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Simulation of SB-LOCA of typical PWR with MELCOR code

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Abstract. Safety of Nuclear Power Plants (NPPs) is one of the main issues in nuclear industry mainly because of the probability of radionuclides release to the environment. Due to this fact nuclear safety is continuously being improved by vendors to face the growing demands especially after the accident in Fukushima. This accident because of its character (station blackout scenario) resulted in putting more emphasis than ever on passive safety systems in NPPs. Such systems are one of main focus points in presented simulation. One of the ways to improve NPPs' safety and check its robustness is to proceed simulations of different events and find the weakest points inside the system. First thing is to create simulation model of NPP which has to be validated in order to prove reliability of future results. This paper presents simulation run of design basis accident for large PWR reactor based on AP1000. Model was developed using publicly available data only, which means that it should be considered only as "AP1000-like model". MELCOR code which was used in presented calculations is a severe accident code which simulates core melting during the worst possible accidents in NPP. Here it was decided to simulate design basis accident namely small break loss of coolant accident (SB-LOCA) and compare obtained results with those from the safety reports. Main motivation to create such simulation was to increase authors' knowledge about MELCOR behaviour and sensitivity in case of thermal hydraulic response to learn what kind of errors and uncertainty could be expect. Second motivation was to check if Melcor is capable to simulate design basis accident with acceptable results. Finally the obtained results are in relatively good agreement with the expectations and some differences that have appeared do not affect general course of the accident. Shape of transient curves and timing of events are reasonable correct.

1. Introduction

In case of nuclear safety investigation lots of variables and points of view have to be taken into account to be as close as it is possible to realistic solutions. One of the most dangerous scenarios for nuclear power plant is one with core degradation, called severe accident. To project management of this kind of accident, mitigation or avoid it is necessary to simulate possible scenarios and observe what could be improved. This simulations are conducted using the code which is developed by Sandia National Laboratory for Nuclear Regulatory Commission in the US. System, severe accident codes like this can simulate and in relatively short time give results to the user however main part of this code is to investigate melting phenomena of the core and provide source term during accident. Thermal hydraulic answer due to fact that it is severe accident code is not precise and usually vary by user to user and it could give slightly different results then thermal hydraulic (TH) or system codes e.g. RELAP5. For proper simulation of Severe Accident (SA) by MELCOR[1,2] is necessary to understand and to know how the investigated system should behave and what uncertainties are expected. To increase knowledge about MELCOR behaviour in this part of accident introduced simulation design basis type Small Break Lost Of Coolant Accident SBLOCA. Such investigation gives a possibility to observe how MELCOR responds for such relatively small accident comparing to severe accidents like large break LOCA, what was the main motivation. Second one is to in some way, validate presented model for this kind of transient.

2. Model description

Model as was mentioned before is developed in MELCOR code [1,2] which is used worldwide for severe accident simulation in NPPs. Whole model was created from the scratch using publicly available data based on AP1000 project, due to this fact lots of assumptions had to be done and model should be considered as a "AP1000 like model" and it is presented as typical PWR NPP. Model itself contains 215 control volumes, 123 heat structures, 320 flow paths. All instrumentation and controls are modelled by 762 control systems in accordance to the documentation used during model development which was DCD [3]. As presented in figure 1, model has relatively dense nodalization. The Design Basis Accident (DBA) such as SB-LOCA propagates slowly comparing to more severe transients and it this case more precise nodalization can be helpful to observe the predicted phenomena. Model also contains containment and related systems: passive air cooling of primary containment and Passive Containment Cooling System (PCCS). All signals, delays and valves opening times are modelled in consistence with data from DCD [3,4]. Pumps characteristics was implemented by tabular function dependent on the pressure also based on DCD data [3]. Core is modelled as 17 axial and 6 radial regions from which 11 axial and 5 radial are active, rest are for down comer or eventual core relocation after degradation. Visualization was developed using graphical interface for MELCOR called SNAP.

