

Application of ASTEC Computer Code for Validation of SAMG Based on LB LOCA Scenario for VVER 1000

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Abstract. In case of unsuccessful application of VVER1000 Emergency Operating Procedures (EOPs) for different reasons during the accident in Nuclear Power Plant (NPP), the operators have to leave the EOPs and to start applying Severe Accident Management Guidance (SAMGs). This paper presents calculation results using computer code for validation of operator actions during severe accident conditions in NPP with VVER-1000 type of reactors. The Kozloduy NPP Units 5&6 has been selected as a reference NPP. As an analytical tool has been used ASTECvp2r3p2 computer code developed by IRSN and GRS.

The work is oriented on investigation of plant behavior mainly after beginning of reactor core heat up, due to simulation of Large Break Loss of Coolant Accident (LB LOCA) with simultaneous loss of AC and DC power. It is assumed that DGs are missing until a certain plant state, and after that they are available.

It have been done calculations without and with operator actions selected based on severe accident management strategies considered in Kozloduy Nuclear Power Plant (KNPP). Based on the SAMG strategies the operator should depressurize primary circuit by gas removing system (YR) if there is a need of additional primary depressurization and to start to cool down the reactor core at 923 K and 1253 K core exit temperatures as it is pointed in SAMG strategy.

The purpose of these analyses is to study the reactor core behavior parameters and to estimate the time available for performing actions. The main goal is to analyze the possibility of preserving the reactor core from damage during a severe accident and to assess hydrogen generation as a result of reflooding of the overheated core.

LB LOCA scenario has been selected with break sizes ID 300 mm in the cold leg between main coolant pump (MCP) and reactor pressure vessel. The cold water is injected by a high pressure pump (HPP) in the undamaged cold leg.

Keywords: severe accident, VVER1000, SAMG, ASTEC.

1 Introduction

The use of SAMGs is required when an accident situation is not handled properly through the use of EOPs thus leading to severe accident (SA) sequences with partial or complete core uncover followed by core heat up and damaging of the reactor core [2,6,9,17,18,20]. Nevertheless, that investigated scenario is with very low probability circumstances, SA sequences may have happened during loss of coolant (LOCA) if simultaneously with LOCA is involved loss of AC power and failure of all diesel generators (DGs) for some period of time. It will cause a core heat up and core melting if there are no involved SAMGs. Loss of alternating and direct current power (AC and DC) will not allow the operators in case of any kind of LOCAs to apply EOPs successfully and it will request further transition to SAMGs. The operator actions in such situations are oriented to create conditions for water injection in primary circuit and further cool down of reactor core with any one of the available high or low pressure pumps to prevent core damage. In case of small break LOCA there is a possibility of limitation of using pumps due to possible high pressure. In this situation there is a strategy in SAMGs for primary depressurization, which will allow the use of high or low pressure pumps. As the selected initiating event of LOCA with break size of 300 mm is large enough to depressurize

primary circuit the operator actions will be limited to run at least one DG and to start at least one pump to cool down reactor core, as soon as possible.

Units 5 and 6 of the Kozloduy NPP equipped with VVER1000/v320 reactors are used as a reference nuclear power plant for this analysis. The main equipment for this type of reactors as reactor vessel, primary circuit, pressurizer, steam generators, main coolant pumps, hydro-accumulators and other important equipment of VVER1000/v320 design could be seen in Figure 1.

The VVER1000/v320 is a pressurized water reactor that generates 1000 MW electric power and 3000 MW thermal power. The reactor has four coolant loops and four horizontal steam generators (SGs). Each one of the loops consists of cold and hot leg, MCP and a horizontal SG. In the model used for this investigation the three of the loops are represented as one but the other one loop is represented as a single loop. A feed-water system fed SGs.

2 Event Description

Large Break LOCA is classified as a postulated accident which is of such low frequency that it is not expected to occur, but it is considered in the reactor design to demonstrate sufficient safety margin in relation to a loss

