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Comparison of the Main Parameters Behavior in VVER 1000 During MSLB Accident for Different Fuel Campaigns

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ABSTRACT

The present paper has been focused on investigation of the plant behaviour parameters in the event of a Main Steam Line Break (MSLB) accident during different fuel campaigns. The calculations have been performed using RELAP5/mod3.3 thermal-hydraulic integral system computer code with the point kinetics approach.

The simulated scenario is "Steam Line Break" with ID 580 mm. The break is located outside of the containment, between the SG #4 and the "Steam Isolation Valve" (SIV). The studied event is characterized by asymmetric cooling in the reactor vessel and reactor core and by significant space-time effects. The investigated scenario is important to investigate core criticality and possible power return.

To simplify the investigation, most of the assumptions applied in the preparation of the executed scenario have been developed based on the research done in the previously performed OECD VVER1000 MSLB benchmark problem. The main objective of the work is to study the response of the main parameters of the plant during a MSLB accident. The other objective is to investigate the reactivity response under conditions corresponding to the beginning of the fuel cycle of 1st campaign with a fully fresh core and 8th campaign with mixed core fuel of a VVER1000 reactor. Furthermore, the important phenomena observed during MSLB such as flow reverse in the damaged loop have been analysed, too.

The work performed in this study is important for improving the code and evaluating the behavior of the VVER1000 plant parameters important to the safety.

1 INTRODUCTION

The MSLB transient is analysed using a RELAP 5 integral system thermal hydraulic code [2] with the point kinetics approach. It has been performed both calculations based on the previously performed OECD MSLB benchmark problem. Although, that the previous OECD MSLB [5] is evaluated for the end of cycle (EOC), it has been decided to perform the analysis using fresh fuel conditions at the beginning of cycle (BOC) at nominal reactor power for first calculation in order to reduce possible source of discrepancies [4].

For the purposes of the study, the reactivity response and plant response were investigated during the two fuel cycles: fresh fuel cycle (1st cycle) and mixed fuel cycle (8th cycle).

2 VVER1000 BRIEF MODEL DESCRIPTION

The Baseline input deck for VVER-1000/V320 Kozloduy Nuclear Power Plant, Unit 6 has been developed by the INRNE-BAS [6]. The VVER1000 RELAP5 model has been developed for analysis of operational occurrences, abnormal events, and design basis scenarios. The model provides a significant analytical capability for the specialists working in the field of the NPP safety. Data and information for the modelling of all main systems and components were obtained from the Kozloduy documentation and from the power plant staff.

The Kozloduy VVER-1000 RELAP5 model has been defined to include all major systems of Kozloduy NPP, Unit 6, namely: reactor core, reactor vessel, main coolant pumps (MCPs), steam generators (SGs), steam lines and main steam header (MSH), emergency protection system, pressure control system of the primary circuit, makeup system, safety injection system, steam dumping devices (BRU-K, BRU-A, SG and pressurizer safety valves), and main feedwater system (FW).

The RELAP5 model configuration provides a detailed representation of the primary, secondary, and safety systems (see Figure1). A hot and average heated flow paths and a core bypass channel represent the reactor core region. The reactor vessel model includes a downcomer, lower plenum, and outlet plenum. The pressurizer (PRz) system includes heaters, spray and pressurizer relief valves. The safety system representation includes accumulators, high and low pressure injection systems, and reactor scram system. The model of the make up and blowdown systems includes associated control systems.