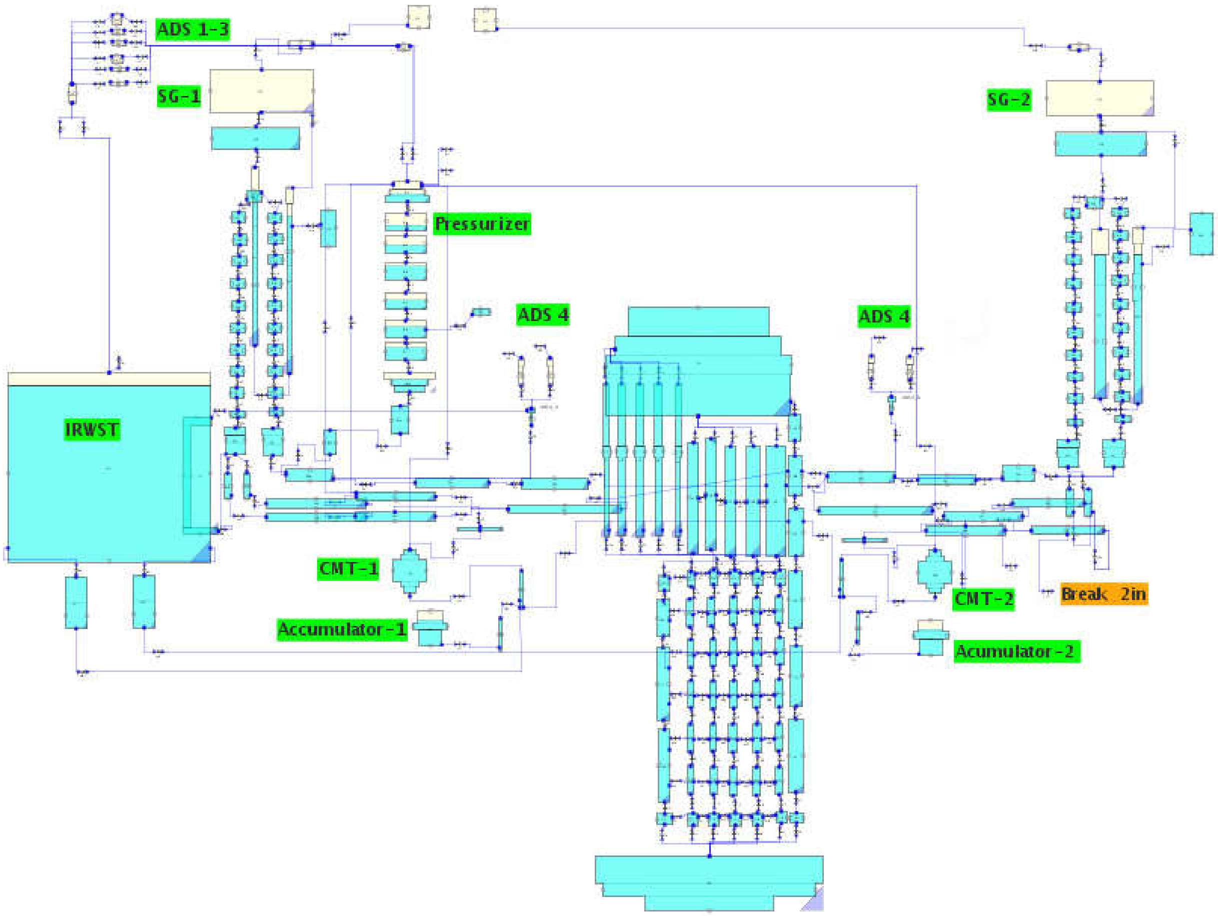

Figure 1. Nodalization of typical PWR in MELCOR code, developed by SNAP.

2.1. Steady state

Steady state was calculated for 1000s before time 0.0 s when accident begins. Values of variables of steady state are presented in table 1 where they are also compared to the reference data. As it is shown values are close to reference, with few percent differences, what allows to assume that initial conditions for accident simulations are good and comparable with those from references. Only mass flow through the core is higher by 5% than acceptable range.

Table 1. Steady state condition.

