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In-Vessel Melt Retention with External Cooling For Vver-1000

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ABSTRACT
In order to identify the sensitivity of the BH model for Melcor 1.8.5 code a comparative analysis of the two strategies was conducted: by using deflector and without it. It allowed estimating the influence of using the deflector on final results. We also compared the results of calculation with those using Melcor 1.8.6 code (Czech Republic) to assess the results reliability and carry out the validation of the BH model for Melcor 1.8.5.
KEYWORDS
Melcor, melt, debris, reactor vessel, reactor cavity, water supply.

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I. INTRODUCTION

Over the past two decades, there have been qualitative changes in the safety requirements of nuclear power plants. During this period, new documents of IAEA on NPPs safety [1, 2] and three versions of the requirements of European manufacturers were published [3].

After the Chernobyl disaster in 1986 and Fukushima in 2011, the issue of controlling the atom splitting became in doubt. It was the problem of losing control over the fission reaction in the rector that became the main issue for developers of a new technology that could operate in conditions of severe accidents with a complete loss of power supply. This technology was labelled as passive.

The in-vessel coolability and retention is based on the idea of external cooling of the reactor lower plenum from the water source. This concept is used at Loviisa VVER- 440 [4] in Finland, where it was approved by the regulatory authority of STUK.

Recently, the IVMR concept of was adopted on all VVER-440 units operating in Central Europe. The concept is also used in GEN-III PWR designs: AP-600, AP-1000 [5, 6], Advanced PWR-1400 Korea [7].

The modeling of severe accidents is mainly dependent on “simulation instruments», which are the calculation codes. Therefore, adequate validation of the simulation models and analytical study of the features of these codes is very actual issue.

The analysis of uncertainties in the calculation codes is essential for the validations of the calculation codes and for evaluating the objectivity and the accuracy of the performed calculations. This is due to the results of standard calculations of emergency processes at nuclear power plants can retain a high degree of uncertainty [8, 9]. Melcor 1.8.5. is the main simulation code of Ukrainian NPPs for SA calculations. It is in this version that models of Ukrainian power units are developed. The comparison of the calculation results of this code with Melcor 1.8.6 allows refining of the simulation model, its validation and reducing the uncertainty of the calculations.

II. THE DESCRIPTION OF THE NATURAL CIRCULATION LOOP

A hypothetical cooling source of two make-up water tanks for filling the reactor cavity (500 m3 each) is used to cool the external surface of the reactor lower head. Tanks are located on the roof of the containment (see Fig. 1). Water supply into the reactor cavity is fed by the ventilation system (TL05) (see Fig. 2, Fig. 3). The driving force of water flow is natural circulation - gravity driven coolant injection.

The vapor formed by water boiling on the outer surface of the reactor lower head shall be removed from the reactor cavity (see Fig. 4), otherwise, it will accumulate in the upper part of the reactor cavity causing pressure surge, and therefore displacing the feed water and forcing it to change direction.

Since the model simulates vapor removal by the Measuring Channels Ionization Chambers (IC), in the BH model. Also, the flow of the debris through the link between the reactor vessel and the reactor cavity is modeled. The cross section of the flow area, in the IC used in the model, was taken as the sum of the areas of all thirty channels comprising the IC. This makes it possible to assess the efficiency of implementing strategies of corium retention in the reactor vessel.

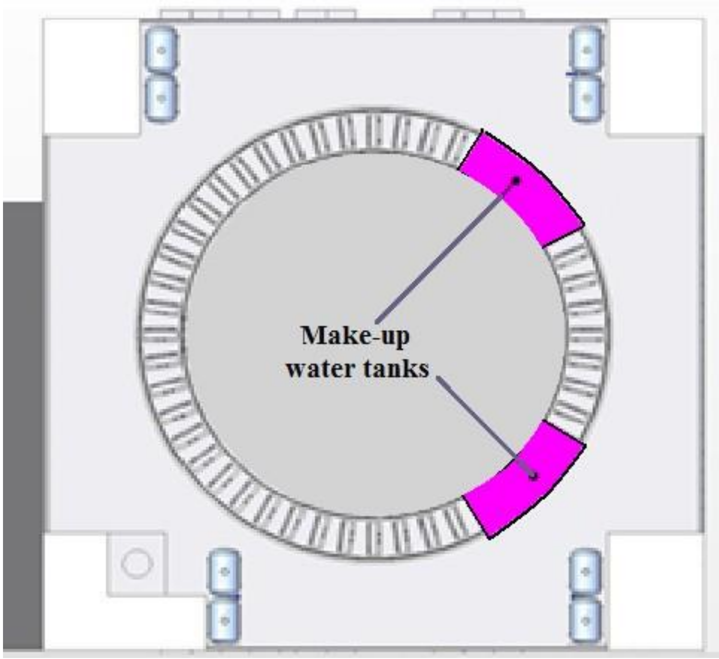

Figure 1. Placement of the make-up water tanks on the containment roof (the top view).