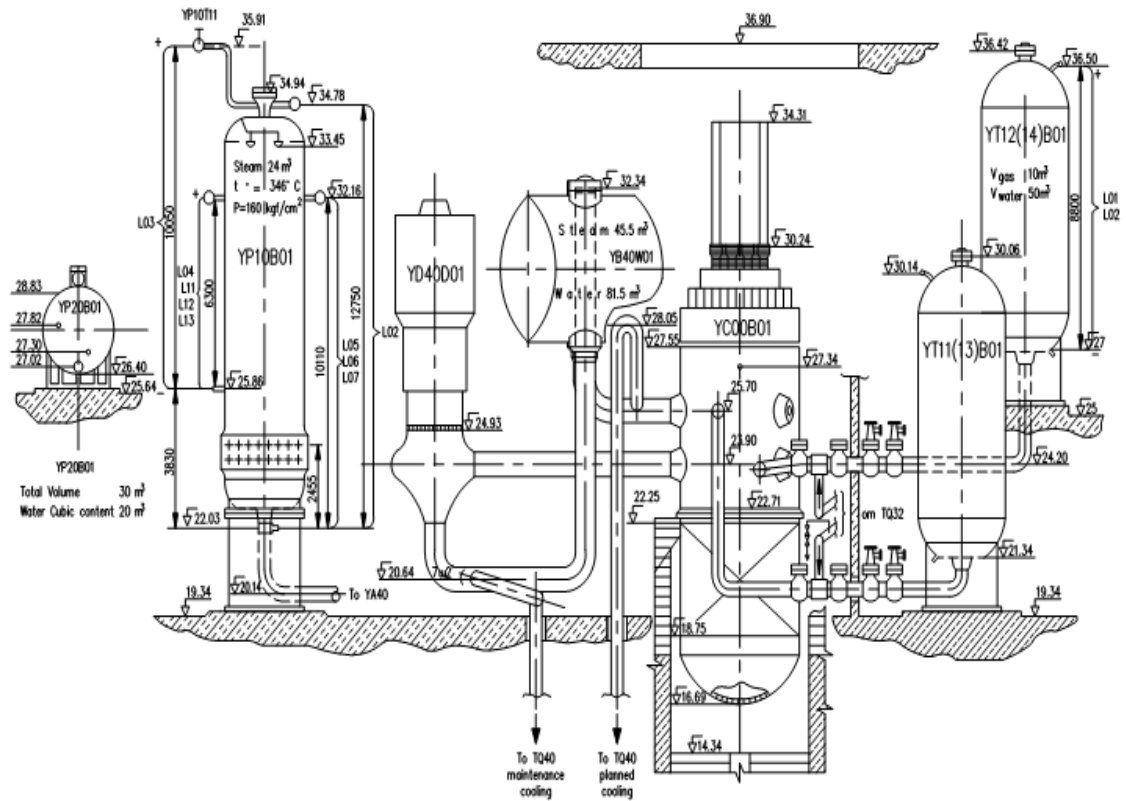


Figure 1. Primary circuit equipment of VVER 1000/v320.

of coolant event [10,14,16,18]. In the technical assignment for VVER-1000/320 reactor LB LOCAs cover the different ruptures of the primary pipelines with blowdown area above 0.00785 m² (equivalent diameter more than 100 mm).

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low pressure trip set point is reached. The safety injection system is actuated when subcooling margin trip set point is reached – $\Delta T_S < 10$ K (between saturated temperature in primary side and coolant temperature in hot leg). At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. The core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection are considered as core heat transfer mechanisms.

When the Reactor Coolant System (RCS) pressure falls below 10.78 MPa (110 kg/cm²), the HPPs begin to inject borated water in the cold legs.

The reactor coolant pumps are assumed to be tripped 0 sec.

Turbine stop valve isolates turbine 5 sec.

When the RCS pressure falls below 5.88 MPa (60 kg/cm²) all four accumulators start to inject borated water.

The LPPs will start to inject borated water after reaching set point of injection – 2.54 MPa (26 kg/cm²).

The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection lead to void formation in causing rapid reduction of the nuclear power to a residual level corresponding to the delayed fission product decay;
2. Injection of borated water ensures sufficient flooding of the core to prevent excessive temperature.

In case of simultaneously occurring of LB LOCA and loss of AC and DC power (station blackout) it will happen severe accident [3,5,7,8,11-13,18,19]. The restoration of at least one DG could prevent the progression of severe accident sequences if it happens at a certain time. Estimation of maximum possible time for operator reaction for a successful determination of severe accident scenarios and plant recovery is one of the main goals of the present work. For this purpose NPP strategies in SAMG are used.

For LB LOCA there is almost no time for operators to respond in the first few minutes of transient and in such situation actuation of passive system – hydro accumulators plays a crucial role of restricting possible damage.

3 Brief Description of ASTEC Input Model for VVER-1000 Reactor

For simulation of the described above initiating event it was used ASTEC integral computer code. The ASTEC computer code is used for severe accidents simulation and aims to predict the main parameters behavior of water cooled nuclear power plants during a severe accident [1, 4]. The code consists of different modules, which simulate particular physical phenomena.

