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MELCOR SIMULATION OF A SEVERE ACCIDENT SCENARIO DERIVED FROM A SMALL BREAK LOCA IN A TYPICAL PWR WITH PASSIVE AUTOCATALYTIC RECOMBINERS

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ABSTRACT

This work presents the simulation of a severe accident scenario in a referential model of pressurized water reactor, which came about from a rupture of 20cm^2 in a cold leg of a reactor cooling system. The simulation was carried out on the MELCOR code using a model elaborated by the Global Research for Safety – Germany, with the passive autocatalytic recombiners implemented in almost every compartment in the containment. The efficacy and effectiveness of this well-known mitigating measure of severe accident management are demonstrated by means of a comparison with the case previously simulated without this measure using the same model. This referential reactor is important and very useful for the independent analysis of severe accidents in the Brazilian Angra 2 nuclear power plant in virtue of the similarity between both of them, so that after some proper modifications on this referential reactor's model, it could be applied for the study of severe accidents in the other. In this sense, the result presented in this work is to be taken as an important reference for the severe accident analysis of Angra 2.

1. INTRODUCTION

Severe accidents [1] in a nuclear power plant (NPP) are those classified as beyond design basis accidents (BDBA) involving a considerable damage in the reactor core as consequence of a series of events of fails, or malfunctions, of some components of the whole system. For most of the pressurized water reactors (PWR), one of the most likely combinations of these events set called 'scenarios' is leaded by a small break in the primary reactor cooling system (RCS). The accident consisting of leakage of coolant through a small break in the RCS is called 'Small Break Loss-Of-Coolant-Accident' (SBLOCA), and the scenario presented here in particular, is a SBLOCA that is brought about by a break of 20cm^2 in a cold leg of the primary coolant system of a typical PWR. The magnitude of this break by itself is too small to ensure a core meltdown in the PWR; hence some other events are assumed to follow, or to take place simultaneously with, the rupture in order to complete a severe accident scenario. Henceforth, these additional events are called 'boundary conditions' of the problem. Moreover, two mitigating measures of severe accident management were implemented in this simulation in order to verify their efficacy and effectiveness during the process of the accident; namely, the Filtered Containment Venting System (FCVS) [2], whose role is of depressurization of the containment and minimizing of the fission products release to the environment; and Passive Autocatalytic Recombiners (PAR) [3], which is for reducing of hydrogen concentration inside the containment to prevent their deflagrations.

Since Fukushima Daiichi disaster [4] in 2011, submission of reports about the computer simulations of some viable severe accident scenarios has become a highly demanding requirement for licensing of any nuclear system by the International Atomic Energy Agency (IAEA). In keeping with this demand, the *Comissão Nacional de Energia Nuclear* (CNEN) has also begun to require the development and implementation of severe accident management programs (SAMP) for the NPPs in operation (or in construction) in Brazil. In order to meet this requirement, Electronuclear (ETN), a utility company in charge of operation of the NPPs in Brazil, presented to CNEN a report about the deterministic safety analysis for Angra 2. This analysis that had actually been carried out by the vendor of the Angra 2, AREVA, was based on simulations performed by means of MELCOR code [5, 6]. Notwithstanding, CNEN decided to make itself an independent analysis using the same code with the collaboration of a German company, Global Research for Safety (GRS). GRS has provided to CNEN a model of a typical PWR; namely, DWR-1300 (DWR – *Druckwasserreaktor*, i.e., PWR), whose design is similar to that of Angra 2. Several changes were made on this model in order to fit it to Angra 2, and the model thus modified is used in this work to reproduce the severe accident of SBLOCA of 20cm^2 in Angra 2 by means of MELCOR code.

2. METHODOLOGY

The modeling of the DWR-1300 that GRS elaborated is shown in two following figures. Figure 1 illustrates the distribution of the compartments inside the containment together with the auxiliary building, while Figure 2 shows the model of the primary and secondary RCS divided into two loops, as a standard practice.

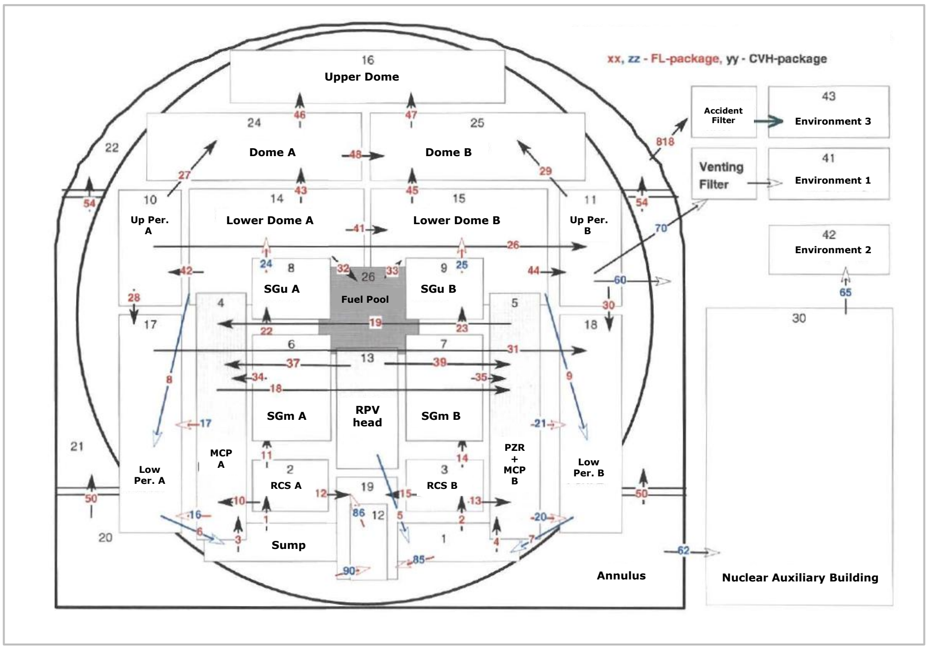

Figure 1: Model of the containment and its auxiliary building. (*source*: GRS)

