulate natural circulation phenomena peculiar to SBLOCAs and transients. Besides the major components, the reactor protection systems and equipment controls, the secondary and various auxiliary systems are included, too. These systems include emergency core-cooling systems, feedwater, condensate and steam systems together with component service systems such as the cooling water, instrument air, water purification, etc.

### 2.4 Scenario description

This test presents a 10% hot-leg break test SB-HL-02, conducted on June 30, 1987 on the ROSA/LSTF [16]. It involved a horizontal break on the hot leg, with an area of 10%, followed by loss of the off-site power occurring concurrently with scram due to a low pressurizer-pressure signal. Both the high-pressure safety injection (HPSI) system and the auxiliary feedwater system (AFW) were assumed unavailable on demand. The primary system depressurized quickly after break and reached the scram signal setpoint pressure of 12.97 MPa at 6 s. The reactor scram signal initiated core-power decay at 42 s, tripped off the main feedwater at 13 s, and initiated the closure of the main steam valves at 11 s. Loss of the offsite power was assumed to occur concurrently with scram. Thus, the scram signal also initiated the coastdown of the coolant-pump speed. The primary pressure reached the accumulator injection pressure (4.51 MPa) at 310 s. After the accumulator injection has been terminated, low pressure injection initiated automatically at 856 s as the injection pressure (1.29 MPa) was reached. The test was terminated at 1941 s.

### 2.5 RELAP5 input model

The RELAP5 input model consists of 212 volumes, 221 junctions and 213 heat structures. The RELAP5 hydrodynamic component input model created by SNAP is shown in Figure 1. This is an example of creating from the ASCII file and editing an RELAP5 input model (pre-processing). All important components of the primary and secondary side of the ROSA/LSTF are modelled up to the turbine. The RELAP5 input model includes also the reactor protection system.

