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# START – Glass Model of PWR

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## Abstract

Recognizing the importance of nuclear engineering in the area of process engineering the University of Maribor, Faculty of Mechanical Engineering has invested in procuring and erecting glass model of pressurized water reactor. This paper deals with description of the model, its capabilities, and plans for its use within nuclear engineering community of Slovenia.

The model, made primarily of glass, serves three purposes:
(a) educational:
(b) professional development;
(c) research.

As an example, medium break loss of coolant accident is presented in the paper. Temperatures within primary and secondary side, and pressure on primary side of reactor coolant system are followed. The characteristic points are emphasized, and commented.

## Introduction

The experimental apparatus described in this contribution was obtained by University of Maribor as a donation from Paul Scherrer Institute of Switzerland where it operated under name of GLAMOUR (see Hudina and Škerget, 1994, Muelli, 1985).

The paper first discusses geometrical characteristics of the setup. Then it moves to possible uses of the setup with an emphasis on its research capabilities. It concludes with description of three sets of results obtained during test runs simulating medium loss of coolant accidents.

## Geometrical characteristics

Figure 1 shows the schematics of experimental setup used to simulate the transients in PWR. It is made primarily of glass, with maximum allowable pressure of 1.5 bar(g) and 35 kW of nominal power. It consists of

(a) primary circuit with the following components:
- reactor vessel containing heating elements connected to transformer;
- hot and cold leg,
- steam generator comprised of U tubes,
- primary pump,
- pressurizer containing spray and heater;

(b) secondary circuit comprising the following components:
- condenser,
- system for secondary water;

(c) auxiliary components such as:
- system of regulation and control,
- water treatment system,
- vacuum and spray pump,
- data acquisition system.

Figures 2 through 5 show most important parts of the experimental setup.

**Possible uses**

There are three main areas in which the experiment can be used, as follows:

(a) educational:

Faculty of mechanical engineering teaches future mechanical engineers, among them those who will be specializing in process engineering among which issues of nuclear engineering are of particular interest. Within the scope of their education they need to be subjected to experimental as well as general simulations to deepen their understanding of the processes encoutered in real world applications.

START offers various issues which can be both measured and discussed. Among those the issues connected to two phase flow such as flashing, instability, flow regimes, inception of boiling are significant, and easily visible in the experimental apparatus.

In addition, START features state of the art data acquisition system through the control panel which permits simultaneous observation of data values on the control panel as well as on attached computer display via applicable software. The students can learn how to use these features and modify them for their needs and purposes.

(b) professional development:

While there is no doubt of how the operators of nuclear facilities perform their tasks using established routines it is also interesting to subject their actions to immediate feedback of the system. In addition, certain responses of the system such as seal loop clearance can be seen as well as behavior of system for control of pressure (pressurizer spray and heaters).

(c) research:

The research which is going to be performed on the experimental apparatus in mainly twofold: curiosity driven, and research for computer program validation and verification.

The curiosity driven research consists of scaling experiments needed for various answers to behaviour of primary and secondary loop during midloop operation such as during refueling. The main problem of the experimental apparatus is in pressure limitation which is significantly different from actual pressure in the PWR. To succesfully overcome this problem either the fluids with appropriate critical pressure ratio should be employed (which is not feasible), or accident sequence portraying actual accident conditions at low (i.e. atmosphere) pressure is foreseen.

The validation and verification program consists of following and developing the methodology well established in other types of mechanical engineering, e.g. for compressible flows as depicted in available literature (see Aechliman and Oberkampf, 1998). This methodology adapted for use in nuclear energy can then be used to sucesfully answer, or attempt to answer the questions arising from verification and validation efforts.

**Medium break loss of coolant accident simulation**

To show possible uses of the system, and its drawbacks, medium LOCA was simulated. In the beginning the system is stabilized at operating temperature and pressure. The secondary side is slightly below ambient pressure (-0.3 bar (g)), and primary side around 0.55 bar (g). At time = 0 the break in cold leg is opened. As soon as the level in the pressurizer drops below preset value the power is reduced to approximately 30% of nominal power. While the target value is 7% (to simulate the decay heat) significant consideration is given to heat losses of uninsulated glass model compared to very well insulated reactor coolant system of PWR. Also, no safety system (e.g. emergency core cooling) engages in order to faster achieve the core uncovery.