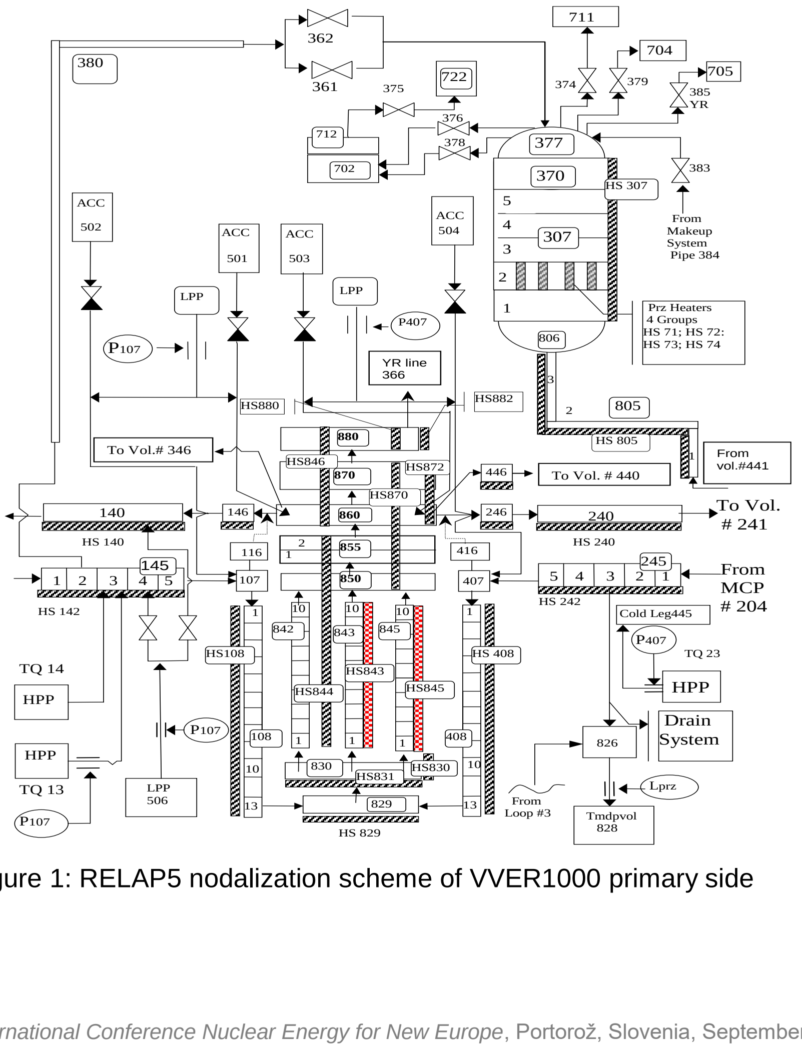

Figure 1: RELAP5 nodalization scheme of VVER1000 primary side

3 BRIEF DESCRIPTION OF MSLB SCENARIO AND NEUTRON PHISICS DATA

3.1 MSLB scenario and main assumptions

1. Size break (ID:580 mm) at steam line #4 between SG #4 and BZOK (SIV).
2. SCRAM is activated after the break initiation.
3. MCP #4 coast down at app. 55 sec.
4. Feed water valve fails and remains open for some time (additional FW into SG). In fact, the feed water flow rates have been taken as the boundary conditions from the OECD MSLB benchmark problem to avoiding additional discrepancy.
5. Turbine stop valves (MSIV) closes 10 sec. after the reactor scram. Time to fully open/close of MSIV is 0.2 sec.
6. Make up and Let down systems activate only during the steady state. During the transient the systems are not used for reducing the possible uncertainty. The both systems isolate for 2 sec.
7. Turbine bypass to the condenser (BRU-Ks) starts to open and switches to MSH pressure control mode after closing MSIV. The all BRU-Ks open, when the P_{MSH} > 6.67 MPa, reducing the pressure and supporting it to P_{MSH} = 6.2807 MPa. BRU-Ks closes when the P_{MSH} drop to 5.79 MPa and re-open again, only if P_{MSH} > 6.67 MPa.
8. BZOK #4 is activats after reaching it's set point and isolate the SG #4 by steam.
9. PRz heaters are switched on after primary side depressurization. They try to support the primary pressure until PRz water level drops to 4.2 m. After that they switched off. The work of PRz heaters was decided to be included additionally to the OECD MSLB scenario.

3.2 Reactor physics data used for the investigation

The reactor physics data used in this investigation is presented below.

The kinetic parameters for BOC of 1st and 8th campaigns [1, 2] have been based on the OECD benchmark. The scram reactivity is assumed to be real NPP value, which is 10.96$.