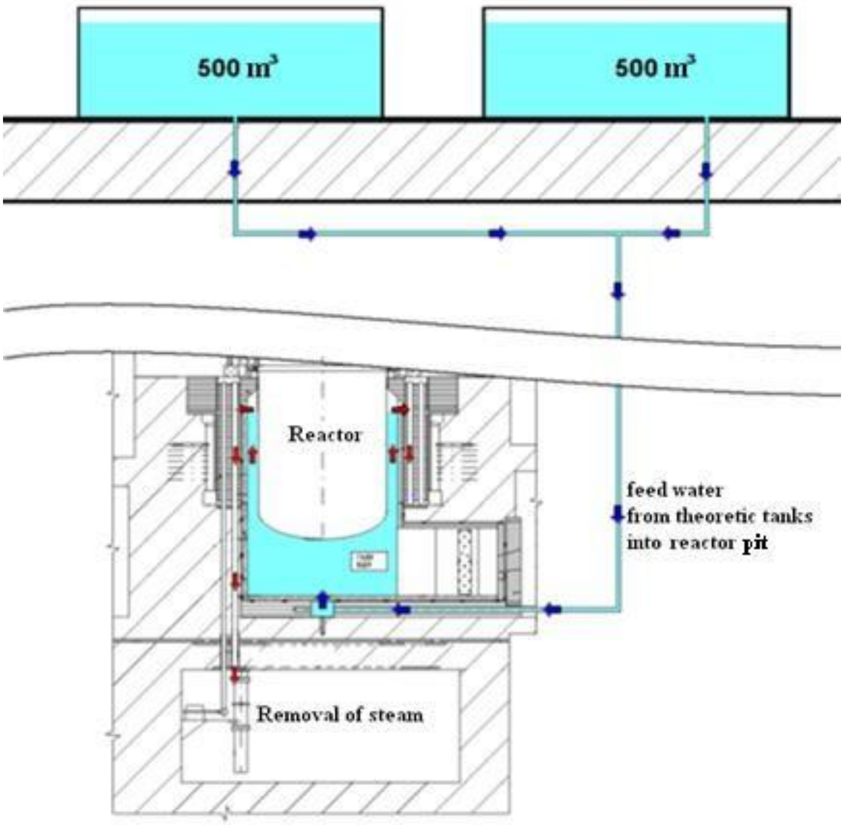

Figure 2. Passive cooling of the external surface of the reactor lower plenum.

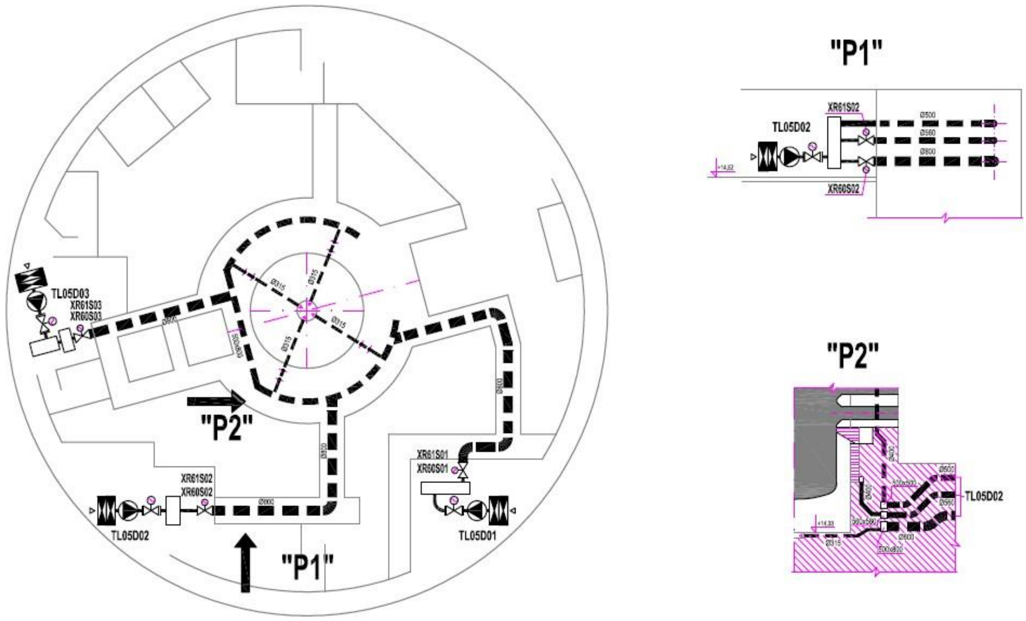

Figure 3. Feed water supply into reactor cavity trough ventilation system TL05.

III. THE MODEL OF WATER SUPPLY INTO THE REACTOR CAVITY

The make-up water tanks are presented by control volume CV021 with total volume of 1000 m3. The flow path FL021 was created to connect CV021 to the reactor cavity. The reactor cavity is presented as CV607 and control functions CF970, CF972, CF197. The control function CF970 determines the conditions of water feed (cooling start time and flow rate) that required for filling the reactor cavity in 30 minutes. The control function CF972 simulates a long-term rector cavity feed, keeping the water level constant. The control function CF197 sets the initial temperature of the water. To take into account heat transfer intensification, the volume CV607 was divided into two control volumes - CV607 and CV677. The control volume CV607 represented the area around the reactor vessel and connected to CV677 by the flow paths FL660 and FL664 (see Fig 4). To remove vapor from the reactor cavity (CV607 and CV677), a control volume under reactor (CV605) was formed (by a flow path FL697). The area of the cross section was taken as the sum of the areas of all thirty channels. The nodalization scheme for Melcor 1.8.5 is represented in Fig. 4.