Three sets of experiments were performed, see Figs. 6 through 8. Temperature of pressurizer, cold leg, and steam generator, pressure in primary and secondary systems, and reactor power are plotted as a function of time.

The temperatures of pressurizer varies significantly, depending on its status. First, the temperature is higher than in the cold leg, which is understandable and predicatble. Following the initiation of the accident, the temperature drops until it reaches the saturation conditions in the primary system, to drop below the value due to high cooling of the pressurizer vessel in the experimental setup.

In all three cases significant drop in reactor coolant system pressure is seen, followed by coasting toward the steady state, ended by core uncovery. After the liquid level falls bellow cold leg rupture level, and after some boil-off period (approximately 900 seconds after the initiation of the accident) the seal loop clears which is indicated by sharp rise in pressurizer pressure as the pressures in the code leg off-sets the pressure difference in hot leg. After that the accident progresses untill the pressure starts to rise again.

The power is not uniformly distributed in time which is due to control of the transformer by pulses. The integral of power peaks, however, can be assumed to be essentially constant.

## Conclusions

This paper deals with overview of capabilities and plans for use of existing experimental facility START at University of Maribor. First, the experimental facility is discussed from the geometrical characteristics point of view. Then, the foreseen uses are described, and briefly commented. Lastly, three test runs of medium loss of coolant accident are presented, and briefly discussed.

To conclude, the experimental facility is believed to offer a lot of possiblities to nuclear engineering in Slovenia, and it should be used as much as possible. Its research capabilities are limited due to its glass composition, but the transparency on the other hand gives tremendous educational and professional opportunities.

## Literature


Aeschliman, D.P., Oberkampf, W. L., "Experimental Methodology for Computational Fluid Dynamics Code Validation", AIAA Journal, Vol. 36, pp. 733-740, 1998.

Hudina, M., Škerget, L., Glass model of pressurized light water reactor/ Stekleni model tlačnovodnega reaktorja, Strojniški vestnik, Vol. 40, št. 7/8, str. 283-285, 1994.

Muelli, R., "GLAMOUR – Ein Glassmodell eines Druckwasser-Reaktors", EIR Bericht Nr. 571, Wuerenlingen, 1985.

Oberkampf, W. L., Blottner, F.G., "Issues in Computational Fluid Dynamics Code Verification and Validation", AIAA Journal, Vol. 36, pp. 687-695, 1998.

