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Use of Simplified Nuclear Power Plant Simulator

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ABSTRACT

Simplified nuclear power plant simulator is a tool for simulating normal, abnormal and emergency operation of a nuclear power plant. The nuclear power plant with two loop pressurized water reactor and with inverted U-bend steam generators and dry containment system is considered. One loop with the pressurizer is modeled separately from the other loop without it. The model deals with 138 main input parameters related with the plant parameters such as pressures, temperatures, levels, power, setpoints, concentrations, capacities, masses and dimensionless numbers. The initial conditions contain 110 parameters. 18 initiating events from the set of internal initiating events can be considered. The objective of the work is to show the applicability of the simplified nuclear power plant simulator for modelling of the selected scenarios from a set of selected design basis accidents for the education purposes. Selected initiating events and scenarios have been identified and the data about them was collected. The simulation of the scenarios was performed. The initial conditions have been determined and the operational characteristics were modelled in sense to timely model plant automatic actuations and manual actions of plant operators. The results have been obtained in sense of time dependent curves of the main parameters of interest for showing the state of the plant itself and its systems and subsystems. The obtained results have been compared with the results of other simulations. The differences and the similarities have been discussed. The comparison of the results with some measurements and mostly with other simulations shows some degree of similarity and some differences, which differ among the parameters of interest. In general, the resulted comparisons show acceptability of simulator for education purposes.

Keywords: *nuclear power plant, simulator, safety, human reliability*

1 INTRODUCTION

Simplified nuclear power plant simulator is a tool for simulating normal, abnormal and emergency operation of a nuclear power plant. The nuclear power plant with two loop pressurized water reactor and with inverted U-bend steam generators and dry containment system is considered. One loop with the pressurizer is modeled separately from the other loop without it. The model deals with 138 main input parameters related with the plant parameters such as pressures, temperatures, levels, power, setpoints, concentrations, capacities, masses and dimensionless numbers. The initial conditions contain 110 parameters. 18 initiating events from the set of internal initiating events can be considered. The details of the modelling within the simulation are collected in ref. [1], [2], [3].

The objective of the work is to show the applicability of the simplified nuclear power plant simulator for modelling of the selected scenarios from a set of selected design basis accidents for the education purposes.

2 METHODS AND MODELS

Simplified simulator contains a reduced-node approach used to model the primary coolant system. A non-equilibrium model of the pressurizer handles its normal controls by the pressurizer sprays, pressurizer heaters and pressurizer relief valves. It allows sudden changes of related parameters and extreme conditions such as two-phase mixture in the reactor core and pressurizer filled rigid with water. The steam generators (two of them in the model) are modeled as homogeneous equilibrium two-phase volumes. Heat transfer from the primary system (reactor coolant system) to the secondary system (power conversion system) is treated rigorously during both forced and natural circulation. A point kinetics model is used for the reactor power calculation. The model of large containment is included. Major plant control systems are modelled. Improved heat transfer correlation for the steam generators is included. The discharge rates of fluids due to breaks use typical critical flow models. A mechanistic model of the reactor coolant flow covering both forced and natural circulation provides temperature distribution in the primary coolant. The conditions of the containment are calculated based on a homogeneous equilibrium model with participation of non-condensable air and hydrogen. If the core would be exposed to steam for extended period of time, the core may become overheated and melted consequently. If the zirconium in the cladding reacts with steam then a calculated amount of hydrogen is generated. The mass and energy balance equations with correlations in momentum and heat transfer are solved for all control volumes simultaneously. The progress of transients is dealt with by using Euler integration over every time step increment [1], [2], [3].

2.1 Work procedure

Selected initiating events and scenarios have been identified and the data about them was collected. The initial conditions have been determined and the operational characteristics were modelled in sense to timely model plant automatic actuations and manual actions of plant operators. The simulation of the scenarios was performed. The timing of operator actions was studied for some of the scenarios in order to assess the time window for the operator action before the core gets damaged. The criteria for the core damage is exceedance of the temperature in the reactor core of 923 K for more than 30 minutes or exceedance of the temperature of the core of 1348 K [4].