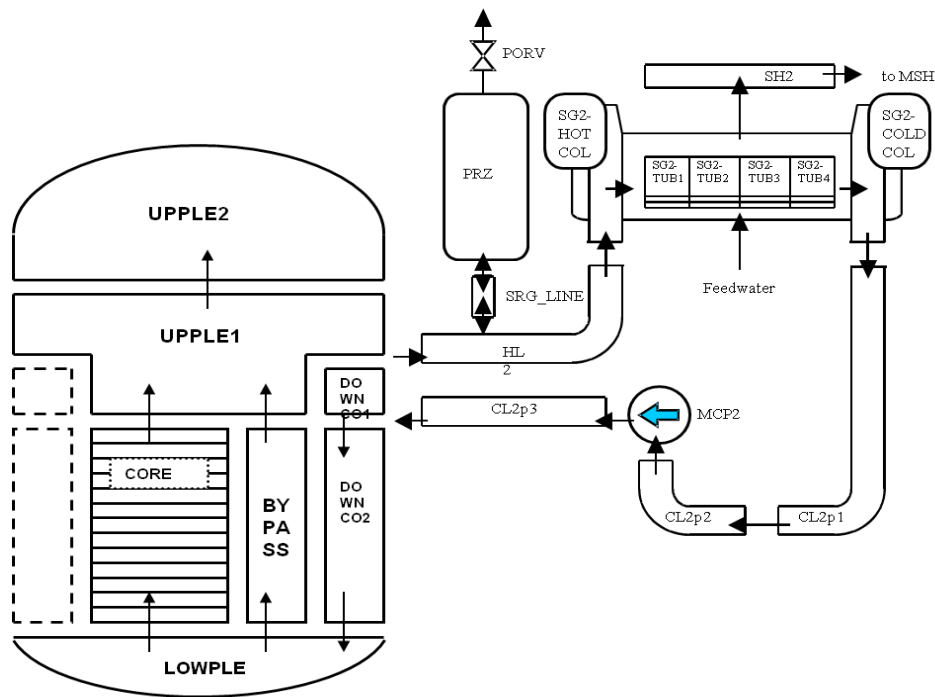


Figure 2. ASTEC input model nodalization scheme – reactor vessel and primary circuit.

An input model for VVER1000/v320 reactor [14] and suitable for ASTECv2r3p2 was developed by INRNE. It was presented on Figure 2 below the nodalization scheme of reactor vessel and primary circuit of the ASTEC input model. As an initiating event has been modeled LB LOCA located on the cold leg of the single loop after the MCP and with equivalent diameter 300 mm. Simultaneously with the LB LOCA it was simulated also Station blackout (SBO). Loss of all AC and DC power at the beginning of transient was assumed.

The modules CESAR, ICARE, SOPHAEROS and CPA of ASTEC computer code are used in a “coupled mode” for our calculation. The study is oriented just to the in-vessel phase of the accident and no other modules are involved.

The thermal-hydraulics in the primary and secondary sides and in the reactor vessel up to the start of core uncover is simulated by CESAR module.

The ICARE module simulates all the reactor vessel structures as well as: reactor core, baffle, the cylindrical part of the barrel, vessel cylindrical part, fuel assembly supports and vessel lower head. The main phenomena arising during the in-vessel degradation phases and the release of

core structural materials, including control rods are also simulated by ICARE module.

The fission products (FPs) and structural materials transport through the circuit are simulated by SOPHAEROS module.

All relevant processes in the containments of reactors are simulated by CPA module.

The VVER1000/v320 input model for ASTECv2r3p2 computer code includes the major components of the primary and secondary sides and the safety injection systems. One of the primary loops has been modeled as a single loop but the other three loops are modeled as one lumped loop. Each one of the loops is represented by hot and cold leg, SG hot collector, SG tubes, SG cold collector and MCP. The pressurizer is connected to the single loop.

The model of the reactor core area has been divided in 5 rings in radial direction and in 10 nodes in axial direction including the baffle and barrel. The pressurizer has been modeled by one volume (79 m³). Two accumulators (ACCU1&2) are connected to the upper plenum and two accumulators (ACCU3&4) are connected to the down-comer.

Table 1. Initial plant conditions

Parameters	Plant design initial parameters	ASTEC v2r3p2 steady-state parameters
Core power, MW	3000	3000
Primary pressure, MPa	15.7	15.713
Average coolant temperature at reactor outlet, K	593	593.56
Maximum coolant temperature at reactor inlet, K	563	563.37
Mass flow rate through one loop, kg/s	4400.0	4363.142
Pressure in SG, MPa	6.27	6.418
Pressure in main steam header (MSH), MPa	6.08	6.382
Steam mass flow rate through SG steam line, kg/s	408	409.613

The initial plant parameters are presented in the Table 1. The ASTEC v2r3p2 calculated steady-state values are compared with plant design initial parameters.

Decay power corresponds to the end of life.

Only one safety system is available and it is the high pressure pump – TQ13. The other pumps for safety injection in the primary circuit are TQ11 – spray pump and TQ12 – low pressure pump.

Operator actions start after reaching the set point of the core exit temperature. After reaching 923 K (650°C) at core exit the operator starts to inject water with TQ13. One more calculation is a study based on SAMG, when if the operator is missing first entrance to SAMG at 923 K will enter at 1253 K and starting pump (HPP) TQ13.

The signal for core exit temperature is calculated as a cladding temperature in the upper axial node of core region. The reason for using this node is that measurement points for core exit temperatures on NPP are located at the upper end of fuel assemblies.