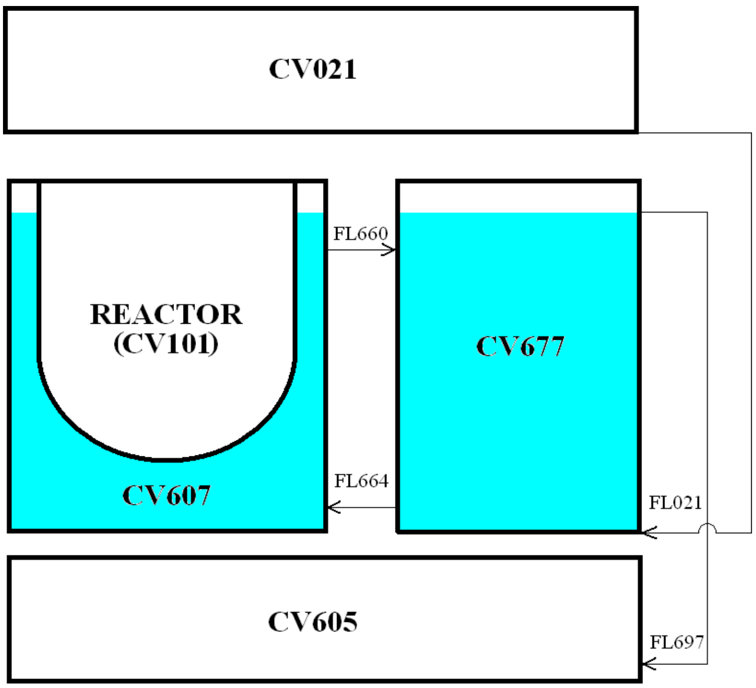

Figure 4. The nodalization scheme of MELCOR 1.8.5 simulation model for a feed water supply passive system.

IV. THE LIMITATIONS OF THE APPLIED SIMULATION MODEL FOR SEVERE ACCIDENT ANALYSIS IN MELCOR 1.8.5 CODE

The Melcor code 1.8.5 simulates the thermo-hydraulic processes in the reactor and the containment, as well as the degradation of the core and the behavior of the corium in the reactor lower plenum [10]. However, the main limitations of this version of the Melcor code are:

– A simplified degradation model of the core: For example, limited thermal- mechanic modeling process of the melting components of the core.

– The inability to set the deflector profile and to take other measures that can improve the heat transfer.

– The inability to create detailed model of the reactor cavity, the support plate of the lower core and other internal devices in BH module.

– Insufficient of experimental data for heat transfer coefficients for internal and external surfaces of the reactor vessel for VVER-1000.

– The lack of data for validation of various components of the model; such as the heat transfer coefficient from the melt to the reactor vessel, the heat transfer coefficient from the external surface of the reactor vessel to water and layers formation of melt in the reactor vessel.

Starting from the updated version of the MELCOR code 1.8.6 and higher, there is the possibility of simulation the hemispherical bottom of the reactor in the COR package, the support plate of the lower core and other internal devices which are located on the way of melt relocation to the reactor lower plenum (LP). It is also possible to simulate the melt formation process in the core, the distribution of the heat flux, the crust formation and the melt convection and its separation into oxide and metal layers.

V. THE MAIN RESULTS OF SEVERE ACCIDENT THERMAL- HYDRAULIC CALCULATIONS USING IVMR STRATEGY

The transition process accident chronology for scenarios V01, V02, V03, LOCA (v06) is presented in Table. 1. In Scenario V01, a large break of the cold leg between MCP and RV is initiating with station blackout. This results in loss of all active systems (LPI, HPI, FW, EFW and the Containment Spray System). The only functioning systems are passive systems (hydro-accumulators, PORV or SV) and systems dedicated for the severe accident management – PARs, depressurization of the primary circuit (opening of all PORV and SV at pressurizer) and reflooding the reactor cavity by water feed supply. Scenario V02 – similar to V01, but a melt localization strategy was used by cooling reactor lower plenum. Scenario V03 is similar to V02, but using a short deflector. LOCA (v06) is similar to V03, but the simulation model was created for the Melcor 1.8.6 code by UJV Rez [11, 12] (Czech Republic).

According to the results of calculations, we can estimate the effect of using feed water supply to cool the reactor lower plenum and assess how the deflector affects the failure of the reactor vessel.

The V03 scenario is one of three calculated options for extending the failure of the reactor vessel. For a comparative analysis of the results of the models Melcor 1.8.5 and Melcor 1.8.6, we compared two scenarios: V03 and LOCA (v06), because these scenarios are identical.

Table 1. The chronology of severe accident events

Time, s				**Event**	**Description**
V01	V02	V03	LOCA (v06)		
0	0	0	0	Station blackout	Initial event
Time, s				**Event**	**Description**
0	0	0	0	Loss of all active systems (LPI, HPI, FW, EFW, Containment Spray),	Station blackout (SBO)
0	0	0	0,11	Reactor SCRAM	Station blackout (SBO)
140	140	140	153,57	Hydro-accumulators 1,2,3,4 – First injection	Pressure in hydro- accumulate is below 60 kgf/cm2
300	300	300	297,64	Hydro-accumulators 1,2,3,4 – end of the injection	Hydro- accumulators is empty
1003 (16 m)	1432 (24 m)	1881 (31 m)	1889,54 (31 m)	Core exit T ≥ 650 deg C	-
2040 (34 m)	1800 (30 m)	2260 (38 m)	1827,21 (30 m)	Cladding rupture due to overheating	Failure of the core structure in the calculation model

-	2040 (34 m)	2040 (34 m)	2381,68 (40 m)	First injection to the reactor cavity	-
3957 (1 h 6 m)	4760 (1 h 19 m)	5096 (1 h 25 m)	5060 (1 h 25 m)	Failure of the core support plate in the calculation model	Loss of the support capability
9449 (2 h 37 m)	10899 (3 h 2 m)	11886 (3 h 18 m)	12705 (3 h 31 m)	Failure of the reactor vessel	Calculation terminated by fail of reactor vessel
272	259	348	298	Full mass of hydrogen, that was generated inside the reactor (kg)	-
20,60	200	20,26	17,7	Decay heat power in debris (106 W)	-

In table 2. the final ingredients of the melt materials in BH model for different scenarios are presented.