Table 1: Kinetic data

Parameters	Value	Value
	1st campaign	8th campaign
Total beta effective (pcm)	727	663
Neutron generation time	0.267E-04	0.271E-04

Table 2: The reactivity coefficients for BOC of 1st and 8th campaigns

Reactivity feedback coefficients	BOC of 1st campaign	BOC of 8th campaign
Fuel temperature (DTC), pcm/K	-1.661	-1.660
Coolant temperature (including coolant density) (MTC), pcm/K	-3.1	-23.32

4 COMPARISON OF CALCULATED RESULTS

In this section is presented the comparison of the main calculated results. The comparison of differential break flow rate is presented on Figure 2. The comparison shows that both calculations predicted identical results. As it is seen the coolant starts to decrease rapidly after the steam line break initiation. At the break initiation it is observed at around 2000 kg/s coolant flow rate, which reducing to 0 kg/s after 250 sec, due to SG#4 pressure decrease. The observed small fluctuations in the period between 20 sec and 100 sec, are due to the code numerical problems. For modelling of break flow is accepted Henry-Fauske critical flow model.

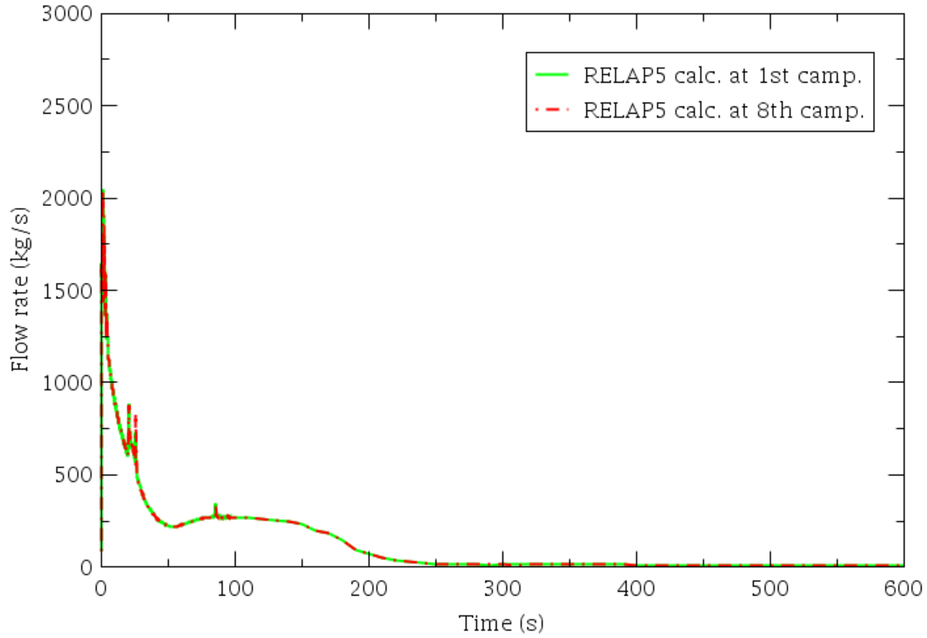

Figure 2: Comparison of differential break flow rate

The comparison of core exit pressure predicted in both calculations show identical behaviour (see Figure 3). As it is seen, after the break initiation in loop#4 and activation of reactor SCRAM, the core exit pressure starts to decrease rapidly in first 10 sec., it decreases rapidly due to the coolant shrinkage (after the reactor SCRAM) and overcooling of primary side from the secondary side. At 200 sec, the primary pressure starts to increase slowly after the secondary system isolation, when the following conditions are reached: P_{SG} < 50 kgf/cm2, dTs (I-II) > 75 °C and $T_{primary}$ > 200°C.

