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Analysis of Influence of DEC Equipment on Severe Accident Development

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ABSTRACT

A hypothetical severe accident in the Krško NPP was analysed with the MELCOR code version 2.2 considering mitigation measures for heat removal from the containment solely by the design extension conditions alternative safety systems. As the initiating external event, a strong earthquake was considered, resulting in a postulated initiating event of simultaneous station black-out and large break loss-of-coolant accident. The following scenarios were analysed: (1) no mitigation, (2) after melt release from failed reactor pressure vessel water injection through containment sprays, (3) after melt release water injection simultaneously through containment sprays and into the reactor coolant system, (4) after melt release water injection into the reactor coolant system, (5) after melt release water injection into the reactor coolant system, without operable alternative residual heat removal system heat exchanger but with operable alternative auxiliary feedwater system.

The five analysed severe accident scenarios and the applied Krško NPP MELCOR code model are presented. The main focus is given to the molten core concrete interaction. It turned out that the molten core concrete interaction in the reactor cavity can be stopped if the molten core is flooded soon after it is released from the failed reactor vessel. The extend of the molten core concrete interaction is very sensitive on the flooding time. The simulation results revealed that the heat transfer through the steam generators by natural circulation of the atmosphere in the failed reactor coolant system is not sufficient to stabilize the severe accident. In this scenario (5) the containment atmosphere was periodically released through the passive containment filtered venting system, like in the unmitigated case (1), whereas in the other mitigated scenarios (2-4) no releases occurred.

1 INTRODUCTION

Following the lessons learned from the accident at the nuclear power plant Fukushima in Japan and according to the Slovenian Nuclear Safety Administration decree, the Krško NPP decided to take the necessary steps for upgrading the safety measures to prevent severe accidents and to improve the means for the successful mitigation of their consequences [1]. Two of the modifications that the Krško NPP implemented are the installation of an alternative safety injection pump and an alternative residual heat removal heat exchanger. These modifications, among the other already existing systems, serve for the purpose of reactor decay heat removal, either from the reactor coolant system (RCS) or from the containment, once the core and RCS are severely damaged.

The purpose of the paper is to analyse a hypothetical severe accident in the Krško NPP considering mitigation measures for heat removal from the containment solely by the design extension conditions (DEC) alternative safety systems (ASS). As the initiating external event, a strong earthquake was considered, resulting in a postulated initiating event of simultaneous station black-out (SBO) and large break loss-of-coolant accident (LBLOCA). The following scenarios were analysed:

Scenario 1: no mitigation (denoted noASS – no alternative safety systems),

Scenario 2: after melt release from failed reactor pressure vessel (RV) water injection through containment sprays (denoted ACI_RV – alternative containment injection),

Scenario 3: after melt release from failed RV water injection simultaneously through containment sprays (50 %) and into the RCS in the RV (50 %) (denoted ACVI_RV – alternative containment and vessel injection),

Scenario 4: after melt release from failed RV water injection into the RCS in the RV (denoted AVI_RV – alternative vessel injection),

Scenario 5: after melt release from failed RV water injection into the RCS in the RV, without operable alternative residual heat removal system heat exchanger ARHR HEX but with operable alternative auxiliary feedwater system AAF (denoted AVI_RV_noHEX_AAF).

The analyses were performed using the MELCOR 2.2 computer code version 15254 [2], [3], with the Krško NPP MELCOR 2.2 standard input deck [4], [5], [6], [7], which was upgraded with the DEC alternative safety systems [8]. Default models and model parameters were applied.

In Section 2 the model is presented, focusing on some model details. The simulation results are presented in Section 3, first some integral results in table form and then in comparison the results of some selected variables for all five considered scenarios in form of graphs. In Section 4 the conclusions are given.

2 MODEL DESCRIPTION

The primary and secondary systems and the containment, including regulation systems and control volumes that represent boundary conditions, consist of 145 thermal-hydraulic control volumes, 197 flow paths and 149 heat structures [7], [8]. In Figure 1 the Krško NPP containment nodalization for the calculation with the MELCOR code is presented and in Figure 2 some more detailed view around the reactor cavity with listed levels is provided. The reactor pressure vessel is located in control volume CV711, and control volume CV704 presents the reactor cavity. The ventilation duct, denoted as flow path FL783, connects the reactor cavity CV704 with the containment lower compartment CV702. The ventilation duct opening is nearly 2.5 m above the floor of the containment lower compartment.