4 Scenario for Investigated Accident

Initial conditions :

1. Reactor Power – 100%;
2. One safety system is available;
3. Cold Leg Rupture with ID 300 mm (LB LOCA) and simultaneous loss of all AC and DC power (station blackout).

Base case :

1. Without operator actions.

Operator action case :

1. Operator starts to inject water in primary circuit after reaching 923 K (650°C) at core exit (second run of calculation);
2. Operator starts to inject water in primary circuit after reaching 1253 K (980 °C) at core exit (third run of calculation).

Schedule of possible events (scenario) :

1. Due to fast primary pressure drop Reactor SCRAM is actuated after 0.4 s plus 1.2 s delay by “pressure ($P_1 < 14.7$ MPa) and power ($N > 75\%$)”. After 2–4 s all control rods drop to the core bottom;
2. Due to loss of all AC and DC power
 - (a) MCPs stop at 0.0 s;
 - (b) Make up system stops at 0.0 s;
 - (c) SG Feed water will be isolated at 5.0 s;
 - (d) Actuation of Reactor SCRAM due to loss of 3 out of 4 MCPs and reactor power $N > 75\%$ (duplicated signal to previous one of reactor SCRAM).
3. Due to primary pressure drop subcooling reaches $\Delta T_S < 10$ K;
4. Turbine is isolated 5 sec after the Reactor SCRAM;
5. Turbine bypass valves failed due to loss of vacuum and secondary pressure will be controlled by steam dump to atmosphere (BRU-A, Russian abbreviation) valves.

5 Main Outcomes from Code Application on Analytical Validation of Operator Actions during the Performance of SAMGs Strategies

The sequences of investigated scenario are presented in Table 2.

The behavior of most important parameters is given in Figure 3 through Figure 20.

The break flow rate is presented in Figures 8, 13 and 18. Coolant blowdown from reactor coolant system amounts more than 20000 kg/s at the moment of the break opening and decreases during the whole transient time. The reactor coolant system is emptying very rapidly during blowdown for the first 30 s as it is seen on Figure 3.

Coolant discharge leads to the immediate drop of primary pressure to the saturation point and then to the slower pressure decrease. Primary and secondary pressures are presented on Figures 3, 9 and 15.

Table 2. Sequence of main events

Events	time, s		
	Base case	Operator action at at 923 K	Operator action at at 1253 K
Cold leg rupture in loop #1	0.0	0.0	0.0
All MCPs are tripped	0.0	0.0	0.0
Reactor SCRAM	1.6	1.6	1.6
Turbine is isolated	5.0	5.0	5.0
Accumulators start to inject	86	86	86
Accumulators isolation	183	183	183
ICARE start	499	499	499
Beginning of oxidation	499	499	499
High pressure injection pumps start (TQ13)	no	660	862
First cladding rupture	679	682	679
Start of FP release	679	682	679
Melting pool formation in the core	3679	no	no
Reactor vessel bottom failure	8737	no	no

Base case results without operator action

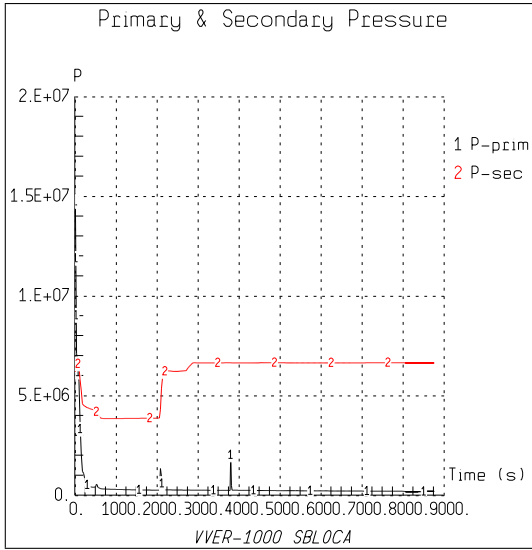


Figure 3. Primary and secondary pressure.

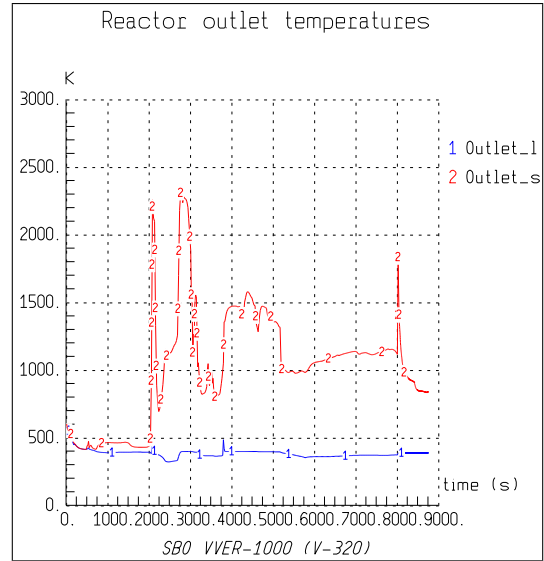


Figure 4. Reactor outlet temperature.