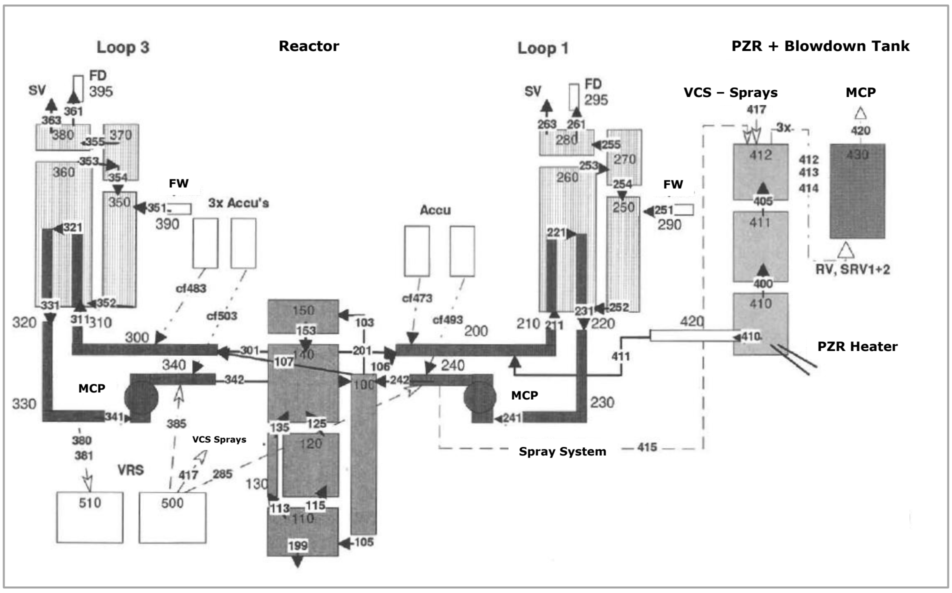

Figure 2: Primary and secondary RCS. (*source*: GRS)

In above figures, the compartments and components of the NPP are illustrated by the numbered control volumes (CV); while some flow paths (FL) and control functions (CF) are indicated by numbered arrows. In Figure 2, the ‘loop 3’ stands for the three loops of the NPP virtually united together to form only one loop; therefore, the associated dimensions had to be properly modified. On the other hand, the ‘loop 1’ is modeled in order to realize the real dimensions. In particular, the pressurizer (PZR) is connected to the loop 1 only, and the rupture is usually assumed to take place there, as it is in this work.

In order to simulate a rupture, an imaginary valve was implemented in the cold leg of the loop 1; more specifically, in the component of number 240 (CV240) in Figure 2. This valve is opened at t = 0s, on simulation time, so that the coolant liquid starts to leak into the compartment represented by the control volume 3 (CV003) in Figure 2. Obviously, the area of the flow path between these two control volumes, CV240 and CV003, is 20cm^2, equal to that of the rupture. It is worthy to mention that the opening of the valve is carried out by mean of a time dependent tabular function (TF), instead of a mere real constant.

2.1. Description of the Scenario: Boundary Conditions

As already mentioned, the rupture by itself may not cause a severe accident due to its small size, so that it will be necessary to take the following boundary conditions into account:

a) Turbine bypass and condenser not available;
b) Emergency core cooling (ECC) injection from the re-fuelling water storage tank (RWST) by safety injection pumps (SIP) and residual heat removal (RHR) pumps available;
c) All accumulators available (cold leg accumulators automatically locked 500s after the ECC signal);
d) Emergency Feed Water System (EFWS) available;
e) Loss of suction from the sump and RHR;

f) Loss of secondary side 100 K/h cool-down;
g) The Primary Bleed and Feed (PBF) not available;
h) The Secondary Bleed and Feed (SBF) not available.

2.2. The Measures for Mitigation: PAR and FCVS

Several units of Passive Autocatalytic Recombiners (PAR) were implemented in different compartments of the containment for the hydrogen concentration decrease during the accident in order to prevent the explosions of significant scale for the containment integrity. A PAR, as the name alludes, does not need external power supply to operate: it just comes into action as soon as the concentration of hydrogen attains certain level. The effect of the presence of a PAR in a arbitrary CV can be quantified by means of Equation 1 below [7], which is an empirical correlation used to calculate the quantity of burnable gases inside the control volume.

$$\frac{dm_{br}}{dt} = N \cdot M_{br} \cdot \eta \cdot \eta_{O_2} \cdot min\left(c_{br}, 2c_{O_2}\frac{c_{br}}{c_{H_2} + c_{O_2}}, c_{max}^{br}\right) \cdot (k_1 \cdot p + k_2) \cdot tanh\left(c_{H_2} + c_{O_2} - c_{min}\right) \tag{1}$$

where:

m_{br}	- mass of H_2 or O_2 [g],
M_{br}	- molecular weight of burnable gas [g/mol],
N	- number of recombiners,
η	- efficienty of burnable gas,
η_{O_2}	- oxygen efficienty,
p	- pressure [bar],
c_{br}	- actual volume concentration [Vol-%] of gas component, H_2 and O_2,
c_{min}	- minimum volume concentration for recombination [Vol-%],
c_{max}^{br}	- depletion-limiting constant,
k_1, k_2	- coefficients of the recombiner, dependeing on recombiner's type, in [mol/s bar] and [mol/s], respectively.