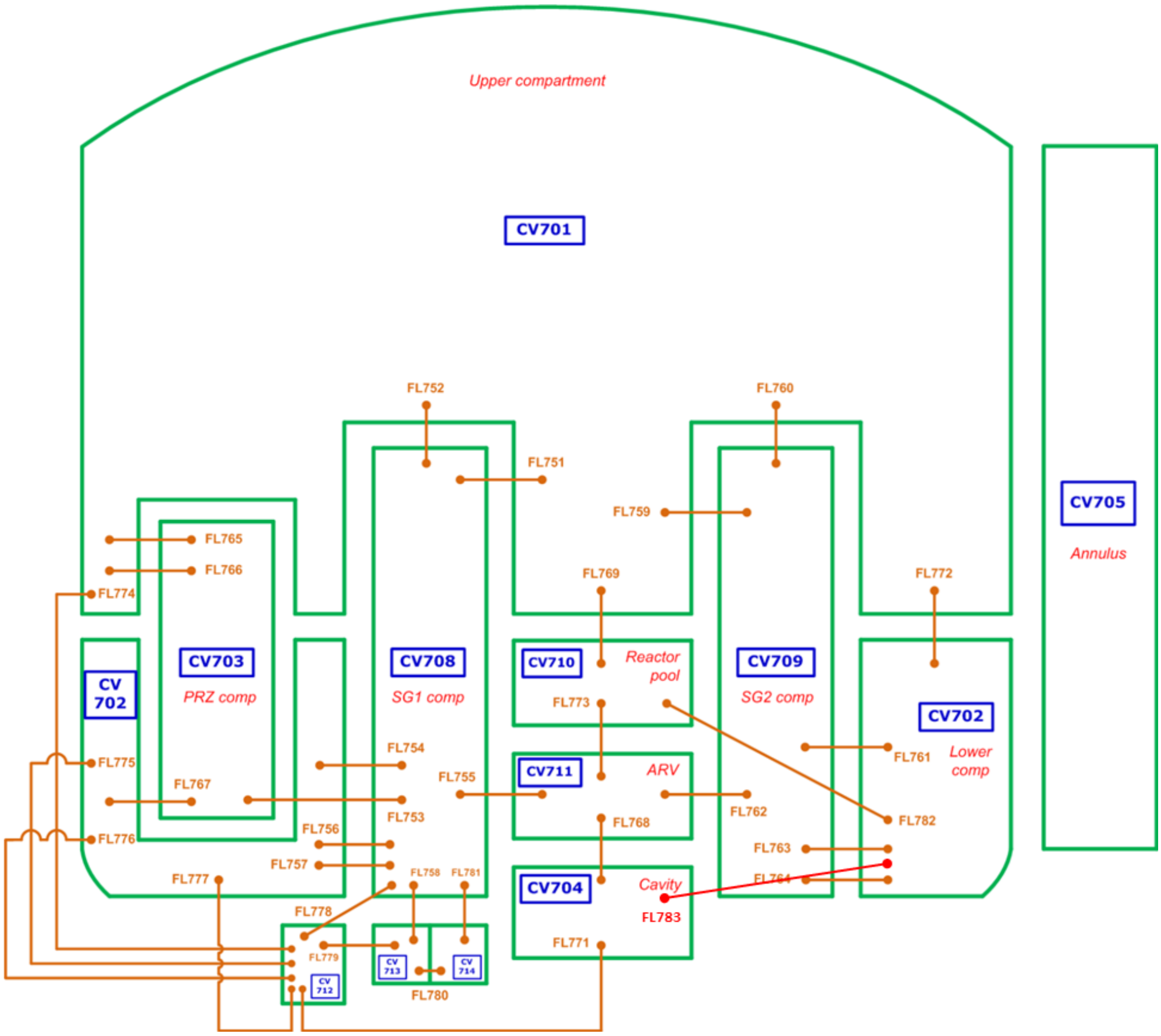

Figure 1: Krško NPP containment nodalization [7], [8].