For some of the simulations, the reference literature has been collected and compared [5], [6], [7], [8], [9].

The small loss of coolant accident, the steam generator tube rupture and the steam line break are selected for presentation of the results.

2.2 Small loss of coolant accident

Initial conditions for the simulation before t=0 were the following: reactor was at full power, 1800 MWt, which suits 600 MW electrical for pressurized water reactor with two loops. The end of fuel cycle is assumed. The location of small loss of coolant accident is in hot leg towards the steam generator A. The extent of the break was 2 inch, where the considering break relate to affected area of 20.25 cm^2 (the simulation software requires the input of the area of the break). It is assumed that the break occurs instantly. Different timings of operator establishing high pressure safety injection are compared assumed that it does not start automatically. When started, both pumps are at times after 30 min, after 40 min, after 50 min and after 1 hour started instantly.

2.3 Steam generator tube rupture model

Initial conditions for the simulation before t=0 were the following: reactor was at full power, 1800 MWt. The end of fuel cycle is assumed. The 100 % break of one tube of steam generator A is assumed ruptured instantly.

2.4 Steam line break model

Initial conditions for the simulation before t=0 were the following: reactor was at full power, 1800 MWt. The end of fuel cycle is assumed. The location of steam line break is outside of containment. The extend of the break was varied as diameter of the break: 8 inch, 9 inch, 10 inch and 11 inch, where the considering breaks relate to affected area of 324.1 cm^2, 410.2 cm^2, 506.4 cm^2 and 612.8 cm^2, respectively. It is assumed that the break occurs instantly.

3 ANALYSES AND RESULTS

The results have been obtained in sense of time dependent curves of the main parameters of interest for showing the status of the plant itself and its systems and subsystems.

3.1 Small loss of coolant accident – results

The results of timely response of 92 parameters was collected. The focus was placed to the water level in the core, high pressure safety injection flow (which gives the timing of success of the related operator action), cladding temperature and the void of reactor coolant system.