Equation 1 is for one type of PAR and two coefficients, k_1 and k_2, depend on type of each PAR. However, some CV associated to the containment compartments have no PAR inside; while there are others that have more than one PAR, even of distinct types. When more than one recombiner are present in a single CV, the coefficients k_1 and k_2 will end up being replaced in Equation 1 by the arithmetic sum of the respective coefficient of each type of PAR, multiplied by its number (N) in the CV. For instance, if a CV is equipped with N_A of PAR of the type A, N_B of the type B, N_C of the type C, and so on, then the sum of the respective coefficients, k_1' and k_2', are given by Equations 2 and 3 below:

$$k_1' = N_A \cdot k_{A,1} + N_B \cdot k_{B,1} + N_C \cdot k_{C,1} + \cdots \tag{2}$$

$$k_2' = N_A \cdot k_{A,2} + N_B \cdot k_{B,2} + N_C \cdot k_{C,1} + \cdots \tag{3}$$

where, $k_{A,1}$ and $k_{A,2}$ stand for the coefficients of the type A; the same for $k_{B,1}$ and $k_{B,2}$ and so on.

Thus, the overall rate inside a CV given by Equation 1 will be calculated putting Equations 2 and 3 into k_1 and k_2, respectively, in the same equation. Table 1 shows the disposition of the PARs in the containment and the respective values of the effective coefficients, k_1' and k_2' for each CV, calculated by Equations 2 and 3. As it can be observed, there are four types of PAR; namely, 1500T, 960, 380T and 320; their associated coefficients (k_1 , k_2) are: (0.137, 0.167), (0.061, 0.074), (0.031, 0.037) and (0.010, 0.012), respectively. It is worth pointing out that the individual coefficients above for each type of PAR, as well as the effective ones for each CV, shown on Table 1, are given in units of [kg/(s bar)] and [kg/s].

Table 1: Disposition of the PARs and the effective coefficients in each CV.

CV	PAR TYPE				k_1' [kg/(s bar)]	k_2' [kg/s]
	1500T	960	380T	320		
001	-	-	-	-	0.000	0.000
002	-	-	-	-	0.000	0.000
003	-	-	-	-	0.000	0.000
004	1	1	1		0.229	0.278
005	1		1		0.168	0.204
006	4	1	1	3	0.670	0.815
007	3	2		2	0.553	0.673
008		2	1	1	0.163	0.197
009	1		1	1	0.178	0.216
010	2	1			0.335	0.408
011	2				0.274	0.334
012	-	-	-	-	0.000	0.000
013	1				0.137	0.167
014	4		1		0.579	0.705
015	5		1		0.716	0.872
016	-	-	-	-	0.000	0.000
017		3		3	0.213	0.258
018	1	2	1	1	0.300	0.364
019					0.000	0.000
024		2			0.122	0.148
025	2				0.274	0.334

Regarding the Filtered Containment Venting System (FCVS) [2], it aims to ward off catastrophic failure of the containment structure by discharging steam, air and non-condensable gases such as hydrogen into the environment in order to reduce the pressure inside the containment. Radioactive release can be significantly reduced by the implementation of filtering system in ventilation line. The FCVS is activated as soon as the increasing pressure attains the set point for loading; and deactivated, when the decreasing pressure reaches the set point for unloading. In this work, the set points were configured as 6bar for loading and 3,5bar for unloading. This kind of system can be implemented in the simulation using the hysteresis function the MELCOR code deals with.

The radionuclide package [5], of the MELCOR code contains a simple filter model whose efficiency is defined by the global decontamination factor (DFG)[1], whose value is chosen by

[1] DFG≡ (mass entering filter) / (mass not removed by filter) ≥ 1. 0

the user. The DFG assumed in this simulation for both aerosol and vapor filters was 1000, which corresponds to 99.9% of filtering of the respective radionuclide.

3. RESULTS

The core meltdown starts at 6h26m, as can be seen in Figure 3, and it is completely melted at 8h16m. According to Figures 4 and 5, the maximum temperature of claddings of fuel element is 2225°C.

