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Severe Accident Simulations Dedicated to the SAMG Decision-Making Tool Demonstration

Piotr Darnowski, Piotr Mazgaj
Warsaw University of Technology, Institute of Heat Engineering, Poland
piotr.darnowski@pw.edu.pl

Ivica Bašić, Ivan Vrbanić
APoSS d.o.o., Croatia
basic.ivica@kr.t-com.hr

Maciej Skrzypek, Janusz Malesa
National Centre for Nuclear Research, Poland
maciej.skrzypek@ncbj.gov.pl

Ari Silde, Jarno Hiittenkivi
VTT – Technical Research Centre, Finland
ari.silde@vtt.fi

Luka Štrubelj
Gen-Energija, Slovenia
luka.strubelj@gen-energija.si

ABSTRACT

The paper presents preliminary severe accident (SA) simulations performed to generate a database of plant states dedicated to be used with Severe Accident Management Guidelines Decision Making Tool (SAMG DM). The novel software is being developed in the framework of the NARSIS Horizon-2020 research project. It is intended to be a supporting tool for the SAMGs implementation, Emergency Preparedness and selection of Severe Accident Management (SAM) strategies. Simulations were performed with MELCOR 2.2 integral computer code for generic Nuclear Power Plant (NPP) with Gen-II Pressurized Water Reactor (PWR). The database covers results for parameters important for both in-vessel and ex-vessel phases of different accident scenarios. Two general types of scenarios are considered in the database: low-pressure and high-pressure sequences. In this paper, a comparison was prepared for two base case low-pressure scenarios, that is hot leg and cold leg LB-LOCAs without safety injection. Sensitivity simulations were performed with and without Containment Filtered Venting (CFV) as it substantially influences the containment performance. Both accidents are characterized by rapid progression with core meltdown within 2 hours and containment failure within 40 hours and eventual venting with CFV after 13-15 hours. It was observed that for the cold-leg break, the Reactor Pressure Vessel failure occurs earlier only by ~5 minutes. On the contrary, the containment failure is predicted to occur three hours earlier for the hot-leg LOCA.

1 INTRODUCTION

In the framework of the NARSIS Horizon 2020 research project [1], the Work Package 5 (WP5) is focused on the development of the Severe Accident Management Guidelines Decision Making Tool (SAMG DM). The software will be dedicated to simplify and support implementation, assessment and execution of the SAMGs, but also Emergency Preparedness and evaluation of different Severe Accident Management strategies.

A generic Gen-II NPP was defined in the framework of the WP5 [2] , and it is used for the SAMG DM tool development and demonstration. A sub-task of the WP5 is devoted to preparing a database with SA progression results to be applied as input data for the SAMG DM tool. The database covers major plant parameters like pressures, temperatures, water levels and timing of crucial events, including a transition to SAM regime, vessel rupture or containment failure. The matrix of ten representative scenarios was defined, and it covers both low pressure and high-pressure sequences. Low-pressure scenarios are variants of Large Break Loss of Coolant Accident (LB-LOCA), cold-leg, hot-leg breaks and different safety injection and Engineering Safety Features (ESF) activation options. For high-pressure scenarios, it covers Station Blackout (SBO) sequences with/without seal LOCA, steam generator tubes rupture, surge line rupture, different valves positions, ESF and recovery setup.

This paper presents the NPP model, results for steady-state validation and preliminary results for base case LB-LOCA simulations. Considered initiating events (IE) are Cold Leg (CL) and Hot Leg (HL) LOCAs without safety injection. Both in-vessel and ex-vessel phases are studied, and the emphasis was put on the assessment of the parameters important for the database. In consequence, thermal-hydraulics, core degradation with RPV failure and containment failure were investigated.