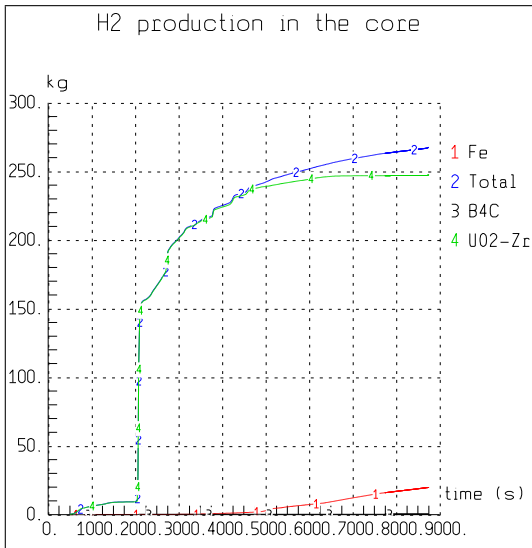


Figure 5. Hydrogen production in the core.

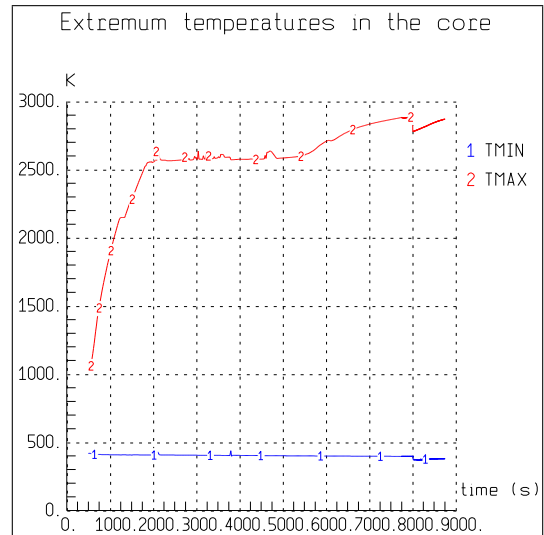


Figure 6. Minimal and maximal temperature in the core.

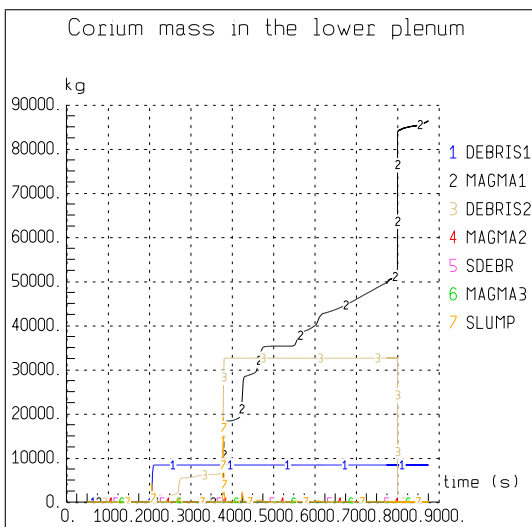


Figure 7. Corium mass into the lower plenum.

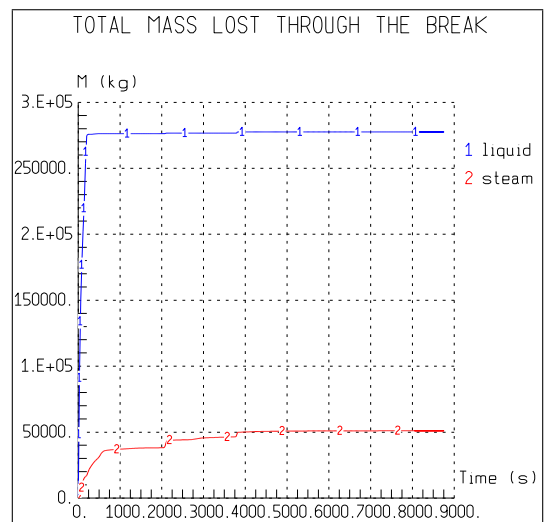


Figure 8. Total mass through the break.

Operator action results with HPP injection at 923 K

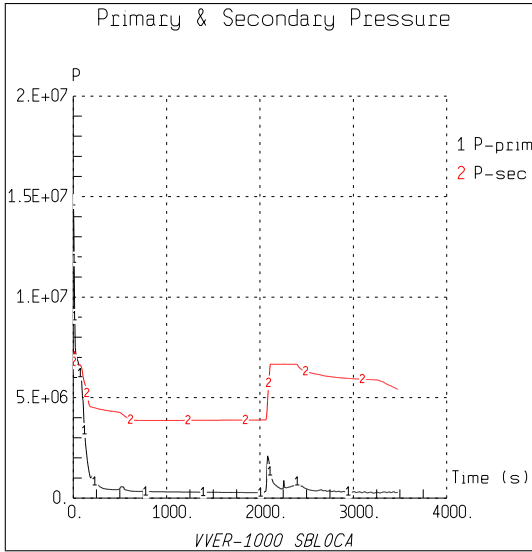


Figure 9. Primary and secondary pressure.