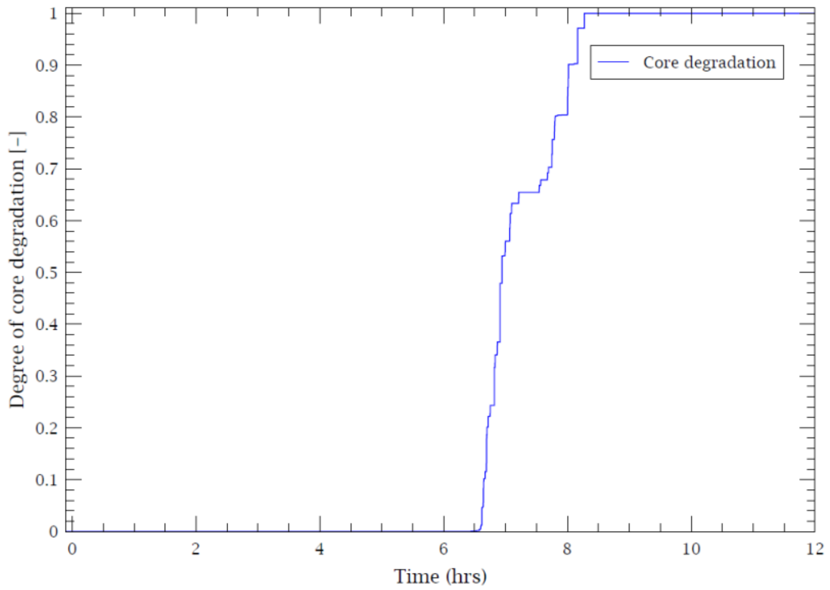

Figure 3: Degree of core degradation.

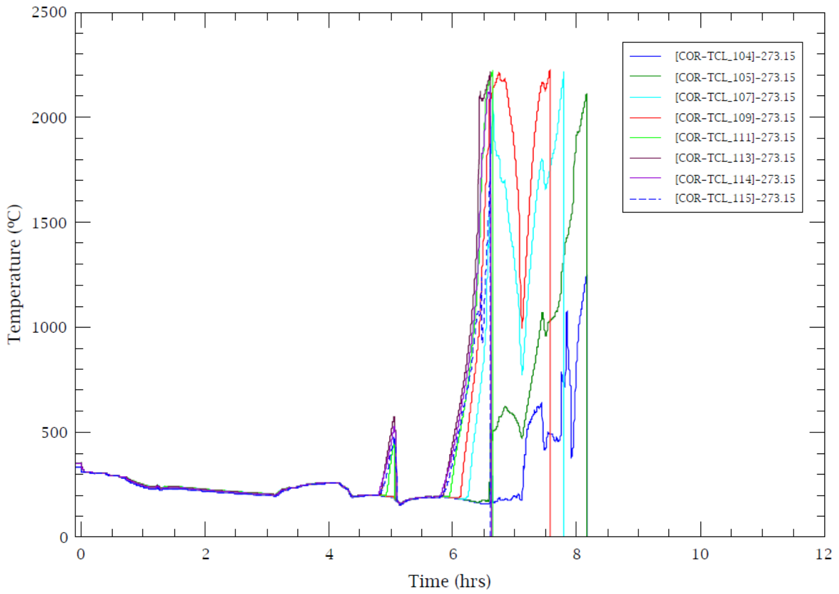

Figure 4: Cladding temperature for the innermost ring.

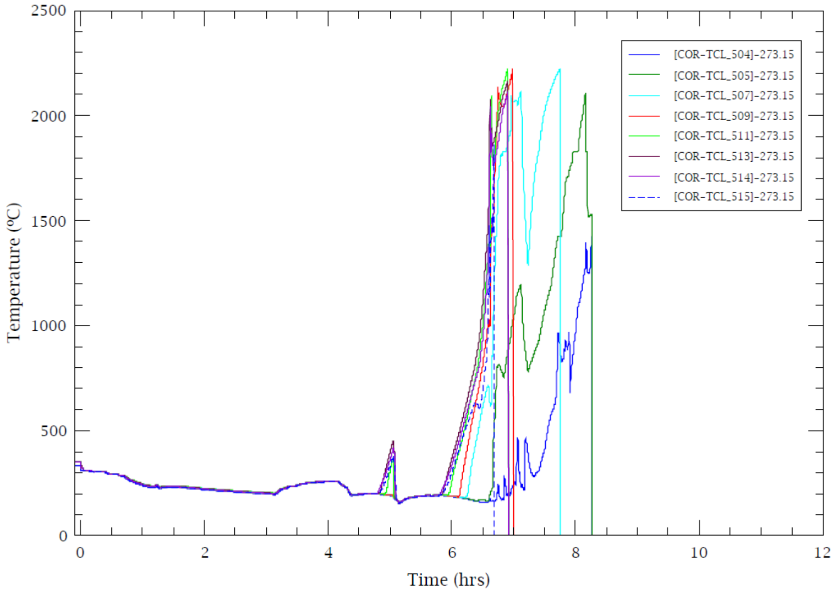

Figure 5: Cladding temperature for the outermost ring.

It can be observed in Figure 6 that the pressure of the RCS rises rapidly around 1h18m and remains stagnant up to 3h7m. This stabilization around 80bar occurs due to the equalization of the leakage rate with the rate of the injection of Safety Injection pumps (SIP), and it continues until the Safety Injection (SI) starts due to the emptying of the Refueling Water Storage Tanks (RWST). Then, the pressure of the RCS drops to 50bar and continues to decrease until 3h25m; at which point, the water saturation begins and, thus, the pressure increases. Since the evaporation rate decreases with the drop of the water level of the core, the pressure of the RCS falls again until it equals the pressure of the accumulators, namely 26.5bar.