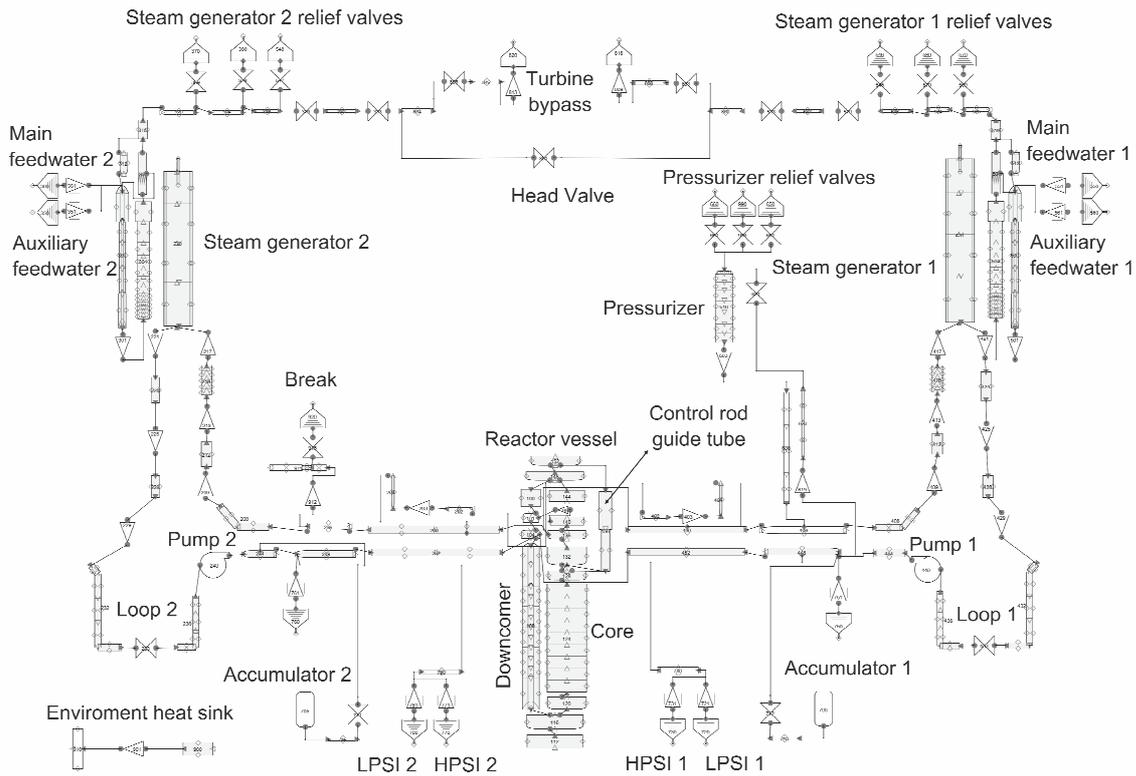


Figure 1. SNAP model editor showing hydraulic component view of the RELAP5 input model

### 3 RESULTS PRESENTED BY THE SNAP GRAPHICAL USER INTERFACE

The first example of results presented by GUI is reactor coolant-pump controller to achieve the desired reactor coolant-system flow during steady-state (normal operation at 100% power). From Figure 2 it can be seen that after a 1000 s long steady-state calculation starting at time -1000 s, at time 0 s, the reference flow was established. Such controllers are built-in into the RELAP5 code.

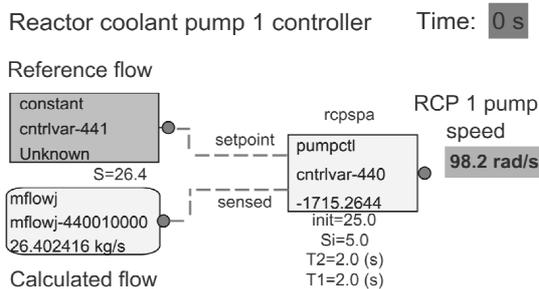


Figure 2. Reactor coolant-pump controller during steady-state

The error signal in the flow is sent to the PUMPCTL proportional-integral controller. The initial value of the controller was set to 25 ('init'). Si is the scale factor and must be non-zero, because we divide the error-flow signal. T1 is the time constant for the proportional part and T2 is the time constant for the integral part. In the

transient calculation, the controller is switched off, because pump coastdown is modelled by table (the values are given as a function of time).

The second example of using GUI for post-processing are the plots of important variables during a simulated SBLOCA test. They are shown in Figure 3. The data were plotted directly from SNAP, using plot batch scripts. They can be created for quick plotting with AptPlot, intended for plotting the graphs. Figure 3(a) shows the pressurizer pressure dictating transient progression. Until the safety injection start, the pressure depends on the break flow. The break causes core uncover (see Figure 3(c)) and core heatup (see Figure 3(b)). The integrated value of the break flow is shown in Figure 3(d). From these plots the analyst can get a general information of the transient. Presentation of the physical phenomena and processes during the test can be improved by using the animation masks and movies (sequence of animation masks). Figure 4 shows the general mask of the ROSA/LSTF at 300 s. There are two colour maps, one for void fraction and the other for rod temperature. For void fraction the blue colour represents the fluid and the white colour represents the gas. According to Figure 3, at 300 s the core is most uncovered. From the void fraction colour map it can be seen that on the primary side the fluid is mostly in the core and a slightly in the loop seals. There is more than 80% of the reactor coolant-system (RCS) mass (total mass 5418 kg) released through the break and the accumulators just started to inject (106 kg of injected

mass; the total is 1944 kg). The RCS pressure is so high that low-pressure safety-injection (LPSI) pump cannot inject. One may also see that all the pumps are stopped at this moment and that loop seals are formed (loop seal phenomenon) on both loops thus causing deeper core uncover. After the loop seal has been cleared, the water stored in the reactor-vessel downcomer will quickly flood the core. In spite of much inventory lost, only small heatup occurs as can be seen from Figure 3(b). One may also note that more mass was injected from accumulator no. 1. The reason is that according to the test specification the flow from the accumulator no. 1 was three times larger than that from the accumulator no. 2.