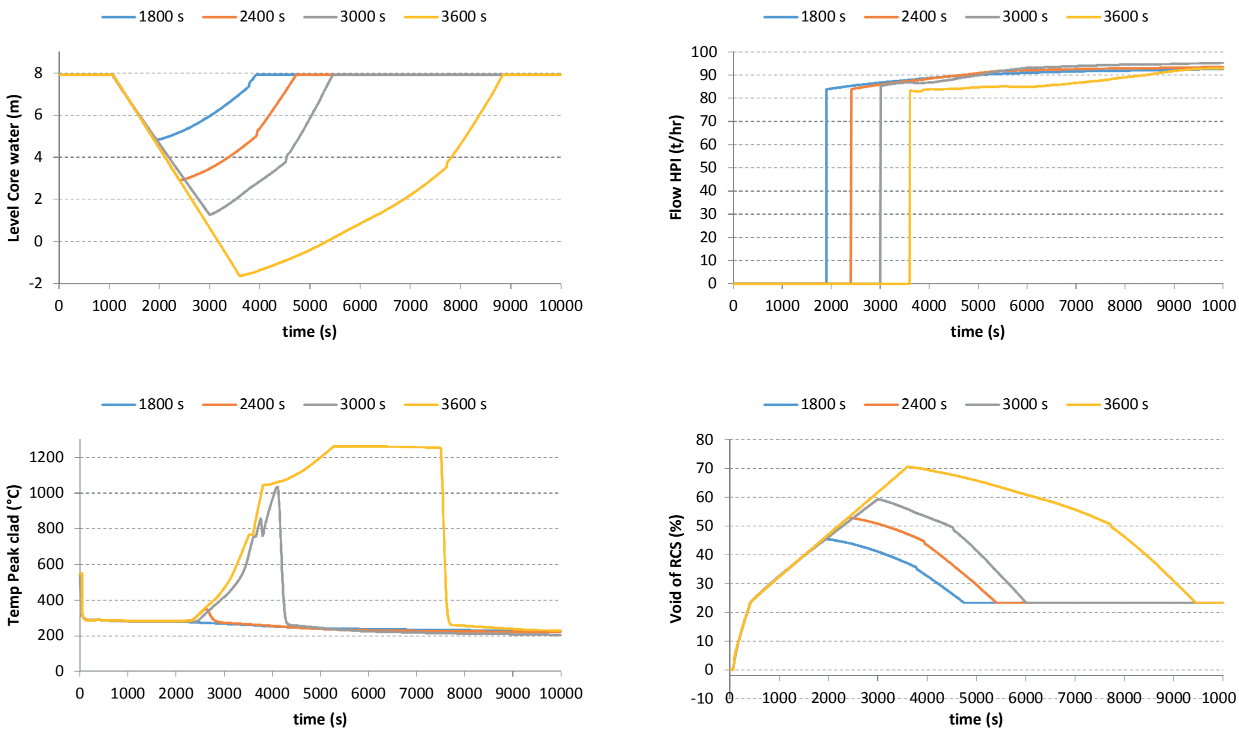

Figure 1: Small loss of coolant accident, 2 inch, timing of establishing high pressure safety injection manually (after 1800 s, after 2400 s, after 3000 s and after 3600 s) vary for 4 cases as indicated at legend, parameters of interest: water level in the core, high pressure safety injection flow (which gives the timing of success of the related operator action), cladding temperature and the void of reactor coolant system

Figure 1 shows the results, which indicate that the lack of the high pressure safety injection for 40 minutes does not result in core damage if soon after that time the high pressure safety injection is restored. The amount of water in the system is assumed rather large (reactor coolant system volume without pressurizer is assumed 180 m^3).

Such sensitivity curves are interesting for strengthening the knowledge about the mutual cross connection of various plant parameters and for feedback to engineering students about behaviour of the plant systems, which can be obtained relatively quickly (the simulation runs 16 times faster than the real time).

Such sensitivity curves may help in determining the operator time windows which are in addition to the determination of the time needed for operator action needed for evaluation of human error probability. The time available equals the difference between the time window and the time needed for the operator action. The larger is the time available, the smaller is the human error probability [4]. The ref. [4] focused to consideration of timing of auxiliary feedwater system actuation if the automatic action has failed.

3.2 Steam generator tube rupture - results

The focus of the 92 observed parameters was placed to the reactor coolant system pressure, high pressure safety injection system flow, rupture mass flow and steam generator levels.

The results depend largely on the amount and the timing of high pressure safety injection to the primary system. The sensitivity analyses for different simulations with different timing of faulted steam generator isolation and injecting water flow revealed difficulties with steam generator level, which can be raised in faulted steam generator far more than expected. Our simulations revealed much larger reactor coolant system pressure decrease and consequently smaller safety injection flow and rupture flow compared to ref. [6] in some of the simulations. Although, if the operator actions are done in the appropriate way the transient is more comparable to the literature [10].