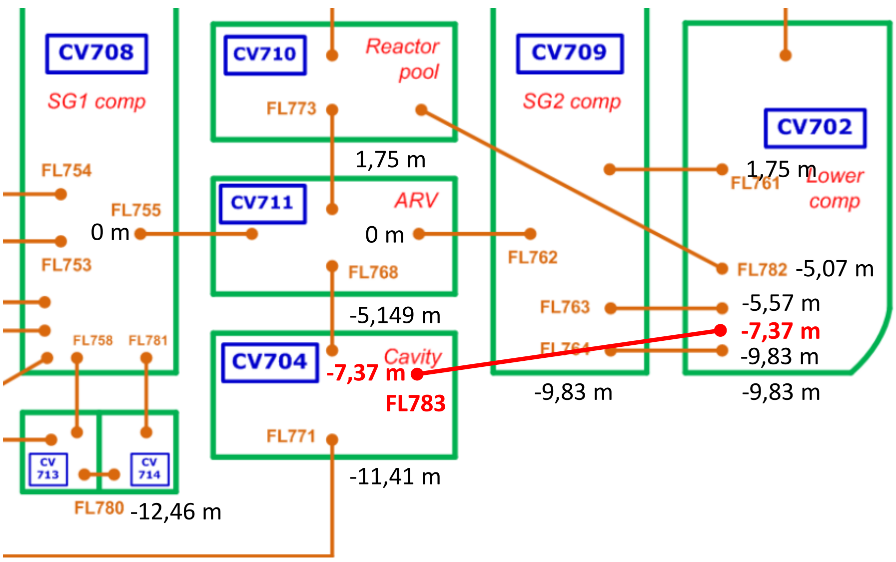

Figure 2: Detail of Krško NPP containment nodalization [7], [8].

The alternative safety injection system injects the water into the RV. Thus, the water flows directly into the reactor cavity CV704 through the failed RV. When injecting the water through the containment sprays, the water level in the containment has to rise first to the height of the ventilation duct opening FL783 before it can flow in the reactor cavity. It was assumed that the large break occurs in the second loop of the RCS at the connection of the accumulator to the cold leg and that the break size is 12”. Consequently, the water from the second accumulator and through the break from the RCS flows into the containment lower compartment CV702. From there it flows through the flow path FL777 into the sump CV712 and then through the flow path FL771 into the reactor cavity CV704. Once the molten core is released from the failed RV into the reactor cavity CV704 it is assumed that the flow path FL771 is choked due to the spread molten core on the reactor cavity floor. Then the water cannot flow anymore from the containment to the reactor cavity through this flow path at the bottom.

The passive containment filtered venting system (PCFVS) has a rupture disc which breaks at a pressure of 6 bar. If the pressure inside the containment exceeds this value, the containment atmosphere is released into the environment through the filter till the setpoint of the containment relief valve closure of 4.1 bar is reached. The containment venting setpoint for the next containment relief valve openings is 4.9 bar.

3 SIMULATION RESULTS

The first 300,000 s (~3.5 days) of the accident were simulated for all five considered scenarios. In Table 1 the chronology of the main events is presented with the starting time of their occurrence. After the decrease of the RCS pressure due to the assumed LBLOCA with SBO the first accumulator starts to inject the water in the RCS, whereas the second accumulator is discharged into the containment due to the break at the connection to the RCS. When the accumulators are emptied (accumulators empty), the core starts to heat up, the integrity of the fuel rods is lost and radioactive gases are released from the gap between the fuel pellets and the cladding (gap release). The core starts to melt (core melting) and relocates to the RV lower head which eventually fails (RV failure). The molten core is released in the reactor cavity, where the molten core concrete interaction (MCCI) starts. 5000 s after the start of the accident, i.e. soon after the molten core is fully released from the failed reactor vessel, some alternative safety systems are activated (ASS activated) in the mitigated scenarios (2-5). Due to the water evaporation and the release of gases during the MCCI the pressure in the containment increases. In the unmitigated scenario noASS and the mitigated scenario AVI_RV_noHEX_AAF the pressure in the containment exceeds the PCFVS opening setpoint and the atmosphere is released into the environment through the filter (PCFVS open). The containment pressure drops and the PCFVS closes. After that the pressure starts again to rise and the PCFVS opens the second time (PCFVS open 2nd time).

The observed differences for the scenarios in the period when they are still identical, i.e. the first 5000 s, are due to numerical variance, which is caused by the numerical errors in solving the system of nonlinear equations for the slightly different input files of the considered scenarios.

Table 1: Chronology of main events with starting time of their occurrence

Event \ Scenario	Time (s)				
	noASS	ACI_RV	ACVI_RV	AVI_RV	AVI_RV_noHEX_AAF
Accumulators empty	83	82	82	83	81
Gap release	303	305	305	303	302
Core melting	888	927	927	888	874
RV failure	3467	3734	3734	3467	3614
ASS activated	/	5000	5000	5000	5000
PCFVS open	72263	/	/	/	111702
PCFVS open 2nd time	92700	/	/	/	135100

In the graphs the simulation results for some selected variables are presented in comparison for all five considered scenarios. In Figure 3 the containment atmosphere pressure is presented. The pressure increases due to the water evaporation and the release of gases during the MCCI after the molten corium is released from the failed RV into the reactor cavity. In the unmitigated scenario noASS, when the pressure reaches the containment venting setpoint of 6 bars for the first opening of the containment relief valve the pressure starts to decrease till it reaches the containment relief valve closing setpoint of 4.1 bar. Then the pressure starts to increase again till it reaches the containment venting setpoint of 4.9 bar for the next containment relief valve openings. The pressure then cycles between the two setpoints. In the mitigated scenarios 2 to 4 the pressure gradually increases but does not reach the setpoint for the PCFVS opening. In the mitigated scenario AVI_RV_noHEX_AAF without operable ARHR HEX the pressure behaviour is similar to the unmitigated scenario noASS, except that the first containment venting occurs later. Thus, the cooling through the steam generators by natural circulation of the atmosphere in the failed RCS is not sufficient to stabilize the severe accident.