Finally, Figure 5 show single-phase natural circulation and reflux condensation for the loop no. 1. Single-phase natural circulation produces heat transfer from the core to the steam generators. When RCS pressure decreases to the value slightly higher than the secondary pressure, the fluid starts to flash at the top. Flushing at the top of the steam generator U-tubes interrupts the single-phase natural circulation flow and causes a decrease in the level of water on the cold side of the steam generator (see Figure 5(b)) in order to maintain the RCS loop-pressure balance. Steam produced in the core rises through the hot legs to the steam generator U-tubes, is condensed there and drains

back through the hot legs to the reactor vessel. The arrow in the hot leg shows that the total mass flow is negative and that the steam flow which passes the U-tubes is small. Some larger flow at the steam generator outlet means that the steam generator is slowly emptying. At later stages when the RCS pressure drops, the break flow and the injection flow equalize and if the break is not too large, the RCS system starts to fill again. In the opposite, the RCS system is filled to the elevation of break with the core completely covered.

All the logic and control systems used for editing the input model can also be used for animation. The pump controller shown in Figure 2 is one such example. All these masks can be used during calculation, for replaying the stored data and to generate separate movies. In this way the physical behaviour during the calculation it may be visually very efficiently shown. So displayed results are very useful also to wider a community not so familiar with nuclear technology and can be used as a training tool for people working in the nuclear field. Finally, such graphical user interface is a convenient tool to train new users of nuclear safety analysis codes. Besides the masks shown, the base nodalization scheme shown in Figure 1 can also be used for animation.