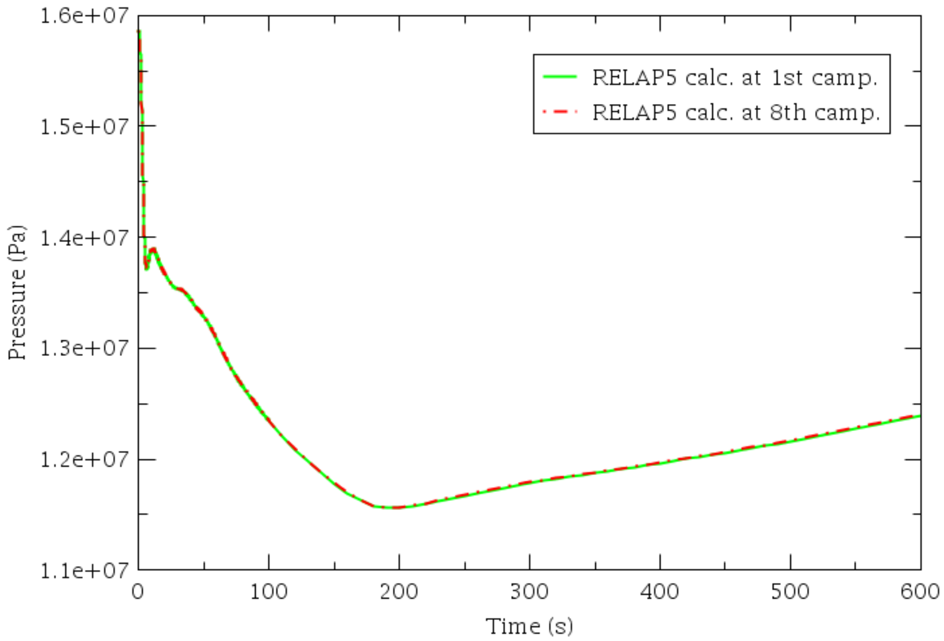

Figure 3: Comparison of core exit pressure

The comparison of PRz water level behavior is presented on Figure 4, The comparison shows that in both calculations are predicted identical results. The PRz water level starts to decrease after the break initiating. It decreases to 1.8 m at app. 190 sec, after that it starts to increase slowly to 2.2 m. The PRz heaters work in correspondence with their set point. After the break initiation the heaters start to work and to support the primary pressure. The heaters switch off when the PRz water level drops to 4.2 m.

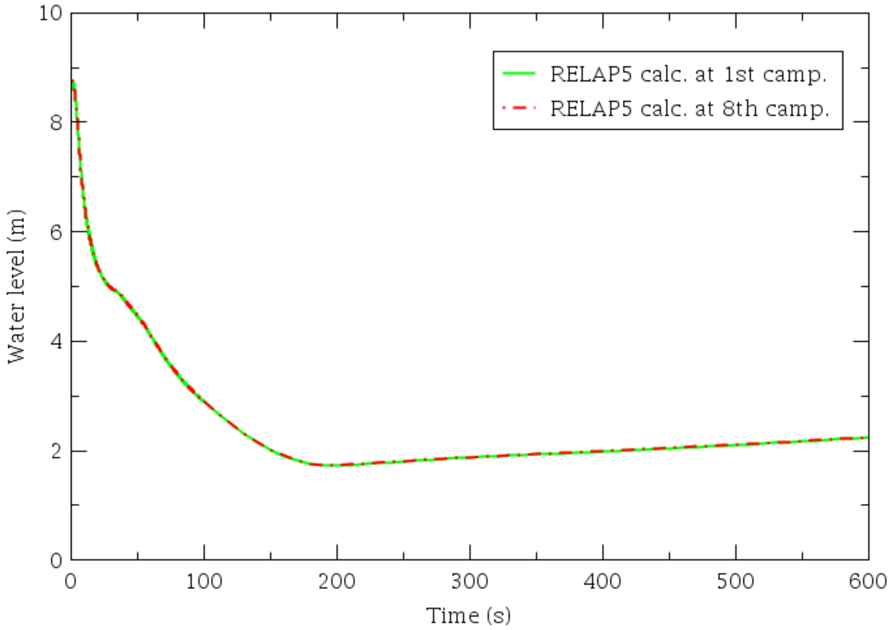

Figure 4: Comparison of pressurizer water level

The comparison of secondary side pressure in non-affected SG#1 is presented on Figure 5. The comparison of secondary side pressure, shows identical results. The secondary pressure increase shortly after the start of the transient, then 10 sec after the beginning of event, it decrease rapidly for short and start to increase again. The pressure increases due to the closing of turbine MSIV and isolating of affected SG#4 with SIV (BZOK). This also leads to pressure increase at the non-affected SGs. After reaching the set points for safety valve BRU-K activation, they open and start to regulate the secondary pressure. The BRU-K opens when the secondary pressure reaches 6.67 MPa, reducing the pressure and controls it to the 6.28 MPa, when the secondary pressure drops to 5.79 MPa the BRU-K closes.