Figure 4 shows the containment atmosphere temperature. The temperature is the highest in the unmitigated scenario noASS and rises all the time, which means that despite the high atmosphere temperature the heat transfer through the containment walls is not sufficient to extract the entire residual heat from the molten core. In all other scenarios the temperature stabilizes, also in scenario AVI_RV_noHEX_AAF because there is enough water available for the cooling of the molten core by evaporation. In this scenario the temperature stabilizes at the water saturation temperature.

The simulation results revealed that in the second loop of the RCS with the large break a natural circulation of the atmosphere develops, which transfers heat from the containment atmosphere to the water in the second steam generator. Therefore, the water level in this steam generator decreases due to evaporation (Figure 5). Only in scenario AVI_RV_noHEX_AAF it was assumed that the water in the steam generator is replenished with the AAF. But as already explained it turned out that this heat transfer by the steam generator is not enough to remove all the residual heat from the molten core and stabilize the severe accident. No natural circulation develops in the intact first loop of the RCS.

In Figure 6 the water level in the reactor cavity is shown. The reactor cavity is initially dry. After the start of the accident it is flooded with the water which flows from the second accumulator and through the break from the RCS into the sump and from there through the flow path FL771 into the reactor cavity. When the molten core is released from the failed RV it thus flows in an already flooded reactor cavity. As the molten core chokes the flow path FL711, no further water can later flow in the reactor cavity through this flow path from the containment sump. In the unmitigated accident (noASS) the water in the reactor cavity soon evaporates and the reactor cavity remains dry till the end of the simulation. In all other scenarios the reactor cavity is flooded before it dries out, except in scenario ACI_RV, where the reactor cavity dries out for more than one hour before it is flooded again. In this scenario the water level in the containment has to rise first to the height of the ventilation duct opening before it can flow thought it into the reactor cavity, whereas in all other mitigated scenarios the water flows directly into the reactor cavity through the failed RV. The time of the reactor cavity flooding has a huge influence on the extent of the MCCI.