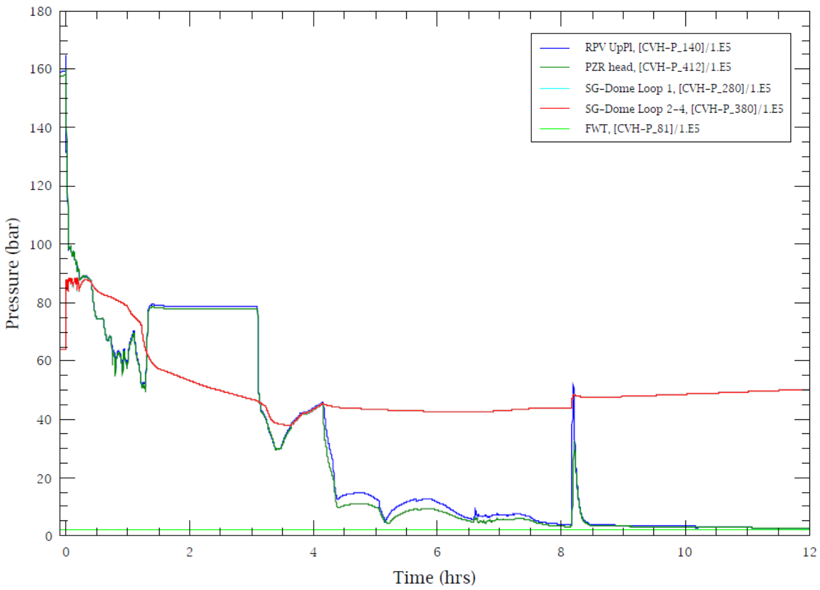

Figure 6: Pressure in RPV, PZR, SG and FWT.

The pressure of the RCS is subsequently determined by the pressure of the accumulators, which decreases slowly from 4h23m to 10h10m, at which time the failure of the wall of the Reactor Pressure Vessel (RPV) takes place. A significant peak of pressure can be observed at 8h13m due to the relocation of the corium into the residual water. In addition, the pressure of the secondary system increases immediately to 88.3bar after the rupture; this happens due to the unavailability of the turbine by-pass and condenser. Afterwards, the safety valve is opened.

The water levels in the RCS, pressurizer (PZR) and steam generator (SG) are shown in Figure 7. The water levels in hot leg of both loops are kept practically constant during the first 3 hours, until the RWST get completely empty; after that, the levels start to decrease.

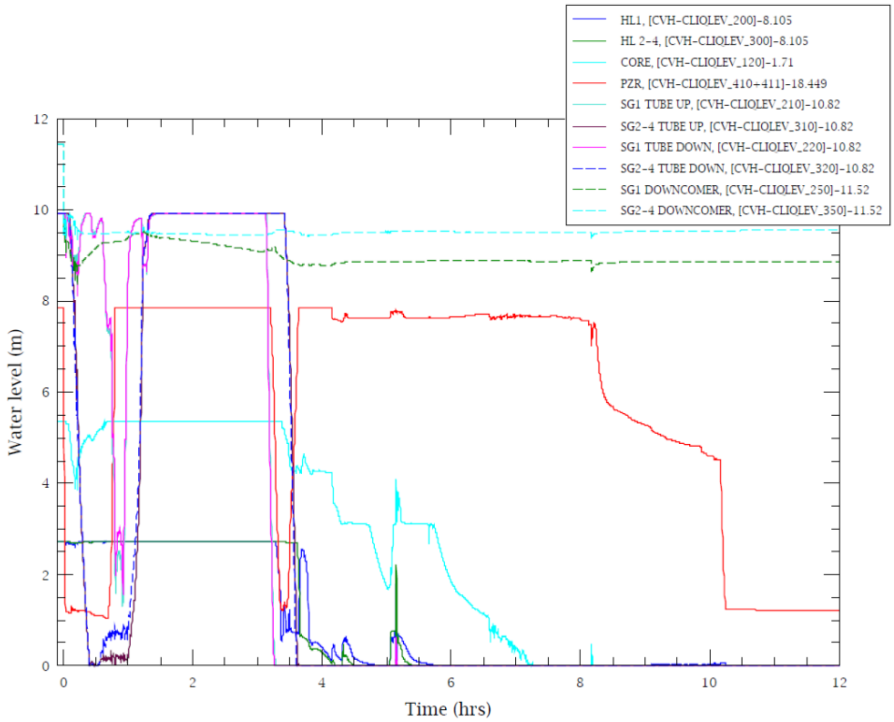

Figure 7: Water level in RCS, PZR and SG.

When the temperature of the hottest cladding reaches 880°C, a significant oxidation take place, and thus, hydrogen production begins, as it is shown in Figures 8 and 9. According to the Figure 8, the total mass of hydrogen produced by the oxidation of the metallic structure of the core by steam is around 733kg. Notwithstanding, most of hydrogen and carbon monoxide are generated by the Molten Core - Concrete Interactions (MCCI) [8], which starts at around 10h10m. On the other hand, the total amount of hydrogen and carbon monoxide recombined by the PARs can be obtained from the Figure 9, namely 1954kg and 2411kg, respectively.