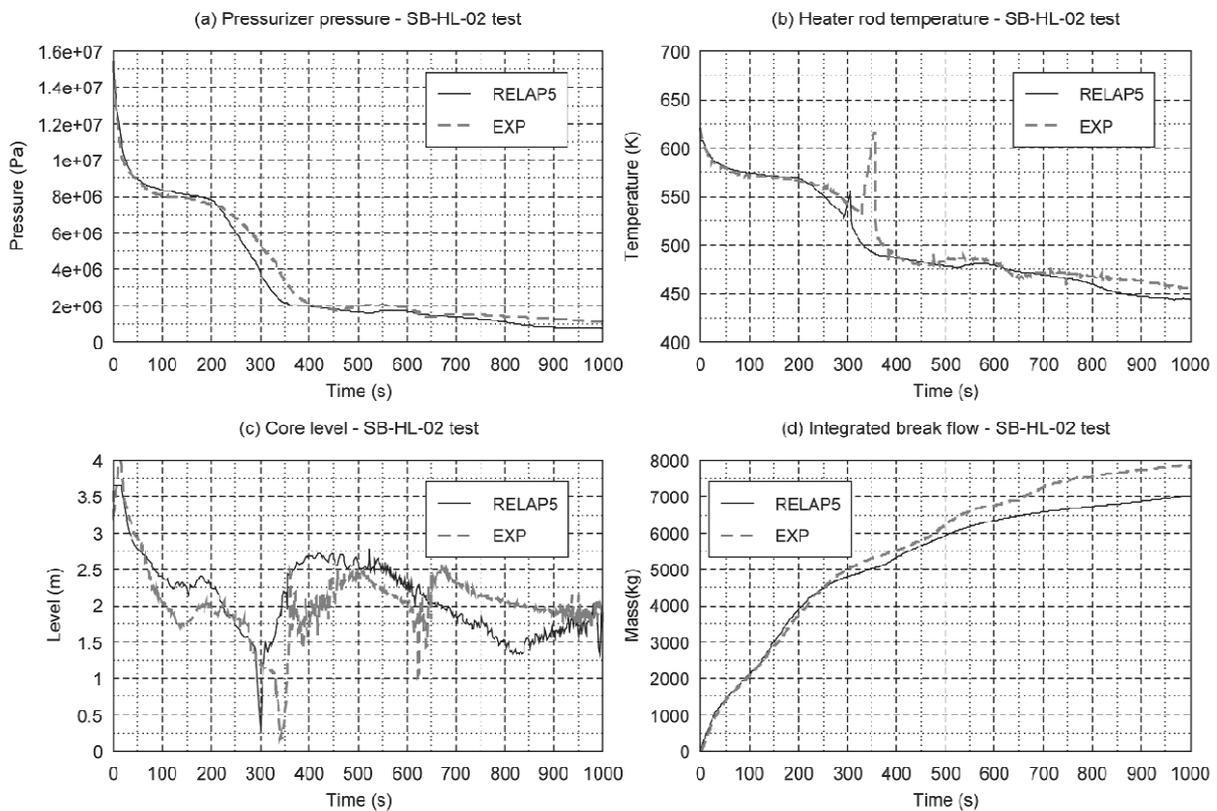


Figure 3. Important variables during the simulated SBLOCA test

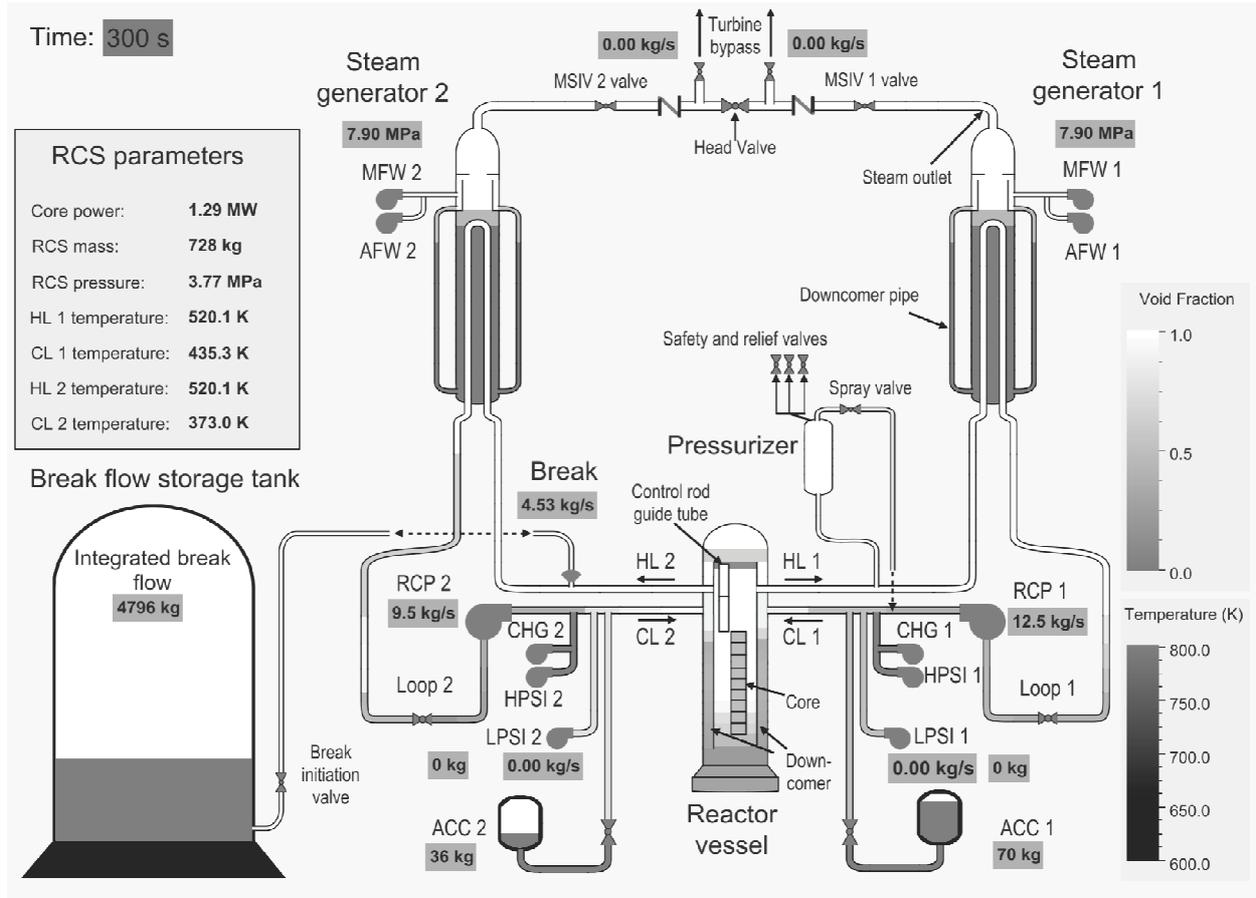


Figure 4. General animation mask of the ROSA/LSTF facility at 300 s

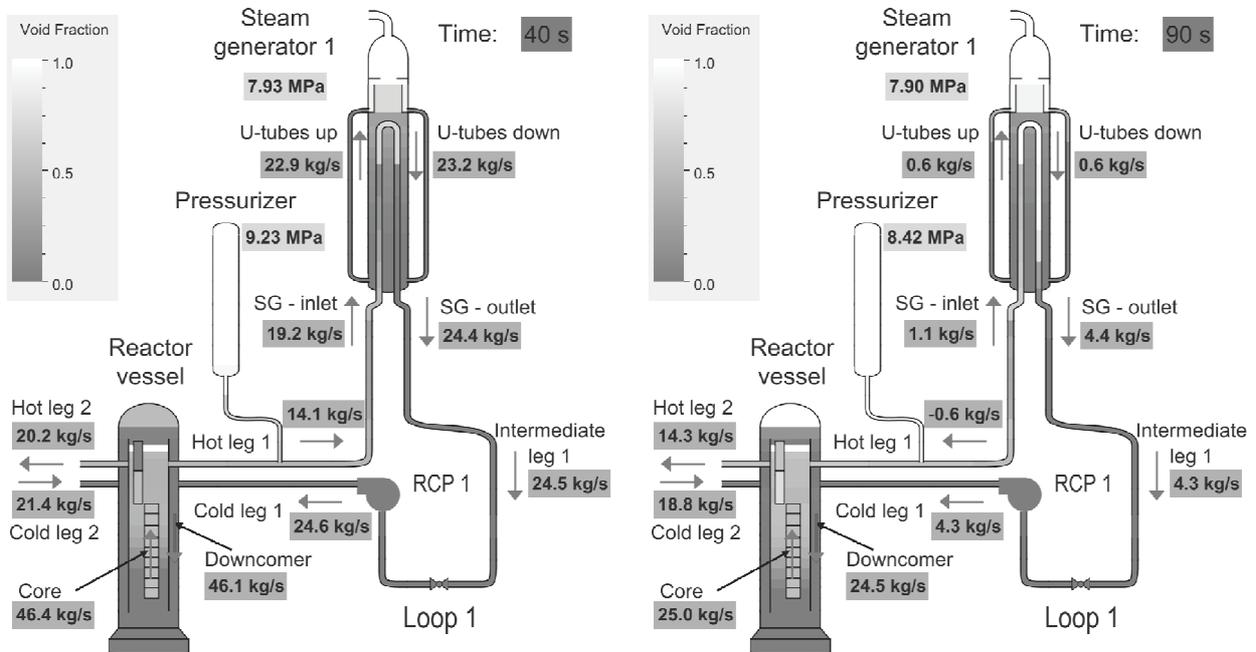


Figure 5. (a) Single-phase natural circulation (left) and (b) reflux condensation (right)

## 4 CONCLUSIONS

The SNAP animation of the ROSA/LSTF facility used for presentation of physical phenomena and processes was developed. The source data needed for animation were obtained by the RELAP5/MOD3.3 Patch 04 computer code. A scenario of a hypothetical small-break loss-of-coolant accident in a hot leg was simulated on the ROSA/LSTF integral test facility. Different animation masks have been developed, including the general facility mask. These masks were used in addition to graphs to describe the calculated scenario. It can be concluded that the developed animation model significantly helps in the analysis and the analysis results can be presented to a wider community in an easy and understandable way. Dissemination of such results may contribute to educating general public about the hypothetical nuclear accidents and their mitigation measures.

## ACKNOWLEDGEMENTS

The author acknowledges the financial support from the Krško NPP and the Slovenian Nuclear Safety Administration within the CAMP program (project no. POG-3473) and from the state budget by the Slovenian Research Agency (program no. P2-0026).