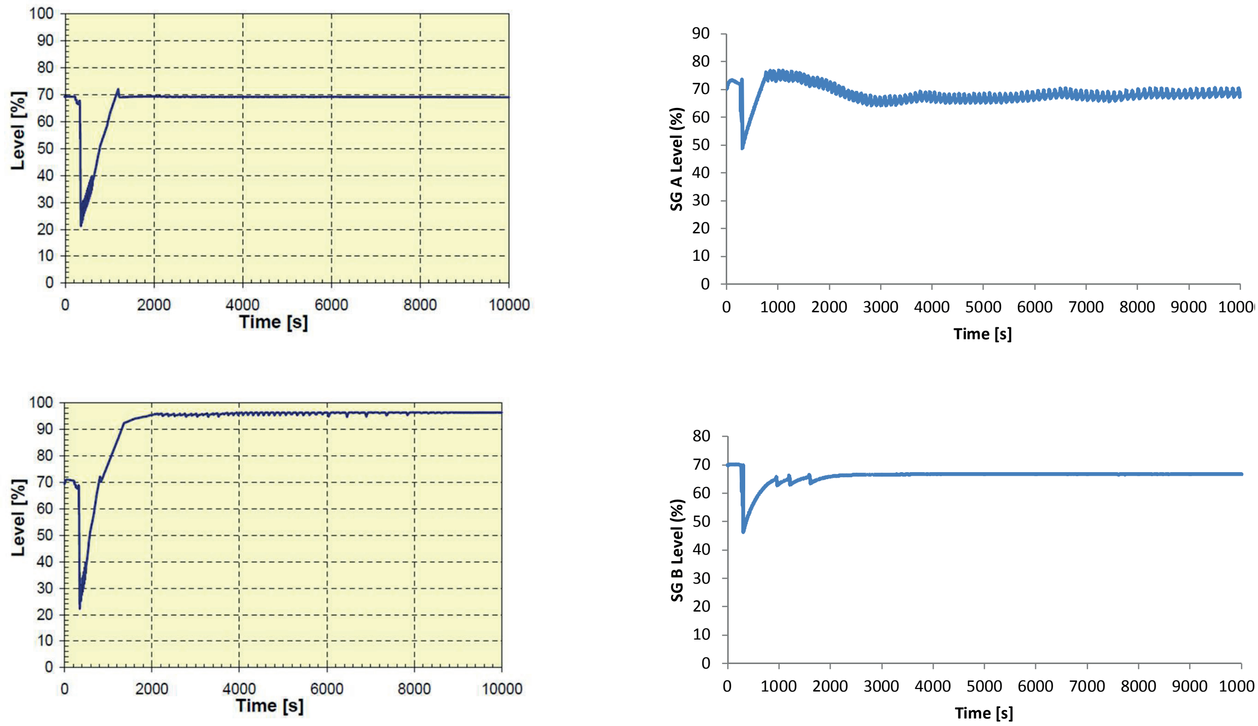

Figure 2: Steam generator tube rupture – steam generator levels, compared with ref. [6]

3.3 Steam line break - results

The parameters such as pressurizer pressure, core thermal power, departure from nucleate boiling, primary side coolant temperatures, steam generator flow rate are selected for presentation of the results of the simulation and for the comparison with the reference literature [5]. The steam line break simulation results differ by the timing of automatic events, which depends on the size of the break. Reactor trip due to high neutron flux (stated at 118 %) occurs at 24 s (116 % of power), 13 s (114 % of power), 9 s (111 % of power) or 7 s (107 % of power) for 8 inch, 9 inch, 10 inch or 11 inch break respectively. Reference [5] reports the reactor trip at 121 % of reactor power after 14.5 s, but the power of the plant is significantly higher 2815 MW of thermal power and two cold legs per one hot leg of each loop, while our plant has one cold leg per one hot leg within each loop.

The following figures show comparison of performed simulations for 9 inch break with the reference [5]. The selection of parameters on those figures was made based on the figures shown in the literature to enable comparison. Figure 3 shows pressurizer pressure. The pressure decreases slowly and with higher rate after the plant trip, similarly as in reference [5]. Figure 4 shows total core power, which increases firstly, and decreases after reactor trip. The power curve is similar as in reference [5].