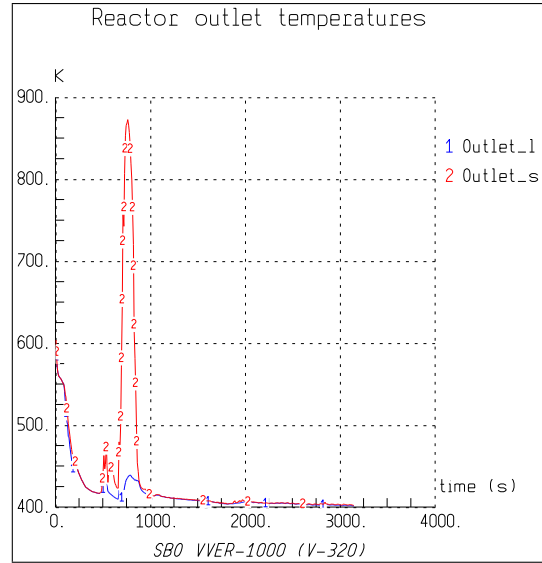


Figure 10. Reactor outlet temperature.

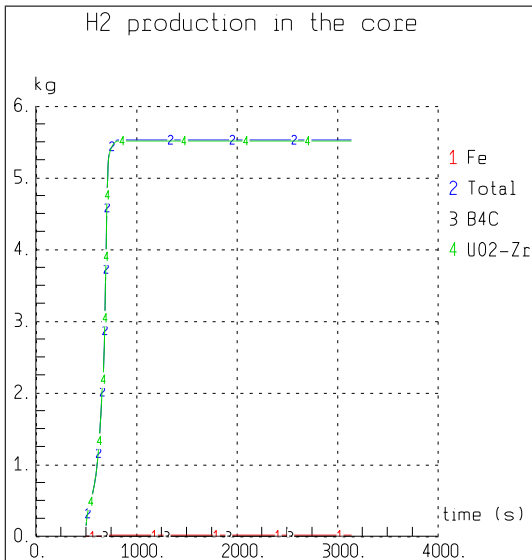


Figure 11. Hydrogen production in the core.

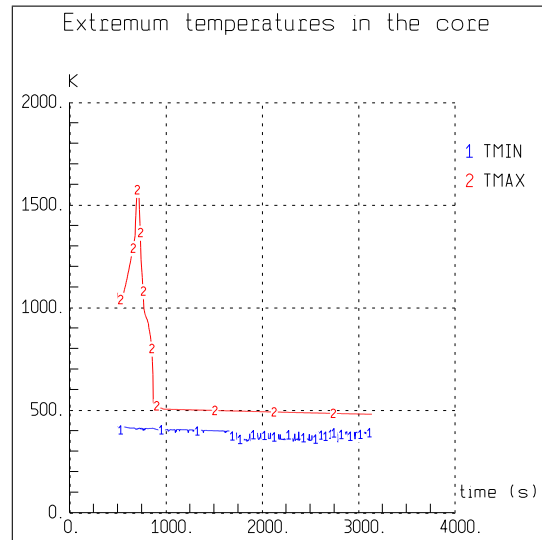


Figure 12. Minimal and maximal temperature in the core.

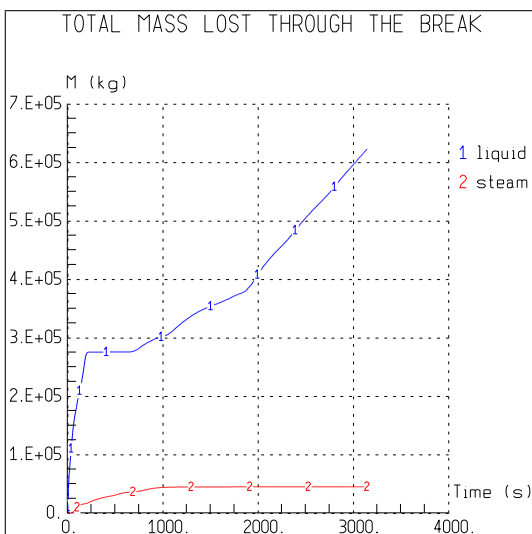


Figure 13. Total mass through the break.