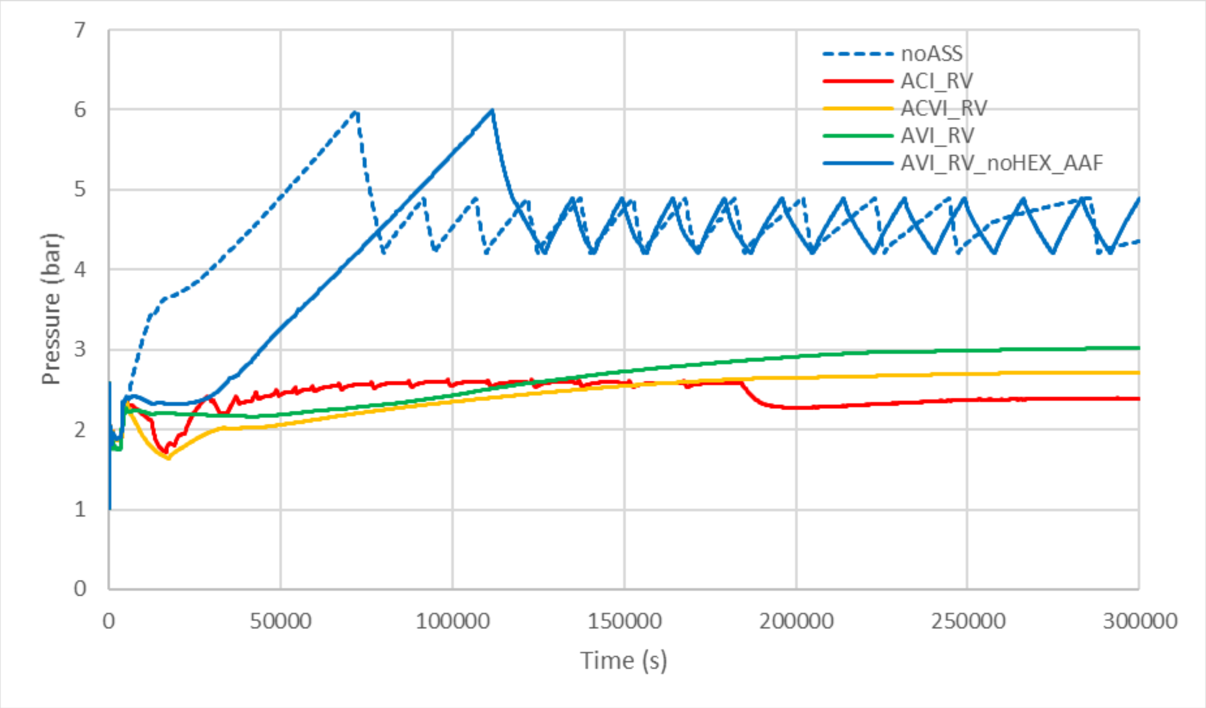

Figure 3: Containment atmosphere pressure (in CV701).

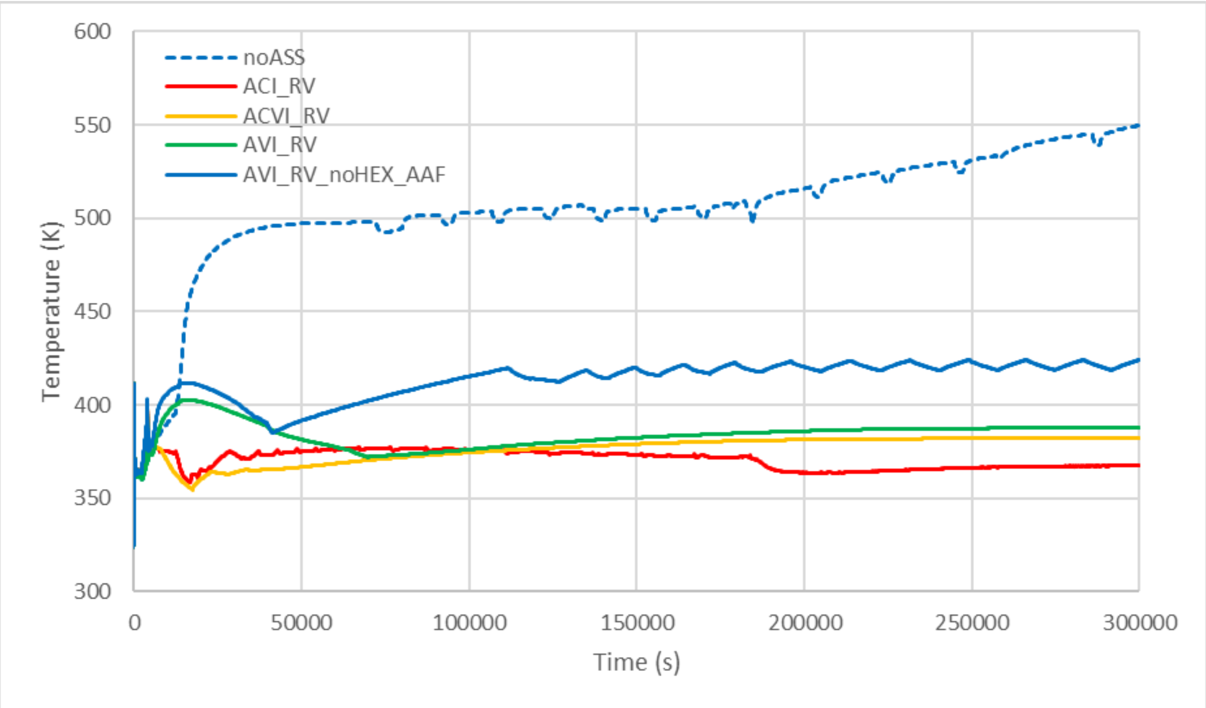

Figure 4: Containment atmosphere temperature (in CV701).