Table 2. The ingredients of melt materials in BH model

Materials	**Mass, kg**			
	V01	**V02**	**V03**	**LOCA (v06)**
Fe	51571.6	46558.5	48273.6	45486.0
FeO	1428.3	1313.8	1869.8	1012.2
Zr	24584.0	24734.6	23221.6	23195.0
Cr	9524.0	9143.7	9602.3	-
Ni	4231.7	4062.2	4266.7	-
B4C	374.2	377.2	352.8	-
ZrO2	6455.4	6169.3	8025.7	8101.5
Cr2O3	385.4	350.0	503.8	-
NiO	133.0	128.4	178.9	-
UO2	80094.2	80093.9	79839.1	85885.0
Total	178781.8	172958.7	176134.3	163682.0

The scenario V03 was created to be as similar as LOCA (V06) [7, 8] to assess the acceptability and to achieve validation of the results.

The first initial event is a large break of the cold leg between MCP and RV with station blackout (at 0 sec). When the station blackout took place, the reactor scram started and all the active systems became out of function (LPI, HPI, FW, EFW, CSS the containment spray system). At that moment the reactor is in subcritical state because of the density effect of reactivity. The decay heat power is 7% of nominal and continues to decrease (see Fig. 7, Fig. 8).

Since there is no enough heat removal from the fuel assembly by the vapor, its temperature becomes high. This resulted in water boiling and a decrease in the water level in the reactor core. The time of the core structure rupture was used as a start time of water supply to the reactor plenum. The results of the V01 calculation showed that this point is 34th minute (39th minute for LOCA (v06)).

Chemical processes occurring between the components of core materials and vapor, high temperature, degradation of the fuel cladding, cracking and shedding of the fuel pallet, cause core materials to melt.

After entering into temperature range 2100–2500 K, fuel elements lose their integrity and collapse immediately. The degradation starts in the upper central part of the core and the process continued by formation of complex debris beds with evolving molten pools. Material of the fuel assemblies melts, candles, refreezes, slumps and successively moves downward.

About 10 tons of core material is transferred to the lower plenum before the massive relocation occurred that initiated by the core plate failure. A part of debris and molten material created in the upper parts of the core relocates downwards and settles on the lower core structures. Hot material during relocation intensively cooled by remaining water promoted steam creation. After the complete core uncovered, the hot material collected on the core plate was no longer cooled. MELCOR due to its modeling features allows molten materials, i.e., remained superheated unoxidized metals or control tube materials transferred to the lower plenum before the failure of reactor core.

The core melt interacts with water, forms debris and settles on the lower head. In this process, water in the lower plenum is intensively transformed into steam which leaves the RPV. The phenomenology described above is the main reason of earlier concealing of the core predicted in MELCOR calculation. The mass of molten material and debris relocated to the lower plenum is presented in Fig. 5, Fig. 6. In LOCA (v06), during the first massive relocation there are approximately 40 tons of materials more transported to the lower plenum. Relocation took more time due to sequenced failure of the core support plate rings. Corium starts to relocate after 1 h 25 m (1 h 24 m in LOCA (v06)) and it takes more than 1 h to RPV breach.

The evolution of component masses in the lower plenum for V03 and LOCA (v06) strategies is shown in Fig. 5, Fig. 6. The figures show that, at the end of the calculation, the accumulated mass of corium for (V03) is 176.13 tons, the oxide part (sum of UO2, ZrO2 and SSOX) is 90.23 tons, the metal (steel and Zr, Cr, Ni) is 85.36 tons, 27.0% of which is Zr. In LOCA (v06), the mass of the corium is 163.68 tons, the oxide part (sum of UO2, ZrO2 and SSOX) is 95 tons, the metal part (steel and Zr) is 68.68 tons, 20.5% of which consists of Zr.

This ratio of Zr concentrations in the melt shows that the oxidation process in V03 and LBLOCA (v06) is not very intense, since the decrease in pressure in the primary circuit, because of LBLOCA of Deq200 mm was initiated, which reduced the vapor content in the core.

For v03 scenario, the heat flux through the reactor wall reached its max of 2.8 MW/m2 at failure point. The average value during the calculation is 0.8 M W /m2 (see Fig. 9). A value of 1.4 MW/m2 is achieved for a short period at the end of the calculation time and an average maximum during the calculation is slightly lower than 1.3 MW/m2 in LOCA (v06) (Fig. 10).

The partitioning of the reactor vessel wall into nodes is a one of the main differences between Melcor 1.8.5 and Melcor 1.8.6, when modeling the LP. The walls are represented in the form of 18 nodes in Melcor 1.8.5 code, and 12 nodes in Melcor 1.8.6 code. Hence, a slight difference in the values is shown in the graphs Fig. 11 – Fig. 16.

Melt convection in each layer enhances heat transfer to the reactor vessel, which prevents overheating of the melt. The distribution of heat flux at the boundary of the melt and the reactor vessel is different and has a maximum value at the top (Fig. 15, Fig. 16).

The lower head failure is predicted to occur after 3 h 18 m (3 h 31 m for LOCA (v06)) since the start of SA as creep rupture process in one of the sideward nodes and it lead to termination of simulation.

