Influence of main variables modifications on accident transient based on AP1000-like MELCOR model

M. Malicki  
AGH University of Science and Technology, Faculty of Energy and Fuels,  
Mickiewicza 30, 30-059 Krakow, Poland  
E-mail: malicki@agh.edu.pl

L. Pieńkowski  
AGH University of Science and Technology, Faculty of Energy and Fuels,  
Mickiewicza 30, 30-059 Krakow, Poland  
E-mail: pienkows@agh.edu.pl

Abstract. Analysis of Severe Accidents (SA) is one of the most important parts of nuclear safety researches. MELCOR is a validated system code for severe accident analysis and as such it was used to obtain presented results. Analysed AP1000 model is based on publicly available data only. Sensitivity analysis was done for the main variables of primary reactor coolant system to find their influence on accident transient. This kind of analysis helps to find weak points of reactor design and the model itself. Performed analysis is a base for creation of Small Modular Reactor (SMR) generic model which will be the next step of the investigation aiming to estimate safety level of different reactors. Results clearly help to establish a range of boundary conditions for main the variables in future SMR model.

1. Introduction

Current situation on energy market corresponds to decreasing usage of coal and therefore forces us to develop new kinds of energy sources. Increasing the amount of energy coming from nuclear power plants could be one of possible solutions for this challenge. Small Modular Reactors (SMR) are a concept which gives an opportunity to produce relatively cheap, clean and safe energy even in remote areas where conventional ways to deliver or produce energy are problematic. In brief SMRs are nuclear reactors with power up to 300 MWe and modular design which can be produced in a factory and built from the parts on site. They are becoming more popular nowadays and the industry and R&D are showing their interest. Unfortunately SMRs in civil usage are new idea and before they can be used in such way it is necessary to prove their safety and reliability. First step could be to compare this new approach with classical well known solution, namely a full power PWR. Simulation of classical nuclear power plant using system code could bring us closer to that point what is the reason why we decide to prepare such work. This kind of analysis is usually conducted for licensing process or wide safety study. In this paper brief sensitivity analysis of classical AP1000-like model is presented to get some knowledge about behavior of well-known nuclear system during accident with different initial
conditions. Due to the fact that authors have conducted such simulations for the first time only it is a very general study and only few cases are taken into account.

2. Problem description

To prepare proper model with any system code it is necessary to have all the data required for the model. Unfortunately in the nuclear industry and especially for new concepts such as SMRs, most of the details are very difficult to find. Due to this fact many assumptions and simplifications are unavoidable. To decrease potential error resulting from this problem in the future SMR MELCOR model it was decided to conduct first calculation on much better known type of reactor. PWR model based on AP1000 public available data only was created. During first calculation was notice the problem with adjustment few variable like value of coefficient of local pressure losses. To check the influence of those few main variables on accident transient this paper was prepare. It should be noted how the effect of scale and modifications of different variables influence such accident simulation. Variables chosen for the analysis are pressure drop in Reactor Pressure Vessel (RPV), pressure losses in Direct Vessel Injection (DVI) and Core Makeup Tanks (CMT) valve opening time. The ranges of these variables are shown in the table 1, below. Three cases were considered.

| variables                        | minimum | maximum | reference value |
|----------------------------------|---------|---------|-----------------|
| Pressure drop in RPV [MPa]       | 0.23    | 0.55    | 0.43            |
| DVI pressure losses [%]          | -50     | +50     | 100             |
| CMT opening time [s]             | 20      | 40      | 31              |

3. Simulation model

The model was created in MELCOR code. MELCOR is a system code for nuclear safety analysis developed for U.S. NRC. It is used mainly for Beyond Design Basis Accidents (BDBA) to analyze core melt in case of severe accident within a frame of simulation model. It allows us to simulate propagation of accident scenario in whole Nuclear Power Plant (NPP) and finally to calculate the source term. Usually it is used for part of Probabilistic Safety Assessment (PSA) as a deterministic thermo-hydraulic answer of NPP for given accident scenario to predict weak and strong points in terms of safety [1]. However, also for research study on severe accident transient during blackout or with core melt. [2] [3]