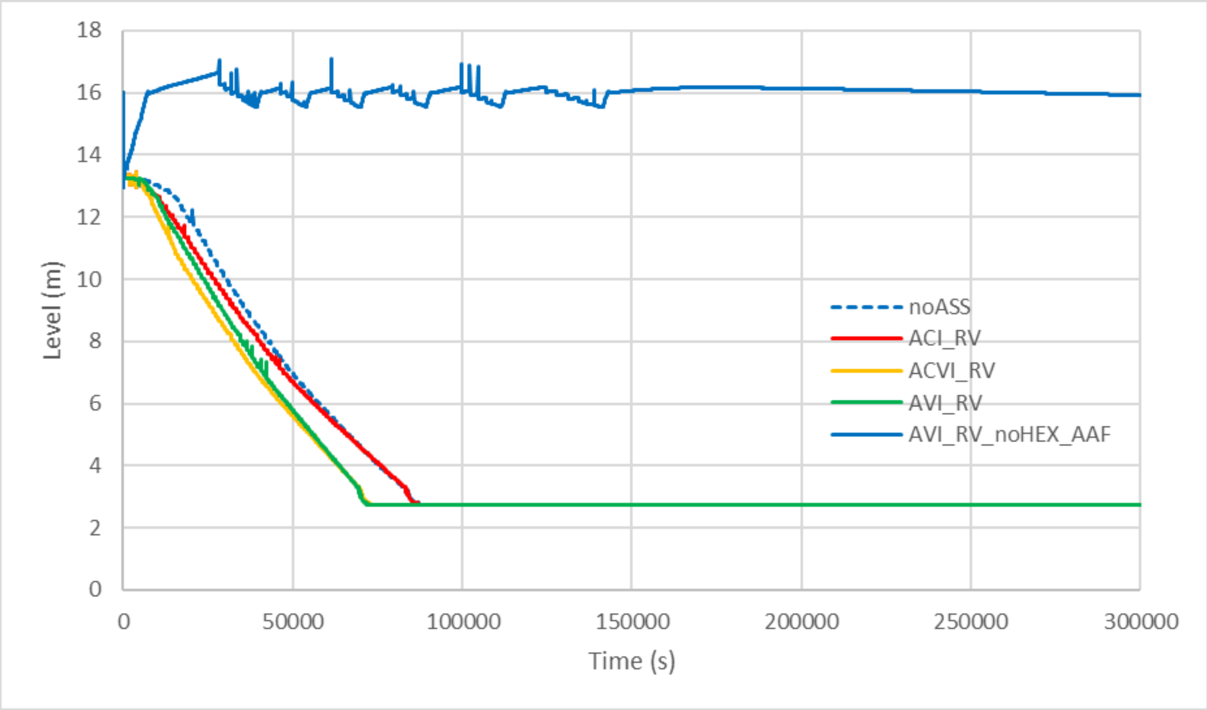

Figure 5: Water level in steam generator two.

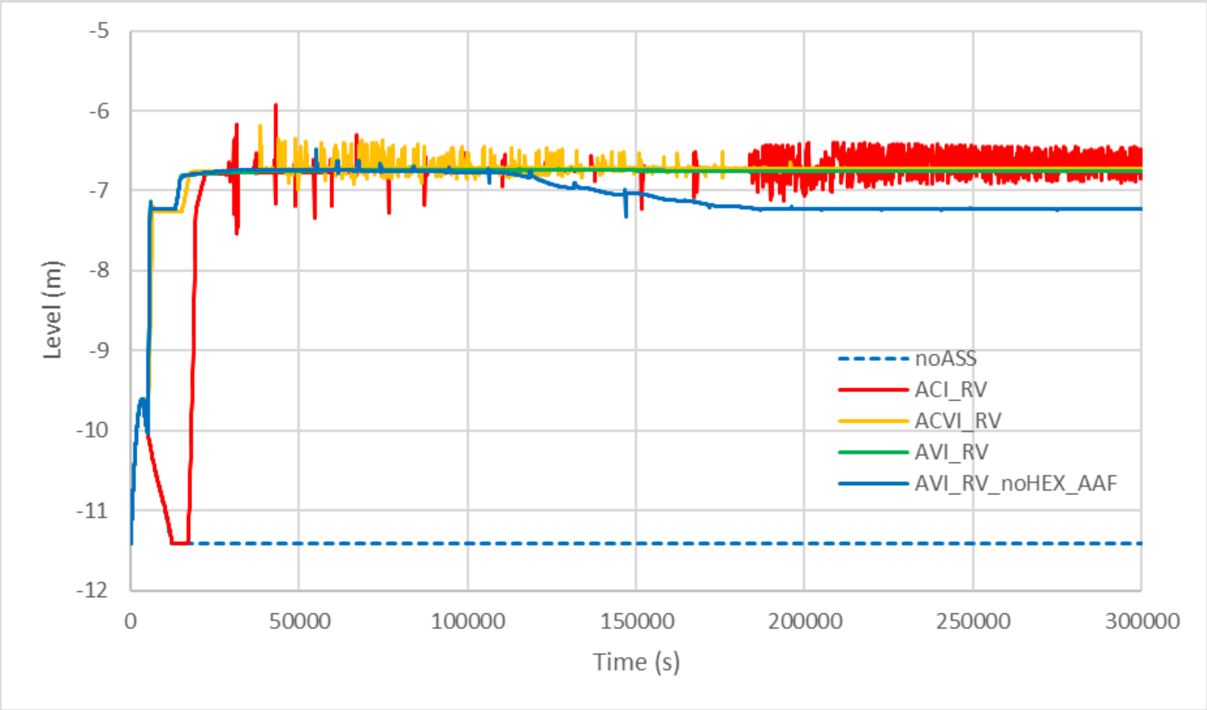

Figure 6: Water level in reactor cavity (CV704).