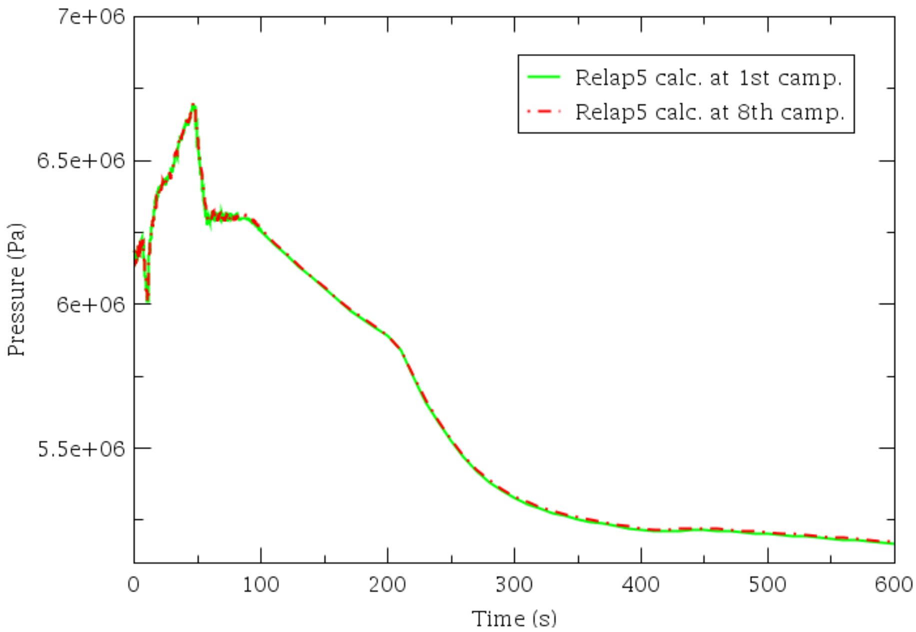

Figure 5: Comparison of secondary pressure in SG#1

The comparison of the secondary side pressure in affected SG#4 is presented on Figure 6. The secondary pressure predicted in both calculations is identical. The pressure decreases rapidly after the break initiation it continues to decrease slowly after 100 sec until approximately 250 sec. The decay heat in the first 50 s is removed mainly from the SG#4, after the depressurization of SG#4 at around 220 s, the decay heat is removing from the work of BRU-K.

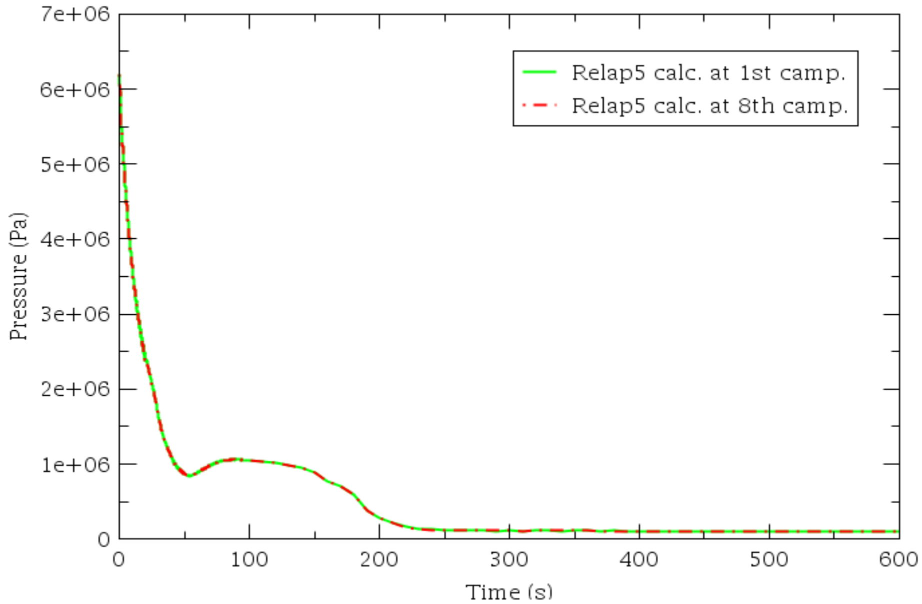

Figure 6: Comparison of of secondary pressure in SG#4

The comparison of coolant temperature feedback reactivity is given on Figure 7. As it is seen from the comparison the inserted positive reactivity in the case with mixed core is significantly larger. The max. value of app 2 $ is reached at 200 sec, while in the calculation with fresh core the reactivity is only 0.23 $. The observed reactivity increases at about 200 sec in both cases due to the coolant temperature decrease.