The side rupture indicates the interaction with a metal layer. Various MELCOR simulations performed for the ex-vessel phase (beyond the scope of this article) predicts the series of the failures of other lower head nodes and effectively continuous corium ejection into the reactor cavity.

Since the beginning of the SA phase, a mass of 348 kg (298 kg – for LOCA (v06)) of hydrogen was generated as the result of exothermic zirconium oxidation reactions.

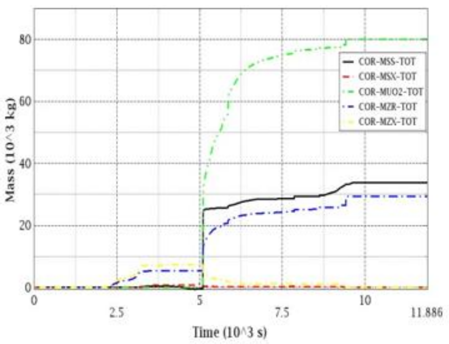

Figure 5. Evolution of component masses in lower plenum in case V03.

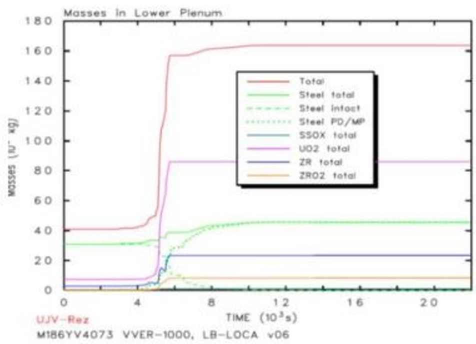

Figure 6. Evolution of component masses in lower plenum in case LBLOCA v06.

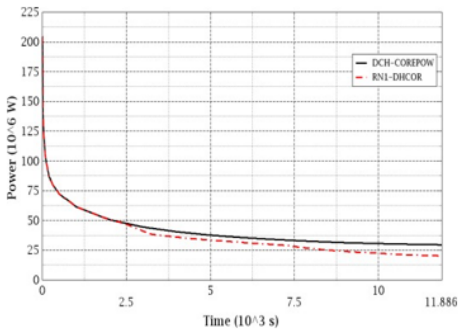

Figure 7. Evolution of decay power distribution for V03 case.

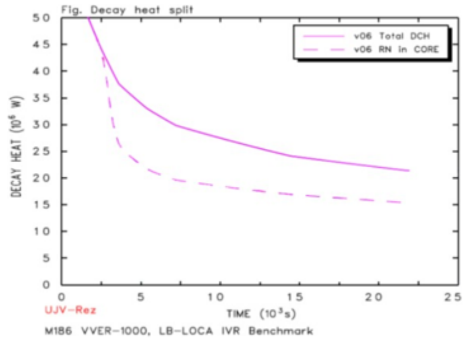

Figure 8. Evolution of decay power distribution LBLOCA v06.

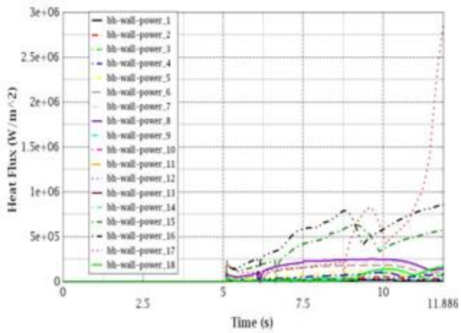

Figure 9. Evolution of heat flux densities in segments of MELCOR 1.8.5 for V03 case

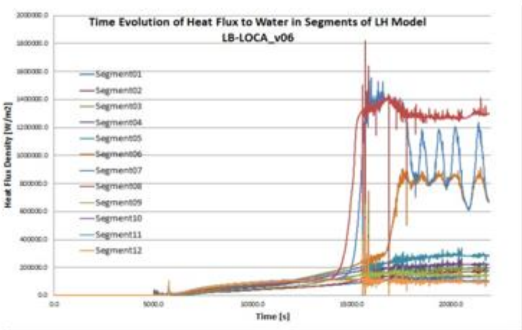

Fig. 10. Evolution of heat flux densities in segments of MELCOR 1.8.6 LBLOCA v06 case