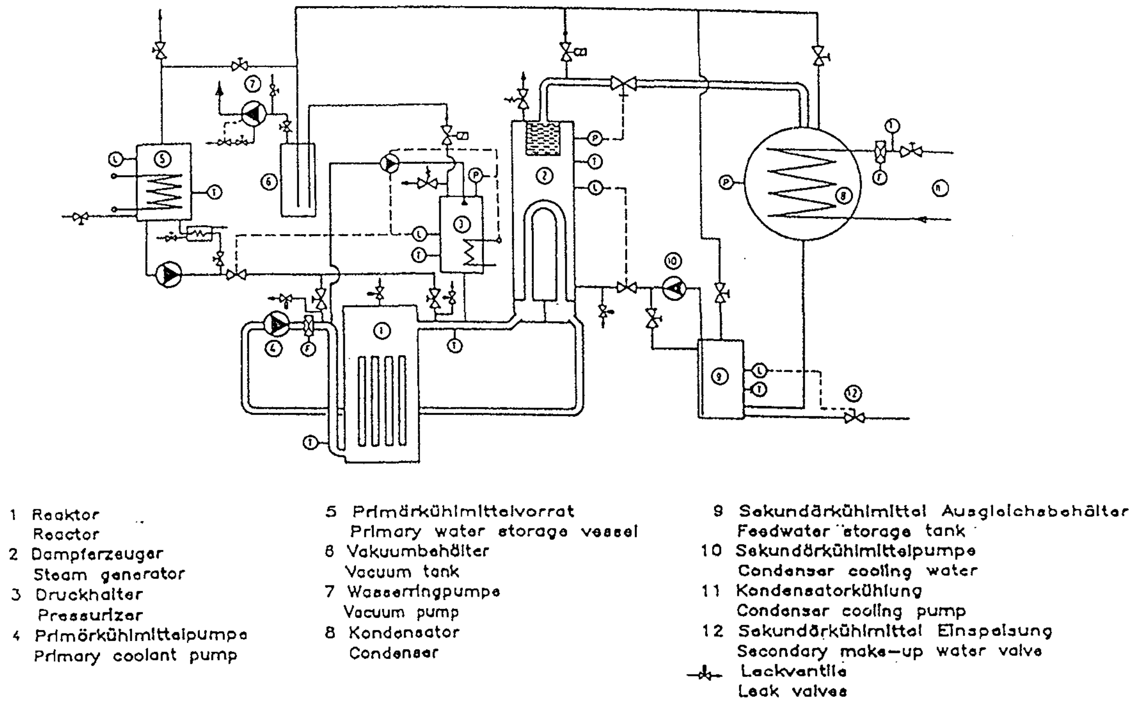


Figure 1. The schematics of START (from Muelli, 1985).

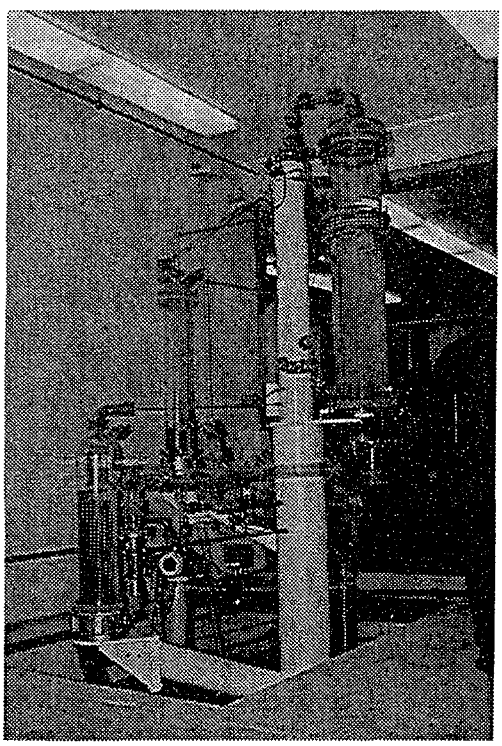

Figure 2. The START.

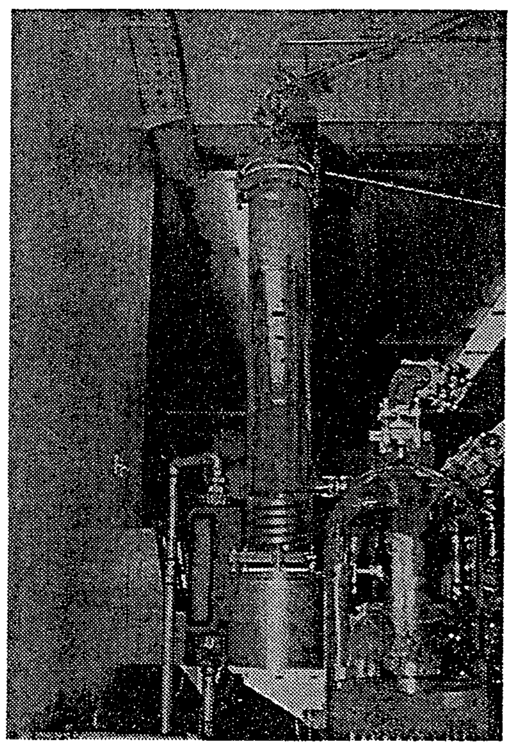

Figure 3. The pressurizer.