2 GEN-II NPP MODEL

2.1 Plant Definition and MELCOR Model

The studied plant is a 1994 MWth two-loop PWR reactor with a large dry-containment building. MELCOR 2.2.9541 was used as a SA integral code [2]. The core model is presented in Figure 1 (Right), the COR package nodalization has six radial rings and 17 axial levels, where five levels are included in the lower plenum model. The lower head was modelled by six segments and each divided into ten layers. A simplified nodalization was used for the RPV model with only four control volumes (CVs), one for lower plenum (LP), upper plenum, downcomer and core region (Figure 1 (Left)). The primary side of the Reactor Coolant System (RCS) was modelled with 28 CVs, including piping, pressurizer (PRZ), safety injection systems (SIS) which cover accumulators (ACC), low-pressure injection system (LPIS) and high-pressure injection system (HPIS). Two ACCs were modelled with dedicated ACC MELCOR package. The secondary side, including steam generators (SG), steam lines, turbine, auxiliary feedwater (AFW) and feedwater (FW), were all modelled with 16 CVs (Figure 2). The RCS model has 63 flow paths (FL package) including MSIV valves, PORV valves for SGs and PRZ, PRZ SV valves, SG safety valves, turbine valves, breaks, SIS and Reactor Coolant Pumps (RCP). The RCS model covers about 50 heat structures (HS package).

The containment building model is presented in Figure 2. It consists of 12 CVs, 25 flow paths and 15 heat structures. The model includes containment failure due to the pressurization but also passive Containment Filtered Venting (CFV) system. Containment sprays, fan coolers, and a set of Passive Autocatalytic Recombiners (PARs) were also modelled. The default MELCOR PAR type was applied with a dedicated flow model given by proper control functions. A single cavity for MELCOR CAV with TP and FDI packages to model ex-vessel phenomena was also applied with CORCON limestone concrete [2]. The corium was modelled as a homogenous mixture with default code's setup. The RN package was not used in this work as radionuclides behaviour was not studied. What is important, deactivation of RN package is considered to be more conservative for the core degradation and to be less conservative for the containment. When the RN package is active, it is the opposite.

Totally, more than 150 control functions (CF) with over 30 tabular functions (TFs) were developed to model all crucial plant control systems, material properties, scenario setup and steady-state simulations.

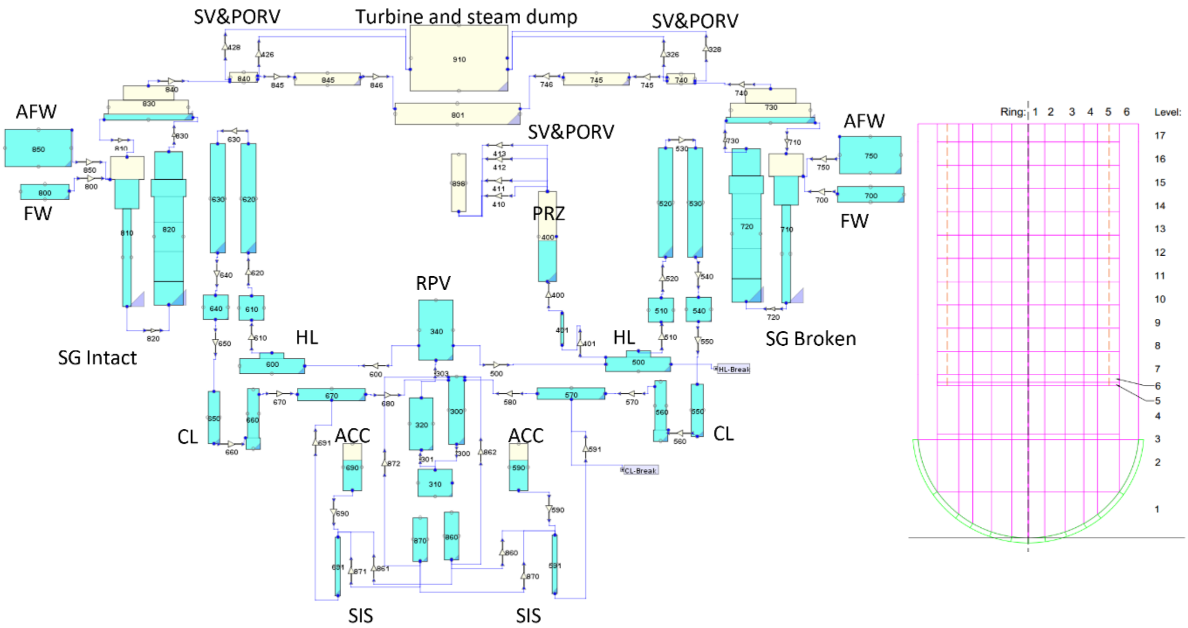

Figure 1 Core model (Right) and Reactor Coolant System (Left) nodalization.