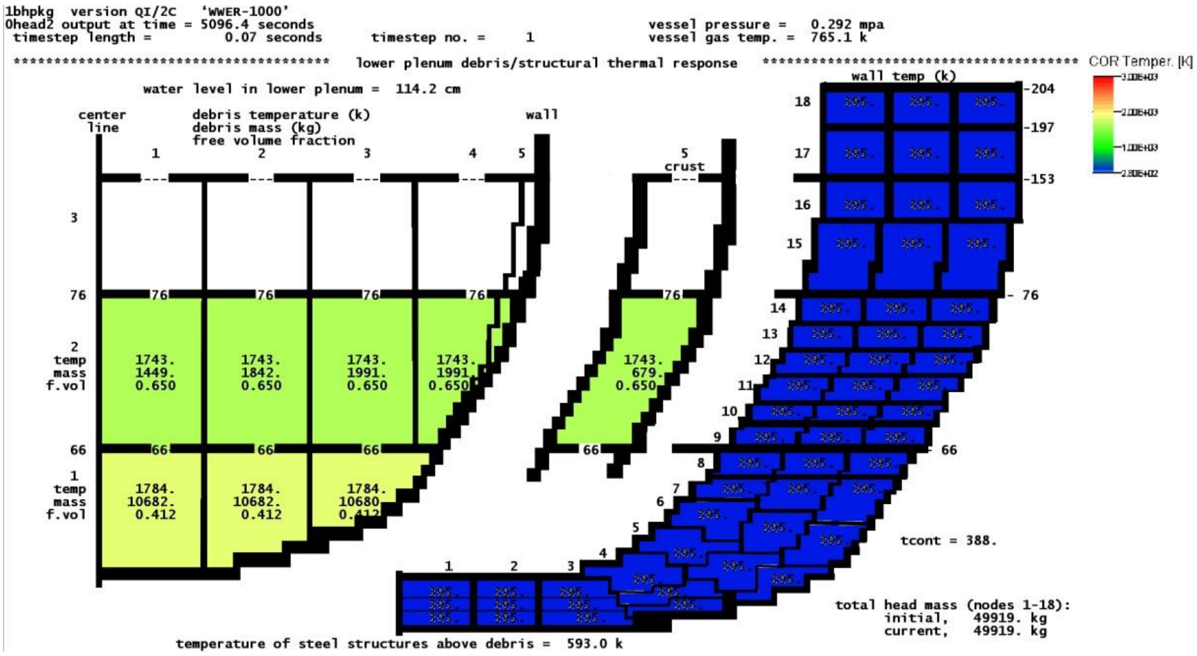

Figure 11. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - V03 case, time 5096 s.

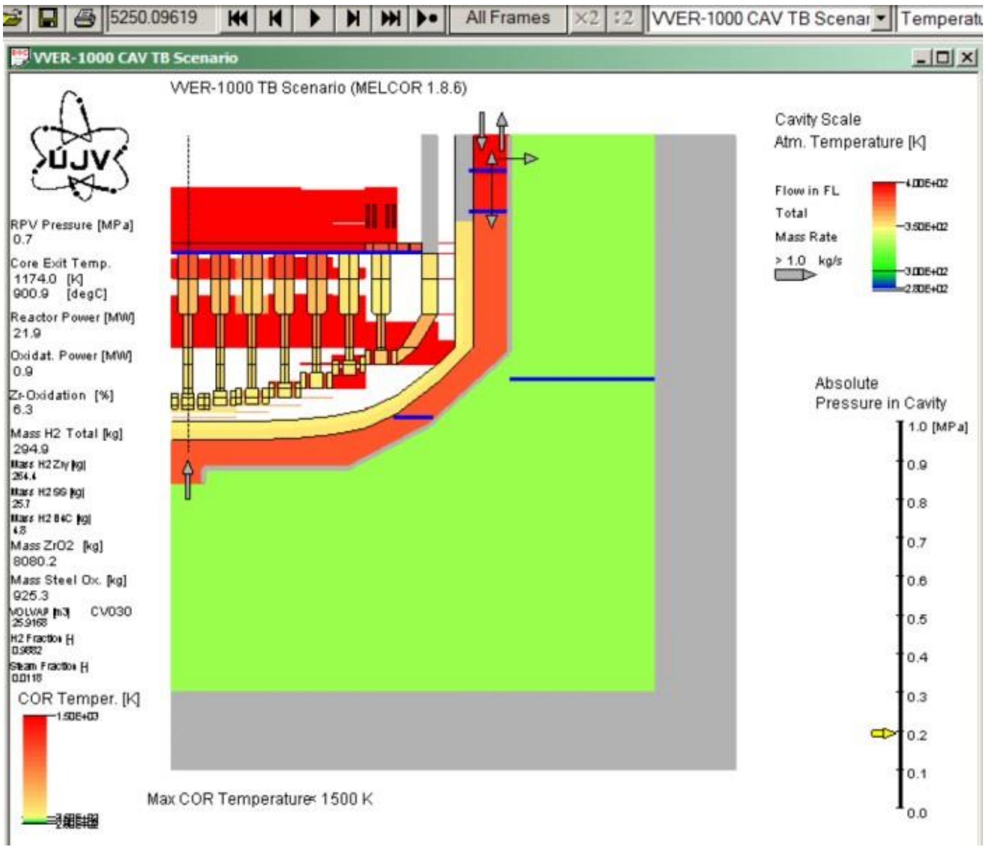

Figure 12. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - case LBLOCA v06, time 5250 s.