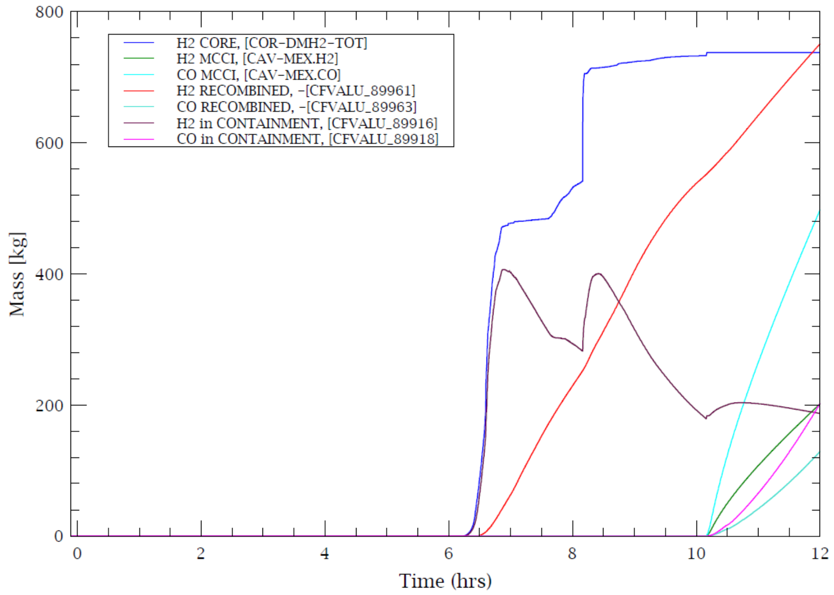

Figure 8: Generated and recombined and residual H_2 and CO mass (short-term)

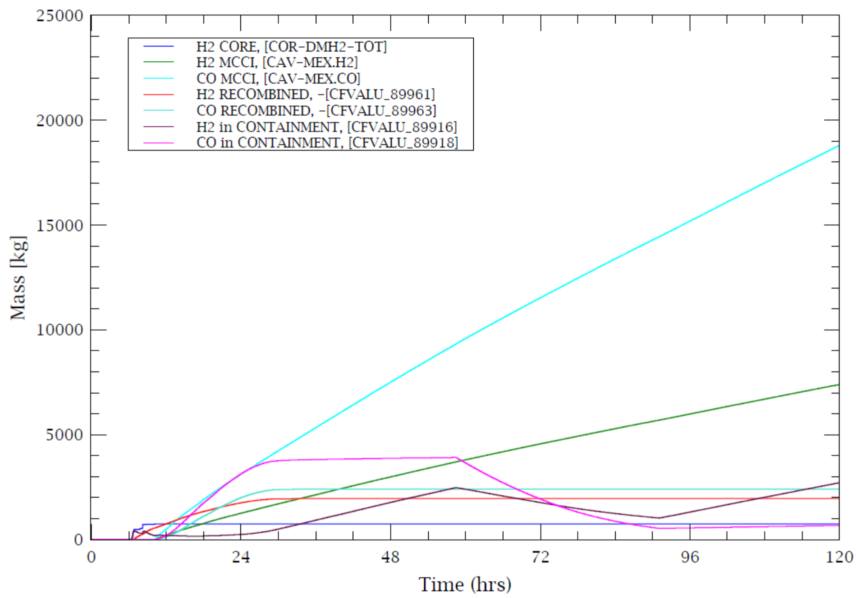

Figure 9: Generated and recombined and residual H_2 and CO mass (long-term)

The atmospheric temperatures of the containment are shown in Figures 10 and 11, and in these figures it can be noted that the temperature of the reactor pit has atypical values, i.e., around 8h19m52, which reflect some deflagrations of hydrogen occurring inside the containment. Obviously, this range would have been much messier if there were not PARs deployed in the containment [9] as it can be seen at the end of this section.

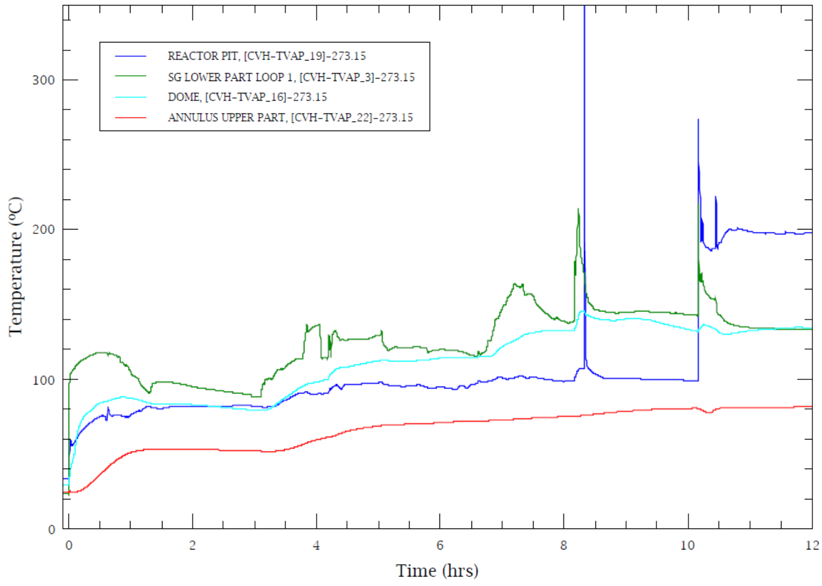

Figure 10: Containment atmosphere temperatures (short-term)