Figure 7 and Figure 8 show the radial and vertical progression of the core melt in the cavity. It may be observed that in the unmitigated scenario noASS the MCCI progresses steadily, whereas in all other scenarios it stops after some time. In the scenarios ACVI_RV and AVI_RV, where the molten core remains flooded all the time the MCCI stops after only about 0.2 m of concrete is eroded in radial direction and about 0.4 m in vertical direction. In scenario ACI_RV, where the molten core dries out for more than one hour before it is flooded again, the extend of the MCCI is significantly larger, especially in vertical direction where more than 1.2 m of concrete is eroded. The results of scenario AVI_RV_noHEX_AAF lie in-between the results of scenarios where the molten core remains flooded all the time (ACVI_RV, AVI_RV) and the results of the scenario when the molten core dries out for some time (ACI_RV). Small differences in the flooding conditions influence the characteristics of the molten core concrete mixture which influence the MCCI. Consequently, the MCCI is very sensitive on the flooding conditions.

4 CONCLUSIONS

An analysis of unmitigated and mitigated station black out with large break loss-of-coolant accident scenarios in the Krško NPP was performed with MELCOR 2.2 considering mitigation measures for heat removal from the containment solely by the design extension conditions alternative safety systems. The main focus was given to the molten core concrete interaction. It turned out that the molten core concrete interaction in the reactor cavity can be stopped if the molten core is flooded soon after it is released from the failed reactor vessel. The extend of the molten core concrete interaction is very sensitive on the flooding time. In the scenarios, where the water is injected into the reactor pressure vessel and flows through the failed reactor vessel directly into the reactor cavity, the molten corium in the reactor cavity remains flooded all the time. In the scenario, where the water is injected solely through the containment sprays there is a time delay before the water flows into the reactor cavity, because the water in the containment has first to rise up to the level of the ventilation duct, which connects the containment with the reactor cavity. Consequently, the reactor cavity dries out for more than one hour before the spread molten core is flooded again, resulting in a much more extensive molten core corium interaction.

In the loop of the reactor coolant system where the large break occurs a natural circulation of the atmosphere develops, which transfers heat from the containment atmosphere to the water in the corresponding steam generator. In one scenario it was assumed that the water in the steam generators is replenished with the alternative auxiliary feedwater system. It turned out that the heat transfer by the steam generators is not enough to remove all the residual heat from the molten core and to stabilize the severe accident. Thus, in this scenario the containment atmosphere was periodically released through the passive containment filtered venting system, like in the unmitigated scenario. In all mitigated scenarios with operable alternative residual heat removal system heat exchanger the severe accident could be stabilized and no releases into the environment occurred.

The performed study revealed that for a successful mitigation of a severe accident it is important to activate the available safety systems as soon as possible.