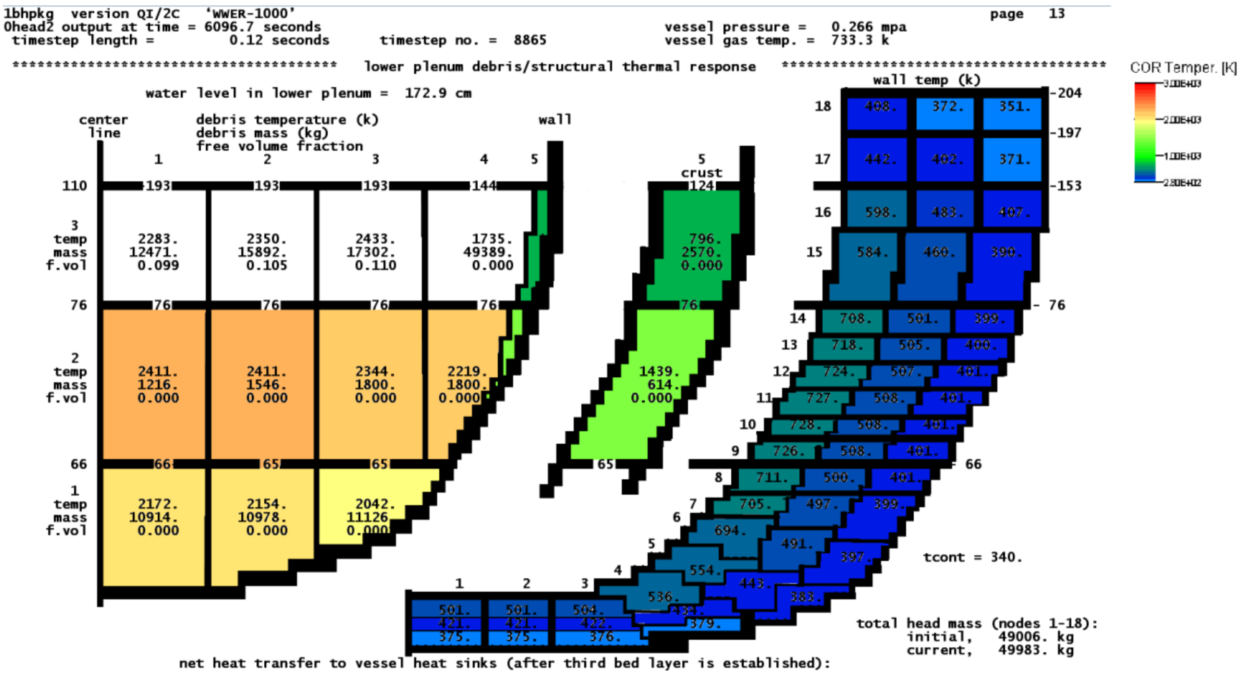

Figure 13. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - case V03, time 6096 s.

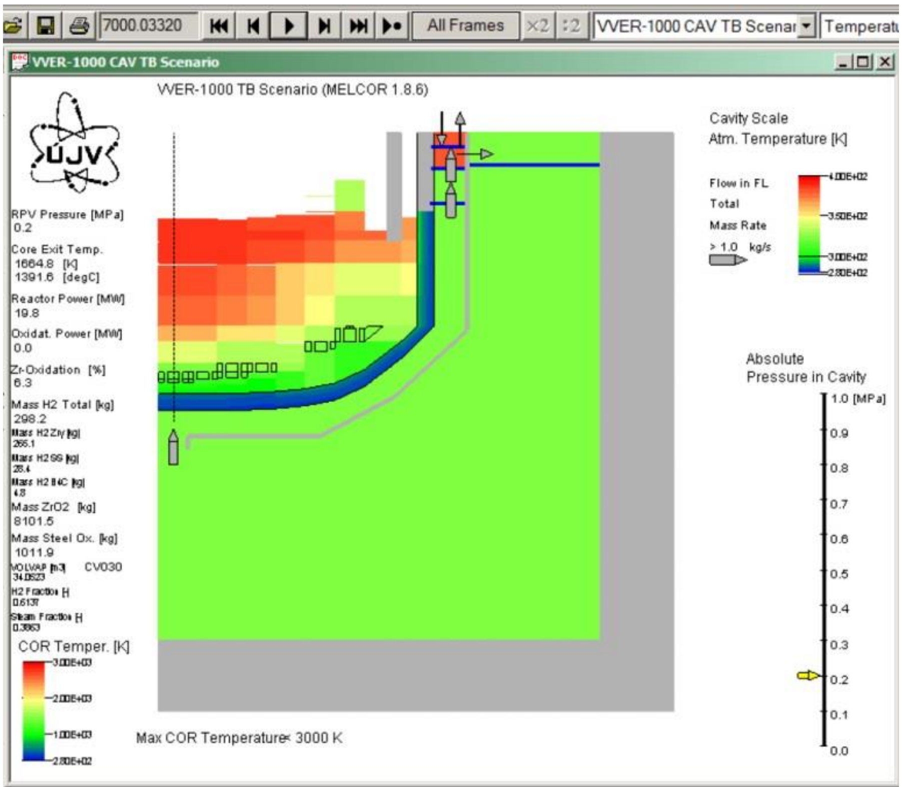

Figure 14. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - case LBLOCA v06, time 7000 s.