	Parameter	Reference value [3]	Calculated value	Acceptable error [3]	error
	Pressure [MPa]			+/- 0.345 [MPa]	
1	Pressurizer pressure	15.85 abs	15.68		0.17
2	SG outlet pressure	5.43 abs	5.65		0.22
	Temperature [K]			+/- 3.5 [K]	
3	RPV inlet	552.2	551.9		0.3
4	RPV Outlet	595.87	593.4		2.47
5	Feedwater	499		-	
	Coolant flow [kg/s]			+/- 2 %	
6	Reactor flow	14101.37	15053		7.26% 951.63
7	Bypass flow	705	651		7.66% 54
	Accident scenario assumptions [3]				
8	Power	102%			
9	SG tubes	10% plugged			
10	Containment pressure	0.104 Mpa			
11	Containment temp	330 K			
12	Decay heat	120% ans 71 standard			
13	Single failure criteria	CMT connected to broken line is separated from system			

2.2. Scenario description
Scenario which authors decided to choose is SBLOCA. It is partial rupture with diameter 0.05 m of one out of four cold legs (connected with Core Makeup Tanks - CMTs). As one of the main DBA scenarios considered during safety analysis, it engages all safety systems in relatively long time and in some way it allows to validate presented model from thermal hydraulic point of view. Reactor trip is actuated by low pressurizer pressure signal. Then reactor power decreases in consistence with decay heat curve applied by tabular function. Values of decay heat are established by using ANS-71 standard and increased by 20% as conservative approach. Authors decided to follow conservative approach from DCD during simulation. Due to mentioned fact that model was developed from publicly available data some assumptions had to be done. Therefore the conservative approach should in some way decrease impact of lack of data and also[3,5].

3. Results and discussion
In table 2, the obtained timing of events during accident transient is compared to the references. As is clearly shown, the MELCOR results in general are in good agreement with references. Differences are expected and could be caused by numerical differences, nodalization or user influence. Of course MELCOR code itself could bring such differences because usually it is used to more rapid and severe transients and computational models used in the code are mostly devoted to severe and not design basis accidents. First difference which creates delay in whole transient is signal of low pressure in pressurizer, which is actuation signal for reactor trip. In MELCOR calculation sit appears at 80.5s

while in the reference at 54.7s. This delay could be caused by voiding effect in hot area of the core. In early stage of accident vapor appears into the hot channel and keeps the pressure high which causes the delay in low pressurizer pressure signal. Probably this could be solved by changing power profiles or nodalization. Such differences also could escalate peak of injection which occurs at the beginning of CMT injection phase (figure 2). The reason of such delay could be also not precise enough pressurizer or surge line model. This problem will be investigated during next simulations and some uncertainties analysis will be done. Another significant differences are related to the accumulator start and stop injections. It could be connected the reactor trip delay mentioned but another reason could be the Direct Vessel Injection (DVI) line pressure losses and piping geometry. Pressure losses in DVI also could be a reason why CMTs are not emptied even at the end of transient. So problem with pressure losses and geometry seems to be the most probable reason for it and it will be the case of further investigations.

Table 2. Timing of events during accident transient.

	Event	Reference time [s] [3]	Calculated time [s]	Differences [s]
1	Break Opens	0	0	0
2	Reactor tripped	54.7	80.5	25.8
3	Steam turbine stop valve closes	60.7	83.5	22.8
4	“S” signal	61.9	83.8	21.9
5	Main feed isolation valves begin to close	63.9	87.8	23.9
6	RC pumps tripped	67.9	89.8	21.9
7	ADS Stage 1	1334.1	1216.8	-117.3
8	ADS Stage 2	1404.1	1264.8	-139.3
9	Accumulators injection starts	1405	1304.6	-100.4
10	ADS Stage 3	1524.1	1384.9	139.2
11	Accumulators emptied	1940.2	1709.5	-230,7
12	ADS Stage 4	2418.6	2573.65	155.05
13	CMT emptied	2895	-	
14	IRWST injection starts	3280	2574.8	-705.2

Next part of analysis is a comparison of main plant parameters during the transient with reference data from DCD [3]. Figure 2 shows injections of CMT1 (left) and CMT2 (right). Beginning of recirculation phase in both cases is very close to references but contains peak, probably caused by water hammer. Second phase injections also have good agreement with reference however in case of CMT 1 it starts earlier than expected and in both cases the injection stops around 100s earlier than expected. Last phase of CMT injections is in good agreement in case of amount of water injected but timing differs by around 200 s what is caused again by accumulator behaviour.