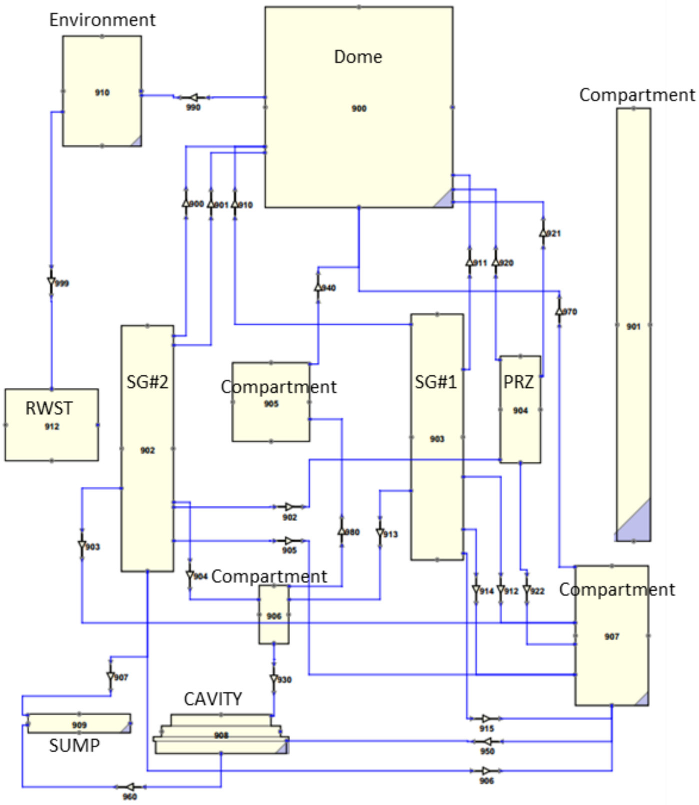

Figure 2 Containment building nodalization.

2.2 Steady-state validation

The dedicated steady-state control system was developed, and it covers mainly PZR heaters, sprays but also dedicated CFs to initiate and accelerate steady-state stabilization. Plant parameters are compared with reference design data ([2]) in Table 1. Steady-state simulations started 10000 seconds before the transient and a stable state was achieved. The highest differences are present for RPV temperatures, and it can be explained by the simple RPV nodalization. Moreover, FW conditions were controlled to obtain proper steady state, and it affects conditions in the RCS.

Table 1 Comparison of the steady-state parameters with the reference design. (value/value) – denotes value for (I) intact loop and (B) broken loop with pressurizer.

Parameter	Units	Calculated	Reference [2]	Difference, [%]
NSSS Power	[MWth]	2000.0	2000.0	0.0
PZR pressure	[bar]	155.00	155.10	-0.064
SG outlet pressure (B/I)	[bar]	65.40/65.41	65.7/65.7	-0.456/-0.448
SG outlet temp. (B/I)	[K]	554.50/554.50	554.35/554.35	0.027/0.028
SG outlet flow, sum	[kg/s]	1091.83	1090.00	0.168
SG water mass (B/I)	[kg]	47921.35/47921.50	48000/48000	-0.164/-0.164
SG FW temp.	[K]	492.95	492.55	0.081
SG FW flow	[kg/s]	1092.00	1088.0	0.368
RPV inlet temp.	[K]	566.32	561.25	0.904
RPV outlet temp.	[K]	603.57	599.85	0.620
Core inlet temp.	[K]	566.33	561.25	0.905
RPV average temp.	[K]	584.95	580.55	0.757
RCS flow (B/I)	[kg/s]	4561.60/4561.00	--	--
Core total flow	[kg/s]	9122.16	9133.6	-0.125
PZR level	[%]	49.86%	50%	-0.287