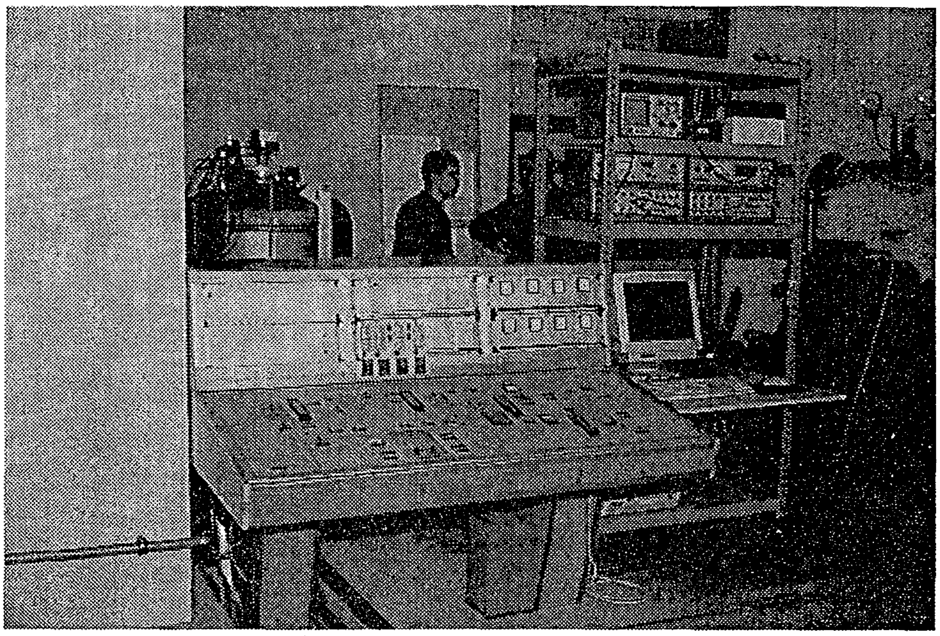

Figure 4. The control panel and data acquisition system.

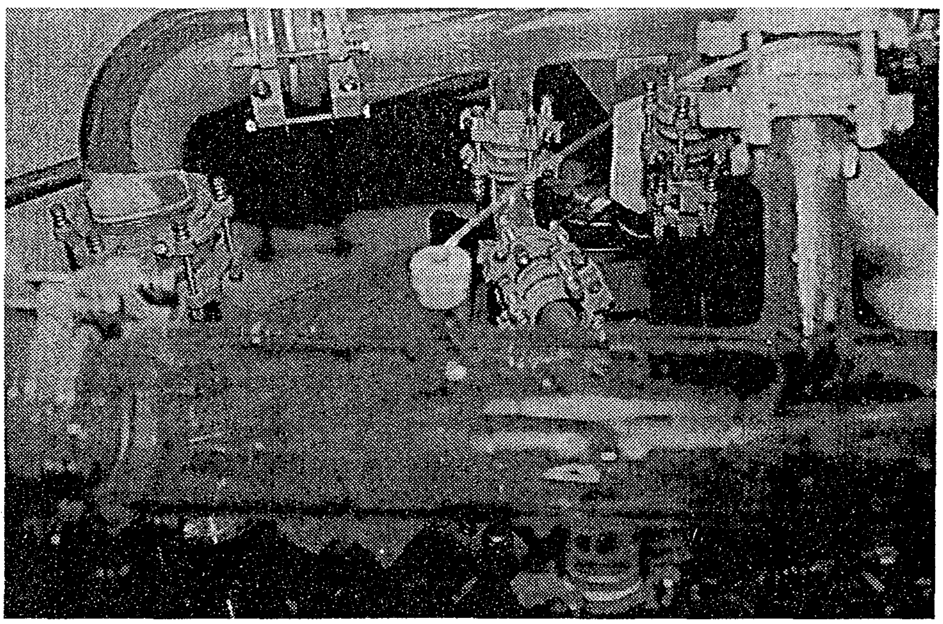

Figure 5. The detail of the cold leg break.