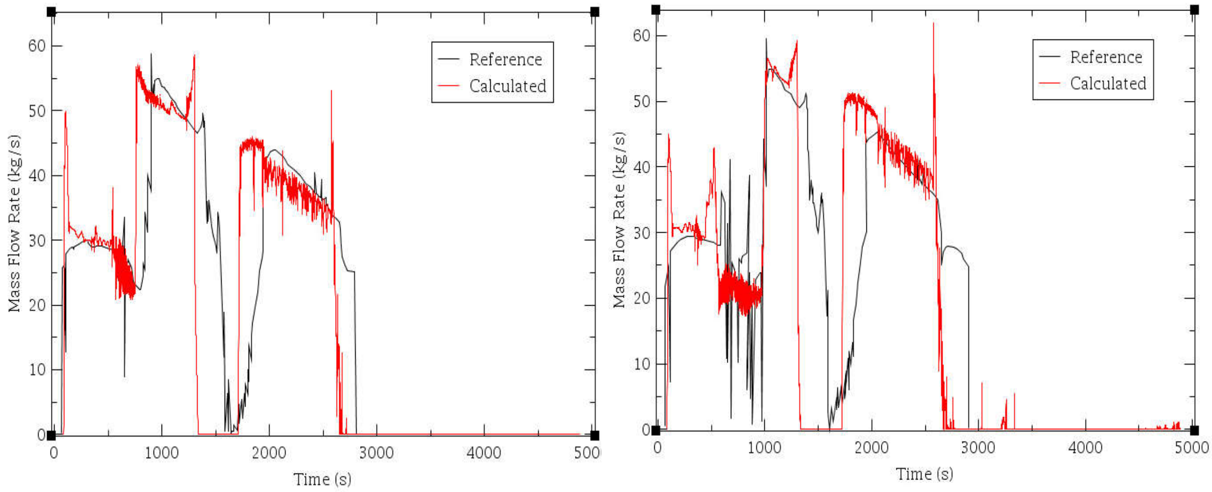

Figure 2. CMT injection during accident transient. CMT1- left, CMT2-right.

Figure 3 presents flow of coolant through break (left) and injection of accumulator (right). Results for only one accumulator are shown because second one is working almost identically and the differences are negligible. Break flow is generally very close to references with the same shape and values of flow except part between 800-1100 s when mass flow in presented calculations is visibly lower than expected. In case of accumulator injection it starts around 100s before references and mass flow is higher than expected. It could by caused by common cause for example mentioned pressure losses in DVI.

Figure 4 shows pressure changes in reactor coolant system (RSC) during transient (left) measured in pressurizer and inventory of RCS (right). Pressurizer parameters have good agreement with references, relatively small variations observed are acceptable in such simulations of design basis accident by severe accident code. Similar conclusion regards RCS inventory, for which the general shape of transient is very close to reference and variations are at the acceptable level.

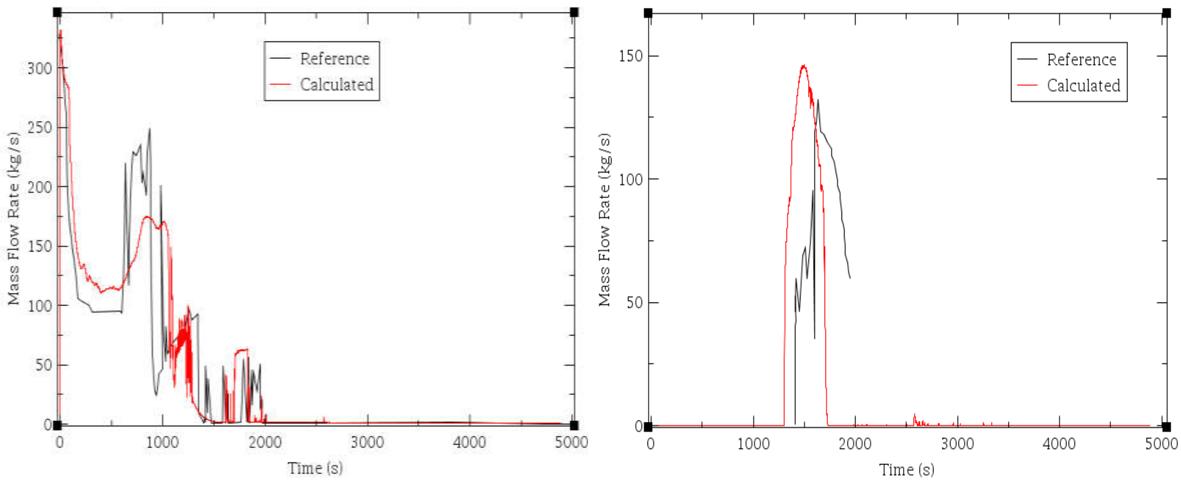

Figure 3. Break flow (left) and accumulator injection (right).