## 5 REFERENCES

- [1] D. M. Snider, K. L. Wagner, W. H. Grush, K. R. Jones, Nuclear Plant Analyzer, U.S. NRC report, NUREG/CR-6291, 1997, Idaho National Engineering Laboratory, INEL-94/0123.
- [2] F. Eltawila, Thermal-Hydraulics Research Plan, Proceedings of The Code Applications and Maintenance Program, Santa Fe, NM, USA, 1996.
- [3] K.K. Murata et al., Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis, NUREG/CR-65, SAND97-1735, Sandia National Laboratories, USA, 1997.
- [4] U.S. NRC, RELAP5/MOD3.3 code manual, Vols. 1 to 8, U.S. NRC report, NUREG/CR-5535/Rev P4, 2010.
- [5] U. S. Nuclear Regulatory Commission, TRACE V5.0p2 User's Manual, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, Washington, DC, 2010.
- [6] B. Mavko, A. Prošek, Users Guide and Description of Demo version of NEK Analyzer, IJS Report IJS-DP-7130, Jožef Stefan Institute, 1995.
- [7] I. Parzer, Possibilities for animation of severe accident phenomena by NPA, Proc. of 4th Regional Meeting Nuclear Energy in Central Europe, Bled, Slovenia, 7-10 September 1997, pp. 37-42.
- [8] L. Cizelj, B. Končar, B. and M. Leskovar, Vulnerability of a partially flooded PWR reactor cavity to a steam explosion, *Nucl. Eng. Des.*, 2006, Vol. 236, pp. 1617-1627.
- [9] Leskovar, M. and Uršič, M., Estimation of ex-vessel steam explosion pressure loads, *Nucl. Eng. Des.*, 2009, Vol. 239(11), pp. 2444-2458.
- [10] A. Prošek, B. Mavko, Animation model of Krško nuclear power plant for RELAP5 calculations, *Nucl. Eng. Des.*, 2011, Vol. 241, pp. 1034-1046.
- [11] M. Čepin, A. Prošek, Contribution of human reliability analysis to the results of probabilistic safety assessment of the nuclear power plant, *Electrotechnical Review*, 2009, Vol. 76(3), pp. 139-144. Available: <http://ev.fe.uni-lj.si/3-2009/Cepin.pdf>.
- [12] A. Prošek, M. Leskovar, Extensions of the fast Fourier transform based method for quantitative assessment of code calculations, *Electrotechnical Review*, 2009, Vol. 76(5), pp. 251-256. Available: <http://ev.fe.uni-lj.si/5-2009/Prosek.pdf>.
- [13] I. Kljenak, B. Mavko, Simulation of containment thermal-hydraulics in the Marviken Blowdown 16 experiment with ASTEC and CONTAIN codes, *Nucl. Eng. Des.*, 2011, Vol. 241(4), pp. 1063-1070.
- [14] APT, Symbolic Nuclear Analysis Package (SNAP), User's Manual, Applied Programming Technology (APT), Inc., 2007.
- [15] ROSA-IV group, ROSA-IV Large Scale Test Facility (LSTF) System Description, Japan Atomic Energy Research Institute (JAERI) report 84-237, January 1985.
- [16] Y. Kukita et al. (22 co-authors), Japan Atomic Energy Research Institute (JAERI): Data Report for ROSA-IV LSTF 10% Hot Leg Break Experiment Run SB-HL-02, Report JAERI-M 90-039, 1990.

**Andrej Prošek** received his diploma degree from the Faculty of Electrical Engineering, University of Ljubljana, in 1987. The same year he joined the Jožef Stefan Institute (JSI), Reactor Engineering Division, and began his postgraduate studies. He received his M.Sc. and Ph.D. degrees in nuclear engineering from the Faculty of Mathematics and Physics, University of Ljubljana, in 1992 and 1999, respectively. Currently he is a senior research associate at JSI. His research interests include nuclear safety, thermal-hydraulic (TH) safety analyses, uncertainty evaluation and accuracy quantification of TH codes.