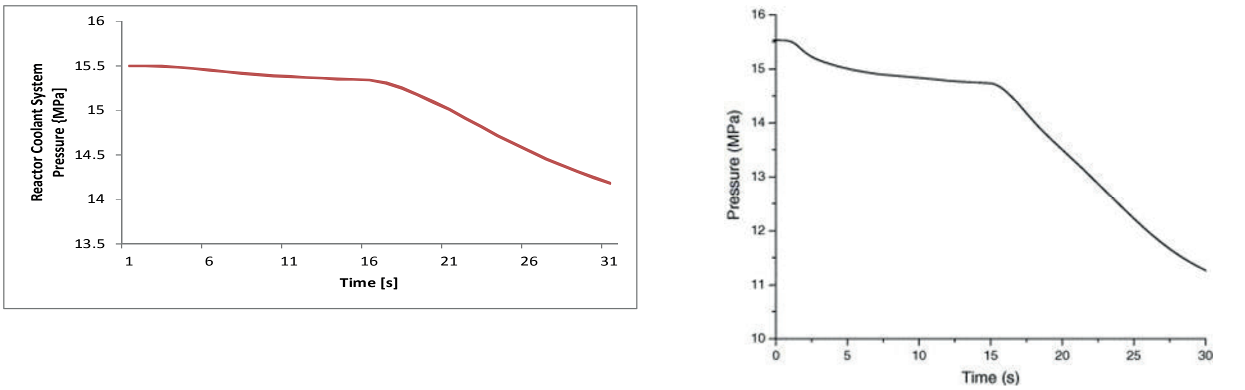

Figure 3: Steam line break outside of containment - pressurizer pressure, comparison with ref. [5]

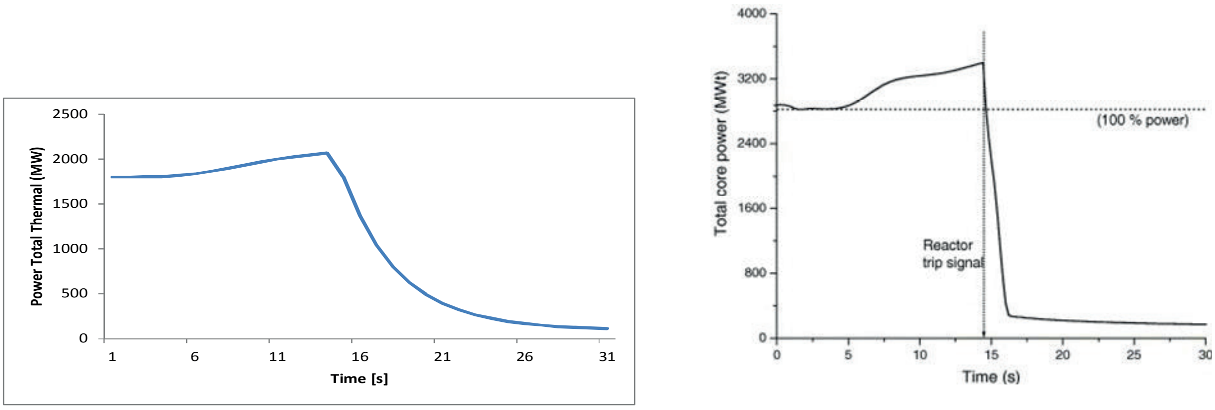

Figure 4: Steam line break outside of containment - power, comparison with ref. [5]

Figure 5 shows minimum departure from nucleate boiling (DNBR), which decrease firstly and increase fast after reactor trip. The curve is similar as in reference [5]. Figure 6 shows primary side coolant temperatures. The hot leg temperatures at our simulations decrease slowly after the reactor trip than in reference [5]. The cold leg temperature at our simulations decrease slowly compared to reference [5].