2.3 Accident simulations - assumptions

Two IEs are considered, namely: Cold Leg (CLL#1) and Hot Leg (HLL#1) Guillotine Double Ended LB-LOCAs without SIS and with loss of other systems to obtain unfavourable conditions. The main assumptions are:

- HPSI and LPSI not available – RWST and recirculation mode
- Containment sprays, AFWs and PRZ PORV not available.
- PARs, Accumulators, SG PORVs, SG SVs, PZR SVs available.
- CFV system was tested for the inactive (base) case and active (sensitivity) case.
- Containment failure occurs at 2x design pressure.
- Transient starts at time 0.0 s. Reactor trip (RT) occurs at proper I&C signal, low PRZ pressure. MSIV, FW system, and RCPs are tripped at RT with delays.
- 60 hours (216000s) simulation time.

3 RESULTS AND DISCUSSION

Crucial events for scenarios are listed in Table 2. Core water levels, water inventories in RCS and integrated accumulator and break flows are presented in Figure 3. RCS and containment pressures are presented in Figure 4.

The water level reaches TAF immediately for both scenarios, RT occurs after three seconds due to the low PRZ pressure. For CCL#1 the core is uncovered shortly after the break, whereas for HLL#1 it takes 12 minutes (see Figure 3, Left). It can be observed that the accident progression for the CCL#1 is faster than for HLL#1 (see Figures 3). This occurs partially because one accumulator is connected to the CL where the break is located, and some water escapes directly to the containment (Figure 3). Moreover, different flow through the core has an impact as a CL break promotes flow reversal.

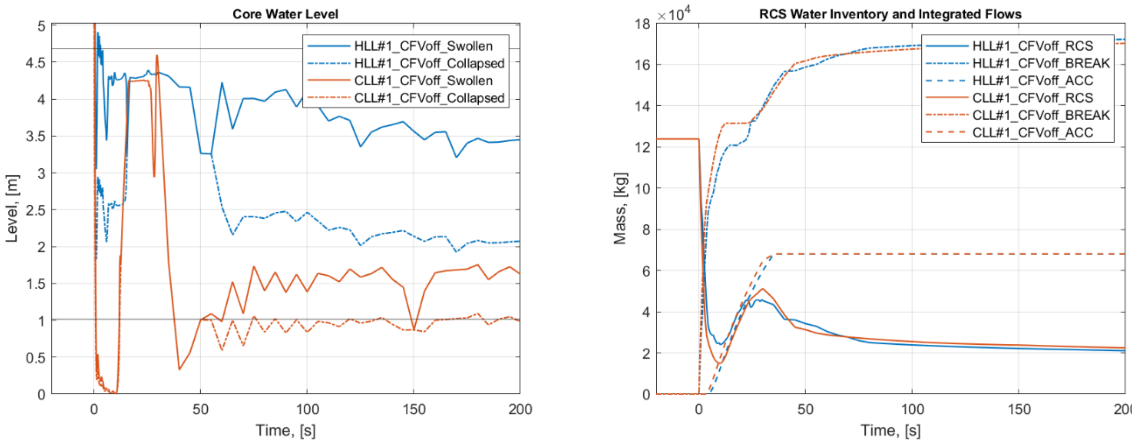

Figure 3 Left – Core water levels, black lines indicate TAF and BAF. Right – RCS water inventory, integrated break flow and water transferred by accumulators.

Table 2 Key events for studied scenarios.

Parameter	Scenario, time [s]	
	HLL#1	CLL#1
Reactor trip	2.84	2.99
Water level <TAF	0.72	0.39
Water level <10% of active core	765.08	0.77
Core exit temperature 650 °C - SAMG signal	515.09	2.70
Core exit temperature 2499 K - Core melting signal	1275.0	382.6
Debris relocation to the lower plenum	3596.6	3188.0
RPV failure	5052.8	4792.1
Containment failure (if no CFV)	144720.0	155598.0
CFV activation (if CFV active)	49680.1	55650.2