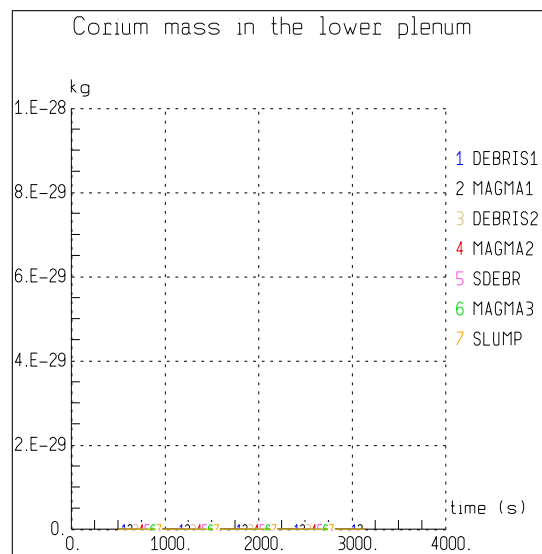


Figure 14. Corium mass into the lower plenum.

Operator action results with HPP injection at 1253 K

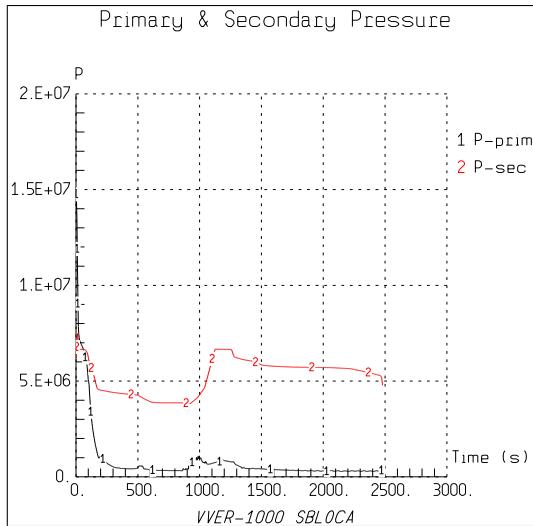


Figure 15. Primary and secondary pressure.

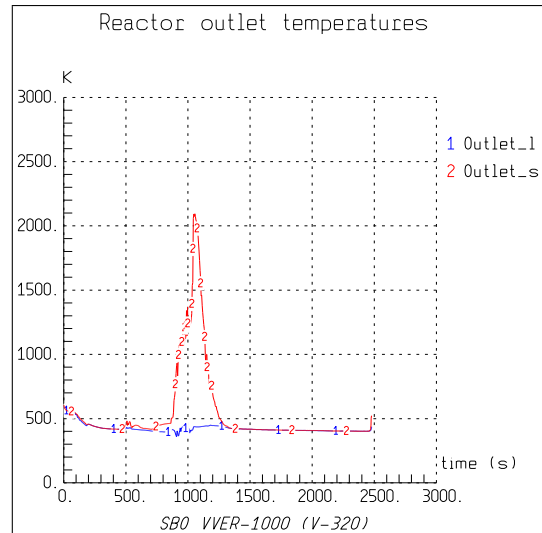


Figure 16. Reactor outlet temperature.

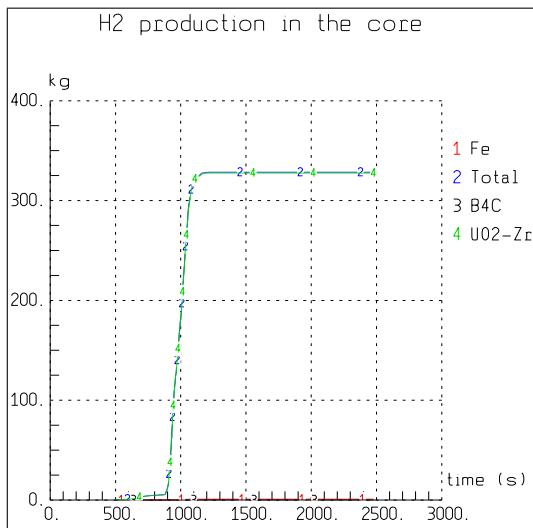


Figure 17. Hydrogen production in the core.

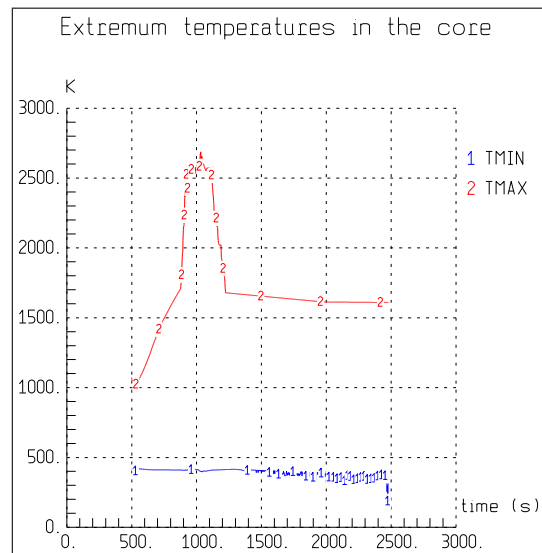


Figure 18. Minimal and maximal temperature in the core.