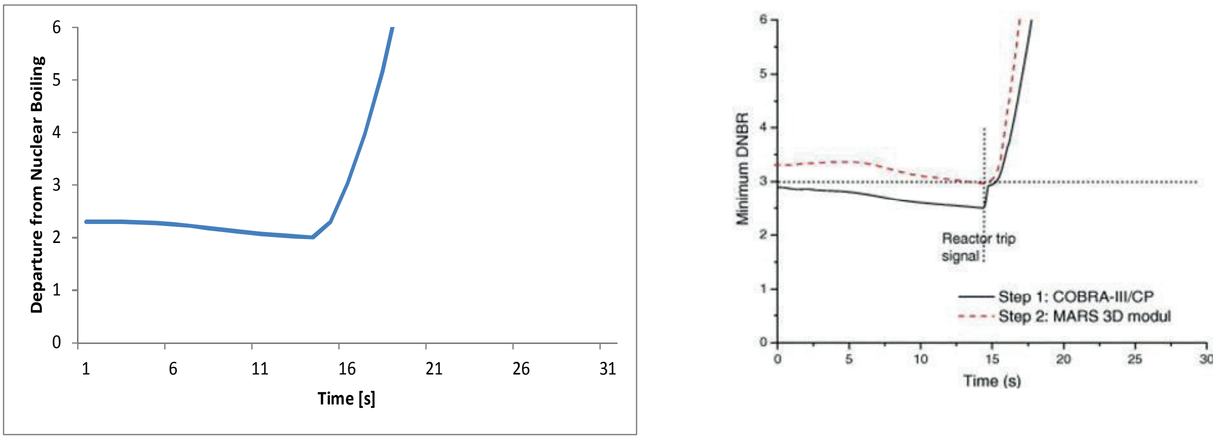

Figure 5: Steam line break outside of containment - minimum DNBR, comparison with ref. [5]

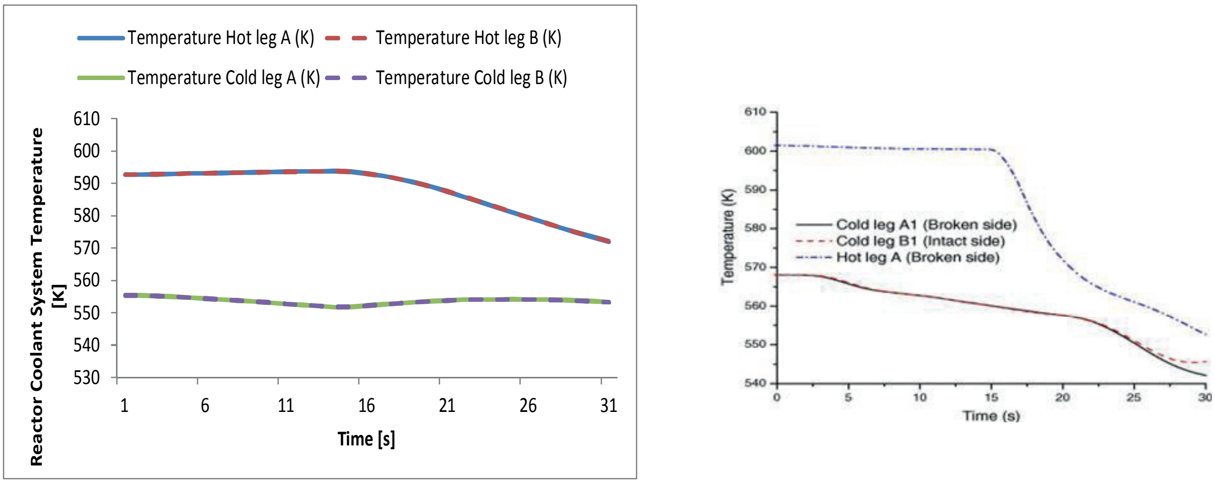

Figure 6: Steam line break outside of containment - primary side coolant temperatures, comparison with ref. [5]