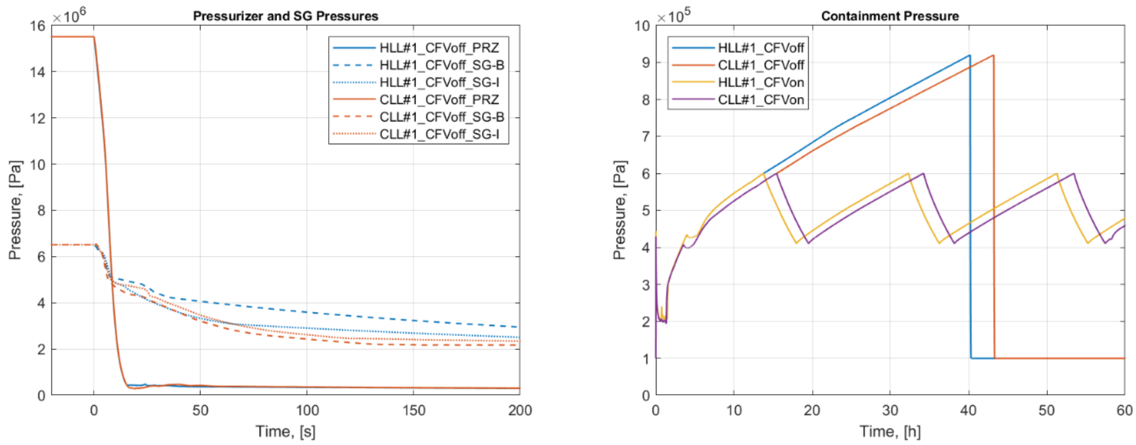

Figure 4 Left – PRZ and SG pressures. Right – Containment dome pressure.

Figure 5 (Left) presents selected fuel temperatures, and we can observe that intensive heating is slightly delayed by water from accumulators for HLL#1, but it is not the case for CLL#1 (also Figure 3 (Right)).

The transition to the SAMG regime is initiated by 650 °C signal at the core exit. For CLL#1 scenario it happens before RT, but for HLL#1 it occurs 8 minutes after the IE (Table 2). Core exit temperature 2499 K signal indicates core melting initiation. For CL break, this signal occurs 6 min after IE, and for HL 21 min. Initially, core signals (650 °C and 2499 K) were measured with the CVH package core or upper plenum. However, for HLL#1 scenario 2499 K signal was observed after massive relocation to the lower plenum, which is not realistic. The issue was solved using COR-TSCV variable, which indicates the fluid temperature in the COR package [3], [4].

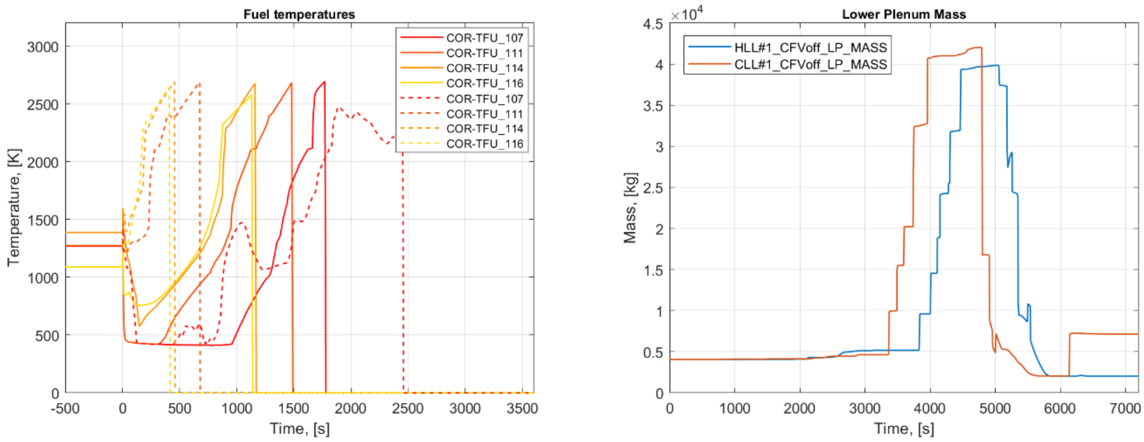

Figure 5 Left – Fuel temperatures for the central ring and selected levels for HLL#1 (solid line) and CLL#1 (dashed line). Right – Total mass of material in the lower plenum (levels 1,2).