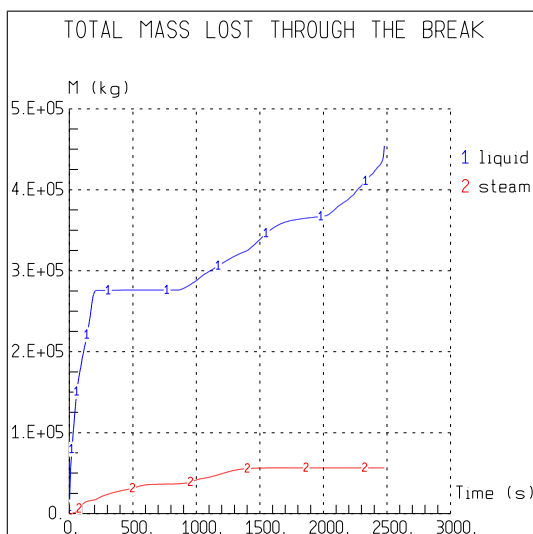


Figure 19. Total mass through the break.

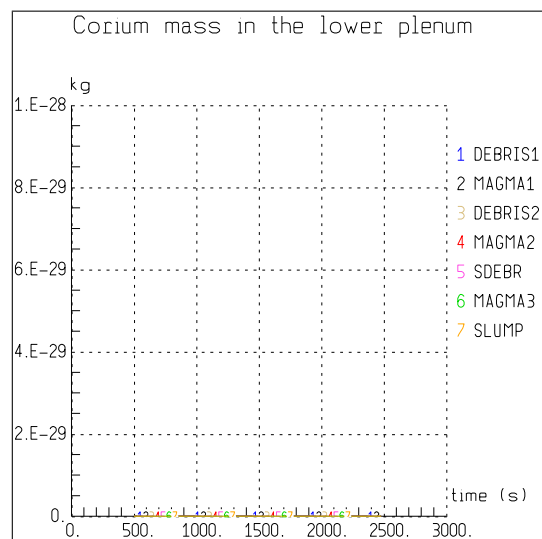


Figure 20. Corium mass into the lower plenum.

Due to fast primary pressure decrease, 0.4 s after the reactor pressure system (RPS) signal appears " $P_1 < 14.7$ MPa and $N > 75\%$ " occurs reactor scram and with a delay of 1.2 s control rods fall to the core bottom in 2–4 s.

Break flow becomes "unchoked" after 30 s.

The rapidly decreasing primary pressure and core coolant stagnation effect in the beginning of transient lead to deterioration of the core cooling.

All four MCPs were tripped at 0.0 s due to station blackout. Turbine Stop Valve (TSV) tripped and isolated Turbine in 5 sec. after the SCRAM. Hydro accumulators (HA) start to inject water into downcomer and upper volume of reactor vessel after reaching 5.88 MPa and do not allow earlier heat up of reactor core above is 1473 K. Due to fast depressurization HA stops to inject due to fast depletion of water. High pressure pump (HPP – TQ13) starts to inject after reaching of 923 K in first calculation with operator actions and at 1253 K in second calculation with operator actions into the core, based on SAMG.

The extremal temperatures in the core – minimal and maximal are given on Figures 6, 12 and 17. As it is seen from the Figures the reached maximal core temperature in the case without operator case is 2800 K, in the case with start of HPP at 923 K, the reached maximal temperature is 1600 K due to early injection, while in case with HPP injection at 1253 K the maximal temperature is 2600 K.

The predicted corium mass into the lower plenum of the reactor vessel is presented on Figure 7 for a case without operator action. The beginning of melting pool formation is observed after 3679 s. The total mass of melt formation on the reactor vessel bottom is 85 000 kg. In both cases with operator action the core is cooled down successfully and there is no melting formation into the core.

The predicted total amount of hydrogen generation during the transient is presented on Figures 5, 11 and 16. In case without operator action the total amount of hydrogen is approximately 260 kg. In the case with HPP injection at 923 K the hydrogen is only 5.5 kg. In the case with HPP injection at 923 K the total amount is approximately 5.5 kg, while in the case with HPP injection at 1253 K the total amount is higher 320 kg due to later cool down of reactor core.

The failure of reactor vessel is observed only in the case without operator actions, which is observed at 8737 s.

6 Conclusions

The performed calculation shows that in case without operator actions the reactor core will melt with further failure of reactor vessel. In this way selected initiating events lead to severe accident conditions.

The additional two calculations with involving operator actions based on SAMG strategies, shows that the operator can stop progression of core melt in both cases. There is no molten pool formation.

In both calculations with operator actions for core exit temperature was used temperature at upper part of struc-

tures in core region. As it is seen there are significant deviation in prediction of max core temperatures in both cases with operator actions nevertheless that the delay of second operator action (at 1253 K) is 200 s compare to the first one at 923 K. Using the core exit temperature calculated in the volume above reactor core region is not appropriate and is not correct.

Hydrogen production in operator action at 923 K was negligible, while at 1253 K hydrogen generation was more than without operator action.

Acknowledgments

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