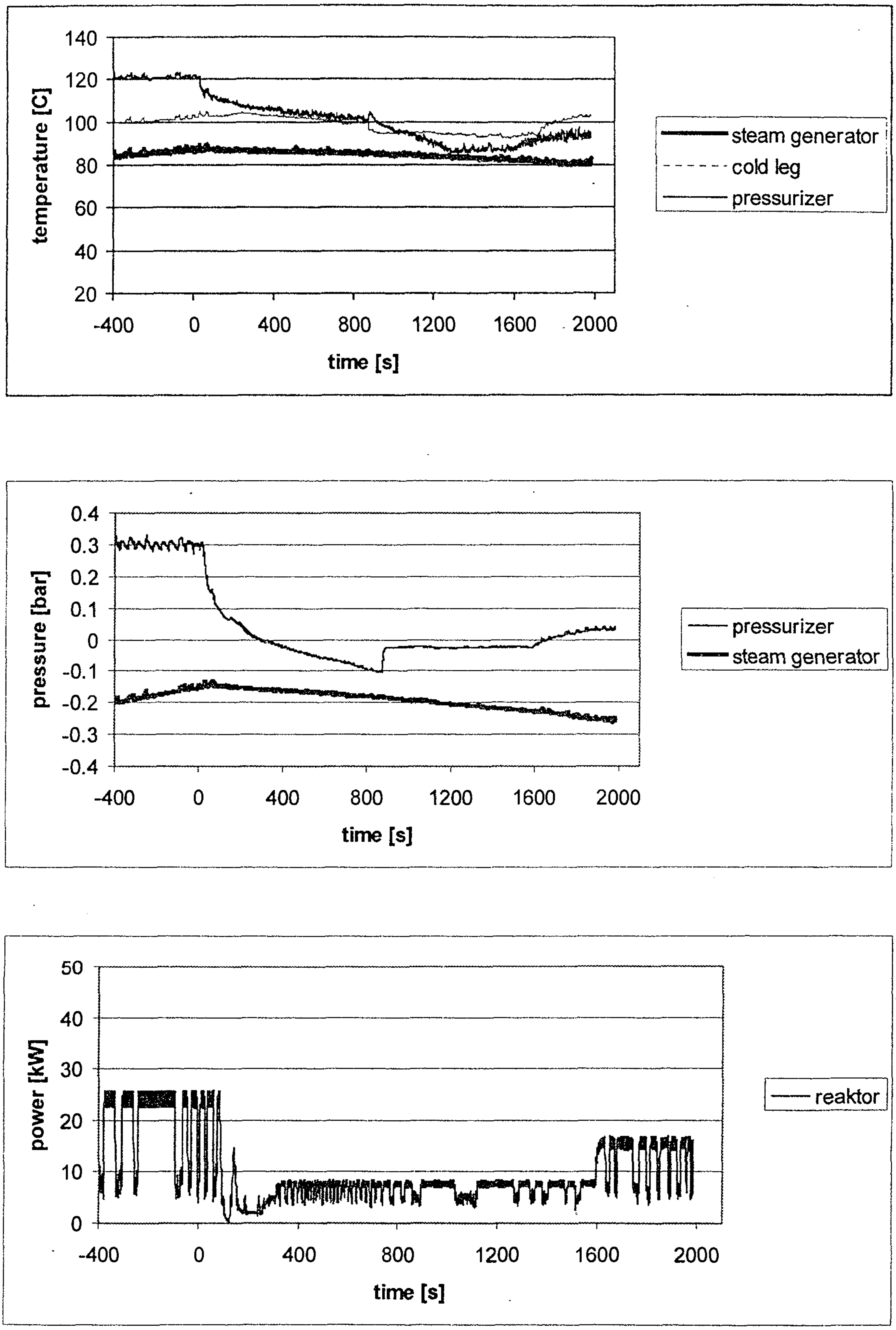


Figure 6. The results for test #1.