After 200 sec the coolant reactivity decreases slowly until the end of the transient and reaches at 1.4 $ in the calculations with mixed fuel core, while in the calculation with fresh core the reactivity feedback is significantly lower, only 0.20 $.

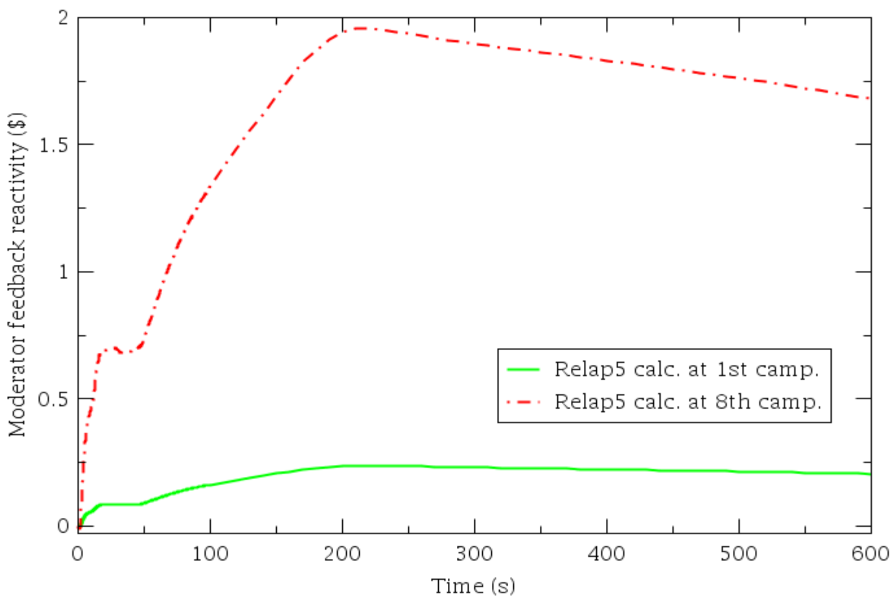

Figure 7: Comparison of coolant temperature feedback reactivity

The comparison of the fuel temperature feedback reactivity is given in Figure 8. As it can be seen from the comparison, the inserted fuel feedback reactivity is very small in both cases. Overall, the predicted fuel reactivity in both calculations is in good agreement. The fuel temperature feedback reactivity in the calculation with mixed core is slightly lower than the predicted fuel reactivity in the calculation with fresh core, after that the reactivity decrease slightly in both cases and reached 0.42 $ in the calculation with mixed core, while in the calculation with fresh core the reactivity is 0.38 $.