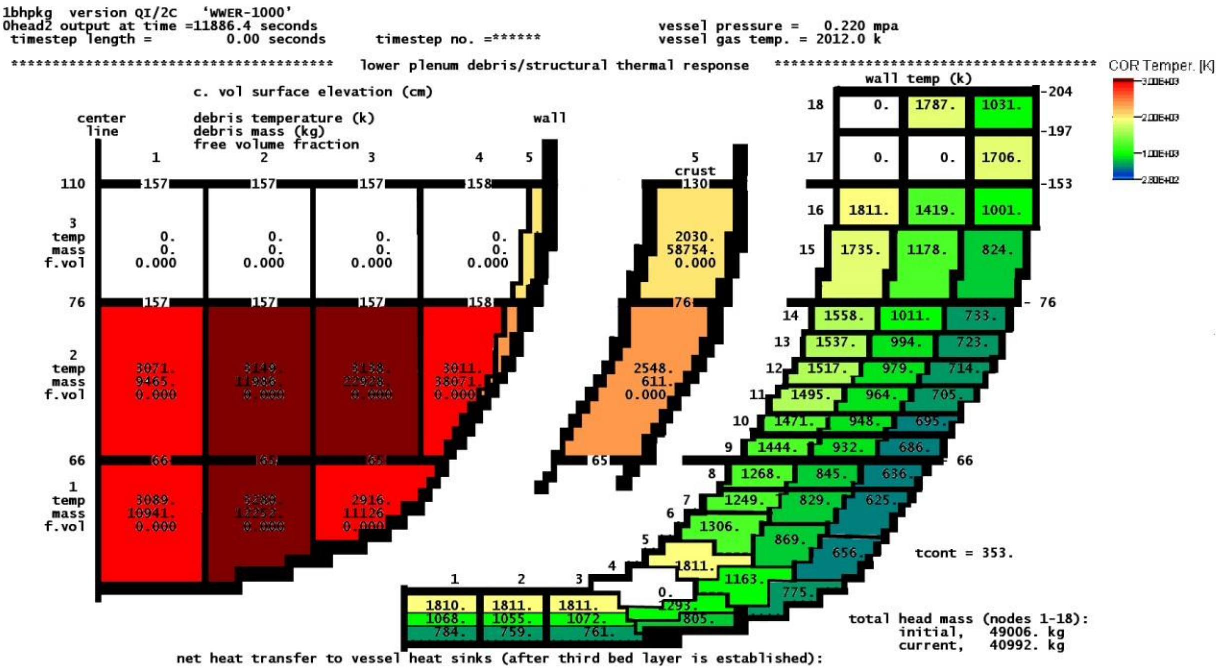

Figure 15. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - case V03, time 11886 s.

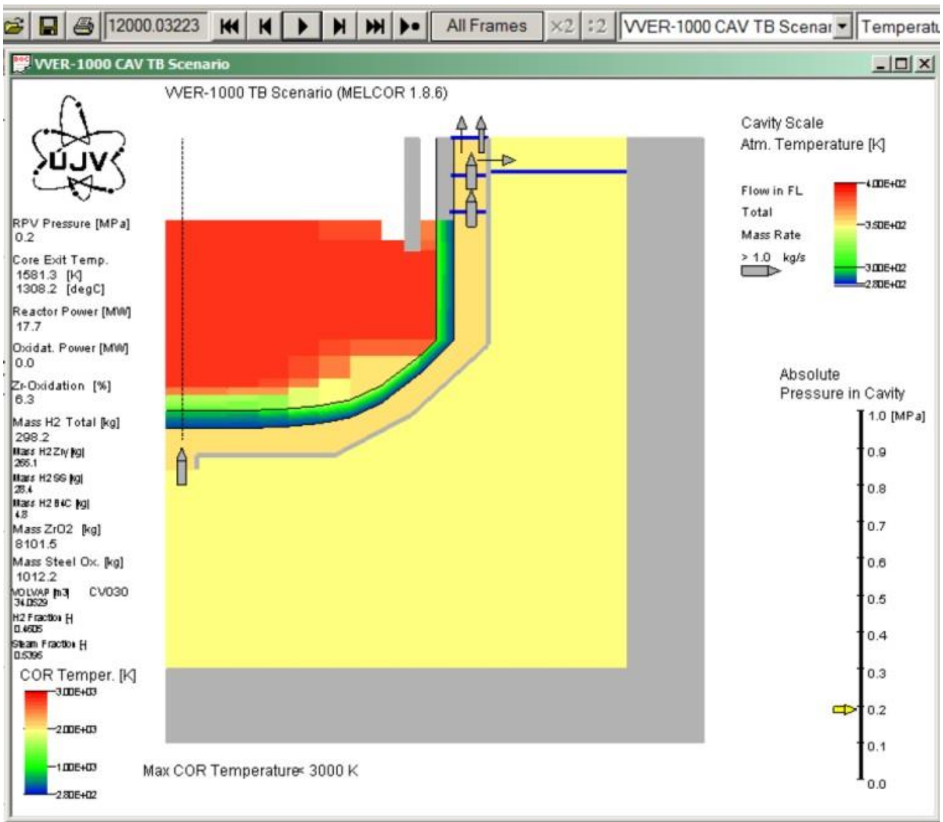

Figure 16. Visualization of lower plenum temperature spatial distribution, location of intact component and corium and coolant outside of RPV wall - case LBLOCA v06, time 12000 s.

VI. CONCLUSIONS

The comparison of calculations of simulated scenarios allowed evaluating the correctness of the simulation by Melcor 1.8.5 code and the convergence of the results. The reason for the divergence of results is due to using different versions of the Melcor code. Undoubtedly, the results of LBLOCA v06 that were calculated using the Melcor 1.8.6 code are more accurate. However, one of the main issues regarding the evaluation of the efficiency of external cooling of the reactor vessel is the accurate determination of the heat transfer coefficients on the inner and outer walls of the reactor vessel, which requires calibration using the experimental data.

The heat transfer conditions on the inner surface depend on many phenomena that may occur in the case of SA (fuel claddings and fuel pellets destruction, displacement and sintering of debris, blocking of the coolant flow area, stratification of the corium and the formation of a layer with high thermal conductivity). For all these phenomena, there are not enough experimental data at the international level, and, unfortunately, these phenomena continue to be one of the main contributors to the uncertainties associated with SA simulation.

As of the present moment, the lack of data on the description of heat transfer through the wall of the BH leads to uncertainty of the heat transfer conditions from the melt through the wall of the BH, which does not allow making unambiguous conclusions about the success of strategy with external cooling of the reactor vessel.

At the moment, the lack of data describing the heat transfer from the melt to BH wall leads to heat transfer uncertainty, which does not allow one to give definitive conclusions about the success of IVMR strategy by external cooling of the reactor vessel.

However, research in the field of this work should continue to use updated codes, considering international experience and new results of experimental studies.

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