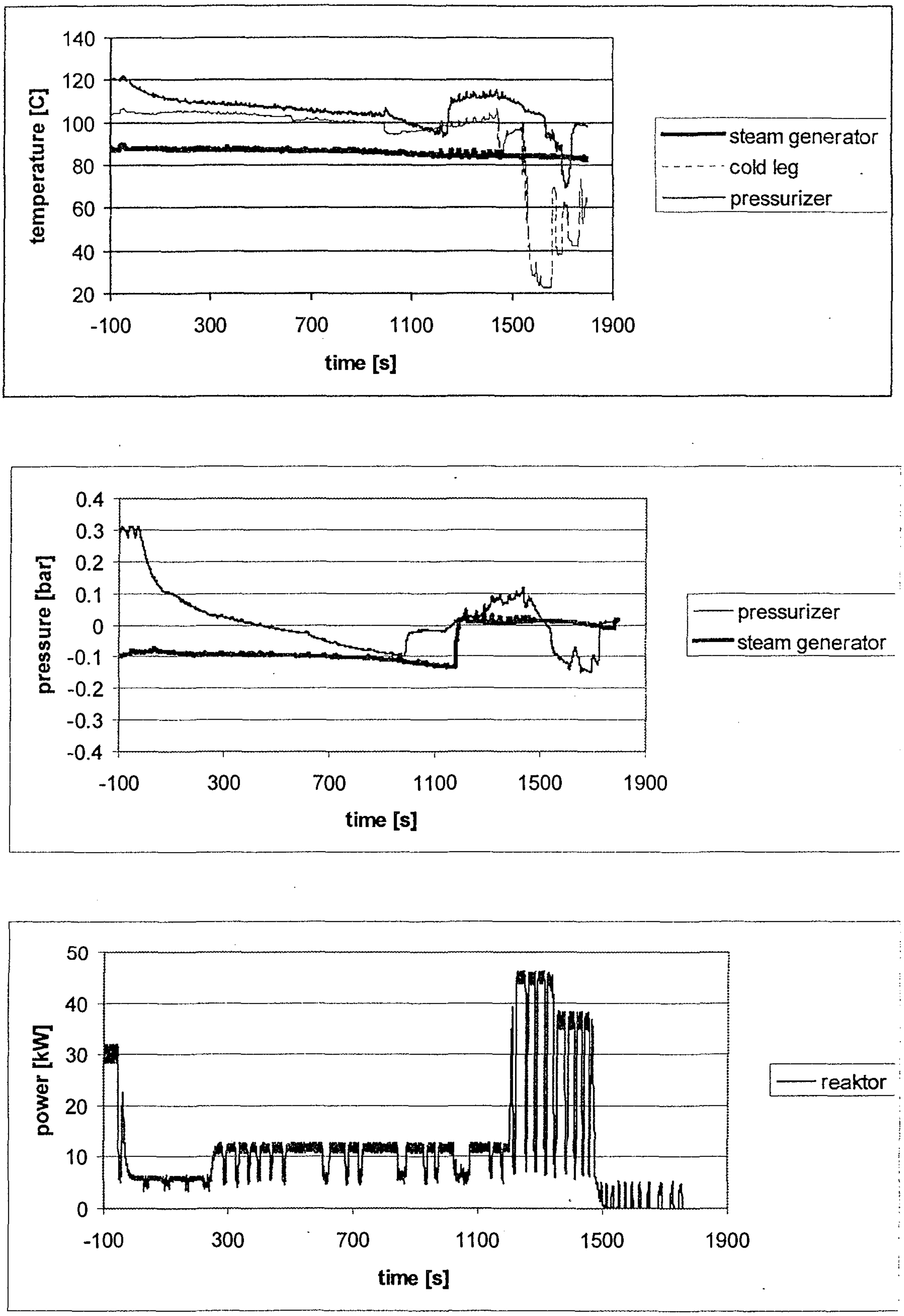


Figure 7. The results for test #2.