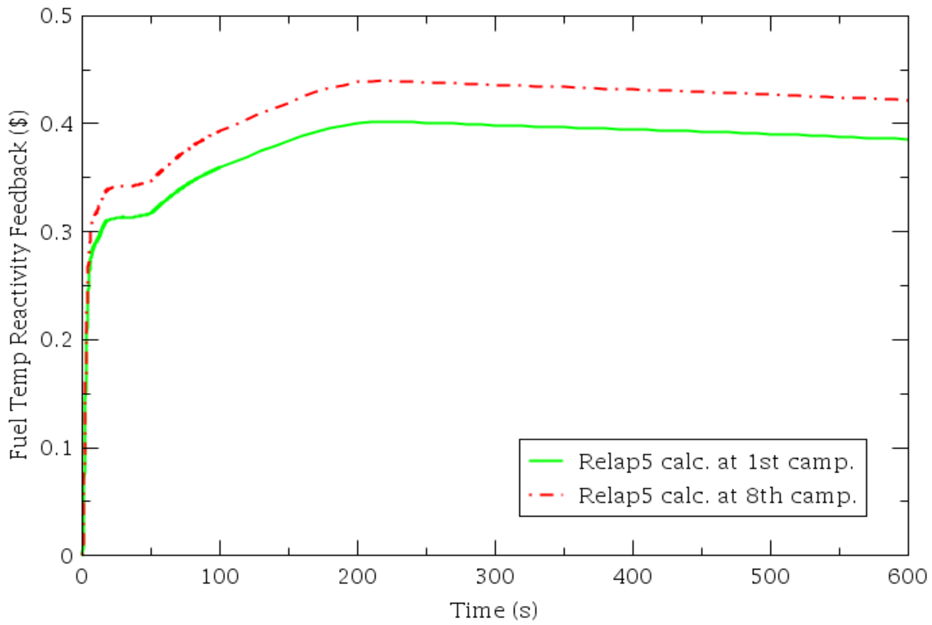

Figure 8: Comparison of fuel temperature feedback reactivity

The comparison of total reactivity is presented on Figure 9.

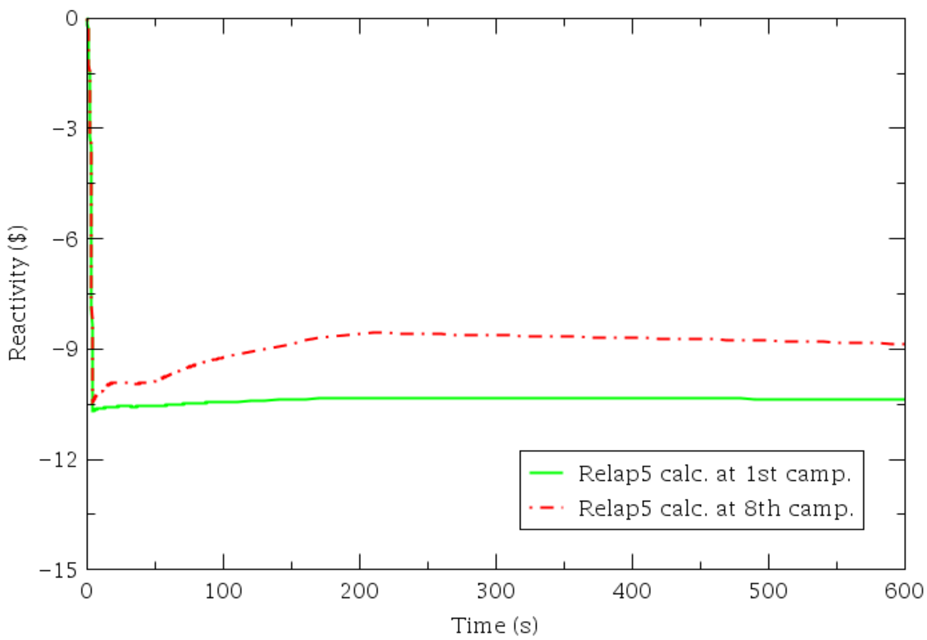

Figure 9: Comparison of total reactivity

The comparison shows a close prediction of the total reactivity between the both calculations. After the water supply from the damaged SG is blown down, the temperature in the primary system begins to rise slowly, which causing the decrease of the reactivity.
The total reactivity is negative in both calculations, it is slightly larger in case with fresh core -

10 $, while in the calculation with mixed load it is -9.15 $. The most significant influence on the total reactivity change is coming from the coolant temperature feedback reactivity. In general, it can be concluded that in both cases there is no risk for power return, due to the reactor subcriticality.

5 CONCLUSIONS

The preformed investigation presented a comparison between both RELAP5 calculations of MSLB accident during different fuel campaigns. The performed analysis is based on a previously performed OECD “MSLB” Benchmark.

The comparison of MSLB results, demonstrate good agreement of the compared parameters. In general, the results of all compared parameters are the same, except the reactivities. The comparison shows that the plant respond is almost with the same behaviour during BOC during the both fuel campaigns: 1st campaign with fresh core and 8th fuel campaign with mixed core.

Generally, it could be concluded that in both cases, there is no risk for power return, due to the reactor subcriticality.

The predicted results in both calculations demonstrates the stability of the reactor system during the accident progression.

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