Figure 5 (Right) shows the mass of materials in the LP (debris, corium and other). In the case of CLL#1, corium relocation occurs earlier, 53 min after accident initiation and RPV failure occurs 24 min later. For HLL#1 relocation occurs 60 min after IE and RPV failure after further 27 min.

In the case of non-active CFV, the containment failure occurs ~39 h and ~42 h after corium transfer to the cavity for CLL#1 and HLL#1 respectively – see Figure 4 (Right). For sensitivity cases, CFV operation starts ~13.8 h (HLL#1) and ~14.5 h (CLL#1) after IE, when containment pressure is ~6 bar, and it reopens (~4 bars) three times during the studied time interval.

Figure 6 (Left) presents hydrogen and oxygen concentrations in the containment dome. The maximum value for H_2 is reached during the in-vessel phase when the production rate is very fast, and oxygen concentration follows the hydrogen. Afterwards, PARs steadily reduce hydrogen inventory, and even during an intensive ex-vessel phase, the H_2 removal rate is higher than production for most of the time. An exception is only a small peak after ~8 h. Figure 6 (Right) presents in-vessel hydrogen generation due to the clad oxidation and ex-vessel MCCI (Molten Core Concrete Interactions) related generation of hydrogen and other gases. It can be observed that for HLL#1 oxidation starts later and more hydrogen (~22% more, 182 kg vs 223 kg) is produced before RPV failure, and it can be explained by larger steam/water availability for the hot-leg scenario.

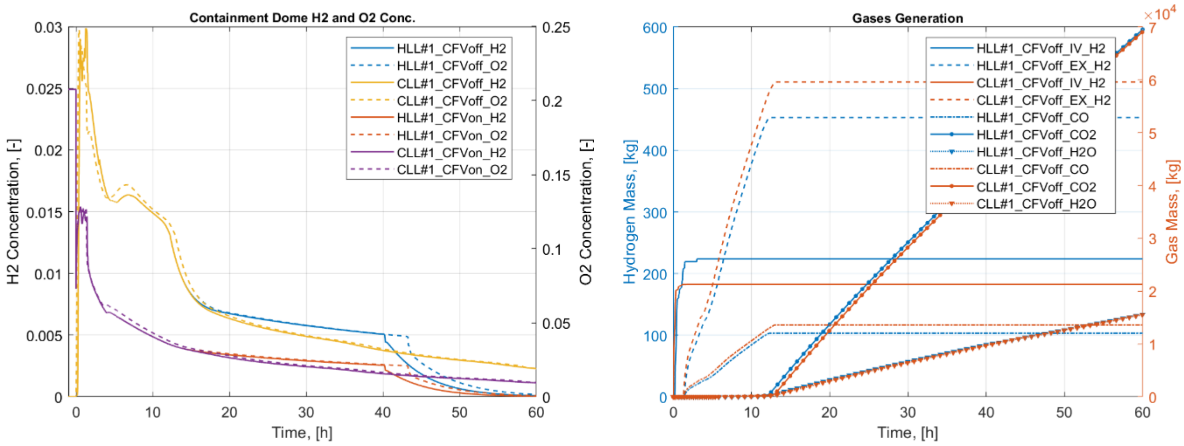

Figure 6 Left - Containment hydrogen and oxygen concentration. Right – In-vessel (IV) and Ex-vessel (EX) H_2 generation and other gases (CO, CO_2, H_2O) produced by MCCI.