Model introduced here was created entirely by the authors and is based only on public available data on AP1000 reactor. Therefore it can be considered as an AP1000-like model and results may differ from the reference for precise AP1000 model. However, as shown further, it was possible to create relatively reliable model to compare results of reference scenario of Large Break Lost Of Coolant Accident (LBLOCA) with original analysis for AP1000 [4]. The emphasis is put on the thermo-hydraulic responses and differences between them, thought release of radionuclides is not taken into account. Whole model consists of 211 control volumes, 108 heat structures which imitating structures for heat conductivity or capacity like steel pipes or concrete, 311 flow paths which are used for heat and mass transport, 483 control systems used mainly for automatic of safety system logic and core package. Reactor core model consists of 17 axial levels and 6 radial rings. All these elements represent nodes and paths within simulation model created accordingly to MELCOR code rules[5] [6].

Nodalization showed in fig. 1 includes: primary loop, secondary loop, CMT, accumulators, IRWST-In-Containment Refueling Storage Tank, PRHR-Passive Residual Heat Removal, ADS-Automatic Depressurization System level 1-4. Nodalization of the containment and related safety systems is incorporated into the model, but they are not presented in fig. 1.
4. Steady State condition

Steady state scenario for LBLOCA is taken from AP1000 European Design Control document, as well as most of data for the model [4]. Some conditions for the NPP with the analyzed reactor under normal operation are presented in the table below. Acceptable error range is also shown together with the actual calculated errors. In all the cases the uncertainties are at the acceptable level.

Table 2. Accident steady state with acceptable errors

| Variable                     | Reference value [4] | Calculated value | Acceptable error [4] | error |
|------------------------------|---------------------|------------------|----------------------|-------|
| Pressure [MPa]               |                     |                  |                      |       |
| In the pressurizer [MPa]     | 15.513              | 15.515           | +0.003               |       |
| Outlet steam [MPa]           | 5.531               | 5.536            | +0.05                |       |
| Temperature [K]              |                     |                  |                      | +/- 3.5 |
| RPV inlet [K]                | 552.89              | 551.762          | -1.125               |       |
| RPV outlet (assumed) [K]     | 595                 | 593.699          | -1.301               |       |
| Feedwater [K]                | 499                 | 499              |                      |       |
| Coolant flow                 |                     |                  |                      | +/- 2% |
| Hot leg [m³/s]               | 9.514               | 9.596            | 0.19                 | +0.082 |
| Reactor core (assumed) [m³/s]| 14.100              | 13.904           | 0.282                | -0.196 |
| Bypass (assumed 5%) [m³/s]   | 0.705               | 0.703            | 0.014                | -0.002 |
| SG tube plugging [%]         | 10                  |                  |                      |       |
There are usually few steady states considered for each NPP depending on the situation, e.g. nominal steady state, accident steady state for LBLOCA or accident steady state for steam generator tube rapture (SGTR). The differences between them refer mainly to pressure or choked part of Steam generator U-tubes. Steady state was simulated for 900 s each time.

5. Reference scenario

LBLOCA is a main accident scenario in such analysis. It is guillotine rapture of cold leg at a given accident steady state condition mentioned before. This was chosen mainly because LBLOCA is proceeded rapidly and it is possible to see the differences between investigated scenarios so it is accepted for presented simulation.

Accident calculation conditions used for this scenario are chosen based on AP1000 European Design Control Document LBLOCA scenario [2]. Results of calculations are compared with those from documentation described on figures 2-4 as a reference one. As it is shown AP1000-like model gives results on acceptable level of compliance corresponding to reference data. Differences between calculations and references most probably come from inaccessibility of some data and resulting assumptions in model development. In figure 2, CMT-1 is one on intact cold leg, CMT-2 is one on broken cold leg. The biggest differences was found in CMT injection especially in final phase of simulation. Our simulation shows injection only at the beginning of accident scenario; the second injection after 140 s does not occur and this is probably caused by lack of information about DVI pressure losses and further investigation was performed.

Figure 2. Comparison of calculated CMT injection with reference one for LBLOCA

Figure 3. Comparison of calculated accumulators injected with reference one for LBLOCA.

Figure 4. Comparison of calculated break outflow with reference one for LBLOCA.
6. Calculations

During model development one of main problems was to estimate pressure losses especially at some parts of piping related to safety systems such as DVI but also inside RPV described on figure 1. Presented calculations show how the changes in this variables influence outflow form accumulators and CMTs. The differences of temperature transient in cells with the highest power ratio were also investigated as these cells were most vulnerable to degeneration.