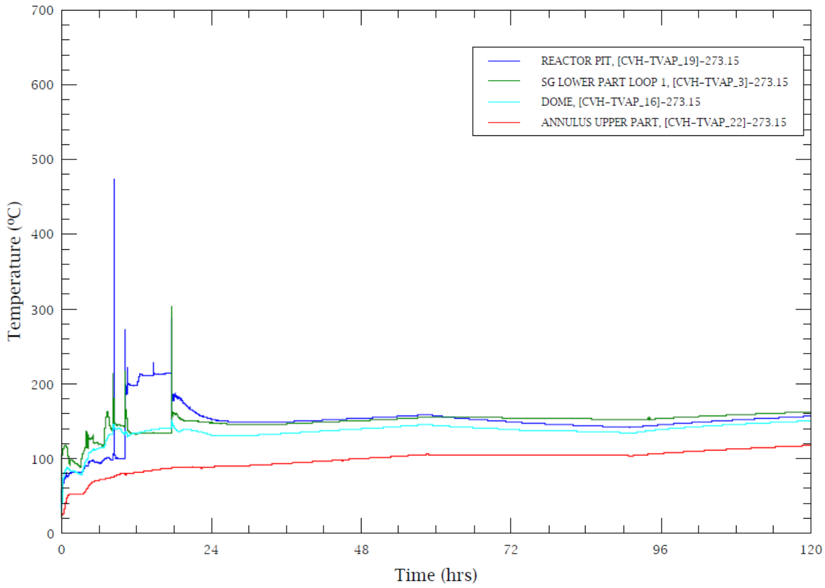

Figure 11: Containment atmosphere temperatures (long-term)

Figures 12 and 13 show that the depressurization of the RCS causes the containment pressure to rise to 1.84bar. After some decrease, during the time when subcooled mass flow through the rupture takes place, the pressure rapidly increases from around 3h. In Figure 13, in particular, the pressure begins to decrease after reaching 6.3bar due to the actuation of the ventilation system (FCVS), whose postulated set point is 6bar for loading and 3.5bar for unloading, as mentioned before. According to the figure, the FCVS is deactivated at 91h6m, at which point the pressure starts to increase again. Of course, the pressure would have steadily increased if there were no FCVS in the NPP so as to endanger the integrity of the containment.

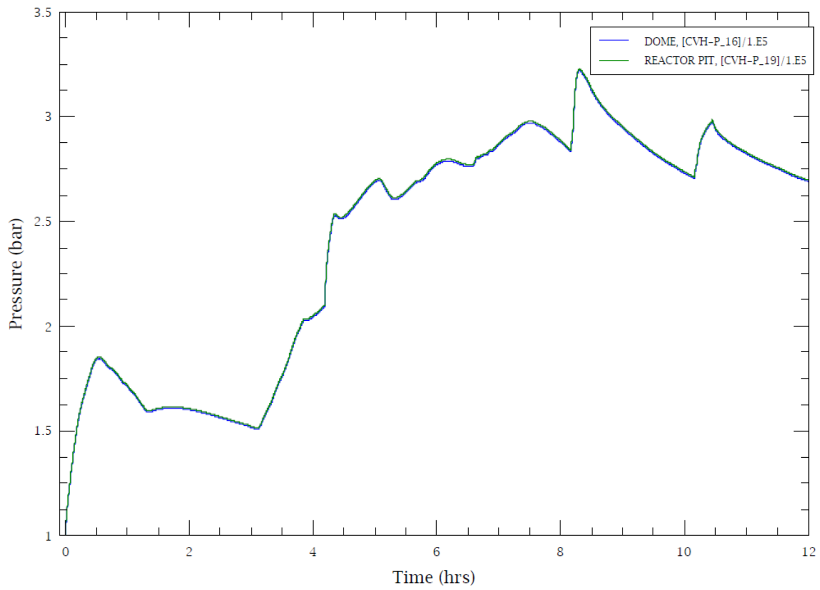

Figure 12: Containment Pressure (short-term)

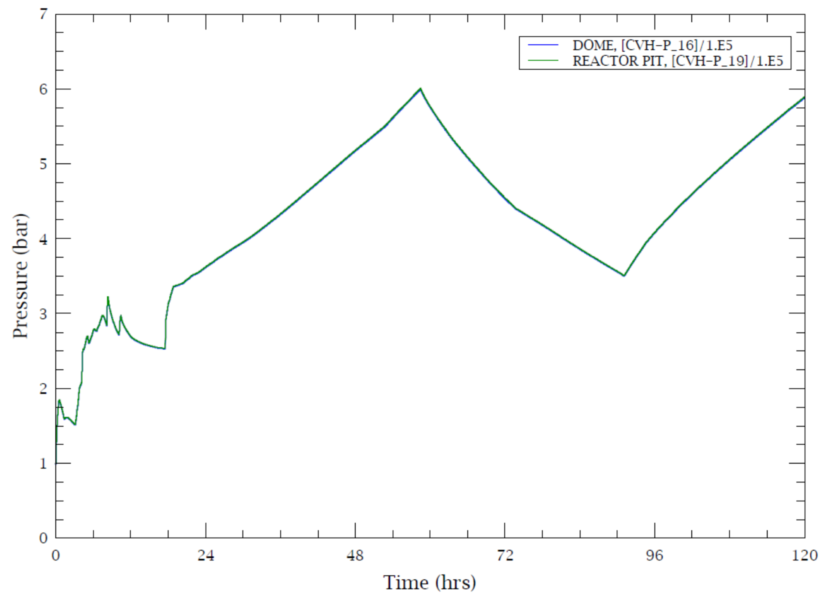

Figure 13: Containment Pressure (long-term)

The results of the simulation are summarized below in Table 2.