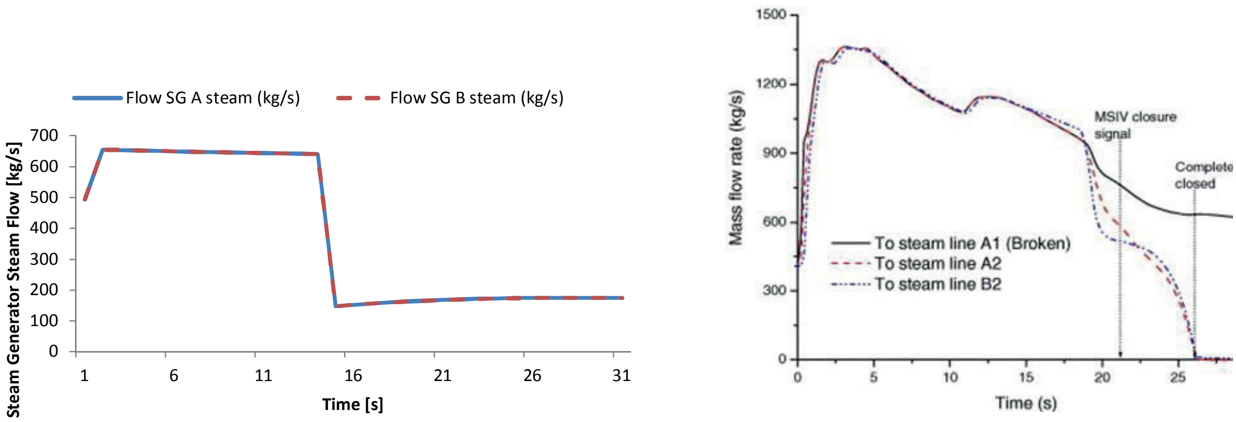

Figure 7: Steam line break outside of containment - steam flow rate at the steam generator exit, comparison with ref. [5]

Figure 7 shows steam flow rate at the steam generator exit. Our simulations shows generally more flat curve, which show significant reduction of flow shortly after trip and final reduction after longer time (see Figure 8) if compared to the ref. [5]. Both times largely depends on the size of the break, which is in more details shown on Figure 8.

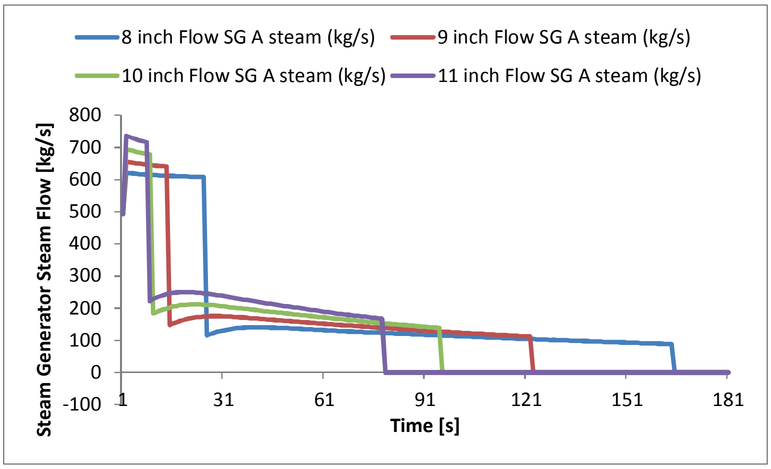

Figure 8: Steam line break outside of containment - steam flow rate at the steam generator exit, extended time to observe the closure time of the main steam isolation valve

Several other simulations of other initiating events were performed [10], [11], including various losses of coolant accidents [12] and the scenario as happened at Three miles island in 1979.

The overall impression about the results obtained is positive. The simplified simulator is simplified enough that engineering students can use it together with studying the theory of reactor systems and their behaviour and can quickly get the results (several times faster than the real time). The differences regarding the simulator results and the scenarios timing from the safety analysis report can differ significantly for some of the parameters and can be very similar for other parameters, so the scenarios need to be carefully selected and prepared. But general connections between parameters of interests support the initial expectation about usability of the simulator.

4 CONCLUSION

The simplified nuclear power plant simulator was tested in sense of observing the timely behaviour of the plant parameters at various plant conditions. Selected initiating events were analysed and the related plant behaviour was observed accordingly and some simulations were compared with the references reporting similar scenarios. Small loss of coolant accident, steam line break and steam generator tube rupture have been selected for analysis, in spite of the fact that several other simulations have been performed. In general, the resulted comparisons show acceptability of simulator for education purposes.

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