6.1 First simulation
First simulation was made for different RPV pressure drop. Changing this variable was established by decreasing and increasing coefficient of local pressure losses on inlet flow paths from cold legs to RPV. Three cases were examined:

1) coefficient of local pressure losses set on 0,
2) coefficient of local pressure losses set on 1.5 (reference value),
3) coefficient of local pressure losses set on 3.

![Figure 5. CMT injection for different sets of local pressure losses.](image1)

![Figure 6. Accumulators injection for different sets of local pressure losses.](image2)

![Figure 7. Temperature transient for different sets of local pressure losses.](image3)

From this calculation it is possible to conclude that relatively big changes in pressure drop through RPV visibly affect temperature transient for the case without pressure losses, figure 7. However, for small pressure drop temperature is much bellow safety margin. For the highest range of pressure drop, 1.5 or 3 for local losses at investigated flow paths almost no changes are visible. It is important to notice that such high temperature, above 2000 K should not appear in this transient. Corresponding to that is known that more precise investigation need to be done, especially for flow through the core region.
Figure 5 shows little influence on presented accident transient to accumulators injection, only at the beginning of transient some differences appear. Variations in RPV pressure drop have also small but visible influence on CMT injection. Case without pressure loses causes bigger differences especially at the beginning of injection but case with higher local pressure losses change injection in second part of accident transient.

6.2 Second simulation
Second simulation was made for different opening time of CMT valve between CMT and DVI volumes described on figure 1. Three cases were considered:

1) opening time 20s,
2) opening time 30 s (reference),
3) opening time 40s.

![Figure 8. CMT injection for different opening time of CMT valve.](image)

![Figure 9. Accumulator injection for different opening time of CMT valve.](image)

![Figure 10. Temperature transient for different opening time of CMT valve.](image)

Changes in opening time of CMT valve resulted in hardly any differences in temperature transient, figure 10. For more precise investigation in this case another hot cell from the very inside region of the core with high power rate was added. However the results were similar and no visible changes were observed. Accumulator, figure 9, caused practically no differences between investigated cases. Only at the beginning of CMT injection it is possible to notice differences between the cases with no significant changes for the second part of the simulation figure 8. However in both cases some small peaks at the end of accident time
are observed what was not present in our reference calculation.

6.3 Last simulation was done for changes in local pressure losses on DVI line. DVI line was simulated as two control volumes and two flow paths for each line described on figure 1. Changes were introduced on paths between those two volumes. Reference value assumed for this calculation was 3.4 and the range of changes was as follows:

1) 50% decrease in local pressure losses,
2) reference coefficient of local pressure losses equal to 3.4,
3) 50% increase in local pressure losses.

Changes in DVI local pressure losses influence safety systems, CMT and Accumulator injections. However the biggest differences are for CMT injections and it is clearly showed in figure 11 especially in area between 140 and 200 s of the accident. This shows that local pressure losses in DVI line have big influence on injection of CMT and the same should be expected also for further accident transient. Accumulators injection are not so much affected by these changes, some effects are visible only at the beginning and at the end of presented transient. In the second part of accumulators working area is possibility to see clearly different time of injection ending, figure 12. From this calculation we see that for increasing DVI coefficient of local pressure losses it is possible to achieve better adjustment of accumulators injection however for better adjustment of CMT injection should be considered decreasing those losses it bring little confusion into results. Due to this fact to get perfect adjustment of calculation in safety systems injection it is needed to prepare more precise investigation.

7. Conclusion

This paper shows the first approach to simulation of accident for AP1000-like model using MELCOR code. Comparison of the results with reference values from AP1000 European Design Control Document shows relatively good agreement.

The main task was to validate AP1000-like model, investigate influence of few variables on accident transient basing on changes in behavior of safety system injections and temperature transient inside the core.
Calculation shows that passive safety systems are relatively sensitive for changes in local pressure losses changes on DVI line and even small variations of those values can lead to significant differences in model behavior during accident. Changes in opening time of CMT valves do not have so considerable influence on presented simulation scenario. However, pressure drop in RPV has visible influence of accident transient especially for temperature of cladding what may change whole accident transient. For further investigation if precise data are not available, brief sensitivity analysis should be done to get the most correct value.

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