Table 2: Summary of results of the scenario

Event / Quantity	Time (hh:mm:ss) / Mass (kg)
Start of core heat-up	05:50:07
Start of core degradation	06:26:40
Start of hydrogen production	06:13:30
End of core degradation	08:16:20
RPV failure	10:09:35
Start of MCCI	10:09:50
Containment pressure > 6.0bar	58:24:10
Produced hydrogen mass until RPV failure (kg)	732.97

Furthermore, by way of comparison, the temperature and pressure inside the containment that had been calculated without PAR are shown in Figure 14 and 15, respectively. As already mentioned above, the graphs of the temperatures of Figure 14 are much messier than those of Figure 10, due to the deflagrations of hydrogen that take place without the mitigating measure; and an additional sharp peak of pressure is observed in Figure 15 at around 9h10m.

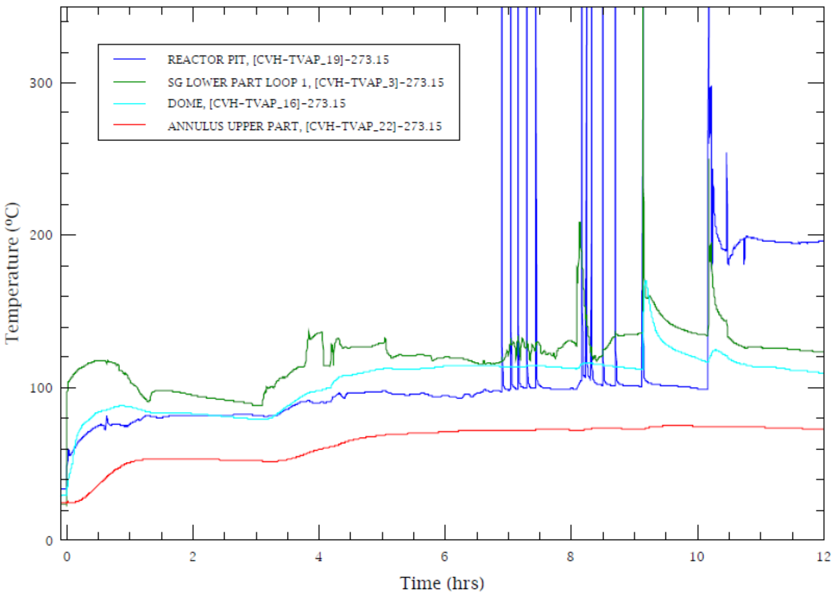

Figure 14 Containment atmosphere temperatures without PARs (short-term).

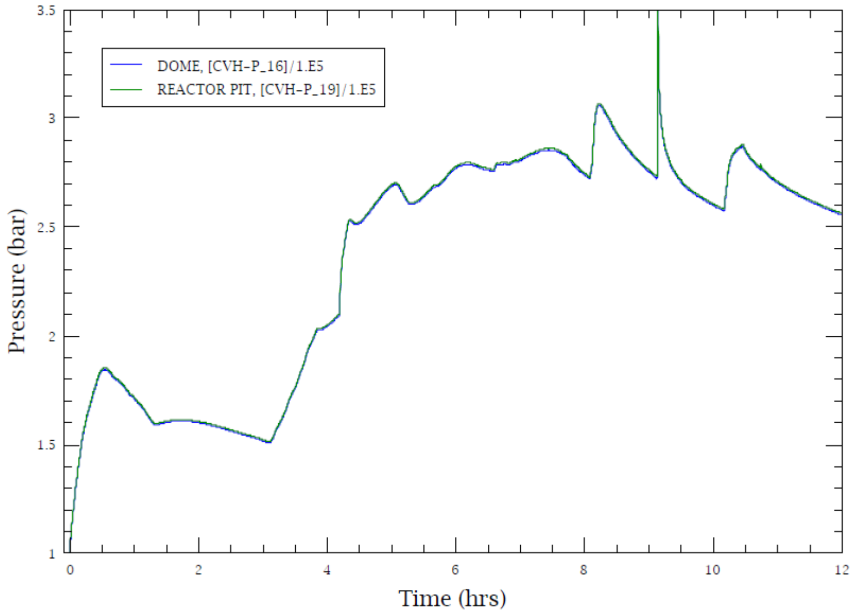

Figure 15: Containment Pressure without PARs (short-term).

3. CONCLUSIONS

The simulation has demonstrated that the postulated rupture on a cold leg and the imposed boundary conditions lead the considered NPP to the core meltdown and subsequent events and phenomena such as RPV failure, generation of considerable amount of hydrogen and its deflagrations, the MCCI, the overpressure inside the containment, and so on, which are characteristic of a severe accident. The simulation has also verified the efficacy and effectiveness of two measures of severe accident management, the PAR and FCVS, in mitigating the consequences of the accident. Further improvement in model, such as subdivision of the reactor cavity, will certainly produce more reliable and realistic result; also, implementation of more data of Angra 2 is still necessary in order to use this model as a proper tool for the independent safety analysis of this NPP. Finally, the uncertainty analysis using tools like SUSA is suggested as a next step.

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