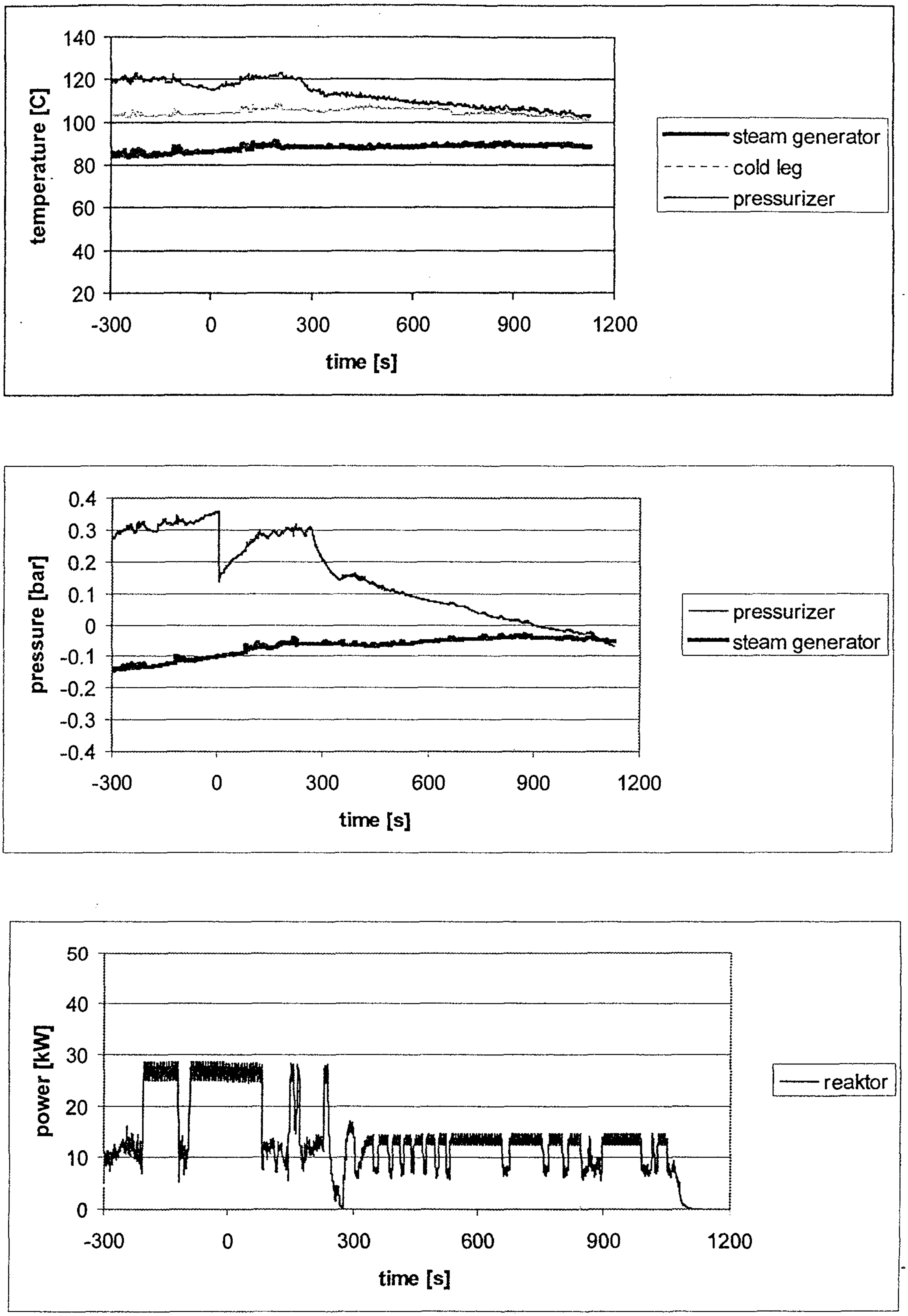


Figure 8. The results for test #3.