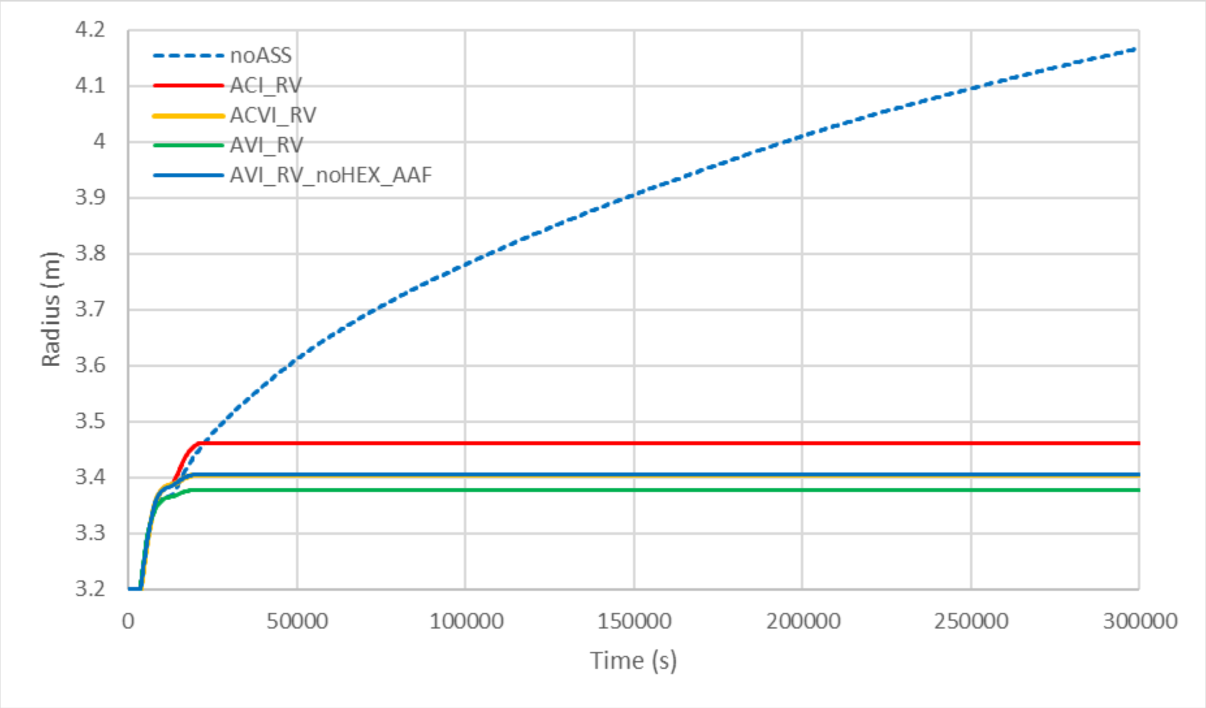

Figure 7: Radius of eroded reactor cavity (CV704).

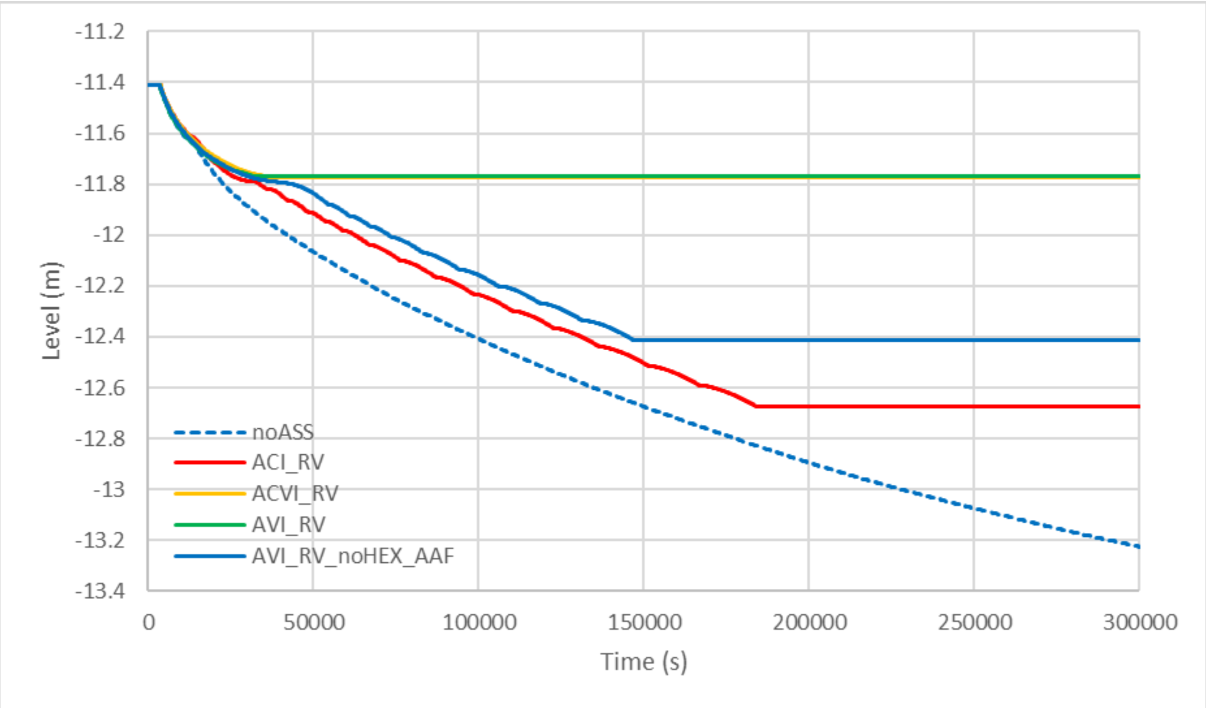

Figure 8: Bottom level of eroded reactor cavity (CV704).

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