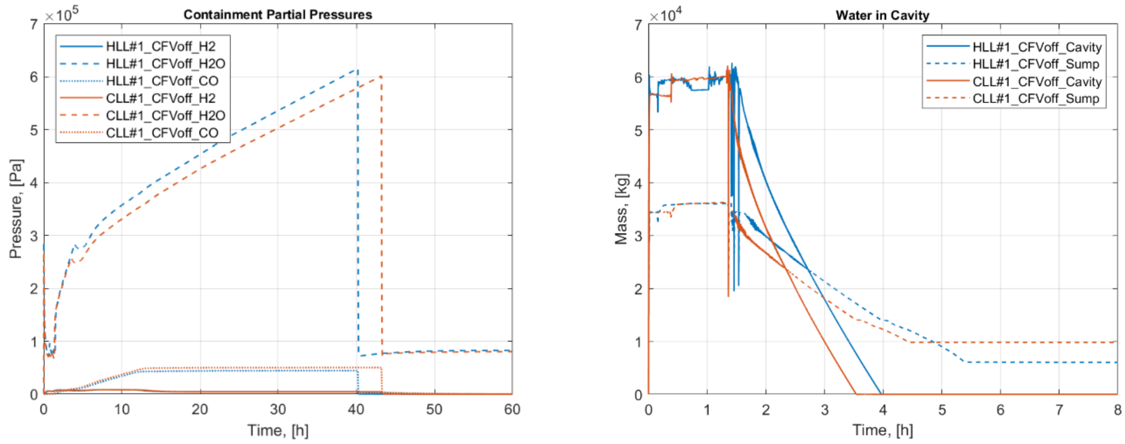

Figure 7 Left – Partial pressures of H_2O, H_2, CO in the containment. Right – Water collected in the cavity and sump.

Differently, in the case of HLL#1, the ex-vessel phase progression is slightly faster, and a 10-15 minutes time difference for the in-vessel phase becomes of secondary importance for the ex-vessel phase, where a 2-3 hours delay to containment failure is present for the base case (see Figure 7 (Left)). After the initiation of the ex-vessel phase, the corium is transferred to the cavity where it interacts with collected water. Before RPV rupture, a similar mass of water is collected in the cavity, but for the cold-leg case, more water is removed from the cavity during a short time period after the transfer process (see Figure 7 (Right)). It is likely due to the violent interactions of water with unoxidized Zr, which is more abundant for CLL#1 (see Figure 6 (Right)). In effect, reduction of the water mass in the cavity leads to earlier dry-out (3.5 h vs 4 h) and less water being turned into steam due to the direct contact with cavity and corium. In consequence, the pressure build-up is smaller, as the steam is the main cause of containment pressure increase. After the dryout, the pressurization rate is similar for both scenarios – see Figure 7 (Left). After about twelve hours of the accident, carbon-oxide and hydrogen generation by MCCI stops, because the corium is fully oxidized. The MELCOR corium mixed-phase (HMX) became pure oxide (LOX), and this occurs slightly earlier for HLL#1, as the transferred corium was more oxidic. Finally, ~700 kg of H_2 and ~13 t of CO were produced – see Figure 6 (Right). In the further part of the accident, only carbon dioxide and water vapour are produced by MCCI until the end of simulations (Figure 6 (Right)). After 60 hours, overall 70 t of CO_2 and ~15 t of vapour were released from concrete to the containment. These conclusions are based on cases without CFV, but the same reasoning can be drawn for cases with active CFV (see Figure 4 (Right)).

4 CONCLUSIONS

The MELCOR 2.2 Nuclear Power Plant input-deck dedicated for the accident progression database for the Severe Accident Management Decision Making (SAMG DM) tool was developed. A stable steady state was obtained with conditions comparable with reference plant data with a reasonable margin. Preliminary simulations for low-pressure scenarios being variants of the Large Break Loss of Coolant Accident (LB-LOCA) for Hot Leg (HL) and Cold Leg (CL) were prepared and discussed in this report. Sensitivity simulations for containment response with and without the Containment Filtered Venting (CFV) were presented. Further research will focus on simulations of other more complex scenarios.

Two considered LB-LOCA accidents lead to very rapid core degradation and meltdown. For the accident phase before the Reactor Pressure Vessel (RPV) failure (in-vessel phase), the CL sequence progression is slightly faster than for the HL sequence, and the observed difference is about 5 minutes. On the contrary, the HL scenario has faster progression during the ex-vessel phase with about 3 hours earlier containment failure. For sensitivity cases without CFV, the containment fails after more than 40 hours, and for variants with CFV available, the venting procedure starts after more than 12 hours. Both scenarios can be considered as possibly among of the most rapid events to be considered in the NPP with a PWR reactor.

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