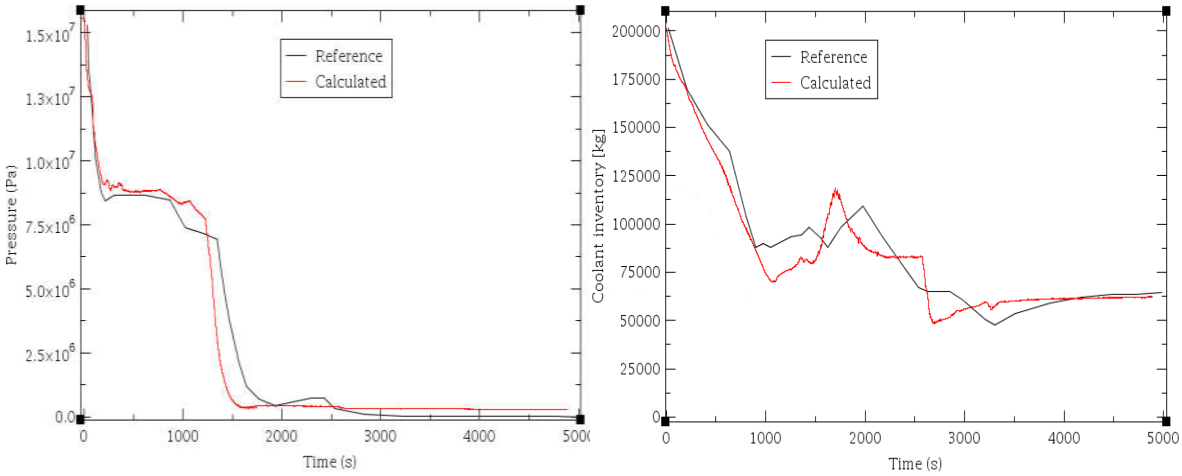

Figure 4. Pressure transient (left) RCS inventory transient (right).

4. Conclusions

Paper describes MELCOR model of large PWR with steady state parameters and DBA scenario simulation. Presented simulation is one of many conducted in order to increase knowledge about MELCOR thermal hydraulic response to Design Basis Accident such as SB-LOCA. Obtained results are generally consistent with the references. As it was shown, differences are visible and sometimes have relatively big influence for whole transient progress. For 5000s transient differences in timing of events are up to few hundreds seconds (in case of accumulator). It is acceptable but for sure needs further investigations. Similar conclusion regards the transient curves. They are comparable with reference and variations are relatively small taking into account that MELCOR code was developed for much more rapid transients and also the fact that some significant information and data on the reactor design was not available for the authors. The paper clearly shows that even without very detailed information about investigated system, MELCOR could give good results and simulate thermal hydraulic response of design basis accident with acceptable error.

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Disclaimer

This article presents the work performed at the *AGH University of Science and Technology in Cracow, Poland* as an academic exercise based on publicly available data, hence Westinghouse Electric Company cannot be held responsible for the details reported.

References

[1] Sandia National Laboratories Albuquerque, NM 87185-0748 *MELCOR Computer Code Manuals Vol. 1: Primer and Users' Guide Version 2.1*, September 2011
[2] Sandia National Laboratories Albuquerque, NM 87185-0748 *MELCOR Computer Code Manuals Vol. 2: Reference Manual Version 2.1*, September 2011
[3] AP1000 European Design Control Document, http://www.westinghousenuclear.com/uknuclear/Documentation (access20.02.2017)
[4] T. Schulz, *Westinghouse AP1000 advanced passive plant,* Nuclear Engineering and Design, p1547-1557, August 2006.
[5] IAEA safety guide, *Deterministic Safety Analysis for Nuclear Power Plants for protecting people and the environment No. SSG-2 Specific Safety Guide*, Vienna, 2009.